WM'95
Conference Proceedings
"HLW, LLW, Mixed Wastes and Environmental Restoration --Working Towards A Cleaner Environment"

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WM'95 PREFACE

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For those needing more than the CD's, we can copy a limited number of full papers from author provided hard copy, reprints may be purchased and we will publish the abstracts and provide a copy to each registrant. Later, we will evaluate putting the proceedings on the "Information Highway".

We are most optimistic about electronic publication but do anticipate problems and questions. Your indulgence and cooperation will be most appreciated.

Roy G. Post
Editor

Session 01 -- Plenary
Co-chairs: R. Theenhaus, Forschungszentrum Juelich GmbH;
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1-1
THE EUROPEAN ORGANIZATION FOR THE CHERNOBYL'S NEW SHELTER
Colette Lewiner
Alliance consortium
Chairman of the Board and CEO of SGN (FRANCE)
ABSTRACT
Eight years ago, the fourth unit of the Chernobyl nuclear power plant was devastated by an explosion. This accident deeply undermined public perception of the nuclear industry as a safe source of energy. Faced by an urgent need to enclose the highly contaminated reactor, Ukrainian authorities succeeded in building in six months a

shelter around it. In the early nineties, the issue of constructing a second shelter to enclose the first one was raised. In 1992, a competition of ideas named Kiev 92 was organized by the Ukrainian government in order to attract suggestions from all over the world on the issue. No outright winner was selected but the Resolution consortium led by Campenon Bernard SGE was second and five other finalists were selected. Soon after, in April 1994, the European Union decided to launch a call for tenders on a feasibility study of the second shelter. It was won by Alliance, a new consortium, comprising of a group of finalists including Campenon Bernard as well as AEA Technology, Bouygues, SGN joined by Walter Bau AG and Taywood Engineering Ltd. Alliance commenced work in September 1994. The project is aimed at assessing the current situation of the first shelter, Ukritiye , and evaluating options for a new shelter. These two fundamental goals have two additional objectives : the Ukritiye dismantling strategy and the management of waste generated by the construction of a second shelter. This is why the project organization is based on four technical task groups working respectively on the stabilization of the Ukritiye, the design of a second shelter, waste management and dismantling. A fifth group, the technical support group, provides a technical consultancy role to the technical task groups. The feasibility study is divided into three phases : first, the evaluation of various options against criteria to be determined by the study. Secondly, developing the options previously approved into a concept definition report on the basis of the recommendations of phase 1 and the advice arising from a consultation process. Thirdly, conducting a validation process among various national and international authorities.

The feasibility study represents a significant step towards the construction for a new shelter for Chernobyl. This momentum must not be lost. Alliance gathers companies that have been working on the new shelter project for the past three years. The expertise gained on the site together with the good working relationship developed with Ukrainian and Russian specialists and the European and G7 funding decided in Corfu and Naples constitute a good grounding for future work.

INTRODUCTION

The Chernobyl accident, with the lives it cost and the damage it caused to the environment, has deeply undermined public trust in RBMK reactor technology and has by extension questioned the reputation of the whole nuclear industry. Since then, much has been done within the nuclear community in order to ward off the risk of a new Chernobyl : RBMK reactors today are undergoing programs of upgrading. More generally, in many countries, nuclear industries have displayed large efforts in order to fight a general public distrust by demonstrating the effectiveness of their nuclear safety approach and developing new methods in related fields (emergency planning, public information, etc.) Today, one of the major challenges on the nuclear industry is to demonstrate our ability to clean up contaminated sites and to treat and safely store nuclear waste.

The Chernobyl accident is now nine years behind us and an assessment of the situation is in process. Our objective, in this paper, is to review briefly past events as well as to describe recent initiatives taking place with respect to the Chernobyl site and specifically the damaged reactor N4.

HISTORICAL BACKGROUND

On April 26, 1986, an explosion occurred in reactor 4 of the Chernobyl power station followed by a fire. The reactor core, primary containment and enclosing structure were all severely damaged. Radioactive materials were projected outside the reactor and released to the environment resulting in an extended contamination of the environs. In the reactor hall open to the sky, highly radioactive and contaminating materials were remaining, which required the urgent reconstruction of a new containment. By December 1986, the damaged enclosing structure was incorporated into a shelter known as the Ukritiye or sarcophagus . It was intended that the Ukritiye should have a design life of at least 30 years. However, it was soon raised that the structure might be unstable.

In early 1992, soon after the independence of Ukraine, a French scientist went to visit the Ukritiye. He came back convinced that France should offer its help to the construction of a new protective shield to be installed around the first one. During the spring, a French civil engineering company, Bouygues, was working with SGN on a proposal for a new Shelter with the cooperation of Ukrainian and Russian organizations and institutes

Given the complexity of the issue and the risks involved, Ukrainian authorities

organized a competition for the project aiming to attract from all over the world the most original and the most advanced solutions. The competition named Kiev 92 was officially launched on July 8, 1992 in Kiev. The winner of the competition was supposed to be awarded the contract for the project.
Proposals were accepted until late April 1993. In total, 394 proposals were presented from Ukraine, Russia and from countries all over of the world. They included impressive proposals from major engineering groups in the world as well as some more modest ones with highly imaginative concepts. At the end of the selection process, on June the 18th, no single proposal was considered as providing a comprehensive solution, therefore, the Jury declared that there would be no winner. Yet, six final candidates were selected by the jury as those representing the best ideas and solutions for a second shelter. These finalists were :
Resolution: (French consortium led by Campenon Bernard SGE) was the best ranked solution : a prestressed concrete building with prefabricated elements remotely assembled above Shelter one.
336: (German consortium led by Kraftanlagen) : a ferro-concrete building incorporating a metallic structure poured in situ with protective screens interposed
Rainbow: (French consortium led by Bouygues and SGN) : a prestressed concrete containment assembled away from the reactor and moved onto it.
Greenleaf: (British consortium led by AEA Technology) : a two layer steel containment built away from the reactor and moved above it.
Monolit: (Russian consortium led by VNIPIET) : concrete filling all the plant's compartments and voids in the sarcophagus
Pluton: (Russian and Ukrainian consortium led by theKurchatov Institute) : a second sarcophagus using a construction similar to Shelter one.
After having provided technical support to the Ukrainian authorities during the Kiev competition, the European Union decided to fund a first study on the reactor 4. A call for tenders was organized in April 1994 by the DG I in the framework of its TACIS program. It called for the conduct of a feasibility study of the Shelter II project and invited to tender the Kiev 92 finalists as well as other European organizations. It was during this period that Alliance came into being. Alliance is a consortium of leading European organizations in the field of nuclear construction and engineering technologies. Alliance consists of the French company who lead the competition of ideas, Campenon Bernard SGE and three companies who were jointly awarded third place in the competition, AEA Technologies, Bouygues and SGN. The Alliance consortium is reinforced by the leading German consortium company Walter Bau AG, and the British consulting engineers Taywood Engineering Ltd.
Alliance was formally awarded the contract in August 1994 and commenced work on the feasibility study in September. In addition to the European organization, Alliance consists of a number of Ukrainian and Russian organizations and institutes. It was recognized during the early stages of the Kiev 92 competition that any feasible and lasting solution to the unique problem in Chernobyl would require a close working relationship with Ukrainian expertise and knowledge. In essence, an alliance of skills and ideas focused upon a single objective. The organizations working with Alliance comprise the Ukrainian Academy of Sciences, the ISTC Shelter, the Kurchatov Institute and the Ukrainian Association of Constructors. This close alliance between Ukrainian, Russian and European organizations has and continues to offer the optimum means of providing a feasible and lasting solution to the Chernobyl legacy.
DESCRIPTION OF THE STUDY
The scope of the study is to determine the optimum concept for protecting the environment from uncontrolled releases of radiation and nuclear materials [...] from the damaged remains of reactor 4 inside the current shelter . In practical terms, the above objective consists of two fundamental requirements : the assessment of the current conditions and the evaluation of options for a second shelter.
If we refer to the Terms of References we read that The existing shelter is to be strengthened and modified as may be necessary to allow the construction of a new shelter to proceed with minimum danger both from structural collapse and from radiation. This is to include securing the remains of the reactor core . Regarding the Shelter II, it is required that A new, structurally independent shelter is to be constructed to contain both the current shelter and the damaged remains of Reactor 4 . The shelter is planned to have a construction period no longer than 5 years and to last during a minimum 100 years. A condition placed upon the Alliance team is that any design must accommodate the continued operation of unit 3 and also

allow for possible decommissioning operations of unit 4.
Before any new shelter can be constructed over the existing Ukritiye, a number of important issues require careful considerations and assessment. First, the stability of the Ukritiye needs to be assessed to ensure that it will remain a viable shelter until the proposed second shelter can be designed, constructed and commissioned. Secondly, stabilization of the Ukritiye and construction of Shelter 2 should be conceived in relation to each other so that either stabilization of the Ukritiye is guaranteed in the long term, or the second Shelter is designed in order to resist possible future failures of Ukritiye. Thirdly, these two strategies should be consistent in particular with two major issues : the dismantling strategy and management of waste inside the shelter or resulting from all future activities.
Given these general objectives, the feasibility study is to be conducted in three distinct phases :

- Phase 1 : option studies ending in February
- Phase 2 : concept definitions due in June
- Phase 3 : concept validation and review

During the first phase of the project, Alliance will undertake to collect, assess, review and document information relating to the Chernobyl unit N4. It is clear that a project of this nature will rely considerably upon the availability of information. The assessment and validation of such information will also provide some of its own particular challenges. Phase 1 of the study is concerned with the generation of options for the design of a second shelter. These options will require evaluation against agreed criteria and will also need to consider the requirements for the eventual decommissioning of unit N4. During phase 1, assessments will be undertaken to review, and where possible, quantify the relative risks associated with the existing conditions and the proposed solutions. At the end of Phase 1, Alliance is required to prepare a report setting out, on the basis of criteria stated in the Terms of Reference, its recommendations and the reasons for those recommendations.
During the second phase, Alliance will review and amend if necessary the recommendations set out in Phase 1. It will also integrate the results of the presentation and consultation process and develop the options previously approved into a comprehensive concept definition report.
Phase 3 will be a validation process for Phase 2 : concepts previously developed in the previous phase will undergo various peer reviews. In the end, Alliance will produce a final report containing all the findings and recommendations of the feasibility study.

PROJECT ORGANIZATION

It is recognized that one of the objectives of the study is to secure a solution for a second shelter which can be demonstrated to be effective on technical, financial, environmental and safety grounds. The project teams are based in a single office location in France and comprise a core team of twenty engineers and specialists. Since the beginning of the study, engineers and experts from various countries have been in an ideal position to exchange information and ideas. The manager of the project is Jean-Louis Le Mao from Campenon Bernard. The study during the first phase is organized into five groups as follows :

- stabilization of Ukritiye
- design of a second shelter
- Waste management
- Dismantling
- Technical support group

It is these 5 groups who will be responsible for establishing the foundations on which the concepts will be developed during the subsequent phases of the study. In support of these groups as already stated will be a number of Ukrainian and Russian organizations who will also provide technical input into the study. The role of the technical support group in the above is to provide a consultancy role to the four main groups; the group comprises of specialists in the fields of nuclear safety, civil engineering, radiological protection, pricing, scheduling and risk analysis. In addition, it is the responsibility of this group to manage the information collected and produced during the project. The technical support group is also responsible for the establishment of the design criteria to be adopted for the second shelter. These criteria will be reviewed and ultimately approved by the Ukrainian safety authorities.

A BRIEF ASSESSMENT OF THE SITE TODAY

The fourth unit of the Chernobyl power plant is extensively damaged. Walls and ceilings in the Central reactor hall were demolished, ceilings in the water separation drum premises were displaced and walls totally collapsed. The premises housing the main circulation pumps oriented to the North were completely destroyed. Two upper floors of the de-aeration stack were demolished, the columns of the building frame were shifted to the side of the turbine room by up to 1.1 meter. The ceiling in the turbine room was destroyed in many places by fire and falling debris, several building beams were deformed and building frame columns were displaced by the explosion wave.

The reactor emergency cooling system was completely destroyed from the North side of the reactor building and buried under debris from the construction elements. The upper plate of the Elena reactor biological shield and joint elements, weighing 2,700 tons together, were thrown on the rib with an almost vertical angle of inclination. The reactor base dropped 4 meters lower opening a space through which melted fuel elements from the reactor could flow and invade the maintenance corridors in the sub-reactor space as well as the rooms of the lower part of the reactor block.

The central reactor hall is filled with debris from the destructed reactor core (fuel assemblies, graphite blocks as well as metal and building constructions). Also in some places are materials which were thrown from helicopters in 1986.

The construction within six months after the accident of a shelter in order to encase the damaged reactor was undoubtedly an impressive achievement. The main parts of this structure are a separating wall, the Pioneer wall , the cascade wall, the West wall as well as several roofings.

The separating wall was built in order to isolate the operating third unit from the damaged unit. It was erected by filling a transport corridor up to the + 12 m mark in the Northern part and no higher than the + 5m mark in the Southern part. The Pioneer protective reinforced wall was erected along the perimeter of the damaged unit. The North protective cascade wall was constructed of concrete in the form of projections. The West wall was sealed from the outside by a wall with buttresses 50m high that were manufactured away from the reactor, erected and moved against the wall.

The roofing of the Central hall was made of thin steel plates covering a row of tubes based on metal beams installed along the central hall. Roofing on the Northern and Southern sides of the central hall are made from large sized metal shields. New roofing was constructed over part of the ruined turbine hall. To date it is known that voids within the Ukritiye still exist.

Waste in the reactor can be divided in four main elements :

- spent fuel assemblies originally stored in the storage pool
- fuel containing masses (lava): a mix of concrete, fuel, steel and zirconium from the melted core
- high dispersal dust
- contaminated water

Concerning the nuclear safety of the fuel containing materials in the Shelter, the Ukrainian authorities have concluded that not all experimental studies and calculation confirm guaranteed subcriticality of these materials located in the encasement, although the probability of a chain reaction is limited by natural and artificial barriers. A concern in this respect would be water penetrating into the Ukritiye. However, it is clear the criticality issue still remains a concern.

There is a lot more information to obtain in order to have a complete assessment of the site today. Retrieving and analyzing this information is one of the main objectives of the Alliance feasibility study.

TECHNICAL APPROACH OF ALLIANCE

The provision for an optimum solution for a second shelter is an extremely difficult challenge, as anyone who knows of the problems at Chernobyl will confirm. It is also evident that there is no single solution to the problem; a number of solutions could be considered as feasible. The approach of Alliance is based on the requirements of both TACIS and MinChernobyl which gives some very clear objectives.

As a start, Alliance considered the present situation regarding the stability of the Ukritiye and the location of radioactive debris both within and around the structure needed to be fully characterized before any decisions could be taken as to long term strategies. As an example, the problem of leakage through foundation slabs

into the ground water needs exploration.
The difficulties in construction on the heavily contaminated site also need to be identified as well as their consequences on building methods and cost. For this general purpose, the consortium has secured the cooperation and support of a number of Ukrainian and Russian organizations with the best knowledge of the state of the Ukritiye.
Criteria to be used in the study pay special attention to :
- stabilization of the structure
- radiological exposure
- costs
- the degree of disruption of the regular operation of the third unit
- Ukrainian supply of skills and equipment
- future technological progresses

Compliance with emerging Ukrainian regulatory framework is also a significant aspect of the study.

STABILIZATION OF THE UKRITIYE

The establishment of existing conditions within the Ukritiye is considered an essential and fundamental objective of Phase One. It concerns ground conditions (ground water levels, contamination etc.), structural stability, radiological conditions, site infrastructure and local working methods and resources.
This data is to be used in establishing the stability of existing structures, proposing concepts of remedial actions. More generally, this data is crucial in the conduct of other activities in the project, in particular the concept feasibility planning of a second shelter, the decommissioning planning and assessment and radioactive waste management, storage and disposal.

CONSTRUCTION OF A SECOND SHELTER

The concept for the design and the construction of a second shelter is required to provide an environmentally safe system and also be compatible with any derived longer term strategies for Unit N4. During Phase One, the technical team are undertaking a review, assessment and evaluation of Kiev 92 entries based upon agreed criteria. A large number of factors are considered including reactor 3 containment, waste management and decommissioning strategies, seismic criteria, ventilation systems, etc.
In Phase II, this team is responsible for the provision for general schematic designs of the recommended concept solution. These drawings are to be supported by schedules and programs of work describing the realization of the second shelter. Method statements together with procurement strategies are to be produced. All aspects of the recommended solution will be quantified in respect to safety, costs, risks and resources.

DISMANTLING

The dismantling team are evaluating Top-Down and Bottom-Up retrieval methods, in other words, methods of dismantling starting from the roof of the building or on the contrary from the ground level. It is recognized that some dismantling operations may be required in the current stabilization of Shelter I and in the construction of Shelter II. This team provides technical expertise to the Stabilization and Shelter II teams during the study. It will give special consideration to factors such as radiological safety, radioactive waste inventories, availability of remote and semi-remote technology, waste disposal routes and base budgets.

WASTE MANAGEMENT

The Waste Management team will undertake the study and evaluate the options for recovering, processing, conditioning and storing the radioactive waste arising from the Ukritiye during the construction of the second shelter. It is also to look at the waste management requirements associated with stabilization of Shelter I and the Shelter II. This will require the development of a basic waste management strategy. During Phase 1, some assessment of waste volumes and types will be undertaken together with a review of process requirements and plant capacity.

CONCLUSION

After a succession of selection processes that lasted almost three years, the feasibility study conducted by Alliance represents a key step towards the remediation of the Chernobyl issue. Both an assessment of the site and an evaluation of options for a new shelter are underway. Beyond purely technical matters, the study is required to give consideration to other issues such as environment, costs,

funding, operation of other Chernobyl units, scientific progress and Ukrainian contribution to the future project. Clearly, the Chernobyl remediation project will be closely tied to the economy and the financial sector, to the legal and regulatory system as well as to large scientific projects. In addition, the project presents several exceptional features :

- it is unique references to this project are rare and limited
- it is located in an emerging country with infrastructures that are continuing to develop
- it entails unprecedented sharing of responsibility
- it will require an exceptional financial structure

These features contribute to the unusual character of the Chernobyl project and explain the prudence of companies and institutions involved. Yet, the accident is now eight years behind and further steps need to be taken in remediating the situation.

Last June, at their Corfu summit, European Union leaders have agreed on a program of assistance and loans to help Ukraine complete and upgrade three VVER-1000 reactors in order to shut Chernobyl reactors and to close the site. Less than one month later, at the summit in Naples, the G7 partners decided to add new actions on top of the European initiative. These two institutions have thus demonstrated their will to help Ukraine deal with its energy problem.

This momentum must not be interrupted and the Chernobyl shelter issue should not drop from the world's agenda. The key elements are gathered to start the project. Alliance has made steady progress in establishing the basic requirements and has a significant expertise to provide in view of its implementation.

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RADIOACTIVE WASTE MANAGEMENT - AN INTERNATIONAL VIEW

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ABSTRACT

The management and disposal of radioactive waste is a matter of great concern in many countries. Even if experience of radioactive waste management and disposal exist, disposal of high-level radioactive waste still remains to be implemented. International cooperation between countries and in international organizations can help in reaching acceptance of radioactive waste management and disposal. Openness and peer review may increase credibility.

INTRODUCTION

Nuclear power has been used to produce electricity for many decades. An inevitable result of nuclear power production is the generation of nuclear waste.

The management of radioactive waste has been, and will continue to be an issue under debate for at least as long as the final disposal of, above all, high-level radioactive waste and spent nuclear fuel have not been achieved and the safety of the disposal of such waste has been convincingly demonstrated. It is a fact that this has not yet been achieved in any country.

The safe management and disposal of radioactive waste and spent nuclear fuel is a prerequisite for the full acceptance of nuclear power production.

Although radioactive waste management is mainly a concern for countries with a nuclear power program, it is also an issue for other countries using radioactive isotopes in research, medicine and industry.

The problem of radioactive waste management is shared by all of these countries and, as we live in a world where international contact and communication are increasing, there is an obvious need for a common basis for solving the problem. Cooperation is needed not only to share resources and know-how but also to improve credibility and acceptance. International cooperation between countries and through international organizations can help to formulate a common basis for measures to be taken in individual countries and enhance public acceptance of national programs.

RADIOACTIVE WASTE MANAGEMENT PERSPECTIVES

Radioactive waste management covers a multitude of disciplines that extend over different activities such as pretreatment, treatment, conditioning, interim storage,

transportation and the final disposal of radioactive waste. These activities should be viewed as separate parts that integrate into one system. Each activity should be seen in relation to each other so that the overall system can be implemented in a logical and balanced manner. All steps must lead to the final goal - the safe disposal of radioactive waste. Emphasis should also be put on the minimization of radioactive waste as well as on the recycling and the reuse of materials with a very low radioactive content (1, 2).

The release of very low-level radioactive waste to the environment is normally accepted. However, the basic concept for radioactive waste disposal is to isolate the waste, for example, by incorporating the radionuclides in matrices such as cement or bitumen and by the encapsulation of high-level radioactive waste and spent nuclear fuel in highly durable canisters. The repository, be it a repository for high-level or low- and intermediate-level waste, will normally be designed as a multi-barrier system in which the waste form is one of the barriers.

Different countries may choose different options: the reprocessing of spent fuel or the direct disposal of spent fuel; shallow ground burial of low- and intermediate-level radioactive waste or the disposal of such waste in rock caverns; the construction of repositories for high-level radioactive waste or spent fuel in granitic rock, in clay, in salt or in other geological media. Although each of these options is different, they all share the same need for basic knowledge in scientific and technical areas. The methods for assessing the long-term safety of a repository are basically the same, regardless of the waste disposal method.

Performance assessment methods are being developed as tools for assessing long-term safety and for demonstrating compliance with safety criteria. However, they can also be of use in identifying areas where further research and development work is needed.

For many years, the assessment of the impact of radioactive waste management activities has been an issue of great concern in many countries. The approaches used in assessment of environmental impact are currently being extended beyond the impact which can be directly linked to radioactivity to include also other possible impact on the environment. Environmental Impact Assessment (EIA) is a well-known concept in the USA and in many other countries, whereas in other countries it is less well-established. However, in Sweden and in other countries of the European Union, this concept is becoming increasingly important.

The management of radioactive waste covers many scientific and technical areas. There are other concerns which are also important. Public involvement is an important part of the Environmental Impact Assessment process. Although this process may be complex and time-consuming, it carries with it the hope that, in the end, not only will acceptance be gained from the technical and legal systems but also from the public.

The need for assessing the impact of a nuclear waste repository over the very long timescales involved is not only a scientific and technical challenge, it also entails ethical considerations related to the impact of the repository on future generations. The uncertainty that is an unavoidable consequence of the long timescales involved must not only be recognized but also treated in a way that provides a basis for credible impact assessments. Ultimately, impact assessments are largely based on the understanding and judgement of experts (3,4). If we maintain a policy of openness including the publication of findings which are subjected to peer review and international cooperation, we can have reasonable assurance of making well-founded, balanced decisions within this field (5).

EXISTING EXPERIENCE OF THE MANAGEMENT OF RADIOACTIVE WASTE

Nuclear power has been used for many decades. In the early days of nuclear power production, radioactive waste management issues were recognized but too little emphasis was placed on these issues for acceptable long-term solutions to be found. Today, concepts which had been previously accepted, such as very simple forms of shallow ground burial and the dumping of radioactive waste into the sea, are questioned or no longer accepted.

We have also had positive experience of the management of radioactive waste which can help us to plan for the future. Much valuable experience has been gained in the areas of treatment, conditioning, transportation and near surface disposal of waste as well as of interim storage of radioactive waste and spent nuclear fuel. Experience in these fields has been gained in many countries, for example, France, Germany, Spain, Sweden, the UK and the USA (6,7).

Regardless of the nature of the experience that has been gained within the management of radioactive waste, lessons can always be learned.
At the same time, we need to move ahead within this field. The interim storage of radioactive waste and spent nuclear fuel is only a temporary solution. A final solution must be found. In general, near surface disposal is accepted for low- and intermediate-level waste and deep geological disposal is regarded as the most promising solution for high-level waste and spent fuel. New concepts such as the partitioning and transmutation of radioactive waste are also being explored, even if there is skepticism about whether these alternatives are realistic or even desirable.

NATIONAL AND INTERNATIONAL ACTIVITIES

The general understanding is that it is the responsibility of each country to take the necessary steps for the management and final disposal of radioactive waste and spent fuel from national programs.
Comprehensive R&D programs on radioactive waste management and final disposal are being carried out in many countries. Options under consideration and methods for implementation and demonstration of safe disposal of radioactive waste can be:

- Reprocessing or direct disposal of spent fuel
- Disposal concepts
- Geological media for the construction of a repository: salt, clay, granitic rock etc.
- Site selection methodology, technical criteria or volunteerism,or a combination
- Performance assessment methodology
- Etc

There is no single answer to the question of which option should be chosen. The conditions which exist in each country as regards the scope and nature of the nuclear power program, the geological media etc. will be of importance. Each country has the responsibility for managing its radioactive waste and the solution chosen for a particular country must take into account the specific conditions which exist within that country.
Even if the management of radioactive waste is a national responsibility, there is a recognized need for international cooperation. Cooperation between individual countries or in international organizations can be effective in utilizing the know-how and facilities which exist and can also help to harmonize views and criteria.

A NATIONAL PROGRAM FOR RADIOACTIVE WASTE MANAGEMENT

The Swedish program for radioactive waste management will be presented here as an example of a fairly comprehensive national program (8). The Finnish program for radioactive waste management is quite similar both as regards the disposal concept and the legal basis. Other countries have comparable programs.
The Swedish nuclear power program comprises twelve nuclear reactors - three pressurized water reactors (Westinghouse) and nine boiling water reactors (ABB Atom). Sweden has decided to phase out nuclear power no later than the year 2010. This means that 7,000 tonnes of spent nuclear fuel will have to be disposed of.
Under Swedish law, the responsibility for radioactive waste management and the decommissioning of the nuclear power reactors lies with the reactor owners. The reactor owners must conduct an R&D program within this area in order to live up to their legal obligations. The reactor owners must also provide the funds required to cover the future costs of nuclear waste management. The reactor owners have formed a jointly owned company, The Swedish Nuclear Fuel and Waste Management Co. (SKB), in order to fulfill their legal obligations within this field. In Sweden, there is general agreement on the direct disposal of spent nuclear fuel.
Every three years, SKB must submit its R&D program to the Swedish Nuclear Power Inspectorate (SKI). SKI reviews the program and submits its evaluation and recommendations to the Government. The program has since long time focused on the KBS-3 waste disposal method, which involves the encapsulation of spent fuel in copper/steel canisters and the emplacement of the canisters, surrounded by bentonite, in drilled holes in a system of tunnels at a depth of about 500 m in granitic rock.
SKB has proposed that the encapsulation facility should be co-sited with the already existing central interim storage facility for spent fuel (CLAB) at the Oskarshamn nuclear power plant. CLAB has been in operation for ten years. 2 000 tons of spent fuel is now stored in the facility. Discussions with the authorities and with the

local municipality and community are already in progress.
SKB is now at a very early stage in the process for selecting a site for the repository. According to the time-schedule, the construction of the repository will start within ten years from now. According to SKB's plans, the first canister will be deposited in the repository in the year 2008. The repository will be constructed in stages. The first stage is intended to demonstrate handling and disposal methods and will include the possibility of retrieving the canisters, if necessary. This first stage may be important not only for validation of the technical aspects of the concept but also for assuring public acceptance and approval of the repository.
The repository for low- and intermediate-level radioactive waste from the nuclear power program is a rock cavern (SFR), situated close to the Forsmark nuclear power plant, and has been in operation since 1988 (7). The capacity of SFR, 60 000 m3, will probably be sufficient for all operational waste from the Swedish nuclear power program. So far, about 15 000 m3 of waste has been emplaced in the repository.
Since 1985, a transportation system, comprising a purpose-built ship, Sigyn, has been in operation for the transportation of spent fuel and radioactive waste. All existing nuclear facilities - the nuclear power reactors, the central interim storage facility and the repository for low- and intermediate-level waste - are located at the coast.
The Swedish nuclear waste management program includes all activities and facilities needed to fulfill the requirements under Swedish law, i.e. the final disposal of radioactive waste and spent fuel, including the decommissioning of the reactors.
The progress in the Swedish nuclear waste program may besides the scientific and technical qualities of the program be attributed to the existence of regulatory system which clearly defines the responsibilities of the nuclear industry and society. Even if there has been some opposition against nuclear waste facilities the general public has accepted these facilities. However the most critical steps remain to be taken, the siting and licensing of the repository for spent fuel.

INTERNATIONAL ACTIVITIES IN RADIOACTIVE WASTE MANAGEMENT

As previously mentioned, international cooperation between individual countries as well as within international organizations is necessary. With regard to radioactive waste management, two international organizations with many activities in this field, the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) within the Organization for Economic Co-operation and Development (OECD), will specifically be discussed. However, there are many organizations which have substantial programs covering areas of importance for radioactive waste management.

THE INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)

The IAEA program on radioactive waste management issues covers many aspects such as the development of safety standards, the development of principles and criteria, technical assistance and advice to developing countries, services and publications on "state of the art" of technology experience in Member States to countries which request such services. An important service offered by the IAEA is the performance of international peer review services on aspects of radioactive waste management programs of its Member States. Such services are provided within the Agency's Waste Management Assessment and Technical Review Program, WATRP.

Radioactive Waste Management Safety Standards (RADWASS)

This program comprises work to develop safety standards for different aspects of waste management, principles of radioactive waste management, a national system for radioactive waste management, pre-disposal management, near surface disposal, geological disposal, decommissioning and waste from mining and milling.
The principles of radioactive waste management, also referred to as the safety fundamentals, comprise the very basic principles of the protection of human health and the environment, as regards present and future generations within and beyond the borders of the country as well as requirements on controlling radioactive waste generation and the safe operation of radioactive waste management facilities.
The safety standard "Establishing a National System for Radioactive Waste Management", defines responsibilities for the establishment of a legal framework and a regulatory body. It also defines the responsibilities of waste generators and operators of facilities.
In addition to the standards, safety guides and safety practices will be established to address relevant issues in more detail.
The RADWASS program was set up in 1990 and many safety standards, guides and technical documents are now under preparation. The RADWASS program is reviewed by an

International Radioactive Waste Management Advisory Committee (INWAC). The INWAC members are senior experts in different fields of waste management, regulators as well as implementors. INWAC also serves as the expert group that provides guidance and advice to the IAEA on the direction and implementation of the Agency's radioactive waste management program. Also, under the new terms of reference for INWAC, the Committee will consider global waste management issues and publish their opinions and recommendations on how such issues may be overcome.

There is also an INWAC Sub-group on Principles and Criteria. This Group discusses, as the name indicates, principles and criteria for radioactive waste management. Discussions concern, for example, safeguards requirements for final disposal as viewed by experts engaged in the safety aspects of disposal, the optimization of radiation protection in radioactive waste disposal and time-frames and safety indicators in assessing the safety of the final disposal of radioactive waste. This INWAC Sub-group serves as the forum for the discussion of issues on principles and criteria as related to waste disposal where international agreement has not yet been reached. The Sub-group usually publishes its finding when agreement is reached.

International Convention on Radioactive Waste Management

The RADWASS documents "The Principles of Radioactive Waste Management" (9) and "Establishing a National System for Radioactive Waste Management" have been very carefully examined since they are also intended to provide the basis for a planned international convention on radioactive waste.

According to plans, the two documents will be approved by the IAEA's Board of Governors in spring 1995. According to present planning, preparatory work will soon be started to establish a text for an international convention on radioactive waste management.

This convention will be a complement to an international convention on the safety of civil nuclear power reactors (the nuclear safety convention) that was initiated in 1991 at an IAEA conference in Vienna. The nuclear safety convention has been signed by a number of countries and is now in the process of being ratified. The implementation of the convention will be through reporting from the countries on reactor safety issues in accordance with the requirements of the convention and through discussion and peer review of the national reports.

OECD/NEA

Within the NEA, the Radioactive Waste Management Committee (RWMC) is directly engaged in radioactive waste management issues. The work within the RWMC focuses on radioactive waste disposal and performance assessment methodology for long-term evaluation.

A few years ago, the RWMC published, in cooperation with the International Radioactive Waste Management Advisory Committee (INWAC) of the IAEA and endorsed by the Experts for the Community Plan of Action in the Field of Radioactive Waste Management, Commission of the European Communities, a collective opinion document entitled, "Disposal of Radioactive Waste: Can Long-term Safety Be Evaluated?" (3). In the document, the answer to the title question is yes, although it is also noted that the methodology for safety assessment can be further developed and that site-specific data are needed to decide whether a specific disposal system provides the level of safety required.

In the RWMC, discussions are now in progress on a new collective opinion document which will discuss the arguments for deep geological disposal of high-level radioactive waste and spent fuel.

As a basis for the new planned collective opinion document, the RWMC in collaboration with the OECD Directorate of the Environment recently organized a seminar on ethical aspects of the disposal of radioactive waste. The discussions at this seminar very clearly demonstrated that there seems to be a wide international consensus on the general principles of waste management and disposal which has developed over the years.

These very basic principles address the protection of human health and the environment with regard to present and future generations within and beyond the borders of the country as well as requirements on exercising control over radioactive waste generation etc., as they are presented in the IAEA Safety Fundamentals (9).

Another observation from the seminar is the need which has been expressed for finding and implementing methods for the final disposal of radioactive waste so as not to rely on interim storage for long periods of time. The possibility of

retrieving, for a certain period of time, the waste deposited in a repository may help to promote public acceptance of the final disposal system.

CONCLUSIONS

The management of radioactive waste, notably final disposal, is an issue which is currently under debate and which will remain so until the final disposal of high-level radioactive waste and spent fuel has been implemented and acceptance has been gained not only from the technical and legal perspective but also by the public.

It is generally agreed that the management and disposal of radioactive waste is primarily a matter of national responsibility. However, there is a need for attaining an international consensus on this issue. Consensus on standards and criteria for radioactive waste management is essential for gaining the confidence of the public.

It seems that it is possible to reach a consensus on the possibility of implementing deep geological disposal for high-level radioactive waste and spent fuel. However, it should be emphasized that R&D work should be pursued in order to further improve long-term safety.

Openness, peer review and international cooperation are means of gaining confidence. An international convention on radioactive waste management may be beneficial.

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Session 02 -- DOE Deputy Assistant Secretary's Panel
Co-chairs: Ed Helminski, S.I.M./EP;
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DAS PANEL

The Department of Energy's Environmental Management (EM) Program held a special session on Monday, February 27, 1995, addressing the responsibilities of each Office within the EM organization. The session focussed on current program status, plans for the future, and specific changes anticipated over the coming year.

Attendees at the special session were invited to submit questions in writing to the session moderator who posed the questions to panel members.

Below are some of the key points made by panel members during their introductory remarks:

Randy Scott (Deputy Assistant Secretary for Compliance and Program Coordination) discussed his Office's key initiatives, which include better definition of EM projects and decentralization to enable more decision making to be done at the Field Offices.

Jim Turi (Director, Office of Program Integration/Office of Waste Management) discussed the reality of budget constraints and the need to meet waste management treatment and disposal objectives.

Jack Baublitz (Acting Deputy Assistant Secretary for Environmental Restoration) addressed the increased flexibility to the program with pilot demonstrations and further involvement of the commercial sector, including privatization.

Carl Cooley (Technical Advisor, Office of Technology Development) described the

"Technology Focus Area" approach to developing and implementing new technologies in five areas: landfill remediation, plume remediation, mixed waste management, underground storage tank remediation, and facility transition issues.
Andy Szilagyi (Team Leader in the Office of Integration and Assessment/Office of Facility Transition) noted that finding ways to make facilities "cheap-to-keep" is the only way DOE can ensure worker safety associated with monitoring and maintenance of surplus facilities now awaiting decontamination and decommissioning.
Following introductory remarks by each panel member, specific questions from the audience were posed. Some of the questions and answers are provided below:
How will privatization be pursued, and what do you expect to result from the initiative?
Decentralization of decision-making authority will enable the Field Offices to make more decisions locally, including what specific activities may benefit from privatization. One example - remediation of Hanford high level waste storage tanks has been offered to private industry. If this initiative is successful, DOE will benefit from the infusion of a new approach and new technologies into the program, as well as potentially accelerate the remediation of these tanks. The successful private company will gain specific experience which can be applied to other DOE sites, thereby increasing their ability to compete for other work at Hanford and other DOE sites. Overall, we expect to see less reliance on the Management and Operating contractors at the sites.
How will EM safely manage the large number of facilities being transferred from other DOE programs?
EM must find ways to prioritize projects in order to save money. By approaching individual facilities and addressing them from start to finish, EM will avoid the cost of addressing numerous facilities at the same time and taking longer to complete the activity at each of these numerous facilities. Additional focus on the deactivation process is also expected to provide cost savings.
What kinds of technologies should be pursued to save money? Will these be funded exclusively from the Technology Development Program, or from all EM programs? And how will you ensure the "new technology development approach" will work?
It is clear that capping or treating in-situ (rather than exhuming and re-disposing elsewhere) saves money in treatment, transportation, and re-disposal costs. Technologies suitable for remediating or isolating groundwater will also save money, since "pump and treat" is relatively ineffective at restoring an aquifer.
Not all technology-development or technology-application funding is provided by the Office of Technology Development, nor will it be in the future. It is currently estimated that approximately $135 million from the Waste Management program and $50 million from the Environmental Restoration program is directed at waste-specific or site-specific technology activities.
To ensure the new approach is successful, the Office of Technology Development will involve the "user organizations" (i.e., Waste Management or Environmental Restoration) in the Technology Development activity. In this way, technologies will be developed to target a specific need, and they will be more likely to be successfully implemented, since the user will have been part of the development and application process.
Is there a future for "baselines"?
The rigor associated with preparing and living to a baseline is very relevant to managing a successful program. However, DOE-headquarters recognizes that development and implementation of baselines must be made simpler. One possibility may be to have the baseline cover a short time period. This would allow better data to be incorporated and reduce the effort required to incorporate changes as they occur. Baselines remain an inherent part of project management, reinforce that projects have a beginning, middle, and end, and must proceed through each stage in the most cost-effective manner.
What authorities have been delegated to the Field, and what might be delegated in the future?
Approval for Safety Analysis documentation, Operational Readiness Review completion, and most National Environmental Policy Act authority has been transferred to the Field Offices. HQ continues to look for other areas in which delegation to the Field can occur, though we have no specifics to offer at this time. However, to further reduce the impact of HQ's review process on the field, the Office of Environmental Restoration has delegated approval of Records of Decision to HQ Office

Directors, which should reduce the amount of time required for document approval. To ensure that lessons learned at one site are shared with other sites, the role of the "site coordinator" within the Office of Compliance and Program Coordination will be strengthened. It will be the responsibility of the site coordinator to ensure inter-site sharing of information. We also look to the Office of Field Management for additional sharing of information.

What is the status of updating or replacing DOE Orders?

There has been talk of changing DOE Orders to "rules", which would require publication of each proposed rule in the Code of Federal Register. We do not know what the timing of this might be. In the meantime, the "Standards/Re-quirements Identification Documents" (SRIDs) will enable program personnel to feel more comfortable about the requirements for conducting their program in a safe manner. It is expected that Orders will be circulated to Field Office personnel for detailed review prior to being finalized.

What will be the process in renegotiating compliance agreements due to budget constraints?

The Office of Compliance and Program Coordination will play a role in renegotiations as will the program offices. EM is hoping the regulators and the public will recognize that not all objectives can be accomplished within the current budget constraints, and is further hoping that the stakeholders will work with the Department to focus these limited resources on the most significant problems, leaving wastes or contaminants that pose less risk to be addressed later.

Session 03 -- Waste Management Systems Costs

Co-chairs: Ivan Vovk, IAEA;
John Christian, Rust Federal Services

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SYSTEMS COST ANALYSIS AND OPTIMIZATION OF WASTE MANAGEMENT FACILITIES

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ABSTRACT

This paper presents general conclusions from application of a system cost analysis method developed by the United States Department of Energy (DOE), Waste Management Division (WM), Waste Management Facilities Costs Information (WMFCI) program. The WMFCI method has been used to assess the DOE complex-wide management of radioactive, hazardous, and mixed wastes. The Idaho Engineering Laboratory, along with its subcontractor Morrison Knudsen Corporation, has been responsible for developing and applying the WMFCI cost analysis method.

The cost analyses are based on system planning level life-cycle costs. The costs for life-cycle waste management activities estimated by WMFCI range from bench-scale testing and developmental work needed to design and construct a facility, facility permitting and startup, operation and maintenance, to the final decontamination, decommissioning, and closure of the facility. For DOE complex-wide assessments, cost estimates have been developed at the treatment, storage, and disposal module level and rolled up for each DOE installation.

Discussions include conclusions reached by studies covering complex-wide consolidation of treatment, storage, and disposal facilities, system cost modeling, system costs sensitivity, system cost optimization, and the integration of WM waste with the environmental restoration and decontamination and decommissioning secondary wastes.

BACKGROUND

To provide a cost basis for evaluating Environmental Management options, the United States Department of Energy (DOE), Office of Program Integration (EM-33), has sponsored the Waste Management Facilities Cost Information (WMFCI) program, which is supported by the Idaho National Engineering Laboratory (INEL) and Morrison Knudsen Corporation (MK).

The WMFCI program has developed a method that is being used to estimate the planning level life-cycle cost (PLCC) of facilities needed for treatment, storage, and disposal (TSD) of radioactive, hazardous, and mixed wastes. The method is based on

dissecting the overall TSD facilities into several smaller functions, called cost modules, and developing a cost versus capacity relationship for the various cost elements of the modules. Using this approach, the type and input capacity of the TSD modules needed for handling a given input waste must be defined. Then, a PLCC estimate is developed by referring to a set of regression curves that contain the cost versus capacity relationship.

The WMFCI data base includes PLCC estimates for over 200 modules that cover six waste types: low-level waste (LLW), mixed low-level waste (MLLW), transuranic waste (TRUW), greater-than-Class C (GTCC) and DOE equivalent special case waste, hazardous waste (HW), and spent nuclear fuel (SNF). For each module, cost versus capacity relationships are provided for four different work breakdown structure (WBS) elements. The WBS elements are: preoperation, facility construction, operation and maintenance (O&M), and decontamination and decommissioning.

The WMFCI cost method provides a documented and flexible modular cost basis that DOE and its contractors can use to analyze what-if questions concerning alternative waste management strategies. This powerful method has been broadly used for assessing life-cycle costs associated with several waste management option evaluation tasks. Applications have included the analysis of DOE complex-wide consolidation of TSD facilities and studies covering system cost sensitivities, cost modeling, cost optimization, and integration of WM waste with the secondary waste from environmental restoration (ER), and decontamination and decommissioning.

The WMFCI cost method has supported DOE strategic planning, environmental compliance, and budget planning. Specific applications have included development of life-cycle cost estimates, staffing estimates for risk assessment, and facility design inputs for the socioeconomic analysis of the Environmental Management Programmatic Environmental Impact Statement; cost guidance documentation for preparation of the Site Treatment Plans required under the Federal Facility Compliance Act; and development and application of the System Cost Model for the Waste Management Baseline Environmental Management Report (BEMR) submittal to the U.S. Congress.

System Consolidation

The WMFCI method was applied during the economic studies of DOE facility consolidation options. This analysis included several alternative scenarios involving a different number and location of TSD facilities in the DOE complex. Options studied were grouped as decentralized, regionalized, and centralized configurations. The decentralized option proposed a dedicated TSD facility for most of the DOE installation. The regionalized option proposed a TSD facility at each major installation. The centralized option proposed one or two TSD facilities serving the entire complex. Some conclusions reached from the system consolidation studies are presented below.

Economies of Scale

The economy of scale from a decentralized configuration to a regionalized or centralized configuration is in the range of -5% to -30% savings for regionalized and -25% to -60% savings for centralized. The greatest economies are seen for MLLW and ER-LLW. (See Fig. 1.)

The greatest economies are based on configurations that regionalize at existing disposal sites (e.g., Hanford) and maximize the use of existing facilities.

Existing WM facilities, particularly under regionalized alternatives, provide significant MLLW treatment and LLW disposal capacity. Because of these existing facilities, fewer new facilities are constructed.

Disposal facilities have greater economies of scale than do treatment facilities because of

- lower fixed administrative costs
- a higher ratio of capital to operating costs (i.e., relatively less O&M).

Transportation Impacts

Transportation costs will offset some benefits of regionalization and centralization, although costs have been relatively small (0.3% decentralized, 1% average, and 5% one-site case) compared with the total life cycle costs. (See Fig. 2.)

Rail transportation costs are 1.5-3.5 times the costs for road transport for TRUW.

Total mileage differences on MLLW (implied transportation risk indicators) between decentralized and regionalized cases (factor of 11), and from regionalized to centralized cases (factor of 7). TRUW total miles are similar for all cases (all

ship to the Waste Isolation Pilot Plant).

Rail transportation may be more suitable for ER wastes because of larger loads (better rates).

Because of relatively low transportation costs, it makes more sense to ship some (special) wastes to more centralized sites.

Impacts of ER Residual Waste Going to WM

ER-LLW disposal costs were 91-97% of the total life cycle cost, indicating that only minor treatment would be required and most wastes would go directly to disposal. ER-MLLW disposal costs range from 65-75% of total life cycle cost. (See Fig. 3.)

Comparatively, the disposal costs for WM legacy and newly generated wastes have been approximately 25-40% of the total life-cycle cost.

Cost Sensitivity Studies

The WMFCI method was used to conduct sensitivity analysis of system costs versus the waste management elements. Conclusions of the sensitivity analysis are as follows.

Privatization of treatment facilities could result in a significant cost savings. For a scenario in which a private sector provides a grass roots treatment facility dedicated to DOE waste, the cost savings could be as high as 30-40%. The savings could be even higher if existing commercial treatment facilities were employed.

System Modeling Studies

A computerized tool, the System Cost Model (SCM), has been developed to apply the WMFCI cost estimation method for system modeling studies. After development was completed, the SCM was calibrated to model planning level life-cycle cost estimates at six major DOE installations: Hanford Site, INEL, Los Alamos National Laboratory, Oak Ridge Reservation, Rocky Flats Environmental Technology Site, and Savannah River Site. System cost modeling conclusions are as follows:

To model system costs adequately, the fixed and variable costs must be clearly understood and delineated. Variable costs are a function of the waste feed rates. Fixed costs are those expenditures incurred even if waste is not treated or disposed of. Figure 4 shows the relationship between fixed costs and variable costs.

Existing aqueous waste treatment facilities may not support future treatment needs of sites. Aqueous waste treatment is generally dedicated to specific waste streams that may be different to support processing future waste streams.

The extent that some high cost WMFCI modules are used may impact cost estimates. The open, dump, and sort; characterization; and receiving modules carry high costs. Waste should be minimized to these operations to reduce costs.

Storage of legacy LLW, MLLW, and TRUW account for a significant portion of the system TSD costs. For the six major installations, the total storage costs are 10-25% of the total TSD costs.

Review of System Cost Optimization Opportunities

Through the WMFCI method and system modeling tool, not only can life-cycle, costs be estimated but also opportunities for costs optimization can also be evaluated. Currently, several waste management operations are being reviewed to identify cost reduction potentials and areas that could be studied in the future. The following are some ideas proposed for such studies:

Maximize the use of existing TSD facilities. Using the WMFCI method, life-cycle cost estimates can be developed for scenarios that make maximum use of existing facilities. This scenario involves analyzing excess capacity and configuring the waste shipments for regional TSD accordingly. Whereas current waste management strategies designate regional treatment sites for all waste streams within a particular waste type category, this modified scenario will consider sending different waste streams to various sites, depending on existing facility capacities. Costs for these modified scenarios can then be compared with the costs previously developed for DOE strategic planning to demonstrate the order-of-magnitude potential for cost savings.

Optimize facility operating periods. The DSTP's reflect a wide range of facility operation assumptions (hours of facility operation per year and the number of shifts). The differences in operating hours will influence the sizing of the facility, and the number of shifts will impact the total work force that must be paid.

Site regionalized and centralized facilities where capabilities exist (e.g., compactable or incinerable materials).

Build disposal facilities as needed versus overbuilding capacity up front.

Use commercial facilities. Many sites are currently evaluating options for commercial treatment versus on-site DOE treatment and disposal.

Build facilities with larger throughput so that the backlog waste can be treated in a shorter time. Savings that result from eliminating backlog waste and shifting those dollars to reduce operations and maintenance cost could prove that this approach is more cost effective than sizing the facilities and processing backlog waste over a 10-20 year operating period.

Combine different waste types to use common TSD facilities and reduce the need for construction and operation of additional waste management facilities. Some waste management strategies under consideration specify separate TSD facilities for each of the major waste type categories (LLW, MLLW, TRUW, etc.). This study will combine waste types, where technically feasible, for treatment, storage, and disposal, and provide cost estimates for these optimized scenarios. Again, the optimized costs will be compared with previously developed cost estimates to demonstrate the potential for cost savings.

Consolidate special waste streams for processing. Small quantities of wastes (e.g., remote handled) at more than one site may be handled best in a centralized location, rather than regional alternatives.

Use treatment technologies that reduce pretreatment sorting and characterization.

Substitute vitrified waste forms (e.g., iron enriched basalt) with shallow land disposal rather than building expensive engineered disposal.

Integration of WM, ER, and D&D Waste

Life cycle activities and waste loads relative to the ER and D&D programs are being defined for inclusion in the Baseline Environmental Management Report. This report will be submitted to the U.S. Congress by DOE in March of 1995. The WMFCI method and system modeling tool are being employed to integrate ER and D&D waste with the WM waste, which is the waste generated during the normal operation of the DOE production facilities. Several options for cost-effective integration of WM, ER, and D&D waste are being considered. Some of these options are discussed below.

Use existing and new WM facilities for processing ER wastes

Use lag storage to allow combined WM and ER waste processing.

Reduce ER (LLW and MLLW) residual waste loads to WM by volume reduction technologies.

Use new ER technologies (e.g., in situ vitrification) that produce less residue for WM to process or dispose.

SUMMARY

The INEL and MK have developed and applied a systems cost estimating approach that has yielded several key waste management cost conclusions. These conclusions have helped to identify the areas of high cost sensitivity that require further evaluation by the DOE to develop effective management policies. The most viable options should also consider risk, socioeconomic, regulatory, and other concerns before adoption.

3-2

A PC-BASED AUTOMATED MODEL FOR ESTIMATING LIFE CYCLE COSTS OF MANAGING MIXED WASTE

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ABSTRACT

The U.S. Department of Energy (DOE) Headquarters and associated contractors have developed an IBM-PC based automated computer model for estimating the life cycle costs of different DOE mixed-waste management alternatives. The model has been designed to estimate the complete cradle-to-grave costs of implementing a waste management alternative. The current version of the model includes capabilities to estimate the cost of managing mixed low-level waste. Capabilities for other waste

types are planned to be included in later versions of the model. The model can estimate the cost of most potential mixed low-level waste management activities beginning with waste characterization before and after treatment, waste treatment by different technologies, pretreatment activities such as inspection and sorting of waste where required, storage of waste, certification and shipping of waste, transportation of waste, and disposal of treated waste in different types of disposal facilities. The model interfaces with a database that is based on the DOE mixed waste inventory projections and facility data. This paper discusses the various features of the automated model related to the estimation of costs of different options for management of DOE's mixed low-level waste.

INTRODUCTION

The Automated Systems Analysis Model (the Model), which is being developed by the U.S. Department of Energy (DOE), is an IBM PC-based computer model that will assist users of widely varying expertise in evaluating and comparing options for managing mixed waste generated and stored at various DOE sites. The Model is based on the premise that comparisons between options should include the entire waste handling process, from initial treatment ("pre-treatment") to final disposal because all of these waste management activities are interdependent.

At present, the Model (Version 2.0) provides the capabilities for projecting the life cycle cost and its yearly allocation (schedule) for evaluating options for contact-handled (CH) mixed low-level waste (MLLW). The term "life cycle cost" as used in the Model refers to the sum total cost of four different phases in the life cycle of a waste management facility. These include the preoperations phase consisting of activities such as studies, demonstration, design, etc., the construction phase, the operations phase, and the decontamination and decommissioning phase. The Model could be applied in the future to address public health, worker risk, and the long-term performance of final waste forms in the disposal facility. This paper describes the current version of the Model with reference to its overall structure, user inputs, steps used for estimation of cost and schedule of a waste management option, available formats for presentation of the final cost and schedule results, and also the hardware and software requirements of the Model.

DESCRIPTION OF THE MODEL

Databases Used in the Model

Figure 1 represents the overall structure of the Model. As shown in Fig. 1, the Model includes two databases that are based on the information in the latest revision of the Mixed Waste Inventory Report published by DOE. The information in these databases assists the user in configuring options for a specific waste stream. The Facility Database contains specifications and capabilities for both existing and potentially new MLLW treatment facilities within the DOE Complex. The Inventory Database contains information on the characteristics of MLLW generated within the DOE Complex on a waste stream basis. This includes the latest available information about the existing and projected volumes and masses of each waste stream and the site at which it is generated, waste matrix (e.g., aqueous liquids, organic liquids, soils, etc.), generalized hazardous waste designations based on the presence or absence of hazardous metals, organics, and PCBs, and the radioactive characteristics of the waste stream (i.e., alpha versus non-alpha, contact- versus remote-handled).

(Put Fig. 1 1 here)

User Input Required for Option Configuration

In general, application of the Model involves user input for configuration of an option or a set of options. The user configures an option by accessing the Inventory Database and selecting the MLLW inventory to be considered in the option (e.g., waste stream, group of waste streams, entire site inventory, or entire DOE inventory). The user then specifies the complete waste management train (i.e., the desired sequence of waste management activities) for each waste stream or related group of waste streams included in an option. Specifically, the Model requires user input regarding the following activities:

Primary treatment (user must either select an appropriate facility from the list of existing facilities in the MLLW Facility Database, or in the case of hypothetical facilities, specify a treatment technology and facility location). The Model can estimate the treatment cost for the following treatment technologies:

- Aqueous waste treatment
- Incineration

- Thermal desorption
- Wet air oxidation
- Vitrification
- Deactivation
- Lead recovery
- Mercury separation and amalgamation
- Soil washing
- Debris washing
- Sludge washing
- Polymer encapsulation
- Cementation

Stabilization (user must either select a stabilization facility from the list of existing facilities in the MLLW Facility Database, or in case of hypothetical facilities, specify a stabilization technology and facility location). The Model can estimate the cost for the following stabilization technologies:
- Grout stabilization
- Polymer stabilization
- Vitrification

Disposal (user must select disposal facility location and the type of disposal facility). The Model can estimate cost for the following types of disposal facilities:
- Above ground disposal in engineered vaults
- Below ground disposal in engineered vaults

Storage (user must specify whether storage is required prior to treatment, between treatment and stabilization, or between stabilization and disposal). The Model assumes that a reinforced concrete structure will be used for storage.

Once the user has configured an option by providing the above inputs, the number of shipments for transportation and the capacity requirements for treatment support activities such as facility administration, receipt and inspection of off-site waste, sorting of waste prior to treatment, maintenance, and certification prior to shipping are automatically calculated by the Model based on the waste inventory for each option and do not require user inputs.

Mass Balance Calculations

After completion of option configuration, the Mass Balance section of the Model performs calculations to determine the total quantity of waste to be processed by each waste management activity at each site. This takes into account the reduction or increase in the waste volume due to either primary treatment or stabilization, as well as the generation of secondary wastes on the basis of treatment technology characteristics. The results from the Mass Balance section are used as input to the Cost Estimation section of the Model.

Cost Estimation

The Cost Estimation section of the Model estimates the cost of each facility on the basis of its capacity requirements in kilograms per hour (except for storage and disposal cost, which are based on cubic meters) and also on the basis of cost data for each waste management activity. Based on the output of the Mass Balance section, the total quantities of waste to be processed by each waste management activity (treatment, storage, disposal, etc.) is divided by an assumed 4,032 hours of operation per year to determine the capacity requirements for each facility. The Model sums the quantities of all wastes to be sent to a particular facility and calculates the required size of each facility (if it is a "hypothetical" or "planned" facility) based on the amount of waste to be treated there. For an "existing" facility, the Model uses the capacity reported for that particular facility in the Facility Database for cost estimation. The cost data are based on the cost curves published in support of the Programmatic Environmental Impact Statement that is currently under review by the DOE (1). For each planned facility, costs are calculated for four operational phases during the life cycle of a facility. These phases are preoperations (facility design, permitting, etc.), construction, operations and maintenance (O&M), and decontamination and decommissioning (D&D). For existing facilities, only O&M and D&D costs are calculated.

Schedule Estimation

Based on the life cycle cost estimated for each facility, and the projected operational start-up dates and the inventory work-off periods specified in the

Facility Database for treatment and stabilization facilities, the Schedule section of the Model automatically determines the projected expenditure for a particular option during each fiscal year. The user also has the choice of estimating yearly expenditure both with or without the use of cost escalation factors.

Report Generation

The Report Generation section of the Model enables the user to specify a desired report format for presenting the estimated cost of an option in either tabular or graphic form. Flexibility is provided by allowing the user to select from six primary presentation formats of which up to three can be included in any one report. The following is a list of the six primary formats that can be selected by the user:

- Cost by facility operational phase (i.e., preoperations, construction, O&M, and D&D)
- Cost incurred by each DOE site
- Cost by fiscal year (i.e., results from the Schedule section)
- Cost by each treatment and stabilization facility used for an option
- Cost by each waste management activity (e.g., transportation, treatment, disposal, etc.)
- Cost by waste matrix (e.g., aqueous liquids).

Thus, depending on the purpose of the analysis, it is up to the user to select a reporting format (or a combination of one or more of the above formats). As an example, the cost can be reported by each waste management activity incurred at a given site during each operational phase. The Report Generation section also enables the user to compare cost results from more than one option in a single report.

HARDWARE AND SOFTWARE REQUIREMENTS

At a minimum, this IBM PC-based automated model requires the following hardware:

- PC with a 486 DX-33 processor
- 4 megabytes of RAM
- Hard drive with 120 megabytes or a network drive
- 1 floppy drive
- VGA video display
- Mouse.

Minimum software requirements include MS-DOS (Version 6.0 or later), Microsoft Windows (Version 3.1 or later), and Paradox for Windows (Version 4.5). Due to the calculational and disk access requirements of the Model, faster processors, additional RAM, and fast access hard drives with hardware disk cache will improve performance (in terms of response/action time) of the Model. Currently, once an option configuration is completed by the user, the Model takes an average of approximately 1 hour for calculating mass balances, estimating cost and schedule, and generating reports for an option.

PLANNED IMPROVEMENTS

Since the development of the Model is an ongoing process, it is in a state of continuous improvement in many areas. The major improvements include the following aspects of the Model:

- Updates to the Facility and Inventory Databases as additional data become available from the DOE sites regarding the precise waste acceptance criteria of each facility and the hazardous and radionuclide characteristics of each waste stream.
- Inclusion of cost data for additional treatment and stabilization technologies, especially innovative technologies that are currently in the research and development stage.
- Restructuring of some of the cost data currently used in the Model to allow more user flexibility in selection of technologies. For example, the cost data for aqueous waste treatment will be subdivided into its subsystems such as evaporation, neutralization, ion-exchange, gross organic removal, etc. so that the user can select only those technologies that are necessary for a particular waste stream.
- Inclusion of facility, inventory, and cost curve data for other types of waste such as transuranic and high-level waste to enable automation of options analysis for these wastes.

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OPERABLE UNIT AT THE HANFORD SITE

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ABSTRACT

Results from two separate studies are presented to illustrate cost savings that can be achieved using detailed planning to support waste characterization activities. Data Quality Objectives were developed for an operable unit and a waste treatment system at the Hanford Site. The 200-BP-11 operable unit is a CERCLA site that contains multiple treatment, storage, and disposal units. Over $4 million was saved in sampling costs alone for the 200-BP-11 operable unit. The 242-A Evaporator is a mixed waste evaporation system designed to concentrate mixed waste solutions associated with high-level radioactive waste tanks. Identification of critical sampling requirements for the 242-A Evaporator resulted in cost savings in excess of $65K.

INTRODUCTION

There is a need for rigorous, defensible, and easily implemented cost savings evaluations of waste characterization activities in environmental restoration (ER) and waste management (WM) programs. Cost savings must be expressed in terms of both the costs of program implementation and the effectiveness of potential program outcomes. Budgetary constraints are forcing programs to interact in such a way that reciprocal cost savings in characterization are realized.

A particular challenge facing all US Department of Energy (DOE) sites is how to plan and implement cost-effective sampling and analysis activities for (i) treatment, storage, and disposal units that must comply with the Resource Conservation and Recovery Act (RCRA), and (ii) RCRA Past Practice waste management units that must comply with the Comprehensive Environmental Remediation, Compensation, and Liability Act (CERCLA). As remedial actions or tank waste retrieval/treatment operations mature, a historical database will be required to identify those sampling and analysis programs that offer the greatest cost-effectiveness in support of program-specific activities.

STATEMENT OF THE PROBLEMS

The 200-BP-11 Operable Unit

The 200-BP-11 operable unit (OU) is comprised of 1,175 acres in the 200 East Area of the Hanford Site. The 200-BP-11 OU must, in general, comply with CERCLA requirements associated with Interim Remedial Measures (IRMs) or presumptive remedies. Specifically, the treatment, storage, and disposal units in the OU must meet RCRA closure requirements. The problem addressed for the 200-BP-11 OU was how to design a sampling and analysis program that would support RCRA closure decisions while simultaneously meeting CERCLA/RCRA past practice decisions.

The 200-BP-11 OU is one of the largest waste management units at the Hanford Site. Performing sampling and analysis to satisfy CERCLA requirements, and then, performing a similar effort to meet RCRA requirements is intuitively unreasonable. Stakeholders identified the need for an integrated sampling and analysis program. All stakeholders needed to be able to clearly understand the relative advantages and disadvantages of several sampling and analysis programs in meeting the multiple requirements.

The 242-A Evaporator

Several waste characterization sampling and analysis programs for the 242-A Evaporator at the Hanford Site needed to be evaluated for mid-FY-95 operations. Characterization of specific radionuclides and chemical constituents was required for selected evaporator process streams. Each program needed to provide sufficient analytical data to meet process control, operations safety, and compliance with the RCRA requirements.

A major purpose of the 242-A Evaporator is to support delisting of various RCRA constituents. Sampling and analysis at the evaporator, therefore, needed to support similar requirements at the Effluent Treatment Facility which receives process condensate from the Evaporator.

TECHNICAL APPROACH

A reasonable method for determining the relative cost savings of alternative sampling and analysis programs is cost-effectiveness analysis (Quade, 1984; Redus,

1981). Such an approach forces a complete and concise definition of the cost parameters and the effectiveness parameters of each program. Both the 200-BP-11 OU environmental restoration group and the 242-A Evaporator operations management needed easily understood information to select among sampling and analysis programs for current and future system or site activities.

Cost and effectiveness were modeled separately. The baseline waste characterization program was first defined. Alternative programs were then identified. The cost and the effectiveness of each alternative were determined and compared to the baseline program. Programs that were dominated, i.e., both the cost and the effectiveness parameters were less than at least one other program, were eliminated from further evaluation. The remaining programs were then evaluated. The program with the greatest effectiveness-cost ratio was selected as the optimal program. Sensitivities to cost thresholds and effectiveness parameters were examined to determine the influence of these factors in the selection of the optimal alternative.

Baseline cost estimates were determined using sampling data, analytical services data, and quality control data from previous sampling events. Baseline effectiveness was estimated based on the statistical factors associated with the number of samples and the type of analyses performed.

The Data Quality Objectives (DQO) process (US EPA, 1994; Redus and Sheriff, 1994) was used to identify critical decisions, the decision variables, and the risk associated with the sampling and analysis program alternatives for the 200-BP-11 OU (Tranbarger, 1994) and the 242-A Evaporator (Von Bargen, 1994).

200-BP-11 OPERABLE UNIT

The 200-BP-11 is composed of five RCRA Past Practice Waste Management Units and two RCRA TSD Waste Management Units. The Past Practice Units consist of two ditches, one contingency pond, and two unplanned releases. The TSDs consist of one main pond and three expansion ponds.

During the period 1945 through 1985, the OU received mixed fission products, heavy metals, and organic chemicals from 200 East Area operating facilities. Active parts of the OU still receive waste water and process condensate from other treatment systems or processes in the 200 East Area.

A sampling plan was developed to combine a RCRA Past-Practice strategy with a CERCLA Limited Field Investigation. The sampling plan was included as part of a Limited Field Investigation Work Plan (Tranbarger, 1994). The plan identified, and described in detail, field screening requirements, sampling locations, and constituents of concern. Stakeholders who participated in the 200-BP-11 DQO to develop the sampling plan were from the US Environmental Protection Agency (US EPA), US Department of Energy (DOE), the Washington State Department of Ecology, and Westinghouse Hanford Company (WHC). Program managers and key personnel represented both RCRA and CERCLA organizational interests.

A phased characterization approach was developed to implement the sampling plan. The objective of first phase is to determine i) if hazardous waste exceeded Washington State Model Toxics Control Act (MTCA) Industrial Cleanup Standards, ii) if radionuclide contamination was greater than that prescribed in the Hanford Site Baseline Risk Assessment Methodology, or iii) both. Phase I supports a clean closure or modified clean closure for the inactive TSD in the OU. Regardless of the results of first phase, Phase II is required. The objective of Phase II is to confirm the results of the first phase to support modified closure or closure in place. Phase II also supports identification of additional remedial actions.

A Limited Field Investigation Report of the sampling results is planned. If contamination concentrations are between Residential and Industrial Cleanup Standards, then Phase II sampling will be required to confirm the absence of hazardous waste and radionuclide contamination above Industrial Cleanup Standards. This supports a modified clean closure or closure in place decision. If contamination concentrations are below Residential Standards for hazardous waste and below Industrial Standards for radionuclides, then sampling is performed to clean close the TSDs. If contamination is above Industrial Standards, the extent of contamination above these cleanup standards is determined and further discussions with the stakeholders will be held to determine if site-wide background concentrations will be acceptable as cleanup standards.

The stakeholders reached consensus on the decision logic for determining if contamination concentrations meet or exceed threshold requirements. If any one observation exceeds the threshold, then the specific decision is made that the

standard is not met. Thus, the parameter of interest is the maximum value of a sample, rather than an average or an upper 95% confidence limit on the average.

200-BP-11 OU Sampling and Analysis Program Alternatives

Two sampling and analysis programs were identified as described below.

Baseline: The Baseline sampling and analysis program was really two programs: one for the RCRA Past Practice Waste Management Units examined under CERCLA and another program for the RCRA TSD Waste Management Units. Each program required an extensive sampling effort of a total of 39 boreholes to groundwater, test pits, surface samples, and samples for lithological changes and for radionuclide identification. The number of samples per location ranged from one to 10. Over 200 total samples were to be collected, and virtually all compounds identified in 40 CFR 264, Appendix IX Groundwater Monitoring List, were to be analyzed and the results validated completely.

Alternative: The proposed sampling alternative to the baseline combined the two programs so that data collected for the RCRA Past Practice Waste Management Units examined under CERCLA would support the data requirements for the RCRA TSD Waste Management Units, and vice-versa. The alternative program required a total of 23 boreholes to groundwater, test pits, surface samples, and samples for lithological changes and for radionuclide identification. The number of samples per location ranged from two to 13. A total of 148 total samples were to be collected under this Alternative. A significantly reduced set of compounds identified in 40 CFR 264, Appendix IX Groundwater Monitoring List, were to be analyzed. Validation was reduced to 20% for all data packages from test pits and surface samples, and validation of results remained at 100% for all borehole samples.

Cost Savings Evaluation

The cost savings evaluation for the 200-BP-11 sampling and analysis program alternatives was performed using cost only. An effectiveness-cost ratio, ECi, is computed for each of the waste characterization sampling alternatives. For convenience, we will denote the effectiveness-cost ratio as the EC ratio. The EC ratio is simply the ratio of effectiveness to cost. The alternative with the largest EC ratio was selected as the preferred alternative. The utility of this approach is that large values of cost, regardless of the effectiveness, will push the ratio close to zero. Small values of effectiveness, regardless of cost, will force the ratio to behave in a similar manner.

Baseline cost information was obtained from the proposed 200-BP-11 OU Work Plan. Direct costs were estimated based on sampling and analysis, sample preparation, and data package costs. Costs were broken out as sampling costs and analysis costs. Several methods were employed to arrive at cost figures when data was either unavailable or suspect. Cost estimating relationships were used to relate costs to operations variables. Specific analogy methods were used to estimate costs by comparison with known costs of similar items or services. Included in this cost factor are estimates for complexity factors or scaling laws. Expert assessment was used as a last resort when cost information was unavailable. Cost estimate uncertainty was determined to be + 10%.

Effectiveness is defined as the difference between number of samples proposed for the Alternative and the Baseline. A larger positive value was preferred. This indicates that a reduced number of samples is required for the Alternative. If the difference were negative, this would indicate the Alternative required more samples than the Baseline. Both the Baseline and the Alternative employed a biased sampling approach, i.e., a statistical sampling plan was not generated. Thus, there is no tolerance for risk in either sampling alternative, and each alternative represents a deterministic sampling approach.

TABLE I.

The Alternative is clearly preferred to the Baseline, since the EC ratio is positive. Examination of relative percent differences indicates the sampling costs for the Alternative are 73% less expensive than the Baseline, and the analysis costs for the Alternative are 64% less expensive than the Baseline. Examining the marginal EC ratio (i.e., the ratio of the effectiveness difference to the cost difference), we see that the Alternative offers a marginal gain of 5.8 times the Baseline. Increasing the total number of samples for the Alternative to 80% of the Baseline, i.e., from 148 to 326, changes the marginal gain by more than 165% to 15.6 times the Baseline. There is no difference between the Baseline and the alternative when the total number of samples for the Alternative is 108 or less.

THE 242-A EVAPORATOR SYSTEM

The 242-A Evaporator, also located at the Hanford Site, uses a conventional forced circulation and vacuum evaporation system to concentrate mixed waste solutions. Candidate waste feed tanks must be characterized, pumped to the Evaporator feed tank and processed. The feed stream is separated into a slurry stream, a process condensate, and a gaseous process exhaust stream. Two non-hazardous effluent streams, steam condensate and cooling water, are produced and discharged to an evaporative pond.

The slurry stream consists of the concentrated bottoms and contain the majority of the radionuclides and inorganic constituents. The slurry stream is recycled until it is concentrated to target levels. It is then pumped to double shell tanks to be stored for further treatment. The condensed boiloff, or process condensate, contains primarily water, trace organic material, and a greatly reduced concentration of radionuclides. The process condensate is stored at the Liquid Effluent Retention Facility awaiting treatment capacity in the Effluent Treatment Facility. The process exhaust, or vessel vent stream, consists primarily of non-condensible gases drawn from the condenser system, is filtered and discharged through an exhaust stack.

Stakeholders from the US DOE, Ecology, and WHC collaborated in identification of data requirements, sampling methods and technologies, and statistical process control decision criteria to support waste characterization, waste management, environmental compliance, and safety decisions required for operations of the 242-A Evaporator and the subsequent discharge to the Liquid Effluent Retention Facility and the Effluent Treatment Facility. The resulting product was a set of decision documents addressing waste analysis planning, sampling and analysis planning, process operations, and scheduling of operations, (Von Bargen, 1994).

242-A Evaporator Waste Characterization Program Alternatives

Four sampling and analysis program alternatives were identified as described below.

Baseline -- The Baseline case was the 94-1 evaporator campaign that utilized an extensive sampling effort to determine the contents of certain feed tanks for processing. The sampling consisted of seven samples in the process streams. The baseline data was used as a historical estimate of constituent variability. A comprehensive suite of analyses was performed on the samples with extensive quality control checks.

Alternative 1 -- Alternative 1 was based on the waste characterization program used in evaporator Campaign 94-2. Five samples were taken in the process streams for this alternative. A comprehensive suite of analyses was also performed on the samples with equally extensive quality control checks.

Alternative 2 -- This Alternative has a reduced set of analytes, reduced number of samples, and fewer quality control checks. This Alternative was derived during the Evaporator DQO. One sample in each of the three process streams was identified as sufficient to meet process control, safety, and regulatory compliance requirements. Quality control checks were not as extensive as in the Baseline or Alternative 1.

Alternative 3 -- This Alternative also has a reduced set of analytes, reduced number of samples, and fewer quality control checks. This Alternative was also considered during the Evaporator DQO. Eight samples in each of the three process streams were identified as sufficient to meet process control, safety, and regulatory compliance requirements. Quality control checks were not as extensive as in the Baseline or Alternative 1.

Alternative 4 -- This Alternative is based on the implementation of DQO Process for Campaign 95-1 of the evaporator. Three samples taken in each of the three process streams were considered sufficient to meet process control, safety, and regulatory compliance requirements. Quality control checks were not as extensive as in the Baseline or Alternative 1.

Cost Savings Evaluation

An effectiveness-cost ratio, ECi, was also computed for each of the waste characterization sampling alternatives.

Baseline cost estimates were determined using sampling data, analytical services data, and quality control data from previous evaporator campaigns during FY-94 and early FY-95. Baseline effectiveness was estimated based on the statistical factors associated with the number of samples taken for the same campaigns. Costs associated with the Baseline and Alternative 1 are based on a comprehensive set of analyses and the maximum amount of quality control checks that would be accomplished by both the field samplers and analytical services personnel. Costs associated with Alternatives

2, 3, and 4 are based on a reduced number of analytes and a reduced quality control checks. Use of the DQO Process helped evaporator operations personnel better define their needs and the value of the data requirements relating to process control, safety, and regulatory compliance.

Effectiveness was determined as a weighted function of (i) the confidence to correctly conclude the requirements are not met , i.e., the power of the statistical hypothesis test, and (ii) the risk associated with incorrectly concluding the requirements are not met, i.e., the Type I error of the statistical hypothesis test, (Winkler and Hays, 1975). The requirements that must be met are process control, safety, and regulatory compliance. The minimum requirement is the concentration of all analytes do not exceed a predefined threshold. The general form of the effectiveness model is presented in Eq. 1.

Ei = w [min{(1-)] + v [1 - max(a)] (1)

where Ei is the effectiveness for waste characterization alternative i, w is the weight associated with the Power of the Test, and v = 1 - w, is the weight associated with the risk . Since there are usually several constituents that must be characterized for an alternative sampling program, the most conservative way to describe the effectiveness of any alternative is to use the minimum power, min (1-b), the maximum risk, max (a), and weights such that w = v = 0.50.

Three activities were performed prior to the computation of the ECi value. First, we attempted to identify and assess any other decision factors that should be included in the effectiveness or the cost measures. Second, all alternatives were examined in terms of any one alternative dominating another alternative for all effectiveness or cost values. Finally, we identified any minimum power required or any maximum risk allowed to screen alternatives that did not meet minimum requirements. This resulted in Alternative 2 not being included in the evaluation.

Results of the waste characterization sampling program effectiveness and cost evaluation are presented in Table II. The effectiveness value, the cost value, the EC ratio, and the marginal change between the baseline cost and the baseline effectiveness for each alternative is provided.

Using only the EC ratio, the preferred alterative is Alternative 4 (EC = 21.1). The next preferred alternative is Alternative 3 (EC = 10.0). Alternative 4 is preferred over all alternatives because it requires the least number of samples (three) and displays the smallest cost.

Alternative 4 offers a relatively small marginal change over the Baseline (DE/DC = 2.0) when compared to Alternative 3, (DE/DC = 4.5). This indicates that Alternative 4 is the preferred waste characterization alternative in the absolute sense, but Alternative 3 offers the greatest marginal return. Interpreted in terms of the effectiveness and cost measures, this implies that the alternative with the smallest risk is preferred given i) equal weights for power and risk, ii) if the power is the same for all alternatives, and iii) if the cost model remains the same.

CONCLUSIONS

We have presented a cost-savings approach based on intuitive effectiveness measures and expected costs of candidate sampling and analysis programs. planning to support waste characterization activities. Over $4 million was saved in sampling costs alone for the 200-BP-11 operable unit. A sampling and analysis program was developed that provided a marginal effectiveness-cost contribution over five times the original 200-BP-11 sampling and analysis program. Identification of critical sampling requirements for the 242-A Evaporator resulted in sampling and analysis cost savings in excess of $65K with marginal benefits of 100% over the baseline 242-A Evaporator sampling and analysis program.

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Session 04 -- Quality Assurance
Co-chairs: Dan J. Swaim, WHC
Ganesan Subbaraman, Rockwell International
4-1
A VALUE OF INFORMATION APPROACH TO DATA QUALITY OBJECTIVES FOR THE HANFORD HIGH-LEVEL WASTE TANKS
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ABSTRACT
This report summarizes a Pacific Northwest Laboratorya review of the organic-nitrate reaction safety issue in the Hanford single-shell tanks. This study employed a decision analytic method known as Value of Information (VOI). VOI analysis is a special form of decision analysis that has an information collection alternative as one of the initial decision choices. This type of decision analysis, therefore results in the ability to specify the preferred information collection alternative, taking into account all information gathering and other relevant alternatives. For example, the risk reduction benefit associated with further sampling to quantify total organic carbon inventory or to improve information on energetics can be compared to the risk reduction benefit of better temperature monitoring, operational restrictions, or mitigation by moisture control. This approach allows freedom from built-in assumptions, e.g., that all tanks must be sampled to some degree or that all tanks must be deemed intrinsically safe by some means or another. It allows for each tank management decision to be judged in terms of risk reduction from the current state of affairs, and for that state of affairs to be continuously updated to incorporate new information on tank contents, the phenomenology of safety issues, or the effectiveness of mitigation schemes.
ORGANIC-NITRATE SAFETY ISSUE IN HANFORD HIGH-LEVEL WASTE TANKS
This report summarizes a study conducted by the Pacific Northwest Laboratory of the organic-nitrate reaction safety issue in the Hanford single-shell tanks (SSTs). Production of nuclear weapons materials began at the Hanford Site in 1944 and continued until 1990. Radioactive wastes from the reprocessing operations were stored as alkaline liquids and slurries in near-surface underground tanks. One hundred forty-nine SSTs, ranging in capacity from 208 m3 to 3,800 m3 (55,000 to 1 million gallons), contain approximately 14,000 m3 (36 million gallons) of waste damp saltcake (predominately sodium nitrate and sodium nitrite), metallic hydroxides, other insoluble metal salt sludges, plus about 2,300 m3 (600,000 gallons) of supernatant liquid.
Organic materials were used in several applications in the separations of nuclear materials at the Hanford Site, and many of the waste types generated included organic materials. These included ethylenediaminetetra-acetic acid (EDTA), N-hydroxy-ethylenediaminetetra-acetic acid (HEDTA), sodium citrate, sodium acetate, normal paraffin hydrocarbon (NPH), tri-butyl phosphate (TBP), and hundreds of miscellaneous compounds used in small-scale applications at the Site. (1) An estimated "average chemical composition" of these organic materials approximates

sodium acetate. (2) The waste materials have been degraded by radiolytic and chemical attacks in the waste tanks but still retain significant potential fuel value. (3)

The presence of the organic materials in the waste tanks is of concern because of the following: 1) saltcake wastes are rich in $NaNO_3$ and $NaNO_2$;, 2) efforts have been expended to remove the bulk of drainable liquids from most tanks, and 3) several tanks contain wastes with significant decay heat. Taken together, these factors could create conditions favorable for an organic-nitrate reaction.

DATA QUALITY OBJECTIVES

Data quality objectives (DQOs) are specifications that describe data that are adequate for a particular purpose. In a typical waste characterization problem, DQOs would specify the analytes to be measured, the sensitivity required (detection limits), the accuracy of individual measurements, the spatial volume of material for which measurements are deemed representative, and other such features. In addition to a generic label for this type of data specification, "DQO" has also been used to describe a specific process for deriving these specifications. (4) While some benefit has accrued from applying this method to tank waste problems, the organic-nitrate reaction risk issue was sufficiently complex in terms of phenomenology, number of options, and statistical issues to require structured decision analysis tools.

These decision analysis and risk management tools enhance the standard DQO methodology and allow it to be more responsive to the needs of tank waste characterization. In addition, these methods allowed an integration of the DQO domain (what and how to sample) into a much broader set of risk management decisions: what criteria are appropriate, how conservatively should they be applied, what mitigative actions are "risk-effective," and what incentives exist for developing more effective mitigative measures.

ORGANIC-NITRATE REACTION RISK

The following section describes the organic-nitrate reaction risk. The chemical reaction is introduced with potential initiators and propagation requirements. The reaction risk model is presented with all components. Consequences of the reaction are discussed as well as mitigation measures.

Reaction Initiation and Propagation

The primary hazard of the organic-laden waste is that both fuel and oxidizer are present and intimately mixed. The key mitigative feature is the moisture content of the waste. Waste energetics are characterized by the total organic carbon (TOC) content, expressed as weight percent carbon in the fuel on a dry-waste basis. The reaction of sodium acetate and sodium nitrate is considered a first-order surrogate for the reaction of actual waste (Equation 1) with an ideal heat of reaction of about 7.5 MJ/kg sodium acetate.

$$NaC_2H_3O_2 + 1.6NaNO_2 \rightarrow 1.3Na_2CO_3 + 1.5\ H_2O + 0.7\ CO_2 + 0.8\ N_2 \qquad (1)$$

Stoichiometry for this reaction corresponds to 11 wt% TOC. The reaction cannot take place unless appropriate energy is supplied as an initiator.

Measurements for dry waste have demonstrated that the reaction does not become exothermic unless the waste is at, or above, the relatively high temperature of about 200C. The reaction will not propagate through a medium unless the TOC exceeds 6 wt% and the temperature is about 300C. Moisture in the waste inhibits reactions. Based on the energy release of a stoichiometric mixture, a heat balance indicates that moisture content of 17 wt% would prevent a propagating reaction in mixtures with TOC less than stoichiometry.

The consequences of a hypothetical reaction of organic-nitrate waste involve the heating and pressurization of the tank headspace by hot reaction-product gases and the entrainment of vapors or aerosols from the waste that may be radiologically active. For a TOC value of 6 wt% (the value at which sodium acetate fuel reactions are observed to propagate), a dry reacting volume of 1 m^3 would pressurize the tank headspace to 1.50 atm, or 7 psig overpressure, which is enough to blow out any filter in the system and release gases and aerosols outside the tank through any tank orifice. A 6 wt% TOC dry reacting volume of 2 m^3 would cause a final pressure of about 2.1 atm, or 15 psig overpressure. Structural analysis indicates that a pressure of 14 psi would cause extensive cracking of the concrete dome of a half-million gallon tank, and the failure limit for a million gallon tank is 11.6 psi. This result is important since it indicates that a relatively small reacting volume could be a significant hazard and, thus, that spatial distribution of fuel

within a tank is important.

Reaction Risk Model

Although risks of organic-nitrate reaction are thought to be low in most SSTs, the consequences of a major release are high enough to warrant significant investment in ensuring safe storage. In this framework, information on the organic constituents, moisture level, and temperature status of tank becomes a risk management tool. By using a risk model that embodies uncertainty about the organic constituents, it is possible to show how risk management for a tank benefits from better information on fuel or moisture. Thus, the basic concept of a reaction risk model is to predict the risk of various significant release events as a function of tank fuel, moisture, and temperature (FMT) status, and the uncertainty about them.

The concept was implemented using estimates for probabilities of reaction initiation events and conditional probabilities of reaction propagation (given an initiator was present) expressed as functions of fuel content and moisture level. The basic probability structure of the model is shown in Equation 2.

Pr(Eij Ik) = Pr[F1 , Mm, Tn]p (2)

where

Pr(Eij Ik) = probability that a reaction event will proceed from stage i to stage j given that initiator Ik has occurred in tank p

Fl = fuel state l (one of several discrete states defined by fuel concentration)

Mm = moisture state m (one of several discrete moisture states)

Tn = temperature state n (one of two discrete equilibrium temperature states).

In practice, the probability of an event is a sum of these conditional probabilities weighted by estimated probabilities that a tank is in a given FMT state:

Pr(Eij) = lmn [Lp(F1, Mm, Tn) x Pr(F1, Mm, Tn)] (3)

where

lmn = summation over all FMT states

Lp = estimated probability that tank p is in a given FMT state.

This model was developed using four initiators believed to account for most of the risk of initiation during storage in SSTs. Four event-severity classes were used to define the range of possible consequences of release events. All of the probabilities in this model were estimated by a group of experts familiar with the following: the organic-nitrate reaction, a series of adiabatic calorimetry experiments on waste stimulant mixtures, and SST safety issues. Elicitation was conducted by a trained elicitor and confirmed in three separate meetings. Even with experimental basis and consistency checks, the resulting model has substantial judgmental content. The resulting risk model is not intended to accurately represent the absolute risk of uncontrolled releases from the Hanford tanks, but to capture the sensitivity of risk to FMT variables, and uncertainty about these variables.

The overall structure and function of the risk model is schematically shown in Fig. 1. This figure illustrates how the probabilities are used in conjunction with statistical estimates of TOC, moisture, and temperature to calculate the risk of a given severity class event. Also shown is the effect of various mitigation measures, which were represented by changing the position of a tank in the FMT space, the initiator probabilities, or both.

Fuel, Moisture and Temperature Definitions

The fuel status for a tank was defined using the following ranges:

0 wt% TOC 2.5 wt%
2.5 wt% TOC 5 wt%
5 wt% TOC 7.5 wt%
7.5 wt% TOC 10 wt%
10 wt% TOC.

The model incorporated uncertainty in the spatial variability of TOC within the tank and the uncertainty in specific energy content by using the concept of Maximum Total Fuel Value (MLFV), defined as the product of three independent, lognormally distributed factors (Equation 4):

MLFVp = TOCp x EDF x SCF (4)

where

TOCp = mean TOC concentration in wt%, dry basis, tank p

EDF = energy density factor, [(cal/g)p/(cal/g)NaAc]

SCF = spatial concentration factor, [TOCmax m3/TOC]
This model structure accounts for the risk that a small volume of waste, enriched in organic content, can sustain a propagating reaction given a sufficient initiator. It allows for a separate measurement of the VOI on average TOC inventory in a tank, the degree of spatial heterogeneity in the TOC concentration, and the speciation (and energy of reaction) of the TOC.
The moisture status of a tank was represented in the risk model as a set of discrete moisture states analogous to the fuel status. Reaction propagation probabilities were estimated using an assumption that the moisture variable was the moisture of the potentially reacting waste. The number of moisture states were limited to three:
M 6 wt%
6 wt% M 17 wt%
17 wt% M.
The temperature status of a tank was defined with two states:
T < 149C
T 149C.
Initiator Definitions and Probabilities
In a wet, low temperature condition, organic-nitrate mixtures of even high TOC content are stable to shock and sparks. It is possible, however, for a small spark to initiate a dry mixture. There is also the possibility for a runaway reaction if a source of heat is supplied to some waste volume. These considerations led to the use of four initiators in the risk model. The initiators and the probabilities assigned to them are shown in Table I.
A life cycle of 25 years was chosen for analysis as it would represent a reasonable upper limit for tanks that are retrieved late in the retrieval and treatment program and be conservative as an average for all SSTs. These estimates reflected the strong belief of the expert group that a very small spark was essentially unavoidable during routine tank operations (e.g., sampling and waste retrieval operations).
Event Severity Classes
An organic-nitrate propagating reaction in a SST, if initiated, could result in a wide range of event severity cases because of the possible variation in fuel concentrations, the spatial extent of fuel concentration to support propagation, and the moisture concentration in and near the reacting region. Also, the amount of free headspace volume in the tank and the configuration of headspace ventilation would be important in determining the pressures reached and the severity of structural damage and release. Five severity classes were defined in the risk model to represent the potential range of events. The event severity classes are shown in Table II.
The five severity classes were used as a framework for estimating reaction probabilities and developing estimates of possible consequences.
Mitigation Measures
The final component of the reaction risk model is the development of potential mitigation activities to achieve control of FMT. Fourteen potential corrective actions were studied. The eight shown in Table III were selected as the range of activities that provided the range of information required for this study.
The total cost includes the number and length of time each intervention would take during the 25-year life cycle.
DECISION ANALYSIS FRAMEWORK
The decisions on how to sample and classify tanks, determine whether any mitigative measures should be applied, and decide which are most effective were represented in a decision tree or decision analytic model. Such a model allows assessment of the best decisions according to some decision rule, based on the probabilities of various outcomes given that certain decisions are made. Data on mitigative measures include effectiveness and cost, where effectiveness is measured by the reduction of either the probability of an initiation event or the probability of a propagating reaction given that an initiator event has occurred; and cost of implementation includes the direct costs of the engineering measures plus, in some cases, indirect or intangible costs. Data required on consequences included at least one measure of value or costs that can be associated with each possible outcome in the decision tree.
The risk model was employed in a decision analysis mode to predict the expected costs of each possible path. Expected costs are calculated using the probabilities of release events given that each of the possible mitigation measures was implemented. All calculations depend on the assessments of fuel and moisture status,

i.e., the probabilities that a given tank is in each of the possible fuel and moisture states. The simplest decision rule for such a problem involves minimizing the statistically expected cost (or maximizing the expected value) of the outcomes resulting from a mitigation decision. This risk-neutral decision rule may not be the appropriate one for actually making risky decisions. The appropriate stakeholder (U.S. Department of Energy, State of Washington, the public, etc.) may be willing to pay a premium beyond the statistical expectation of reduced cleanup and social costs saved. However, the focus in this study was on defining the value of different types and qualities of characterization data. For this application, using an expected cost minimization (risk-neutral) decision rule resulted in a lower boundary for VOI estimates, which allows us to determine the minimum that should be spent on efforts to ensure safety through characterization or mitigation.

The value of a given set of information is the decrease in the optimal (minimum expected) cost resulting from making a decision with the information rather than without it (Eq. 5).

V(I) = Min[E(C(xi) I0 - Min[E(C(xi) (I0,I)] (5)

where

V(I)	=	value of the information, I
Min	=	minimum operator (over all possible decisions)
E	=	expected value operator
C	=	cost of option xi
xi	=	decision options
I	=	information to be valued
I0	=	base case or starting information.

In the case of the organic-nitrate reaction problem, the reference information I0 corresponds to the historical assessments of TOC, moisture, and temperature, and the judgmental evaluations of energy density factor (EDF) and spatial concentration factor (SCF) in this study. The right hand term contains both I and I0, indicating that the base case is not discarded in making the least cost decision after the prospective information, I, is available. In practice, some synthesis of base case and new information is attempted to take maximum advantage of both. In this VOI application, this was accomplished with a Bayesian updating calculation in which the information set I0 is the prior distribution and the set (I0,I) is the posterior distribution.

Figure 2 shows the schematic decision tree structure of the model for evaluating the value of characterization about each tank's organic and moisture contents to support the management of the organic-nitrate safety issue. This tree follows all of the same conventions of any decision tree model. Reading left to right, the first decision is whether and what information to gather. This decision is followed, for choices that involve information gathering, with reports on the tank's moisture and fuel (TOC or MLFV) contents. Next comes a decision node that contains the possible mitigative actions, which range from no action to an emergency retrieval and transfer of the tank's contents. Each mitigative action is followed by an uncertain event node of the tank's actual moisture and fuel contents (before mitigation). The final node is an uncertain event node of outcome severity, which ranges from no event to a major release of the tank's contents.

VALUE OF INFORMATION RESULTS

The model described was used to prepare VOI assessments under several sets of assumptions during the study. The general strategy was to begin with analyses of the value of "perfect" information (e.g., no statistical uncertainty) of various kinds. Following these studies of VOI for perfect information, studies were performed for "realistic diagnosticity" measurements. These studies calculated VOI for MLFV and moisture measurement sets with parametrically defined standard errors (10% to 50%) to assess the sensitivity of VOI to measurement quality.

In general, the results indicate substantial VOI for high TOC, indeterminate moisture tanks, and higher VOI for high-heat tanks than low-heat tanks for a given moisture level. An important observation concerns the degree of discrimination between tanks of significant interest for detailed characterization and those for which tank safety factors and associated risks may not warrant extensive characterization studies. The top 50 or so tanks studied in the case where prior distributions are adjusted for both spatial and energy of reaction factors have a high VOI such that the significant investment in sampling and analysis is a good investment for a risk-neutral stakeholder. The bottom 60 to 70 tanks may not warrant

detailed study for this issue. There are 30 tanks in the questionable region. Thus, the study discriminated fairly well among those tanks with high payoff to sampling and those not worth the cost.
Studies separating the VOI for MLFV and moisture showed that perfect information about TOC alone is always worth essentially as much as that about both variables together. However, this should not be taken to mean that priority should go to MLFV determination over moisture measurements. The factors determining MLFV (via its components TOC, SCF, EDF) are much more difficult to measure accurately than the minimum moisture content. Also, from a starting point of considerable uncertainty about both MLFV and moisture, definitive information about either variable will dramatically improve decision-making and is, therefore, of high value.
The VOI statistic for the entire population of 149 SSTs is illustrated in Fig. 3 for each of the TOC inventory values and the pertinent adjustments. This figure also shows that the upper limit to the risk-reduction benefit of information is the cost of the least-cost, but substantially effective mitigative action.
Imperfect diagnosticity cases were studied to determine how measurement precision affected the VOI, and, therefore, arrive at acceptable laboratory measurement curves. The standard errors studied are for the tank-scale indicators of MLFV and minimum moisture, and had to be translated to results for individual assays. In general, tanks with high VOI for perfect information retained a high percentage of this VOI as the standard error of MLFV and moisture estimation increased. For standard errors of 20% of the true value for both MLFV and moisture, the VOI was typically at, or near, 90% of the value of perfect information. The results showed that there was not much incentive to estimate MLFV more accurately than about 20% standard error (as a fraction of true mean). Moisture measurements were shown to be tolerated as crudely as 30% to 40% standard error.

CONCLUSIONS

The following bullets summarize the findings of using the VOI approach for applying DQOs to the organic-nitrate safety issue in Hanford high-level waste tanks.

- The decision-making value of tank waste information on moisture and fuel value is substantial.
- Almost all of the information value is concentrated in the top 50 tanks.
- The lowest-cost mitigation option limits the VOI for a tank.
- Extremely accurate predictors of MLFV and minimum moisture (at the tank scale) are not required.
- Fuel information alone is worth almost as much as fuel and moisture information together.
- Moisture information alone is worth about three quarters of the information on fuel value and moisture together.
- It may not be economical to conduct fuel assessments in wet tanks.
- The uncertainty about spatial distribution of fuel and specific reaction energy warrants investment in tank characterization.

In addition to these specific conclusions, it became clear during and after the study that the VOI model is a general risk management paradigm. With this model, decisions about developing and deploying mitigative measures, restricting operations, and other factors can be integrated and addressed in terms of overall risk reduction and cost minimization.

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REQUIREMENTS

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ABSTRACT

The Department of Energy (DOE) is currently proceeding with actions to provide cost-effective, safe, long-term interim storage of DOE-owned spent nuclear fuel (SNF) in preparation for permanent disposition. Integral to these actions is the identification, implementation, and verification of the quality assurance (QA) requirements associated with the management of this DOE-owned SNF. The potential for multiple and, perhaps, conflicting QA requirements from varying sources complicates this process. Requirements contained in 10 CFR Part 830.120, 10 CFR Part 72, DOE/RW-0333P, and DOE Order 5700.6C all have potential application to SNF activities during the course of proceeding from existing conditions to final disposition. As a result, the DOE Office of Spent Fuel Management (EM-37) has undertaken an evaluation of these different QA source documents to assess the equivalency of their requirements. This paper presents the methodology of this evaluation and the resulting conclusion that, with minor modifications, a QA program based on the requirements of DOE/RW-0333P meets or exceeds the requirements of the other QA source documents. The implementation of a DOE/RW-0333P-based program can be used to ensure compliance with all current QA requirements and ensure the ability to comply with potential requirements for long-term interim storage and final disposition of DOE-owned spent nuclear fuel.

INTRODUCTION

The Office of Spent Fuel Management, EM-37, has recognized that there are multiple quality assurance programs that may be applicable to the management of DOE-owned SNF. All DOE activities, including SNF management, are subject to DOE Orders including the QA requirements specified in DOE Order 5700.6C, Quality Assurance and the analogous requirements for contractors provided through 10 CFR Part 830, Nuclear Safety Management, Section 120, Quality Assurance Requirements. The increasing concern of the nation, in both government and private sectors, over the disposition of DOE-owned SNF has resulted in a growing constituency in favor of subjecting DOE-owned SNF storage facilities to the same management and technical standards imposed upon commercial fuel regulated by the Nuclear Regulatory Commission (NRC). The QA requirements associated with NRC regulation of interim storage facilities are promulgated in 10 CFR Part 72, Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation. Additionally, DOE-owned SNF could eventually come under the jurisdiction of the Office of Civilian Radioactive Waste Management (OCRWM) requiring a QA program governed by Quality Assurance Requirements and Description, DOE/RW-0333P. This QA program would be applicable if the fuel is to be disposed in a geologic repository. Thus the potential exists for multiple and perhaps conflicting QA requirements associated with the management of DOE-owned SNF. As a result, EM-37 has undertaken an evaluation of the QA requirements that may be applicable to DOE-owned SNF management.

This evaluation compared the requirements contained in four source documents, 10 CFR Part 830.120, 10 CFR Part 72, DOE/RW-0333P, and DOE Order 5700.6C, and assessed the equivalency of these requirements. The results are intended to support, through formal evaluation, the policy decision to utilize the applicable portions of DOE/RW-0333P as the baseline for developing and implementing a DOE-owned SNF QA program. The evaluation further identified modifications that would be needed for a QA program based on the requirements of DOE/RW-0333P to ensure the requirements of other QA source documents are, as a minimum, met.

EVALUATION

To perform a realistic and meaningful evaluation of the source requirements, it was necessary to first review the basis, intent, and background of the four source documents. This is especially relevant since each was developed to meet the needs of different organizations and activities. Additionally, equivalency of QA requirements can be assessed more meaningfully if the requirements are evaluated in the context of the regulated activities of interest. For DOE-owned SNF, a systems engineering

approach1 has been used to establish technical functions, and thus regulated activities, that define all aspects of managing the SNF. These technical functions are defined as "Ensure Safe Existing Conditions," "Achieve Interim Storage," and "Prepare for Final Disposition." Addressing the potential applicability of the four QA source documents to the three technical functions assisted in the evaluation. The relationship of the four QA source documents to each other and the SNF Program's technical functions is depicted in Fig. 1.

DOE Order 5700.6C

The most recent major revision to this DOE Order was issued on August 21, 1991. The revision of DOE Order 5700.6B emphasized the establishment of a QA culture to improve the safety and reliability of the Department's programs, projects, and facilities. This revision represented a departure from a NQA-1-based rigorous compliance program and reflected the concept that all work is a process that can be planned, performed, assessed, and improved. This culture encourages setting and maintaining high standards, identifying and resolving problems, continuous improvement, and fostering effective communication between DOE and its contractors. The Department is responsible for meeting the requirements of this Order and to continuously pursue enhancements to safety and reliability. Compliance with this order for DOE-owned SNF requires a QA program that meets or exceeds 5700.6C requirements for all SNF activities and technical functions (see Fig.1).

The Order's ten basic criteria are broken into the three categories of Management, Performance, and Assessment. These three categories capture the range of activities associated with work processes from the initial planning and organizing activities to continuous process improvements. The requirements specified in the Order were established to assist all levels of management and workers to focus on their unique responsibilities in carrying out the provisions of the QA program.

The Order's requirements apply to virtually all DOE activities. This application includes planning and conduct of basic and applied research, scientific investigation, engineering design and operations, maintenance and repair of facilities, project close-out, and environmental restoration. The requirements and guidance provided in the Order apply to DOE and are applied through contract mechanisms to management and operating (M&O) contractors. Exemptions to activities under this Order are limited to work associated with nuclear weapons administered by the Assistant Secretary for Defense Programs, work done by the Energy Information Administration, work conducted under the Naval Nuclear Propulsion Program, NRC-licensed work, and research and development work results that undergo peer review for publication.

10 CFR Part 830.120

Quality assurance requirements for nuclear facilities were promulgated on May 5, 1994 as part of the Final Rule on Nuclear Safety Management, 10 CFR Part 830. The Quality Assurance Requirements, section 120 of this rule, mandate the development and implementation of a formalized QA program. These requirements are subject to enforcement under the Price-Anderson Amendment Act, which includes civil and criminal penalties. Compliance with this rule for DOE-owned SNF requires a QA program that meets or exceeds the 10 CFR Part 830.120 requirements for all SNF activities and technical functions (see Fig.1). The QA requirements of the rule apply to DOE management and operating (M&O) contractors at DOE nuclear facilities. Exemptions to these requirements are limited to NRC-regulated activities, Naval Nuclear Propulsion Program activities, and activities conducted under the Nuclear Explosives and Weapons Safety Program.

The rule establishes three QA categories; Management, Performance, and Assessments, which contain ten criteria that are identical to DOE Order 5700.6C. Use of a graded approach is specified for the application of the requirements to facilitate a value-added process. The implementation of the rule requires M&O contractors to develop and submit to DOE for approval a QA Program applying the criteria specified in the rule.

10 CFR Part 72

The requirements for obtaining a NRC license for an Independent Spent Fuel Storage Installation (ISFSI) are promulgated in 10 CFR Part 72. Subpart G of this rule describes the QA requirements necessary for obtaining the license. These QA requirements apply to all aspects of the design and operation of an ISFSI that are important to safety. Since DOE-owned SNF may potentially be subjected to NRC licensing or licensing reviews for interim storage, 10 CFR Part 72 has potential

applicability to SNF activities associated with the technical function, Achieve Interim Storage.
Each licensee is required to develop, maintain, and implement a QA program satisfying the requirements delineated through 18 criteria. These criteria are based on the 18 criteria of NQA-1. The rule further specifies that a QA program approved by NRC under 10 CFR Part 50, Appendix B, is acceptable (see Fig. 1). The QA program must cover all activities throughout the life of the licensed facility from site selection to decommissioning. The NRC must approve each licensee's QA program prior to receipt of spent fuel.
DOE/RW-0333P
The Quality Assurance Requirements and Description (QARD), DOE/RW-0333P, was issued on December 18, 1992 to support the OCRWM Program. This includes the transportation, interim storage, and disposal of civilian spent nuclear fuel after it has been transferred to DOE custody. The OCRWM Program also includes the disposal of defense high-level waste (HLW). The QARD requirements are based on regulatory requirements from 10 CFR Part 50, 60, 71, and 72. Since DOE-owned SNF may eventually be disposed in a geologic repository, DOE/RW-0333P has potential applicability to SNF activities associated with the technical function Prepare for Final Disposition.
All organizations performing work for OCRWM must use and comply with the QARD to develop and implement QA programs. The QARD applies to: acceptance and transportation of spent nuclear fuel and defense HLW; the Monitored Retrievable Storage facility; the Mined Geological Disposal System, including site characterization activities; and HLW immobilization and waste form acceptance. The line organization responsible for the implementation of the QA program shares responsibility with the QA organization for the verification of quality activities. The OCRWM Program Director retains ultimate responsibility for the QA program.
The QARD is organized into 18 sections, four supplements, three appendices, and a glossary of terms. The 18 sections contain requirements common to all OCRWM activities. The supplements contain requirements for specialized activities, such as software, sample control, scientific investigation, and field surveying. The appendices contain requirements specific to individual OCRWM Program elements. The glossary defines the terms common to the QA Program.
Methodology
This evaluation reviewed and assessed the equivalency of the QA requirements from the four source documents in a series of three comparisons. The first comparison assessed the equivalency of DOE Order 5700.6C requirements with analogous requirements contained in 10 CFR Part 830.120. A similar comparison was then performed between the QA requirements specified in 10 CFR Part 72 and DOE/RW-0333P. Finally, a comparison was made between 10 CFR Part 830.120 and DOE/RW-0333P. These three comparisons allowed for identifying, comparing, and documenting areas of difference between the various source documents. This approach resulted in identifying applicable QA requirements which would not be implemented for DOE-owned SNF through a QA program based solely on strict compliance with DOE/RW-0333P. The process of assessing equivalency consisted of several activities. First the requirements contained in a given criteria were evaluated and summarized. This summary was then cross referenced to the analogous requirements of another document. The cross-referenced requirements were then evaluated, summarized, and compared to the evaluated requirements of the initial criteria. The process was repeated for all three comparisons.
The evaluation compared the source documents at the level of detail necessary to establish differences between the criteria. This allowed determination of the differences and an assessment of equivalency with the minimum appropriate level of detail. A line by line compliance evaluation was therefore not performed. This technique of comparing requirements at a criteria level was adopted based on previous experience with similar comparative analyses that used both criteria level and line by line approaches (2, 7). These previous evaluations indicated that although line by line evaluations will provide detailed compliance verification, such level of detail was not essential in determining the equivalency between directives.
Results
The evaluation showed that the requirements specified by 10 CFR Part 830.120 are, at a minimum, equivalent to those of DOE Order 5700.6C and that all requirements in the DOE Order have corresponding requirements in 10 CFR Part 830.120. A similar

conclusion was obtained that the QA requirements for commercial spent nuclear fuel under DOE/RW-0333P meet or exceed those for storage under 10 CFR Part 72. The evaluation between the ten criteria of the DOE QA directives with the 18 criteria of the NRC-based programs revealed that the QA requirements of the different programs were essentially equivalent. Only the requirements of DOE/RW-0333P associated with quality improvement would need modification to meet or exceed all 10 CFR Part 830.120 requirements. Additionally, numerous examples were found exemplifying the more rigorous, activity-based nature of the requirements of DOE/RW-0333P. For example, DOE/RW-0333P provides detailed records protection requirements, including design criteria for the facilities in which they are stored, whereas 10 CFR Part 830.120 provides only general requirements on retention of records. The evaluation results between these two programs are summarized in Table I.

The requirements of DOE/RW-0333P establish the need to perform trend analysis to detect and prevent problems associated with product or service quality. While these activities are essential elements of any continuous improvement program, they emphasize only the negative aspects of quality management, such as nonconformances and deficiencies. A 10 CFR Part 830.120 continuous improvement program includes observing, tracking, and acting on both positive and negative results. In short, the basis of the RW-based QA program is to ensure achievement of designated levels of performance, whereas, the 10 CFR Part 830.120 also seeks perpetual evaluation to increase the level of performance, i.e. continuous improvement. Thus a RW-based program would require modification to expand the aspects associated with trend analysis and forecasting to include the broader aspects of quality improvement contained in 10 CFR Part 830.120.

SUMMARY

The Office of Spent Fuel Management compared the source QA documents and assessed the equivalency of their requirements. This evaluation concluded that implementation of a RW-based QA program, with minor modifications, meets or exceeds all current QA requirements. Furthermore, this modified RW-based QA program would ensure the ability to comply with all foreseeable QA requirements for long-term interim storage and final disposition of DOE-owned spent nuclear fuel.

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SOME QUALITY ASSURANCE ASPECTS OF TRU WASTE CHARACTERIZATION

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INTRODUCTION

The Waste Isolation Pilot Plant (WIPP) is a planned geologic repository for permanent disposal of defense transuranic (TRU) waste generated by the U.S. nuclear weapons production program. Waste retrievable stored at defense facilities since 1970 is planned for shipment to WIPP, and about two-thirds of the waste intended for WIPP has yet to be produced. Both contact-handled (CH-TRU) and remote-handled (RH-TRU) waste are currently stored at a number of Department of Energy (DOE) facilities and are intended for disposal at the WIPP. CH-TRU is in 0.21 m3 mild

carbon steel drums that have a maximum surface dose rate of 200 rem/hour. RH-TRU waste will be emplaced in 0.85 m3 capacity cylindrical canisters with unshielded surface dose rate greater than 200 rem/hr but no greater than 1,000 rem/hr. The 50-hectare repository has been designed to hold 176,000 m3 (850,000 drums, nine million curies) of CH-TRU waste and 7,100 m3 (7,500 canisters, five million curies) of RH-TRU waste. Quality assurance (QA) programs for the characterization of these mixed wastes pose some challenging problems.

WASTE SAMPLING AND ANALYSIS

The WIPP is intended as a repository for approximately a million waste containers, most of them 55-gallon carbon steel drums. Invasive sampling of every container is not only prohibitively expensive, but poses considerable risk of occupational radiation exposure, and therefore contravenes ALARA. On the other hand, adequate characterization and assurance that data quality objectives will be achieved when only non-invasive characterization methods are used is difficult at best. Quality objectives can be met if a robust statistical sampling procedure can be combined with non-invasive analysis. Such a robust system has been developed in the Quality Assurance Program Plan (QAPP) (1). for the WIPP.

Specification of a single standard sampling procedure to be used at all sites is not only logical but cost-effective. However, a single procedure may not be appropriate for all waste streams. The waste streams at some sites like INEL are relatively homogeneous, while others, like the Hanford mixed wastes, are heterogeneous. In cases where a single standard sampling procedure may not be appropriate, consistency in quality assurance and the planning for quality assurance is important. The TRU waste characterization program is trying to achieve standardization or, at least, a normative measure for different sampling procedures. Even though different waste streams at different sites require different sample preparation methods, the trend toward a normative procedure can compensate for standardized methods or one of a standard suite of methods. The goal of the program is to make measurements at one laboratory comparable with those made at another laboratory.

Contamination of CH-TRU waste by volatile organic compounds (VOCs) includes a wide range of VOC concentrations. In addition, non-radioactive metals and inorganic compounds may be present in very small quantities, and these quantities may be so small that existing chemical methods and instrumentation cannot detect them. The minimum detection limit (MDL) is a function of the instrumentation and analytical methods used. Measurements smaller than ("below") the MDL can only be reported as such, and incorporation of less-than-MDL data into statistical analysis poses a problem. Although it is possible and relatively straightforward to substitute half the MDL for a measurement below the detection limit, such a substitution may not be suitable. Appropriate statistics such as those given by Helsel2 might be considered. Helsel plots a lognormal distribution of measurements above the detection limit and determines the summary statistics that best fit the measured data. The reporting limit is then the concentration below the median and mean that occurs with the maximum frequency. In applying a method like this one, it is important that all sites define the method detection limit consistently.

Sampling techniques for solid process residues and soils are critical for the accurate determination of total volatile organic compounds (VOCs). Inaccuracies are much more likely to occur, and be taken into account, in sampling of heterogeneous mixtures than in well-established, repeatable analytical procedures on homogeneous material. A method for relating sampling errors to analytic inaccuracy is important to any quality assurance plan, as well as being part of good laboratory practice. Although a quality assurance objective (QAO) for total uncertainty could be determined from an evaluation by an expert panel, use of expert panels is fraught with pitfalls (e.g., panel bias, questionable expertise, questions about completeness of the information presented to the panel) and is undertaken only very carefully and as a last resort. Because total uncertainty is propagated throughout the sampling system, the decision to use an expert panel is critical. In the event that a panel is used, the process can be assisted by describing in detail how the panel is selected, how it operates, what information will be presented to panel members and in what format, and how the results of the expert panel elicitation will be aggregated. Alternatively, total uncertainty could be defined as the propagation of all quantifiable uncertainties, and the result would be benchmarked against a quality assurance objective.

Reporting data in fundamental units is preferred to other reporting protocols. For

example, reporting gas concentrations in mole percent, partial pressure, mole/liter, or grams (micrograms)/liter is more readily understood than reporting in volume percent or parts per million by volume (ppmv). Moreover, the precision of the analytical method used might well serve as a goal or the overall method precision. For example, gas chromatography and mass spectroscopy have a considerably better potential precision than +/- 25%, given as the relative standard deviation or difference, and should have considerably better recovery than +/- 30%. The imprecision and inaccuracy inherent in the sampling methods used may limit the precision and accuracy of the overall method. The error band specified for any method properly does not include extreme outliers or excursions. To include them defeats the purpose of quality control and quality assurance.
The equilibrium vapor pressure of volatile organic compounds (VOCs) provides a quick and easy check on the measured concentrations: i.e., the measured headspace concentration of the gas phase of any particular VOC should be about the same as that given by the Raoult's Law calculation from the measured concentration of the total VOC in question. This check could be incorporated into the data validation system. VOC analysis generally includes an organic solvent blank as well as a water blank, since not all analyses are done in or from water solution. High concentration samples are diluted to bring them within the maximum instrument detection limits (IDL). Dilution is the time-honored method for making concentrations to be measured congruent with instrument capability. Measurements that exceed the maximum IDL are usually meaningless; most instrumentation is reliable only well below the maximum IDL.
DATA HANDLING
Quality assurance includes, at least by reference, methods and documentation for qualifying "old" or existing data. For the WIPP, as for other projects, measurements were made and data accumulated before any quality assurance procedures were in place -- indeed, in some cases, before quality assurance had been well defined. Quality assurance is not quality control, and the lack of quality assurance procedures is not a comment on the scientific validity of past investigations and existing data. A number of methods are available for qualifying old data that are considerably short of repeating the experiments. We should remember that quality assurance is a relatively new protocol, and that most of the valid and valuable scientific discoveries of the nineteenth and twentieth centuries were made in the absence of quality assurance. Moving forward with science calls for a rational application of methods to ensure quality for past, present and future data development.
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LEISA - VALIDATING THE MODEL OF THE REFERENCE CONCEPT FOR THE EMPLACEMENT OF ILW(Q) AND HTGR FUEL ELEMENT WASTE PACKAGES IN VERTICAL BOREHOLES
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ABSTRACT
The semihydrostatic model has been developed in order to assess the mechanical loads acting on heat-generating ILW(Q) and HTGR fuel element waste packages to be emplaced in vertical boreholes according to the borehole technique. In this model, the borehole filling composed of emplaced packages and crushed salt is treated as a homogeneous medium with a density averaged according to the volume fractions, in order to derive the basic equations. The calculated pressure components depend both on the axial and the radial borehole coordinate.
For the experimental validation of the theory, laboratory test stands simulating the bottom section of a repository borehole are set up as part of the LEISA test program

(Lastabtragung durch EInbettung in SALzgrus - load diversion by embedding in crushed salt). For a unique interpretation of the measured data the test stands are optimized for minimum thermal and mechanical self-deformation.
A comparison of the measurement results with the data computed by the model, a correlation between the test stand results, and a systematic determination of material-typical crushed salt parameters in a separate research project will serve to derive a set of characteristic equations enabling a description of real conditions in a future repository.
Initial experiments at a prototype test stand confirm the tendency to load reduction by embedding in crushed salt derived in the semihydrostatic model.

INTRODUCTION

Apart from the safe operation of nuclear power plants, the disposal of radioactive wastes is an essential prerequisite for the civil uses of nuclear energy. It must be ensured that neither present nor future generations will be jeopardized by a release of radioactivity. The central component of the safety criteria established by the Federal German Government implies that the definite isolation of radioactive substances in a repository must be ensured by a multistage system of natural (rock formations containing the repository) and technical barriers (treatment, packaging and solidification of wastes, final closure of the repository).
According to the emplacement concept currently applicable in the Federal Republic of Germany, it is intended to emplace heat-generating high- and intermediate-level wastes in vertical boreholes or, alternatively, in horizontal drifts in underground rock salt mines. For emplacement according to the borehole technique the waste packages are lowered from the floor of drifts at 800 m depth into 300 m deep boreholes. Each package having been emplaced is completely covered by crushed salt so that the packages completely rest in crushed salt. This type of emplacement constitutes the current reference concept for final disposal according to the borehole technique.
Crushed salt has to fulfill the following functions:
Complete embedding of the waste packages in crushed salt is to ensure that, with progressing emplacement, the stacking forces acting on the bottom packages emplaced are limited due to load diversion to the surrounding borehole wall.
Backfilling with crushed salt of the cavities left should prevent any uncontrolled dispersion of radiolytic hydrogen and krypton-85 hy free convection into the upper borehole section not yet filled.
The danger of propagating flame fronts in the borehole by an assumed ignition of radiolytically formed hydrogen should be restricted by the quenching effect of crushed salt.
In the Federal Republic of Germany, the Bundesamt fr Strahlenschutz (BfS Federal Office for Radiation Protection) is responsible for the construction and operation of plants for the final disposal of radioactive wastes. The BfS may use the services of third parties (university institutes, national research centers, industry) to fulfill these tasks.
Within the framework of the LEISA experimental program (Lastabtragung durch EInbettung in SAlzgrus - load diversion by embedding in crushed salt) as part of the ILW(Q) and HTGR fuel element experimental program financed by the BfS, it is incumbent on the KFA to experimentally validate the reference concept, the semihydrostatic model (1) for calculating the pressure distribution in vertical repository boreholes filled with crushed salt and waste packages.

THE SEMIHYDROSTATIC MODEL

The semihydrostatic model is mathematically based on the assumption that there are only steady relationships between the components of the pressure tensor with progressing borehole filling. Unsteady gliding and slipping motions occurring in real beds are not contained in the model. Steady relationships between the axial, radial and tangential pressure components pzz, prr and pzr may only be assumed for steady borehole filling with constantly plane bed surface. Analytical methods such as the semihydrostatic model can only be used on these assumptions. However, this does not imply a restriction in the sense of a restricted transferability of the model calculations to real beds. The semihydrostatic model rather provides the theoretical frame for experimental studies at laboratory test stands reduced in size.
In order to derive the basic equations, the filling of crushed salt and embedded waste packages is approximately regarded in the model as a homogeneous bed with a

density rm averaged according to the volume fractions of crushed salt and packages. Rotational symmetry is assumed. With the radial and axial borehole coordinates r and z, the balance of forces in the axial direction (1) reads:

r dr df [pzz(r,z) - pzz(r, z + dz)]
+ r df dz pzr (r, z + dz/2) - [r + dr] df dz pzr (r + dr, z + dz/2)
+ r dr df dz rm g=0 (1)

Assuming that:

prr = kpzz, (2)
pzr = mprr, (3)

which is confirmed by experimental studies (1), expanding the functions in eq. (1) to their Taylor series and neglecting fourth- and higher-order differentials, the following partial differential equation is obtained:

See Eq. (4)

where k is the lateral pressure coefficient and m the coefficient of friction between crushed salt and the borehole wall, which are regarded as parameters typical of crushed salt.

Assuming certain physically meaningful boundary conditions, the solution of the differential equation for the vertical pressure component pzz from eq. (4) reads:

See Eq. (5)

Figure 1 shows the variation in pzz(r,z) for a 400-1 waste package mk = 0.054, rm = 2366 kg/m3) for a borehole radius of RB = 0.5 m.

The maximum pressure in the borehole:

See Eq. (6)

is limited and reached at a depth of:

See Eq. (7)

The depth z0 is called conversion depth because the pressure in the borehole remains constant at pzzmax from depth z0 to the borehole floor according to eq. (6) despite further progressing emplacement. The waste packages are thus not subjected to any further increasing mechanical load from depth z0 (= 9.3 m with the parameters selected above) downwards.

Conversely, this means that the mechanical load on the bottom package emplaced in the borehole only increases until the emplacement horizon has risen by z0 = 9.3 m during borehole filling. The pressure acting on the bottom package will then remain constant at pzzmax despite further borehole filling. This statement for the bottom package can be transferred to any other package in the borehole.

STRATEGY

In order to experimentally validate the semihydrostatic model, laboratory test stands reduced in scale are constructed and set up to simulate the bottom section of a repository borehole. The aim is to formulate a transference law by means of which the experimental results obtained for smaller models (laboratory models) can be transferred to real conditions (repository) without knowing the exact boundary conditions and geometrical dimensions in a future repository. This will be achieved by comparing and matching the experimental data with the model calculations and by correlating the test stand data.

The experiments are carried out varying all decisive test parameters (borehole depth, borehole diameter, crushed salt grain distribution, crushed salt temperature, borehole wall surface, time etc.) in order to define an optimum parameter combination for low package loads.

REQUIREMENTS FOR THE TEST STANDS

The LEISA I and II test stands consist of tube segments with wall thicknesses decreasing from 12 mm (bottom) to 5 mm (top) placed into each other to form a tubular column. The following requirements are fulfilled by selecting carbon-fibre-reinforced plastic (CFRP) as the wall material:

- The thermal deformations of the wall are almost zero in the entire temperature range up to 200C examined.
- Mechanical wall deformations due to bed pressure can be structurally minimized.
- The inner surfaces of the CFRP tube segments can be reconfigured after construction.

The stability of the test stands with respect to thermal deformation is of particular significance for tests at high temperatures (HAW packages reach surface temperatures of about 200C (2)) and also for long-time experiments at room temperature over several days or weeks. Even slight dimensional changes of the tube

segments due to temperature variations in the laboratory will cause considerable pressure changes in the bed. This was shown by experiments at a prototype test stand made of steel. Such dimensional changes should be avoided as far as technically feasible since otherwise no unique interpretation of the test results is possible. It must be ensured under all circumstances that variations in the pressures measured are not caused by self-deformation of the test stands, but solely by pressure changes in the crushed salt bed.

The inner structure and inner configuration of the CFRP test stands were reproduced in a finite-element computer program prior to construction in order to simulate the most extreme thermal and mechanical loads acting on the tube segments. In comparison, calculations were performed for segments made of INVAR steel (steel variety with very low thermal expansion coefficient). Other materials such as quartz glass, graphite, silicon nitride and aluminum titanate were also under discussion as wall materials with low thermal expansion coefficient. They were not used, however, for technical or practical reasons.

The calculations show a deformation of the CFRP cylinder structure (ovalization), which is lower by about a factor of 5 compared to INVAR steel for the assumed load case of 3 bar internal pressure and 150C temperature difference. Figure 2 shows the radial deformations of the CFRP tube segments for the above load case. The CFRP segments are reinforced by a circumferential stiffening ring of CFRP in the region of the flange mounting for the sensors. A cylinder wall thickness of 10 mm was assumed for the steel segments and of 12 mm for the CFRP segments in the FEM calculations.

Compliance with the maximum permissible deformation of the CFRP segments according to FEM calculations was experimentally confirmed in preliminary specimen tests. They only give a linear thermal expansion coefficient of:

See Eq. (8)

This low value cannot be achieved with any other candidate wall material so that the higher technical and financial outlay for CFRP is justified.

The prototype test stand of steel (diameter 250 mm, height 2.5 m) has been completed and put into operation, the test stands of CFRP, LEISA I and II, with diameters of 250 mm and 600 mm and heights of 2.5 m and 5.5 m, respectively, are under construction.

MECHANICS, MEASURING ELECTRONICS AND DATA ACQUISITION

The test stands rest on steel platforms of 800 mm and 1200 mm height and are integrated into accessible working platforms. They are open at the top and closed by bottom plates.

Specifically designed pressure gauges directly measure the pressure components in the axial direction towards the bottom plate and in the radial direction towards the cylinder walls. The pressure plates accommodating the bed pressure are circular and their surfaces are adapted to the cylindrical inner wall of the tube segments. The bed pressures are transmitted to electronic sensors by sensitive mechanics. The mechanical load caused by the bed is converted into an analog electrical signal by a device which is elastically deformed by the load to be measured. The deformation is recorded by strain gauges (3).

Temperature sensors are installed on the inner and outer walls of the tube segments to continuously record the temperature within the crushed salt bed. The pressure and temperature data are registered and processed by a multi-digit measuring instrument. The instrument functions are controlled and the measured data stored by an external IBM-compatible computer.

Crushed salt is reproducibly filled into the test stands in a defined manner using a feeder. For the LEISA II test stand of CFRP (diameter 600 mm) it is planned to emplace electrically heatable inactive dummy packages reaching a surface temperature of 200C from above into the test stand using a crane.

Measuring results from the CFRP test stands are not yet available. However, all essential technical components of the test stands (mechanics and electronics) have already been successfully tested, matched and optimized under operating conditions at the prototype steel test stand. The measuring results presented below are taken from this test stand.

EXPERIMENTAL RESULTS

Crushed salt with a grain size of 0 - 10 mm is used for measurements at the prototype test stand made of steel (Fig. 3). Investigations are to reveal the maximum possible pressure in the test stand filled with crushed salt. The fine

particle fraction, XF, which is defined here as the bed fraction in the range of 0 - 2 mm, varies between 0 - 90 wt%.
The test material is produced by sieving out the 0-2 mm fine particle fraction from the total 0 - 10 mm fraction and adding it proportionally to the 2 - 10 mm fraction, depending on the requirement (10, 30, 60 and 90 wt%). The test material is then homogeneously mixed using a turbulence mixer and filled into the tube segments using the feeder. All experiments are carried out at room temperature.
Figure 4 shows the measured axial pressures pzz as a function of depth z below the crushed salt surface at the radial position r = 0.095 m for increasing fine particle fractions XF. As can be seen, the measured axial pressure initially increases almost linearly with depth z for all grain fractions. After reaching a certain depth (0.2 - 0.3 m), the increase in axial pressure decreases with increasing depth. From a depth of z = 0.5 m the pressure pzz assumes a constant value. At maximum depth of 0.5 m, load diversion for the substitute borehole of steel with a diameter of 250 mm is so effective that nearly unrestricted emplacement is possible above this depth. No further pressure increase is observed on the borehole floor. Waste packages are thus not subjected to any further increasing mechanical load in this location.
Without load diversion, a vertical pressure of about 35 kN/m2 would arise on the bottom plate for a test stand height of 2.5 m. The vertical pressures are 12 to 15 times smaller in the case of load diversion. The measurements are tendentially in full agreement with the theoretical predictions of the semihydrostatic model.
Figure 4 also shows that the measured axial pressures increase with growing fine particle fraction from pzz = 1.4 kN/m2 for XF = 0 wt% to about pzz = 3.0 kN/m2 for XF = 90 wt%. This is only attributable to a minor extent to the larger packing density of crushed salt with elevated fine particle fraction. The density only increases by 10% when the fine particle fraction is increased from 0 to 60 wt%. It is even lower again for XF = 90 wt% compared to a fine particle fraction of XF = 60 wt%.
The increase in pressure with growing fine particle fraction is probably caused by a decrease of the friction coefficient with increasing fine particle fraction. A smaller friction coefficient causes less wall friction and thus a higher axial pressure. In order to quantitatively elucidate this phenomenon, the friction coefficients of crushed salt are systematically determined in a separate research program.

CONCLUSION

The measurement technique used and the mechanical setup are suitable for determining the pressure distributions within a crushed salt bed in tube segments simulating the borehole wall. The experimental studies show good agreement with the calculations according to the semihydrostatic model. The equipment components used prove to be extremely reliable. Measurements to date have shown that the pressure distribution in vertical boreholes decisively depends on the grain distribution of crushed salt and on the filling process. After construction and commissioning of the CFRP test stands (LEISA I and II) extensive further studies varying all essential parameters (borehole radius, borehole depth, grain distribution, mechanical crushed salt parameters, time, temperature, etc.) will be carried out to obtain a more comprehensive picture concerning load diversion in vertical boreholes backfilled with crushed salt.

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CONVERGENCE BEHAVIOR OF PRESSURIZED SALT CAVITIES

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ABSTRACT
A possibility for underground disposal of low- and medium-radioactive waste is emplacement in a dry salt cavity. In connection with the long-term safety and the quality of the natural sealing, it is advantageous to consider the use of deep-lying cavities. To limit the convergence of such cavities during the filling-up phase it is necessary to maintain an internal gas pressure for the duration of this phase. This paper gives results of numerical calculations on the convergence behavior of a cavity as a function of depth, internal gas pressure and cavity radius during the filling-up phase. Results for a cavity radius of 30 m are, that for depths of 1500 and 2200 m the necessary internal gas pressures are about 10 and about 30 MPa, respectively. Furthermore it was established that the convergence rate increases with increasing cavity radius. The convergence behavior in the abandonment phase, during which the gas pressure will be atmospheric, has also been investigated. Calculations show that in this phase the convergence rate changes from very rapid in roughly the first 2 years to very low after more than about 10 years.
INTRODUCTION
An option for underground disposal of solid, non-heat-generating radioactive waste is emplacement in a salt cavity. Both a brine-filled cavity and a dry cavity can be considered. The brine-filled cavity has two disadvantages: i) potentially a large amount of liquid can be contaminated and ii) after cavity abandonment and sealing the permeability of the cavity roof can be enhanced considerably (1). Therefore the dry cavity is to be preferred above the brine-filled cavity. Unfortunately, the dry cavity has an important disadvantage. As a consequence of, mainly, the increasing convergence rate with depth, only depths down to about 1000 m can be reached. Larger depths could be attained, however, if the internal cavity pressure is increased by compressing the air in the cavity during the filling-up phase. During the abandonment phase the cavity is depressurized.
A paper on this disposal technique was presented at the Waste Management '94 Conference (2). The advantages and drawbacks were discussed and preliminary results on cavity convergence and stability were presented. The main advantage of the technique is, that disposal of waste would be realized at a relatively large depth. It can be argued that the quality of the isolation increases with depth. Moreover, at such a depth the natural sealing of the waste within the salt body occurs much more rapidly. Disadvantages are that the cavity has a considerable internal pressure for a period of a few years and that the waste has to be emplaced in a pressurized cavity. Details of the emplacement technique and procedure have still to be worked out (2).
In the current paper the attention is focused on the convergence behavior of the cavity. Numerical calculations have been carried out on cavity convergence at different depths, both for the filling-up phase and for the abandonment phase. This paper summarizes the most important results of these calculations.
SYSTEM PARAMETERS
Depth range, in-situ stress and geothermal gradient
Depths between 1500 m and 2200 m have been investigated. Within this range two specific depths received special attention: 1700 m and 2000 m.
The in-situ stress at a depth of 1400 m was taken as 30 MPa. Below this depth the principal stresses far away from the cavity were supposed to increase linearly with depth according to:

$$\sigma_{xx} = \sigma_{yy} = \sigma_{zz} = 30106 + \rho g D h \ [Pa]$$

The rock-salt temperature was assumed to increase with depth according to:

$$T = 281.7 + 0.03h \ [K]$$

Cavity pressure
The internal cavity pressure during the filling-up phase was taken as 10, 15, 20, 25 or 30 MPa. During abandonment the cavity pressure was assumed to be atmospheric.
Cavity shape
The cavity shape was supposed to be spherical. This shape was chosen because, in an isotropic lithostatic stress situation, the sphere is the most favorable (i.e. the most stable) configuration. Most calculations have been carried out for a radius of

30 m, a rather arbitrary choice. However, also calculations with cavity radii of 10, 20, 40 and 50 m were performed.

CALCULATION PROCEDURE

The simulation program FLAC

The convergence calculations were performed using the finite difference code FLAC (Fast Lagrangian Analysis of Continua), version 3.2. This software package has been developed by Itasca Consulting Group, Inc., Minneapolis, Minn.

To simulate the creep behavior of salt the "WIPP creep model" (WIPP: Waste Isolation Pilot Plant, Carlsbad, N. Mex.) is available in FLAC. This model is based on a reference creep formulation developed in earlier waste isolation studies (4). FLAC also incorporates an experimental model which simulates volumetric hardening and non-linear irreversible compaction. This "double-yield model" was used to represent the mechanical properties of the solid waste.

Filling-up phase

During the filling-up phase the convergence behavior of the cavity is determined by the system parameters (depth, cavity diameter, internal pressure) and the mechanical properties of rock salt. The shear and the bulk modulus of rock salt were chosen as 9.84 and 18.1 GPa, respectively (3). These values correspond with a modulus of elasticity of 25 GPa and a Poisson ratio 0.27. For the convergence calculations it was convenient to use the WIPP creep model (3,4). Two creep regimes can be distinguished:

$$\dot{\varepsilon} = \dot{\varepsilon}_s + (A - B\varepsilon_p)\dot{\varepsilon}_s \quad \text{for } \dot{\varepsilon}_s \geq \dot{\varepsilon}^*_{ss}$$

$$\dot{\varepsilon} = \dot{\varepsilon}_s + [A - B(\dot{\varepsilon}^*_{ss}/\dot{\varepsilon}_s)\varepsilon_p]\dot{\varepsilon}_s \quad \text{for } \dot{\varepsilon}_s < \dot{\varepsilon}^*_{ss}$$

The secondary strain rate is a function of stress and temperature:

$$\dot{\varepsilon}_s = D\sigma^n e^{-Q/RT}$$

Since the cavity is assumed to be perfectly spherical and to be embedded in an infinite isotropic medium, this cavity and its immediate surroundings can be simulated by an axisymmetric configuration. This two-dimensional model is thought to be rotated along a vertical central axis, is 300 m wide and 600 m high (thus having boundaries at about 10 times the cavity radius) and is divided into 200 concentric zones.

Prior to the calculations proper the model is brought to an elastic equilibrium with negligible displacements. Then creep calculations are started, with initial time-steps of 100 seconds. These time-steps increase automatically when the creep rate decreases.

Abandonment phase

During the abandonment phase the convergence behavior of the cavity is determined by the system parameters (depth, cavity diameter and internal pressure; this last parameter is now assumed to be atmospheric), the mechanical properties of salt (see above) and the mechanical behavior of the solid waste, simulated by the double-yield model. This model is intended to represent granular (lightly cemented) materials that show irreversible compaction when placed under pressure, and increase in stiffness as the material becomes more compact. To our knowledge no information is available on the convergence behavior of solid waste in a salt cavity, but is seems reasonable to assume that the waste will decrease with, say, 20-25 vol. % before fully withstanding the converging cavity walls. Therefore the mechanical properties characteristic for 30:1 sand-cement fill material are used here, viz. bulk and shear modulus of 450 and 600 MPa, respectively, a friction angle of 40 and zero cohesion and zero tensile strength. Table I gives data on the assumed, strongly non-linear, stress-strain behavior of the cavity contents.

In the abandonment phase the calculation procedure is simular to that in the filling-up phase, the only difference being the different boundary condition at the cavity/salt interface, which is now a strain-dependent (and, consequently, time-dependent) pressure.

CONVERGENCE BEHAVIOR DURING THE FILLING-UP PHASE

Figure 1 gives the results of the convergence calculations for a depth of 1700 m. The volume convergence decreases significantly with increasing internal gas pressure. E.g. after 5 years the volume convergences for pressures of 15 and 25 MPa are 9.9 and 1.1 %, respectively. Corresponding values for a depth of 2000 m are 34.0 and 7.3 %, respectively, see Fig. 2. The relationship between volume convergence and time is non-linear. The tendency is that the increase in convergence decreases with time.

Fig. 1. Convergence of a spherical cavity, radius 30 m, with the cavity center at

1700 m depth, for different internal gas pressures.
Fig. 2. Convergence of a spherical cavity, radius 30 m, with the cavity center at 2000 m depth, for different internal gas pressures.
For the operation of a pressurized cavity it is required that the total volume convergence is kept below a certain limit, so that during the filling-up phase the cavity volume is kept reasonably constant. If for this limit a convergence of 10 % in 5 years is chosen, this implies the existence of a specific minimum pressure for each depth. For the chosen depth interval the calculations yield that these minimum pressures vary from 9.2 MPa (cavity center at 1500 m) up to 29.1 MPa (cavity center at 2200m).
The results given above are valid for a cavity radius of 30 m. The influence of the cavity radius on the convergence rate has been investigated. Figure 3 gives the results of the calculations, which show that the volume convergence increases with increasing radius. We cannot provide a satisfactory explanation for this scale effect. Possibly it is caused by the rock-mechanical properties of rock salt. For a small cavity the zone around the cavity undergoing significant deformation is relatively small and growing only slowly. This may imply that, for a small cavity, there is a larger tendency of the rock-salt formation as a whole to resist deformation, leading to a smaller cavity convergence than in the case of a large cavity.
Fig. 3. Convergence of a spherical cavity, with the cavity center at 1700 m depth and with an internal gas pressure of 15 MPa, for different cavity radii.

CONVERGENCE BEHAVIOR DURING THE ABANDONMENT PHASE

Once the cavity has been completely filled up, it is no longer necessary to maintain the internal pressure. In practice the pressure will be gradually decreased by letting the cavity air slowly escape during a period of, say, a few months. It may or may not be necessary to filter the escaping air. For the purpose of our calculations it was assumed that, at the end of the filling-up phase, the internal pressure was instantaneously reduced from the prevailing pressure to atmospheric pressure.
Figure 4 gives the data for the cavity convergence during the abandonment phase for a cavity depth of 1700 m. Again it was assumed that the cavity radius was 30 m. Because the internal pressure has been drastically reduced, we see initially a very rapid convergence in the first 2 years of cavity abandonment. This behavior corresponds with the rapid reduction of the porosity of the solid waste. With decreasing porosity it will become more difficult to compact the waste even more, with the result that the convergence rate decreases significantly with time in the next years. After about 10 years the convergence rate has become relatively small. The convergence rate keeps decreasing in time, but does not approach the value zero in the investigated time period (0-20 years).

DISCUSSION

The calculations presented here were based on the mechanical properties of WIPP salt. It is well-known that the mechanical properties of rock salt vary with formation type and location, see e.g. Lux (5). Therefore the results of the calculations must be regarded as being indicative rather than uniformly valid.
The influence of the cavity radius on the convergence rate has been investigated (see Fig. 3), and this yielded that this influence is surprisingly strong. A possible explanation has been suggested above, but more fundamental insight concerning this phenomenon is necessary. In future activities attention will be given to this aspect.
In this paper no specific attention was given to the stability phenomenon. The calculation of the convergence rate makes only sense in cases where one has to do with a stable cavity. In the WIPP creep model this aspect is not incorporated. This means that - in a specific situation of cavity depth, cavity radius and internal pressure - the convergence rate is evidently of importance, but that always must be checked whether one has to do with a stable cavity in a rock-mechanical sense. Also to this aspect attention will be given in future activities.
Finally we point out that, in the framework of waste disposal in a pressurized salt cavity, also attention should be given to subjects not mentioned up till now: subsidence as a result of cavity operation, final cavity and borehole sealing, and analysis of the safety consequences of this disposal design.

CONCLUSIONS

1. For a spherical cavity with a radius of 30 m the minimum gas pressure necessary during the filling-up phase varies from about 10 MPa (1500 m depth) to about 30 MPa (2200 m depth).
2. The cavity convergence rate increases with increasing cavity radius.
3. During the first 2 years of the abandonment phase the cavity convergence rate is very rapid, but this rate decreases quickly and becomes very low for times of about 10 years and more.
4. Some aspects of the disposal method treated in this paper, such as waste handling and abandonment procedure, have still to be worked out.

LIST OF SYMBOLS

A constant [numerical value 4.56 (3)]
B constant [numerical value 127 (3)]
D constant [numerical value 5.7910-36 Pa-4.9s-1 (3)]
g acceleration of gravity [numerical value 9.81 m.s-2]
h depth [m]
n exponent [numerical value 4.9 (3)]
Q activation energy [numerical value 5.02107 J.kmol-1 (3)]
r cavity radius [m]
R universal gas constant [8320 J.kmol-1K-1]
S deviatoric stress [Pa]
T absolute temperature [K]
Dh depth below 1400 m [m]
ep primary creep strain [dimensionless]
r specific mass of rock salt [numerical value 2200 kg.m-3 (3)]
sxx principal stress in x-direction [Pa]
syy principal stress in y-direction [Pa]
szz principal stress in z-direction [Pa]
f creep rate [s-1]
fs secondary creep rate [s-1]
f*ss critical steady-state creep rate [numerical value 5.3910-8 s-1]

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OPTIMIZATION OF ENGINEERED BARRIERS FOR DEEP DISPOSAL

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ABSTRACT

United Kingdom Nirex Limited (Nirex) is owned by the major organizations in the UK nuclear industry. The company is responsible for developing and operating a deep repository for disposal of solid intermediate-level and some low-level radioactive waste. Nirex is concentrating its investigations on a site close to the BNFL Sellafield works in Cumbria in the North-West of England.

Nirex in common with many other disposal agencies has developed a concept of deep geological disposal for radioactive waste which uses a multi-barrier containment system. The concept makes use of both engineered and natural barriers, working in conjunction, to achieve the necessary degree of long-term waste isolation and containment. An important component of the engineered system is the disposal package comprising waste container and solid wasteform.

The principal type of container for ILW is an unshielded stainless steel drum of

nominal 500 liters capacity. Such containers are currently being filled with immobilized waste in readiness for eventual disposal to the Nirex repository. Research has demonstrated that such waste packages will be suitable for 50 years on-site storage before transport to the repository. Once emplaced in the repository and surrounded by cementitious backfill grout, waste packages will make a significant contribution to the engineered component of the multi-barrier containment system.

INTRODUCTION

United Kingdom Nirex Limited (Nirex) is owned by the major organizations in the UK nuclear industry. The company is responsible for developing and operating a deep repository for disposal of solid intermediate level and some low-level radioactive waste (ILW and LLW). Nirex is concentrating its investigations on a site close to the BNFL Sellafield works in Cumbria in the North-West of England.

Nirex in common with other disposal agencies has developed a concept of deep geological disposal for radioactive waste which uses a multi-barrier containment system. The concept makes use of both engineered and natural barriers, working in conjunction to achieve the necessary degree of long-term waste isolation and containment. An important component of the engineered system is the disposal package comprising waste container and solid wasteform.

For the Sellafield site, attention is being focused on disposal at a depth of about 650m below Ordnance Datum (about 750m below the surface), in the Borrowdale Volcanic Group (BVG) of rocks. Nirex aims to bring a repository into operation by the year 2010.

MULTI - BARRIER CONTAINMENT

The Nirex concept for deep geological disposal is similar to that being considered in several other countries. Caverns will be excavated at depth in a suitable geological environment. Wastes, cemented into steel or concrete containers, will be placed in caverns which will inturn be backfilled with a cement based material. The multi-barrier concept is illustrated schematically in Fig. 1.

This concept makes use of both engineered and natural barriers to provide the necessary long-term isolation and containment of the waste. The steel or concrete packaging is of importance during the operational phase and for several hundred years after closure. However, the long-term containment properties of the engineered system with regard to radionuclides dissolved in groundwater will stem from the establishment of uniform chemical conditions and high sorption capacity across the repository. This will be achieved by surrounding waste packages with the required amount of special cement based backfill (1). The Nirex vault backfill has been carefully specified to fulfil a number of requirements, namely:

a) long-term maintenance of alkaline pore water chemistry in order to suppress dissolved levels of key radionuclides under the prevailing conditions of groundwater flow and geochemistry;

b) long-term maintenance of a high active surface area for sorption of key radionuclides;

c) relatively high permeability and porosity to ensure homogeneous performance, in order that localized concentrations of materials in wastes do not exhaust the desired chemical conditioning and thereby locally reduce the containment performance.

With this concept a source term for dissolved radionuclides in repository pore water can be derived in a manner that is relatively simple and therefore easily understood. The natural barrier provided by the deep geological setting serves to control the rate of access of groundwater to the waste, to delay and spread radionuclide migration to the biosphere and to limit the probability of disruption by human-initiated or natural events.

Long-term containment properties of the engineered system therefore rely on the establishment of appropriate chemical conditions in the near-field. The physical barriers, provided by the disposal container and wasteform, do not contribute to containment in the long-term but are important in the short-term, during the operational period of the repository and in the years immediately following closure. Thus the performance of the waste package (ie container and wasteform) has to be specified to provide adequate containment in the repository environment whilst also being suitable for waste processing, interim storage, handling and transport.

WASTE PACKAGE SPECIFICATIONS

Working in conjunction with producers of radioactive waste, Nirex has defined a

range of standard containers suitable for accumulations and future arisings of waste in the UK. The standard containers specified by Nirex are listed on Table I. Each has been justified on the basis of an identified need for packaging a particular range of wastes.

Waste Package Specifications have been produced for each of the standard packages. Each Specification defines dimensional, functional and performance criteria for waste containers and wasteform. These criteria include activity content, dose rate, surface contamination, heat output, dimensions, shape, handling features, venting and filtration, impact performance, integrity, stackability and identification. The Specifications incorporate the various requirements for disposal and transport and are compatible with the requirements for waste packaging, storage and handling at the sites of origin. In the absence of Conditions for Acceptance, which cannot be issued until much closer to repository opening, Waste Package Specifications provide an essential link between waste package design and repository design. The Specifications have enabled waste producing organizations to design containers and build waste packaging plants. At the beginning of 1995 some 6,000 standard 500 liter drums had been filled with conditioned waste and are in store awaiting disposal to the Nirex repository. For Nirex, the Specifications have provided a key element of the basis of design for the repository and transport system and also for the various safety cases to be made to the regulatory authorities.

500 LITER DRUM

The 500 liter drum is the principal type of container for ILW. It will be used for the greater part of operational wastes arising from day-to-day operations of nuclear facilities. Within the UK, intermediate level operational wastes mainly include fuel cladding and fuel element debris, sludges and ion exchange materials, miscellaneous solid waste and plutonium contaminated materials. The 500 liter drum can also be used for the packaging of decommissioning wastes, but larger containers such as the 3m3 box or 4m box may be more appropriate.

A typical 500 liter drum is shown in Fig. 2. The drum has a diameter of 800mm and an overall height of 1200mm. Its nominal capacity is 500 liters. A cylindrical shape was chosen to facilitate in-drum mixing using a paddle. This shape ensures that there will be no 'dead' volumes where mixing of the cement and waste will not take place. Likewise it has been found that a ratio of height to diameter of 1.5 is good for in-drum mixing although this ratio may be varied considerably. An overall diameter of 800mm is considered optimum on transport efficiency grounds. It will allow transport within the British Rail W6A loading gauge with the required amount of shielding. In addition to in-drum mixing, the drum may also be used for in-drum grouting of solid waste items.

The drum is not intended to provide radiation shielding. This was a decision taken as a result of an analysis of the options of disposable shielding and reusable shielding for ILW containers (2). It was shown that relatively cheap materials for disposable shielding, such as concrete, would mean high disposal charges because of the large volumes being disposed of along with the waste. Denser materials such as steel, although incurring lower disposal costs owing to their smaller volume, would have higher waste container costs. Reusable shielding was thus the most economical solution and Nirex is developing a family of reusable transport containers to meet this need (3). The transport container will carry four 500 liter drums in a handling stillage or one 3m3 box or drum. The transport containers are being designed to meet IAEA Type B requirements (4) and will be manufactured in a range of nominal shielding thicknesses up to a maximum of about 285mm. One consequence of the decision to adopt reusable shielding for drummed ILW is that the waste containers will need to be handled remotely both during and after unloading at the repository. A transport container carrying four 500 liter drums is illustrated in Fig. 3.

It has not proved possible to standardize on one single design of 500 liter drum because the processes for immobilising different wastes require variations to the drum, mainly in the lid area and in the internal drum 'furniture'. A limited number of drum shapes has been adopted, each shape allowing variations on lidding and internal arrangements, but all having common lifting and handling arrangements. As shown in Figure 2, the lifting feature is provided by a rim of diameter 800mm at the top of the drum. The dimensions of this lifting feature are specified to permit all drums to be lifted by the same handling equipment.

Typically drums are filled with waste to within 100 - 200mm of the top of the drum.

A gap is left at the top to prevent the spread of contamination by splashing and spilling of the waste or grout introduced for immobilization. An inactive grout cap is then placed above the waste, leaving only a relatively small air gap at the top of the drum. The size of this air gap varies between a few millimetres to about 40mm.

Methods of closing drums include bolting and welding. For wastes which generate gases a vent is provided. If it is expected that sufficient quantities of gas will be produced to entrain particulate material, the vents will be filtered. This allows gases to escape, preventing pressurization and build-up of flammable gas mixtures.

All current 500 liter drums are manufactured from AISI 316L stainless steel, typically using grade 316S11 (5) for the body and sometimes grade 304S11 in the lid area. The thickness of the stainless steel body is between 2 and 3mm. The use of thin-section stainless steel sheet permits drums to be fabricated using conventional spinning and pressing technologies. The actual wall thickness adopted is not a Nirex specified requirement but is the choice of the drum designer in response to the Nirex specified design lifetime for the container.

DRUM INTEGRITY REQUIREMENTS

The Waste Package Specification defines minimum requirements for waste package integrity. Requirements cover a period of interim above-ground storage and the period post-emplacement in the repository when the package contributes to the engineered barrier.

The specification is worded so as to provide flexibility to drum designers, to enable designs to be optimized to suit the requirements of particular wastes and storage conditions. The specification calls for drum material, thickness, corrosion properties and sealing method to be such that following storage at the producer's site for 50 years, the drum will provide containment during:

- transport and handling at the repository
- post-emplacement for a minimum period of 50 years in an alkaline environment

The 50 year interim storage period is specified to account for the fact that wastes packaged now may have to be stored for that length of time before they can be accepted for disposal. Following emplacement in the repository, the specification calls for a minimum lifetime of 50 years to ensure that containment is maintained during the repository operational period. For wastes containing short-lived soluble activity, best practical means are to be employed to extend package integrity beyond the 50 year operational period.

In practice, packages manufactured from grade 316L stainless steel and backfilled with the Nirex backfill material, are expected to exhibit corrosion characteristics significantly better than the minimum requirements specified.

WASTE CONTAINER CORROSION PERFORMANCE

In order to determine the corrosion performance of waste packages following disposal and to assess the contribution of waste packages to the engineered barrier, Nirex has carried out a substantial research program to examine the performance of grade 316L stainless steel under a range of conditions.

During interim storage the waste containers will be held in purpose-built storage buildings in which the temperature and humidity levels will be controlled. Some storage buildings will be located near the coast and consequently there is a possibility of chloride-containing particulates being present in the atmosphere. 316L stainless steel is resistant to atmospheric corrosion; long term exposure to marine conditions may lead to slight staining but no significant metal loss. The most likely form of corrosive attack is crevice corrosion around the lids of the containers, but this should be slight provided condensation can be prevented.

Repository Environment

Provided the correct environmental controls are maintained during storage, the waste containers will be free of significant corrosive damage at the end of the storage period, at which time they will be transported to the repository, where they will be emplaced and covered with the cement based backfill described previously. The corrosion performance of stainless steel drums, and hence their contribution to the engineered barrier will largely be determined by the environmental conditions within the backfilled waste stack. The principal environmental factors controlling the extent of corrosion are oxygen availability, presence of chloride, temperature and the chemical conditioning provided by the backfill.

Concentrations of oxygen and chloride in repository porewater will change as a

function of time. The oxygen concentration will fall as a result of consumption by microbial activity and corrosion of structural carbon steel and eventually the repository will become anaerobic. On the basis of current models it is expected that anaerobic conditions will be achieved after a period of ten to a few hundred years. The chloride concentration in the repository will increase as it resaturates with groundwater. The chloride concentration measured in the groundwater depends on the site where it is sampled, but is typically up to 15,000 ppm (6). It is expected that resaturation will occur gradually as groundwater enters the pores of the backfill. The groundwater has a very low oxygen content and resaturation will not therefore lead to an increase in the oxygen inventory of the repository. Groundwater entering the backfill will be chemically conditioned and the porewater will be alkaline. The temperature in the repository will peak at around 80oC shortly after closure, then decay back to the background temperature of approximately 50oC (7). Corrosion behavior of stainless steel during this period will depend on the relative timescales for oxygen depletion, water availability, chloride ingress and temperature variation.

Corrosion Mechanisms

Several different forms of corrosion can affect metals when they are immersed in a corrosive environment. General corrosion refers to removal of material over a broad surface area. The general corrosion rate of stainless steel is determined by the properties of the oxide film on the surface (the passive film) and is less than 1mm/year in both aerobic and anaerobic conditions. Measurements of passive current densities indicate that the corrosion rate may be as low as 0.01mm/year (7). Localized corrosion includes such phenomena as pitting, crevice corrosion and stress corrosion cracking. Although these forms of corrosion do not lead to the loss of large amounts of metal they can lead to rapid penetration, which may result in the formation of leakage paths if experienced in waste containers. In order to examine the potential for these mechanisms to occur under expected repository conditions Nirex has carried out extensive experimental and theoretical studies.

Pitting Corrosion. Studies have shown that the chemical conditioning provided by the Nirex vault backfill plays an important role in limiting the occurrence of pitting corrosion, even in environments where this mechanism would normally be anticipated. The controlling factor appears to be the ratio of chloride ions to hydroxyl ions. Nirex studies to understand the mechanism are continuing but initial results indicate that pitting corrosion is not expected where this ratio is less than 10. This result suggests that chloride levels of 20,000 ppm could be tolerated in the repository groundwater where this has been conditioned by the Nirex vault backfill.

Crevice Corrosion. Crevice corrosion is the most difficult form of corrosion to study and predict because it is sensitive to such parameters as crevice geometry and surface finish. The most likely regions to be affected are the crevices formed around the lids of waste containers. The restrictions on the mass transport of materials into the crevice result in the development of micro-environments inside the crevice which can be highly acidic, even when the external environment is alkaline. Further experimental research is planned to investigate this phenomenon. Mathematical models (8), have also been developed as an aid to assessing whether a crevice of a particular geometry will activate in particular environmental conditions. The model simulates the evolution of the chemical and migration processes within a passively corroding rectangular crevice.

Stress Corrosion Cracking. The likelihood of Stress Corrosion Cracking (SCC) occurring in waste containers has also been assessed (9). From a literature review it is concluded that at temperatures below 100oC, and therefore in the absence of concentration effects caused by boiling, alkaline-induced SCC can be ruled out. However, chloride-induced SCC is considered to be a possibility and the likelihood of its occurrence predicted to increase with increasing temperature. Further research work in this area is in progress.

Microbially Influenced Corrosion. In conditions where microbial growth can occur, for example where there is a supply of organic nutrients and trace elements, microbially influenced corrosion (MIC) of metals can sometimes occur. Such corrosion often takes the form of bottle-shaped pits. A number of species have been implicated in MIC and each has its own specific nutritional requirements. MIC is unlikely to occur in the high pH and high radiation flux conditions which will prevail in cemented ILW wasteforms (10) but it cannot be ruled out for uncemented LLW wastes with a high proportion of organic wastes.

Corrosion Performance

From the above discussion of corrosion mechanisms it can be seen that stainless steel waste containers will experience very low rates of general corrosion. At the time of transport to the repository, perhaps after a period of 50 years above-ground interim storage, stainless steel waste containers should be unaffected by corrosion given appropriate storage conditions. Following emplacement in the repository, waste packages will initially experience aerobic conditions and even in the presence of chloride, general corrosion rates of less than 1mm/year are predicted. The typical stainless steel drum shown in Figure 2 can therefore expect a lifetime in excess of 1000 years.

The effect of localized corrosion is however more difficult to predict. Stainless steel can be susceptible to localized corrosion mechanisms under the combination of aerobic conditions and chloride. The alkalinity provided by the Nirex vault backfill is beneficial and will control pitting by the provision of hydroxyl ions. Nirex work to examine the effect of localized corrosion and its variation with oxygen availability, chloride level, pH and temperature is continuing. When the repository environment becomes anaerobic, localized corrosion mechanisms will cease, leaving corrosion to proceed at the general rate.

The choice of stainless steel as the material for the manufacture of 500 liter drums has a number of advantages. The drums can be made from thin-section sheet enabling manufacture using conventional spinning and pressing techniques. The drums can be used as process vessels during waste conditioning and because they are unshielded, enable efficient use to be made of interim storage facilities. When waste packages are despatched to the repository, they should be unaffected by corrosion, given appropriate conditions during the storage period, and following emplacement and backfilling will corrode very slowly. The long-term containment properties of the repository do not rely on a physical barrier but the drum will nevertheless provide a significant barrier to the movement of radioactivity.

CONCLUSIONS

The Nirex concept for deep geological disposal makes use of both engineered and natural barriers to provide the necessary long-term isolation and containment of waste. The principal element of the engineered system is the establishment of appropriate chemical conditions in the near-field. This is achieved by surrounding waste packages with a special cement based backfill.

The physical barriers provided by the disposal container and wasteform do not contribute in the long-term but are important in the short-term during the operational period of the repository and in the years following repository closure. The principal container for ILW, the 500 liter drum, fullfils various functions throughout its life, from process vessel and interim storage container through to eventual disposal container in the repository. The selection of grade 316L stainless steel for the manufacture of drums has permitted wastes to be packaged in advance of the opening of the repository and to be interim stored in a cost-effective manner. Research studies indicate that these packages will be materially unaffected by corrosion during interim storage and can make a significant contribution to the engineered component of the multi-barrier containment system following disposal.

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5-4

UNDERGROUND CONCEPTUAL REPOSITORY DESIGN AT YUCCA MOUNTAIN, NEVADA

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ABSTRACT

The Yucca Mountain Project, which is located approximately 175 km northwest of Las Vegas, Nevada, is the site that Congress has directed the U.S. Department of Energy to characterize to determine its suitability for disposing up to 70,000 tonnes of radioactive waste. This amount is comprised of approximately 63,000 tonnes of Spent Nuclear Fuel (SNF), and about 7,000 tonnes of High Level Waste (HLW). The Department of Energy (DOE) is scheduled to begin taking possession of SNF in 1998, but is not scheduled to actually dispose waste packages until the year 2010 if the site proves suitable for repository purposes.

The project schedule as shown in Table I is based upon a number of major products or milestones. High Level Findings (HLF) of a scientific nature will collectively support the determination of Technical Site Suitability (TSS) scheduled for 1998. TSS will supply an initial reading on the technical suitability of the site, and, if found suitable, will allow further development of program activity.

The overall project is comprised of two major phases: the Exploratory Studies Facility (ESF), and the potential repository. Both phases will make extensive use of mechanical excavation techniques in the development of main accesses, test alcoves, long term underground test rooms, and ultimately, emplacement drifts.

PROGRAM APPROACH

The Site Characterization Plan issued in 1988 contains an extensive testing, design, and performance assessment program that provide a comprehensive understanding of Yucca Mountain. The original intent of the Nuclear Waste Policy Act and of the Nuclear Regulatory Commission was to create a site characterization program that provides sufficient information for decision making, realizing that some uncertainty must exist. The series of decisions to be made in the various licensing stages imply an increasing knowledge base much of which is gained after the construction authorization. In fact, the 1990 report, "Rethinking High-Level Waste," by the National Academy of Sciences stressed that it is unrealistic to assume that all information can be available before constructing a repository. The Program Approach represents a fundamental change in the site characterization approach. Whereas the Site Characterization Plan described a program that attempted to provide answers for all possible questions related to the site, the Program Approach recognizes that the resources required to carry out such a program have not been and are unlikely to be available. The Program Approach is consistent with the original intent of the legislative and regulatory framework, as well as recommendations of the National Academy of Sciences.

Under the Program Approach, the site characterization program initially will focus on those tests and analyses most critical to suitability evaluation and licensing issues. If the site appears to be suitable, the focus of the Program Approach will shift emphasis to acquiring the additional site characterization data and developing the analyses needed to submit an application to the Nuclear Regulatory Commission for a Construction Authorization. These data and analyses will seek to provide confidence in the safety of repository operations and waste package containment. The data and analyses needed to support compliance with requirements related to longer-term radionuclide release and transport will likely be of a probabilistic nature, conservative, and flexible enough to accommodate many possible site conditions. Additional confidence in the certainty of these data and the demonstration of the long-term performance of the site will be achieved through the performance confirmation program. The repository will be designed to permit waste retrieval for up to 100 years from the start of waste emplacement, twice as long as the 50-year retrieval period required by the Nuclear Regulatory Commission.

The D.O.E. must develop those necessary data and analyses to enable the Nuclear Regulatory Commission to make findings with reasonable assurance with regard to the performance of the site. This process is manifested by a maturing understanding of the performance of the natural and engineered components of the system as well as the system's sensitivities to the performance of its individual components. Iterative performance assessments will serve to drive and document the maturation of this understanding.

WASTE PACKAGE DESIGN

Waste Package/Engineered Barrier (WP/EB) system design options are being developed and evaluated for applicability to the component design, performance, and MGDS design requirements. The EB Segment is comprised of the emplaced WPs and any other engineered devices or material required to provide waste isolation. Several different concepts for WP/EB packaging were considered for evaluation: Multi-Purpose Canister (MPC) and Disposal Container, Uncanistered Spent Nuclear Fuel (SNF) WP, and defense high level WP. These were subjected to parametric evaluations which included thermal response of the WP and repository with respect to a number of variables such as: age of SNF, burnup, initial U-235 enrichment, repository thermal mass loading and Area Power Density (APD), drift spacing, WP spacing, properties of the WP and repository material, and SNF receipt rate. Included in the evaluations was the attempt at understanding the long term criticality behavior, which investigated different initial enrichments and configurations, and ultimately contributed heavily to the understanding of long term disposal requirements.

Of the different options available, the MPC concept has received the most attention for immediate development and implementation. This concept has refined the method of packaging the waste for transportation and disposal. The MPC is currently undergoing design and fabrication, which will allow SNF to be packaged at the source location and handled using various overpacks for transportation and disposal. (See Fig. 1). This concept greatly affects design and operations for transportation, interim storage, surface repository facilities, and ultimate underground disposal. As structured, this "cradle-to-grave" strategy reduces the necessity for extensive hot cell facilities at the repository surface location, since the packages will not have to be opened at the site before disposal

Several different WP capacities have been evaluated, but recent interest has centered on the large multibarrier WPs with capacities up to 21 Pressurized Water Reactor (PWR) assemblies or 40 Boiling Water Reactor (BWR) assemblies. PWR assemblies are considered limiting because of their higher decay heat output compared to BWR assemblies. Higher capacity WPs are more likely to exceed thermal goals than smaller ones in the same repository thermal environment. The choice of design-basis fuel is important because it will directly limit the number assemblies that can be loaded into a WP or MPC and still meet thermal goals; the limiting thermal goal for large WPs is 350 degrees C at the SNF cladding. For the MPC conceptual, design-basis fuel characteristics of 10 years aged with 40 GWd/MTU burnup, the 21 PWR capacity is considered at or above the maximum allowable temperature for a metallic multibarrier WP such as the MPC with disposal container. All these factors affect the timing of peak temperatures as well as the magnitude. Host rock temperatures will peak between 20 and 500 years depending on the thermal loading, but is largely independent of the individual WP design. The WP will experience its peak temperature between initial emplacement and the repository peak depending on the design-basis fuel and the basket/container design. For the large WP, higher conductivity SNF baskets will lower and delay the peak temperatures experienced. The choice of the design-basis fuel is of key importance to the timing of peak temperatures. Younger fuel types produce high peak temperatures within the first few years, but then temperatures drop off quickly. Older fuel (at the same APD) produces lower temperatures and later peaks with more stable and higher long-term temperatures.

REPOSITORY DESIGN

Repository design on the Yucca Mountain project is made up of two primary components, surface and subsurface, and is currently in the Advanced Conceptual Design (ACD) phase. This phase will define major program elements which are necessary for integration with the ESF, testing, MPC development, transportation and environmental concerns. Significant concepts will be established for future development and design definition in the License Application Design (LAD) and Final Procurement and Construction Design (FPCD). The various phases of design are

scheduled to coincide with major program milestones as indicated on Table I. Information from products developed during these design phases will be used to support High Level Findings (HLF), Technical Site Suitability (TSS), the Environmental Impact Statement (EIS), and License Application (LA). An appropriate amount of information that is necessary to define systems, subsystems and components is scheduled for completion prior to each of the major milestones.

Surface repository facilities consist of two major areas: the general support facilities area, and the more closely controlled and secure geologic repository operations area (Fig. 2). The general support facilities area will provide normal service functions such as offices, warehousing, maintenance shops, utilities substations, and miscellaneous installations. These will be of normal commercial quality, and will not involve any nuclear related design or subsequent interface with nuclear materials.

The geologic repository operations area is adjacent to the general support facilities, and will be the center for receipt and handling of nuclear waste shipments. Conceptually, the area contains the following basic structures: performance confirmation building, waste handling building, waste treatment building, cold canister storage building, and possible lag storage area. As the program has developed, significant changes to the functional concept, size and location of these facilities have been made, due mostly to the incorporation of the MPC concept.

As an example, the Waste Handling Building (WHB) design is affected by factors involving waste forms, transportation mode, shipments, function, and operations. Current approaches have drastically reduced the amount of bare spent fuel to be handled at the site. Transportation has shifted from primarily truck transport to almost exclusively rail. The use of the MPC has reduced the number of casks scheduled to be received per year to about 25% of the previous estimate by increasing the number of assemblies contained in each container. Functionally, the effect of using MPCs minimizes the need for rod consolidation, making the resulting facility smaller, cleaner, less expensive, and safer.

The surface operations area dealing with receipt and handling of waste is immediately adjacent to the north ramp portal, thereby facilitating transfer of waste packages to the underground storage area. Underground accesses consist of two ramps and two shafts, with the ramps serving as main access for personnel and materials, and the shafts primarily serving ventilation needs. Ramps are designed to be less than three per cent grade from portal to the emplacement area, and are 7.62 meters in diameter. Each ramp is outfitted with concrete invert and double rail sets allowing efficient and safe handling of large heavy waste packages.

Conceptual layout of the subsurface emplacement area can be seen in Fig. 3. Main features include two primary areas that are crossed by parallel emplacement drifts which will be used for final disposal. The emplacement drifts will be excavated by tunnel boring machines (TBM) at approximately five meter diameter. The target horizon for the emplacement area is in the TSw2 formation comprised of low lithophysal welded tuff. Rock properties of the area are shown in Table II, and are anticipated to be consistent throughout the block. Most major structural features have been identified and marginally characterized from surface investigations. Remaining features will be delineated during the course of subsurface site characterization and construction, with specific interest being focused on the identification of major fault structures.

Subsurface repository layout has been affected to the greatest extent by physical limitations brought about by MPC size and weight, and the issue of thermal loading. With MPC weight approaching 125 tons (with emplacement overpack), previous concepts of vertical and horizontal borehole emplacement became impractical, and led to the concept of in-drift emplacement. Variations on this theme were considered (in- drift on center; in-drift off center; in-drift in alcove; etc.); however, the current schemes favor the in-drift on center concept. Emplacement is relatively fast and efficient, and waste packages can be left on rail cars or pedestals as shown in Fig. 4. Using this method, careful consideration must be given to drift size. Heat generated by the waste package is dissipated through the wall rock which acts as a heat sink. However, if too closely confined, the rocks efficiency to dissipate heat is maximized, and ultimately may reflect heat back toward the package. As a requirement, the core temperature of the package must not exceed 350 degrees C, resulting in the need for an accurately engineered near field environment. This

constraint, coupled with the large size and weight of the waste package, effectively eliminated consideration for borehole emplacement and small confined openings. Another significant driver of subsurface design is the issue of thermal management. As mentioned before, limits exist on temperatures both within the waste package and in the surrounding rock. Some of the physical parameters that affect both near field and total system thermal profiles include: fuel age, burnup, number of assemblies per MPC, waste package spacing, drift spacing, and drift size. It can be readily understood that while package spacing may be altered, drift spacing cannot. Further, the optimum drift spacing for thermal loading is not consistent across multiple scenarios, thereby eliminating the flexibility of alternate drift emplacement. Other means of thermal management may be employed in the use of ventilation, aging of the waste, and rearranging the emplacement spacing after a certain period of time in the pre-closure period. These management tools are currently under investigation as to their effectiveness in achieving uniform thermal loading.
Current strategies for thermal loading are focusing on a "minimal disturbance" scenario which means that thermal loads are low enough not to cause any temperature induced changes to the rock and hydrologic ambient conditions. This would result in loading at the rate of approximately 25 MTU (Metric Ton Units) per acre. As such, the defined repository emplacement area that is currently targeted for site characterization would only be able to accommodate about 27,000 tonnes of the total inventory. This would trigger the need for characterization of additional area for repository expansion. Another alternative is under investigation to apply for initial license to construct based upon the maximum known thermal loading limit to date. After confirmation testing is completed in actual underground environment, the license application may be updated to increase the loading limit based on new confirmed data, resulting in an increased capacity for the repository, and decreasing the need for expansion area. With this in mind, it is conceivable that underground emplacement designs may be altered up until the final application for license to operate is submitted.

Session 06 -- C-14 Interactions and Issues with the Surface Environment
Co-chairs: Marsha I. Sheppard, AECL Research;
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6-1
NEW PUBLIC HEALTH AND SAFETY STANDARDS FOR YUCCA MOUNTAIN AND THEIR IMPACT ON THE CARBON-14 ISSUE
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ABSTRACT
Preliminary performance analyses of a geologic repository in the unsaturated zone have indicated that carbon-14 releases and release rates may exceed limits set by the Environmental Protection Agency (EPA); however, those releases appear to pose negligible health risks to individuals in the vicinity of the repository. The carbon-14 issue is discussed in the light of the Energy Policy Act of 1992 and the associated study by the National Academy of Sciences (NAS) on health-based standards for a repository at Yucca Mountain. First, the technical and regulatory background of the carbon-14 issue is summarized. The health risks posed by carbon-14 released from a potential high-level waste repository at Yucca Mountain are discussed, along with the rationale for the cumulative release limits in the EPA's regulation 40 CFR Part 191. Second, the ongoing process to develop health-based standards for a repository at Yucca Mountain is reviewed, including the roles of the NAS Committee on Technical Bases for Yucca Mountain Standards, the EPA, and the various affected organizations. Third, the recommendations by four specific organizations - the Department of Energy (DOE), the EPA, the Nuclear Regulatory Commission (NRC), and the Electric Power Research Institute (EPRI) - to the NAS committee are examined. The potential impacts of those recommendations, if adopted, on meeting regulatory limits related to carbon-14 releases are evaluated. The DOE and EPRI recommendations are different, but either, if implemented, would address the compliance concerns associated with carbon-14 releases. The recommendations from the NRC and the EPA input are less specific in nature than the input from the DOE and EPRI, so the implications with respect to carbon-14 are not as straightforward. The NRC notes

that releases of carbon-14 pose a problem in complying with 40 CFR Part 191, and states a preference for a health-based regulation, as opposed to one based on technical achievability. The EPA appears to favor following the current precedent of 40 CFR Part 191; that "status quo" approach would not address carbon-14 related concerns.

BACKGROUND

The ability for a geologic repository in the unsaturated zone to comply with the carbon-14 release limits has been a controversial issue for many years. Pflum (1), Park (2), and others have pointed out the potential high costs and negligible health benefits of complying with the EPA regulations on the carbon-14 radioisotope. The ongoing process to develop health-based environmental standards for a geologic repository at Yucca Mountain has focused renewed attention in this area.

In 1985 the EPA promulgated 40 CFR Part 191, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (3). At that time the regulation applied to all United States geologic repositories for high-level nuclear waste. The EPA regulation established cumulative isotopic release limits into the accessible environment for 10,000 years after permanent closure of a repository. The limits were based on the number of curies released from a hypothetical 100,000 metric ton heavy metal (MTHM) repository that would correspond to 1,000 premature cancer deaths in 10,000 years, or 0.1 fatality per year. The EPA justified the limits being achievable by comparing this level of health effects to simplified performance assessments of several hypothetical geologic repositories in different geologic media. The EPA also analyzed the health risks from the amount of unmined uranium ore that would be required to produce 100,000 MTHM of reactor fuel. The unmined uranium ore analysis indicated between 10 and 100,000 premature cancer deaths over the 10,000 year period of regulatory concern.

In 1987, the U. S. Court of Appeals for the First Circuit responded to challenges to 40 CFR Part 191 by vacating and remanding the regulation. Later that year, the court restored administrative and preclosure portions of the regulation on appeal. In 1992 the Congress of the United States, through the Waste Isolation Pilot Plant (WIPP) Land Withdrawal Act (LWA) of 1992 (4), reinstated the postclosure cumulative isotopic release limits of the remanded regulation. The WIPP LWA also stipulated that the EPA would repromulgate the remaining portions of the regulation, those related to individual and groundwater protection, within a year. Finally, the WIPP LWA specified that 40 CFR Part 191 would not apply to repositories being characterized under the Nuclear Waste Policy Act, i.e., Yucca Mountain. In 1993 the EPA repromulgated the remaining parts of 40 CFR Part 191. The resultant 1993 version of 40 CFR Part 191 retains the cumulative release limits from the 1985 version; however, the 1993 regulation specifically does not apply to a potential geologic repository at Yucca Mountain. The Energy Policy Act of 1992 (5) directed the EPA to promulgate environmental standards specifically for Yucca Mountain, based on the recommendations from a National Academy of Sciences (NAS) study.

As previously noted, the cumulative release limits in the 1985 and 1993 versions of 40 CFR Part 191 are based on analyses performed by the EPA on a set of hypothetical repositories and equivalent uranium ore bodies. Klett (6), Duguid (7), and others have pointed out that the EPA analyses, and the resulting isotopic release limits, were significantly affected by the simplified EPA model and its assumptions. The EPA assumed repositories and ore bodies located in the saturated zone, discharging to a large, diluting river near the site; all releases were aqueous in nature. By contrast, the potential repository at Yucca Mountain is located in the unsaturated zone, far from any diluting surface water; gaseous releases must be expected from such a site. Therefore, the link between the 40 CFR Part 191 cumulative release limits and health effects for a Yucca Mountain repository are tenuous at best. While the current version of 40 CFR Part 191 does not apply to Yucca Mountain, William Gunter of the EPA indicated at the December 16-17, 1993 meeting of the NAS Committee on Technical Bases for Yucca Mountain Standards that the EPA would like to make the Yucca Mountain standards as consistent as possible with that regulation. Therefore, it is prudent to consider the 40 CFR Part 191 cumulative release limits as potential requirements for Yucca Mountain.

40 CFR Part 191, Appendix A, prescribes a cumulative 10,000 year release limit of 100 curies of carbon-14 per 1,000 MTHM in the repository, assuming that no other radionuclides are released. For a 70,000 MTHM repository, that corresponds to an

average yearly release of 0.7 curie. Compliance with the carbon-14 cumulative release limit is not unduly difficult for a repository located in a favorable saturated zone environment; once radioactive carbon, in the form of carbon-14, is released from the waste package, it is dissolved in the groundwater, and significant releases to the accessible environment do not occur. However, a repository located in the unsaturated zone (like Yucca Mountain) offers faster pathways for gaseous releases of carbon-14, and compliance with the EPA limit is not assured. A 1993 total system performance assessment of the potential Yucca Mountain repository indicates that the total normalized release (based on all radioisotopes) may exceed the EPA limit, and that virtually all of the normalized release is attributed to carbon-14 (8). Compliance with the cumulative release limit at an unsaturated site like Yucca Mountain will require exceptional long-term performance from virtually all waste packages in the repository; demonstrating such performance would be very expensive and may not be practical (9).

While it may be challenging for a repository in the unsaturated zone to comply with the cumulative EPA release limits due to the gaseous release pathway for carbon-14, it is not at all clear that a failure to comply with the limits would translate to any significant health risk to future populations. In contrast to the average yearly release of 0.7 curie allowed for a 70,000 MTHM repository, the annual releases from a typical nuclear power plant and a typical reprocessing plant are approximately 10 curies and 850 curies, respectively (9). If the entire repository inventory of carbon-14 were released in one year (an extremely conservative assumption) the inhalation dose to an average members of the global population is estimated to be less than 0.0003 mSv/yr (0.03 mrem/yr). In that same scenario, a maximally exposed individual near the repository would receive less than 0.005 mSv (0.5 mrem). In contrast, the average individual dose due to only the carbon-14 constituent of natural background is estimated to be 0.013 mSv/yr (1.3 mrem/yr), out of a total United States average background dose of 3.6 mSv/yr (360 mrem/yr), including the dose from radon (10). The National Council on Radiation Protection and Measurements has recommended a Negligible Individual Risk Level that corresponds to an annual effective dose equivalent of 0.01 mSv (1 mrem) (11).

Due to gaseous releases of carbon-14 it is difficult for a repository located in the unsaturated zone to comply with the cumulative EPA release limits in 40 CFR Part 191. However, gaseous carbon-14 releases well in excess of those limits would lead to very small doses to the public. Postulating health effects from carbon-14 releases can only be done by integrating very small doses (typically considered negligible) over very large numbers of people and over very long periods of time. The 40 CFR Part 191 cumulative release limits for carbon-14 should not preclude the use of otherwise attractive waste disposal sites in the unsaturated zone.

DEVELOPMENT OF HEALTH-BASED STANDARDS FOR YUCCA MOUNTAIN

As noted earlier, pursuant to the Energy Policy Act of 1992 the NAS is conducting a study to provide findings and recommendations on reasonable standards for protection of the public health and safety from the radiological hazards associated with a geologic repository at Yucca Mountain. The 1992 law also directed the EPA, based on the recommendations of the NAS, to promulgate public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site. Furthermore, the NRC was directed to modify its requirements to be consistent with the EPA standards.

It is apparent that the U. S. Congress did not intend for the EPA to reinstate a slightly modified version of 40 CFR Part 191 for use at Yucca Mountain. If that had been the congressional intent, then there would have been no need for separate pieces of legislation (WIPP LWA and Energy Policy Act) addressing repository standards. The carbon-14 issue was one of the concerns that prompted the Congress to initiate the NAS study and to direct the EPA to issue a dose-based standard, consistent with the recommendations of the NAS.

The NAS formed the Committee on Technical Bases for Yucca Mountain Standards to prepare its recommendations to the EPA. The committee held a series of public meetings between May of 1993 and April of 1994. A number of interested and affected organizations made recommendations to the NAS pertaining to standards for a geologic repository at Yucca Mountain. The NAS committee is expected to issue its report, containing recommendations to the EPA, during the first half of 1995.

SPECIFIC RECOMMENDATIONS

The recommendations of four organizations, the DOE, the EPA, the NRC, and EPRI, to

the NAS are considered below, in the context of how those recommendations would, if implemented, address the concern that carbon-14 releases are regulated at an unreasonably low level.

U. S. Department of Energy

The DOE submitted six specific recommendations pertaining to a standard for the potential repository at Yucca Mountain (12). Of those six, the following three are particularly pertinent to the carbon-14 issue.

1. The standard should be health-based.

2. The standard should focus on protecting those people who will be living in the vicinity of the potential repository.

6. A standard expressed in terms of individual dose would be appropriate, provided that certain aspects are clearly specified in a regulation or associated guidance. Those aspects are:

a. The individual dose limit should be commensurate with current dose limits for individual members of the public as specified in 10 CFR Part 20, and with recommendations of national and international advisory groups.

b. The dose limit should be applied to an average individual in the population living in the vicinity of Yucca Mountain.

c. Assumptions related to the future biosphere should be specified. These should include a water use scenario for the average individual which is based on the current practices of people living in the Amargosa Valley.

The DOE recommendations, if implemented, would address the concern that carbon-14 release limits are overly stringent, without providing commensurate benefits to the health and safety of the public. Yucca Mountain standards would not include cumulative radioisotope release limits that attempt to provide some arbitrary level of protection to the entire population of the world for the next 10,000 years. The current overly stringent carbon-14 release limits (as well as all of the other release limits) in 40 CFR Part 191 would be removed. However, the DOE would have to demonstrate a sufficiently low probability of significant adverse health effects from carbon-14 (or any other radioisotope) on the people who might be living near the repository. One means of doing so might be demonstrating compliance with a dose standard that is commensurate with other present-day radiation protection limits. Most studies have indicated that carbon-14 doses to individuals near a repository located in the unsaturated zone would be negligible compared to such limits.

The DOE recommendations are based on the fact that Yucca Mountain is a non-diluting site, located far from any significant bodies of surface or ground water. Gaseous releases provide the only reasonable pathway for dispersion of radionuclides to the world population, and bounding calculations indicate that the doses that would result from carbon-14, the major gaseous radioisotope, would be negligible.

U. S. Environmental Protection Agency

The EPA did not provide specific recommendations to the NAS committee. This is understandable, since the Energy Policy Act requires the NAS to provide its recommendations to the EPA, not vice versa. However, the EPA did provide the NAS with the EPA views on the context within which the NAS advice will be used (13). Two points that are pertinent to the carbon-14 issue are discussed below.

The EPA restated its concern that a standard for Yucca Mountain must deal with the precedent set by 40 CFR Part 191, the general standard for geologic repositories. The EPA maintains that if it sets standards for Yucca Mountain that are different from the existing 40 CFR Part 191 standards, they will be legally required to explain the difference.

The EPA also expressed the opinion that having only an individual dose standard is not sufficiently protective and cannot replace the need for other requirements. One of their concerns is that an individual dose standard could encourage dilution by failing to cap total releases.

The implication of these points is that the EPA would like to retain the 40 CFR Part 191 release limits in a standard for Yucca Mountain. This would remove the need for them to defend the differences between a Yucca Mountain standard and the general standard for repositories.

While the "status quo" approach that is apparently preferred by the EPA may be the easiest for them to implement, it does not address the concern that carbon-14 limits might disqualify an otherwise suitable repository site, or significantly increase the cost of compliance with no commensurate public health benefit.

U. S. Nuclear Regulatory Commission

The NRC did not provide the NAS committee with specific recommendations on Yucca Mountain standards, but it did provide its staff views on environmental standards for the disposal of high-level waste (14). The NRC first discussed the "Societal Pledge to Future Generations," a philosophy that this generation should do what is necessary to ensure an adequate level of radiation protection for those in the future. Then the NRC discussed seven major issues, three of which pertain to the carbon-14 issue.

First, in the area of health-based versus technology-based standards, the NRC specifically cited the failure to recognize the potential for gaseous release of carbon-14 from an unsaturated zone repository as one of the problems with technology-based standards like 40 CFR Part 191. The staff recommended that more emphasis be placed on health-based reasoning in the development of high-level waste standards.

In addition, in the area of individual versus population protection, the NRC noted that the EPA's decision to use cumulative release limits based on the impacts to the entire world population was "... EPA's most significant departure from the traditional concepts of radiation protection, from the recommendations of advisory groups like the International Commission on Radiological Protection, and from the practices of other nations." The NRC further noted that "There are strong arguments in favor of an individual protection standard, either as a supplement to EPA's cumulative release limits, or as a replacement for those release limits."

Finally, in the area of fundamental versus derived standards, the NRC noted advantages in a standard expressed in terms of a derived quantity, such as the cumulative release limits in 40 CFR Part 191. However, the NRC also observed that the EPA's 1985 cumulative release limits "... were controversial, at least in part, because the release limits were derived using a 'world-average' biosphere that bore little resemblance to the biosphere likely to exist near Yucca Mountain."

The NRC made no recommendations to the NAS on the specific characteristics of a standard for Yucca Mountain. However, their views noted some of the problems with 40 CFR Part 191, and the carbon-14 issue was specifically cited as one of those problems. The NRC expressed a preference for health-based standards over a technology-based approach. It was clearly implied that the NRC is not satisfied with the way that 40 CFR Part 191 treats potential health impacts from carbon-14 at a potential Yucca Mountain repository.

Electric Power Research Institute

EPRI provided the most detailed recommendations of any individual or organization that provided written input to the NAS committee. EPRI proposed a two part standard, with each part pertaining to a specific time period in the lifetime of the repository. A strict, quantitative release limit would be applied for approximately 1,000 years after emplacement. For the time period after 1,000 years, the standard would be a design objective that the health risks from the repository do not constitute a significant incremental addition to other sources of health risk with which future average individuals in a local population group will have to contend. Probabilistic analyses would be used to assess performance with the second part of the standard, which would be treated as a guideline, not a strict licensing requirement. In addition to their detailed recommendations, EPRI provided a report containing supporting analyses and discussions of important issues (15).

Section 4.2.1 of the EPRI report discusses release versus dose criteria. EPRI states "Release-based standards are appropriate during a time scale for which there is a very low probability that radionuclides will even enter the biosphere, let alone provide much of an opportunity for individual dose accumulation. This time scale at Yucca Mountain is the first few hundreds to perhaps thousands of years after loading of the repository." EPRI supports a strict release standard during that time frame "... when the health risk is already vanishingly small ..." because it "... provides an additional factor of safety." For longer time periods, EPRI favors dose minimization over release minimization.

The EPRI recommendations, if implemented would effectively address the carbon-14 concern. The EPRI proposal would apply strict release limits on all radioisotopes (including carbon-14) during the first approximately 1,000 years after waste emplacement in a repository. However, that time frame is short enough so that it should be feasible to rely on robust waste packages to preclude significant releases to the environment. After 1,000 years, carbon-14 releases would be a concern only to the extent that they contribute to projected doses to individuals in the local

population near a repository. Since most analyses indicate that carbon-14 is not a significant contributor to the dose (or health risk) to nearby inhabitants at any time following emplacement, the carbon-14 issue would not pose significant limitations on a Yucca Mountain repository.

SUMMARY

40 CFR Part 191, the current environmental standard applicable to geologic repositories other than Yucca Mountain, imposes a release limit on the radioisotope carbon-14 for a period of 10,000 years following permanent closure of the repository. A repository located in the unsaturated zone, like Yucca Mountain, may have great difficulty meeting the carbon-14 release limit. However, the consequences from exceeding the repository carbon-14 release limit have been shown to be exceedingly small. Current activities such as operating nuclear power plants or fuel reprocessing plants expose the world population to much greater amounts of carbon-14 than are anticipated to be released from a Yucca Mountain repository.

A number of organizations and individuals provided input to the NAS Committee on Technical Bases for Yucca Mountain Standards. The recommendations from four organizations - the DOE, the EPA, the NRC, and EPRI - were considered, in the context of how those recommendations would, if implemented, address the concern that carbon-14 releases are regulated at an unreasonably low level. The EPA and the NRC did not make specific recommendations to the NAS. The input from the EPA indicated that they would prefer to apply the existing general environmental standard for repositories (40 CFR Part 191) to Yucca Mountain - an action that would not address the carbon-14 concern. The NRC expressed some dissatisfaction with the application of 40 CFR Part 191 to Yucca Mountain, specifically citing the carbon-14 issue as one of the problems with that regulation. The DOE recommended that standards for Yucca Mountain should focus on the health risk to individuals in the vicinity of the repository. If implemented, this approach would effectively address the carbon-14 concern. EPRI recommended a two-part standard for Yucca Mountain, with strict release limits for approximately 1,000 years, followed by a probabilistic design objective, based on individual risk, thereafter. Since it is anticipated that demonstrating compliance with the release limits for the first 1,000 years should be feasible, the EPRI recommendations would also address the carbon-14 concern.

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MASS TRANSFER OF CO2 TO GROUNDWATERS FROM A NEAR-SURFACE WASTE DISPOSAL SITE

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ABSTRACT

Gaseous 14CO2 originating from buried Low-Level Radioactive Wastes (LLRW) in a near-surface disposal site are released to the environment via two major paths: gas-phase diffusion through soils to the atmosphere, and dissolution in groundwater, followed by aqueous migration. Aqueous migration is the path giving a higher dose to an individual, especially if C-14 is converted to an organic form and ingested. Gaseous diffusion would give a lower dose, largely because of atmospheric dispersion and dilution.

The objective if this study was to develop the capability to estimate which of the two paths will likely be dominant for typical near-surface disposal facilities. The mass transfer coefficient (KL) of 14CO2 to groundwaters was determined experimentally using a large sand box. The KL thus determined was approximately 10 to 20 times smaller than for an open liquid surface, thus creating a potential resistance to mass transfer. The value obtained was incorporated into a simple model of CO2 transport around a typical near-surface disposal site. The model suggests that CO2 transport via both gaseous release and aqueous migration paths are of similar magnitude for a repository located ~2 m above the water table.

INTRODUCTION

Most of the Carbon-14 present in near-surface Low-Level Radioactive Waste (LLRW) repositories is expected to be as carbonate and released as 14CO2 (1,2). The latter is displaced by CO2 from microbial degradation of the wastes and volatilization. Carbon dioxide can escape the repository via two main pathways: gas-phase migration through soil and into the air, and aqueous transport, which involves the dissolution of 14CO2 and its mass transfer to groundwater. Conservative assumptions are applied for dose assessment where the total inventory is assumed to enter both the gaseous and aqueous pathways. Using that approach, a significantly lower dose would be expected from gas-phase diffusion (dose mostly due to inhalation) as opposed to the aqueous migration (dose caused by ingestion). A more realistic estimate can be generated if the split between the two pathways is quantified.

Gaseous migration of CO2 is the dominant transport mechanism in the vadose zone because of the high gas diffusivity (3). In a humid climate, however, aqueous transport may be important if the 14CO2 source is located near the water table. The capillary fringe, located above the water table, may create a resistance to mass transfer to the groundwater (4,5), thus possibly favouring gas-phase migration. Conventional gas-liquid exchange with an open liquid surface applies to some extent, but specific values must be obtained experimentally in soils because the process is complex due to the capillary fringe acting as an interface.

The primary objective of the work reported here is to determine which path, gaseous migration or liquid-phase migration, is dominant for CO2 release from a typical near-surface disposal site. A secondary objective was to determine experimentally the mass-transfer rate of CO2 to groundwater using a large sand box, as this parameter is currently unknown and is needed to make C-14 transport calculations.

In an earlier study (4), the mass transfer rate to groundwater was determined using bench-scale sand columns. In the columns, the porewater was static and there was no fluctuations in the position of the capillary fringe. Subsequently, the experiment was scaled up using a large sand box with a moving aquifer. We are reporting preliminary results from this experiment. We have also incorporated scoping calculations of carbon-14 transport around a near-surface repository as an example. The example is applicable to AECL's proposed IRUS (Intrusion-Resistant Underground Structure) waste disposal vault (6). The facility features an impermeable roof and walls, with an engineered open bottom to allow drainage of any water that may enter the vault by infiltration (Fig. 1). The open bottom also allows gas release. The vault will be located below the ground surface in unconsolidated material (sand), and the foundation will be approximately 2 m above the average level of the water table. Although the calculations performed here are specific to IRUS, equivalent calculations can be performed for any near-surface storage or disposal facility.

EXPERIMENTAL

Description

A steel box (3.6m long 2m wide 1m high) was used to determine the mass transfer coefficient of CO2 to a moving aquifer (Fig. 2). The box was filled with sand to approximately 0.8 m. Probe sets consisting of pre-cleaned porous ceramic samplers [4] were placed in the sand along the length of the box to sample the vertical profile of dissolved inorganic carbon (DIC). In addition, plastic probes were installed on the sand surface and 15 cm below the surface to monitor the gas composition. An adjustable water inlet and outlet allowed the establishment of a saturated zone which controlled hydraulic gradients and flow rates. A 100 L plastic tank equipped with a float valve served as a source of water to the box. The tank was sparged with CO2-free N2 gas to minimize and maintain a constant background DIC.

A gas mixture of fixed composition (10% CO2 and 10% O2 in N2) was fed under a polyethylene sheet which was placed and sealed above the sand surface.

Sampling and Analysis

A plastic line was connected to a peristaltic pump from each ceramic cup sampler. One dead volume of solution was slowly withdrawn from each probe and discarded, and a 1-mL water sample was collected and analyzed immediately for pH and DIC. The DIC was measured with a Dohrman DC-80 carbon analyzer. Only a few probes were pumped at one time to preserve the vertical profile of the aquifer, and to optimize analysis time with the instrument. The DIC concentrations were monitored at the inlet and the outlet of the box at each sampling episode.

TREATMENT OF THE EXPERIMENTAL RESULTS

A fitting routine to model the DIC vertical profile is very complex because the vertical groundwater velocity is not constant throughout the aquifer, especially in the tension-saturated region. A complete solution will be published at a later date.

An alternate approach is to calculate a mass-transfer coefficient KL (in metre/annum, or m/a) using a mass balance approach. The basic equation of mass transfer through a gas-liquid interface is given by (7):

Eq. 1

where FGW is the direct mass transfer flux of gaseous CO2 to groundwater through the capillary fringe (mole/time), Cg and Cl are the CO2 concentrations (mole/volume) in the gas phase and the liquid phase, respectively, Hc is the pH-dependent Henry's law coefficient (dimensionless; (4)), and AR is the diffusional area under the repository (m2). The value of FGW can be obtained independently using the difference in the total DIC mass in the box at two time periods, plus the total mass that has left the box through the outlet in the same time period. This amount has to be equal to the influx to balance the mass passing through the system. The other parameters in Eq. (1) are known, thus KL can be calculated (Table I). Note that Hc was calculated for pH 6.1 and 6.4, representing the two extremes observed near the diffusion interface, at the top of the capillary fringe. This is strictly an interface condition and it does not have any chemical significance. If KL is a measure of mass conductance across the capillary fringe, then its reciprocal is analoguous to resistance. A dimensionless resistance factor Rf is defined here for convenience:

Eq. (2)

where KL (open surface) 20. This value is smaller than the Rf of ~20-50 obtained for columns (4), but larger than for an open water surface (Rf = 1). This

difference is attributed to the moving aquifer.

TRANSPORT OF 14CO2 AROUND THE REPOSITORY

The information needed to calculate the fate of 14CO2 around the repository is readily available from the literature except for KL, which was obtained in this experiment. The approach presented here constitutes a preliminary estimate of the relative contributions of different transport mechanisms.

Gases expected to be produced in a LLRW vault are CO2, CH4 and H2. We will simplify gas generation by assuming that only CO2 is produced from degradation of organic material. Methane production can be neglected because its assimilation rate is high in dry environments (8,9), such as expected in the vault. We also assume that CO2 gas is unreactive with the soil components. There is a small retardation factor for gaseous CO2 in wet soils (10), but this is unimportant since the source of CO2 is constant.

The transport model is based on a steady-state flux of CO2 generated in the vault (ST; Figure 1), equal to the flux released to the atmosphere (FA) and the total flux to groundwater, FGW and FWO, for the direct mass transfer to groundwater and the flux of CO2 equilibrating with the infiltrating water, respectively (in moles/a):

See Eq. (3)

FA can also be represented as:

See Eq. (4)

where FGD is the flux of CO2 (moles/a) due to gas diffusion in the unsaturated soil. A term for advection caused by gas generation is assumed to be negligible as CO2 accounts for most of the gas produced. Each one of these fluxes can be expressed with simple equations:

Eq. (5)

Eq. (6)

Eq. (7)

The definition and the values for most of these terms are given in Table II. Equation (1) is the flux to groundwater for which KL was determined using the sand box. The term Hc is used here assuming a groundwater pH of 6. Equation (5) is analoguous to Fick's Law, using the gradient between the concentration at the bottom of the vault (Cg) and the air above ground (Ca). To simplify the calculations, we have assumed that the lateral diffusion area under the vault AD (that is 2 m above the top of the capillary fringe and the vault foundation the total length of the four walls) is also equal to the diffusional area from the vault to the surface. In other words, CO2 will diffuse out to the atmosphere via the shortest path through a "chimney" made of unsaturated sand. The term Dg is the gas diffusion coefficient of CO2 in soils corrected for tortuosity (11). In Eq. (6), the washout term, Vc is the volume of the dispersion "chimney", and (I/nl L-1) constitutes the fraction of the infiltrating water reaching the aquifer per annum. The term CT is the DIC upper limit in the pore water, equal to:

Eq. (8)

In this situation again, the pH of the infiltrating water is assumed to be 6, giving the same value for Hc as above.

We have assumed that Cg in the soil volume below the repository is constant, because the distance from the bottom of the repository to the water table is short. By combining Eqs. (1) and (3-7), the variable Cg can be determined manually on a spreadsheet. Using a KL value of 4.1 (Rf = 10), the CO2 gas released from the vault would travel via gas phase diffusion and aqueous migration in a relative proportion of 35-65%, respectively (Table II).

DISCUSSION

The influence of the capillary fringe as a barrier to diffusion is an important component of the model. The KL value used corresponding to Rf = 10 still allows for a large amount of CO2 to transfer to groundwater. A relatively small change in KL leads to a fairly large change in the flux to groundwater because of the large diffusion area under the repository.

The lateral area under the repository is the principal restriction for gaseous transport. It was assumed that diffusion towards the surface was in one direction and the cross-section of the soil "chimney" was the same as the lateral area under the vault. This is somewhat restrictive as diffusion takes place in a 3-dimensional pattern, and CO2 has the tendency to settle due to its molecular weight. This limitation suggests that the current calculation underpredicts gas diffusion. On the other hand, unrestricted diffusion in soil would probably give a higher figure

for gas-phase migration but this, along with CO2 layering, would imply a larger contact area with groundwater, thus increasing CO2 transfer to groundwater. The overall result of these competing effects is not clear. Increasing the distance between the repository bottom and the water table would allow a higher proportion of gas to diffuse to the atmosphere. The use of drain pipes or channels instead of an open-bottom configuration would have a similar effect due to a smaller contact area with groundwater. More detailed calculations would be needed to obtain a better estimate of these fluxes.

The washout flux from groundwater infiltration has a small impact relative to either of the other two fluxes. This parameter is not very sensitive, as also noted elsewhere (11). In practice, this term may be neglected.

This approach for determining the major CO2 path around the repository is essentially independent of the CO2 production rate, ST, and its concentration below the vault, Cg. A 10-fold increase in ST does not appreciably change the relative proportion of FA and FGW. Other estimated parameters such as Dg, etc., (Table II) could vary by a factor of 2-3, but in our example, one would not be able to clearly demonstrate the dominance of one path over the other. Consequently, for the purpose of performance assessment calculations, the full 14CO2 inventory may have to be considered for both pathways (gas-phase diffusion, liquid phase migration) in the situation of a near-surface site, located near the water table.

Finally, the approach given here is generic. The capillary fringe provides a physical barrier, and the findings of this study should be applicable to other volatile unreactive contaminants. Only H (Henry's law coefficient) needs to be known to apply this approach.

CONCLUSIONS

The mass transfer coefficient obtained experimentally for the transport of CO2 across the capillary fringe gave a value of 1.9 to 3.2 m/a, which is approximately 10-20 times smaller than for an open water surface.

This mass transfer coefficient was incorporated into a scoping model for CO2 transport around a near-surface repository. Our calculations indicate that the gaseous and the aqueous paths for 14CO2 release from a repository are of similar magnitude. The calculations would have indicated a near-quantitative transfer to groundwater if the open-surface value of the mass transfer coefficient had been used. In our example, even in extreme but realistic cases, one would still expect a significant proportion of 14CO2 to transfer to groundwater.

The distance between the contaminant source and the groundwater is important. If the source is remote from the groundwater, then there is a gradient to the groundwater and lateral diffusion through soil is higher. Both these factors would favor gaseous migration.

This calculated result is nearly independent of the gas generation rate (i.e., source term) and the CO2 concentration in the repository. This approach is generic and it could be extended to other unreactive volatile contaminants.

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6-3
RETENTION OF INORGANIC 14C IN SURFACE SOILS

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ABSTRACT
The 14C reaching the biosphere from deep disposal facilities is likely to be inorganic, and in many cases will be lost rapidly in the gas phase from surface soils. This is especially true when the surface soils are acidic. However, even in these soils, the potential exists for photosynthetic fixation of some of the 14C into relatively stable forms of soil organic matter. In non-acidic soils, there is also the potential for retention of the inorganic 14C, either sorbed or isotopically exchanged with indigenous carbonate minerals. Isotopic exchange could lead to the formation of relatively stable pools of inorganic 14C. In previous studies, the volatilization rate of 14C in soils was determined, and a stable fraction was observed. The present investigation was directed toward this stable 14C. Soils spiked with inorganic 14C were incubated up to 24 months in buried outdoor containers. Along with incubation time, we included treatments that allowed comparisons of the original soil with soil where carbonates or organic substrate were added, and where the microbial population was left unaltered or was decreased with gamma irradiation to 10 kGy. A range of analyses were performed, centered on a sequential extraction method designed to investigate isotopic exchange of 14C. The results indicate a small fraction of the 14C was incorporated into organic material, perhaps by non-photosynthetic autotrophs. Isotopic exchange appears to have progressed throughout the incubation times, and was likely not complete even after 24 months. Analyses are continuing.

INTRODUCTION
Carbon-14 is one of the most important radionuclides in used CANDU fuel. Sheppard et al. (1) describe the biosphere model developed to assess the impacts of geological disposal of 14C. In soil, 14C is largely regarded as a transient, and Ewing and Sheppard (2) and Sheppard et al. (3) report half-times ranging from minutes to 62 days. However, not all of the 14C may be transient, some may be retained in the pools of stable carbon (C) in the soil.
The objective of this study was to investigate the rate of incorporation of inorganic 14C into stable forms of soil C. In calcareous soils, mineral carbonates reside with half-times of up to millennia, depending on both soil and climatic conditions. In all soils, some amount of organic C resides, and the mean ages range from days for the labile, newly added materials, to centuries for the more recalcitrant fractions. There has been almost no research effort to measure the rate at which inorganic14C becomes incorporated into these pools of stable C in soils. Sheppard et al. (4) investigated the plant uptake of 14C from two soils, an acidic sand and a calcareous sand. Inorganic 14C was applied and the gaseous loss rate was almost identical for both soils. The half-time for gaseous loss was 17 d. The difference between the soils was that this rapid gaseous loss was operative on the

full inventory of 14C in the acid soil, but only on 98% of the inventory in the calcareous soil. The remaining 2% in the calcareous sand stayed in the soil, and contributed 14C to the plant tissues, over the two years of the field study. It is likely this 14C had undergone isotopic exchange with the native carbonate minerals. The present study investigates this process more fully.

METHODS AND MATERIALS

The key aspects of the methods are presented here, with more details given by Sheppard and Evenden (5). The experiment was a factorial design of three soil amendments and two levels of sterilization repeated over four incubation times. There were duplicates of every treatment. The soil amendments were none, calcium carbonate (laboratory grade at 10% by weight) and ground alfalfa (at 10% by volume). The sterilization treatments were none and irradiation (to 10 kGy with 60Co). The Gleyed Rego Black (Aquic Udic Calciboroll) soil was the same as the calcareous sand used by Sheppard et al. (4). The amended soils were spiked with inorganic 14C at 1000 Bq g-1 dry soil and placed in 1-L glass jars. The irradiated jars were sealed, and the unirradiated jars were open and covered with a mesh. All jars were buried 20-cm deep, upside down, outdoors for their respective incubation intervals. The jars were prepared and buried at six-month intervals so that incubation times were 6, 12, 18 and 24 months. Exhumation of all jars was at the same time, and they were stored frozen until extractions could be done.

On exhumation, the open jars were sealed. When each jar was processed, the first step was to obtain two samples of the headspace gas. One was used for gas chromatographic analysis and the other for 14C analysis. The 14C gas sample was collected in contact with NaOH, which trapped the 14C ready for analysis by liquid scintillation counting (LSC). After the jars were opened, samples of soil were collected aseptically for microbial assay by biological activity reaction tubes (BART). These confirmed that the microbial populations were still impeded by the irradiation treatment even at the end of the incubations.

The soils were extracted using a slow, sequentially sampled titration procedure. Under a CO2-depleted N2 gas stream, acid was added to a sample of soil and the offgas was trapped in NaOH. The rate of acid addition was adjusted so that the procedure took over two hours for the soil to change from pH > 7 to pH 1. The strategy was that the most labile inorganic soil C would be degassed first, and the most occluded carbonate minerals dissolved and degassed last. By measuring both 14C and 12C in the offgas, it was possible to determine how much of the soil inorganic 12C had undergone isotopic exchange with the applied 14C. After the acidification series was complete, the acid infusion was replaced with 30% peroxide, the NaOH trap was renewed, and the procedure repeated. A similar strategy was involved, with the idea that the more labile organics would be oxidized first and the most recalcitrant oxidized last, if at all.

Another set of extractions was done to investigate the organic C. Extractions were done with distilled water and NaOH, and with KCl with or without prior treatment with chloroform. The difference in the KCl extractions was a measure of the14C in microbial tissues. All the extracts were acidified below pH 2 to remove inorganic C, and a sample of the unacidified NaOH extract was also analyzed to measure humic acid 14C. All analyses were by LSC.

Analyses of total inorganic 14C on the soil solids followed the method of Sheppard and Schwartz (6). Analyses of total organic 14C in the soils was with a Packard Oxidizer. These methods of analysis were confirmed by independent methods, including thermal release of organic C at 500C and of inorganic C at 1100C in a tube furnace.

RESULTS AND DISCUSSION

Mass balance calculations showed large losses of the original spike in all jars, including the irradiated/sealed jars. This was because of degassing in the few minutes these soils were open after spiking. The degassing losses were largest for the alfalfa-amended soils (>98%), but still as high as 50% for the carbonate-amended soils. The various analyses made throughout the study were sufficient to develop a two-phase first-order loss-rate model to describe the degassing losses (equation 1). This is the same model used previously (4, 5). The fraction of 14C remaining in the soil, F, was described as:

$$F = A\, e^{-(0.693 \times \text{time} / \text{thalf1})} + (1 - A)\, e^{-(0.693 \times \text{time} / \text{thalf2})} \quad (1)$$

where A is the fraction of 14C lost rapidly with the half-time thalf1, and the remaining fraction (1 - A) was lost slowly with the half-time thalf2. The relationship was fit iteratively to the data, and the R2 value was 0.86 (P <

0.0001). The degassing rate constants were assumed equal for all the soil amendments, but the fraction of inventory subject to each rate constant varied. The two phases had half-times of 2.4 min (thalf1) and 1720 d (thalf2). The rapid loss affected 98% of the 14C in the alfalfa-amended soils, 76% in the unamended soils and 56% in the carbonate-amended soils. The balance of the 14C in each soil was lost at the slow rate, if at all.

The effect of incubation time was most evident in the 14C concentrations in the headspace gas. For the open jars, there was no meaningful trend with time, and very little 14C was present in the headspace gas. For the irradiated/sealed jars, the concentration of (and fraction of total) 14C was significantly ($P<0.05$) less in the jars incubated the longest time (Table I). This is consistent with our initial hypothesis that, with time, the 14C becomes more fully incorporated into the stable C pools in the soil. The lower fractions of the total 14C found in the headspace gas indicated that with time more of the 14C was isotopically exchanged with stable C in the soil.

The sequential extraction method clearly showed that isotopic exchange with the mineral 12C was not complete. The release of 14C was relatively rapid and usually reached an asymptote before half of the final amount of acid was added to the soil. In contrast, the 12C was often still being released at the end of the titration when all the acid had been added. This indicates, as expected, that the 14C was present in the more labile fractions of soil inorganic C. If isotopic equilibrium was complete, then the relative release rate of 14C and 12C would be the same. Sheppard and Evenden (5) used linear interpolation to estimate the fraction of the total inorganic 14C in the soil that was released when 20% of the total inorganic 12C was released. This value, REL20, could in theory range from 100%, implying little isotopic exchange, to 20% at full isotopic equilibrium.

The value of REL20 was lowest and was not significantly ($P > 0.05$) different from 20% in the alfalfa-amended soils at any incubation time (Table II). This indicates that isotopic equilibrium was essentially reached in these soils, and within the shortest incubation time of 6 months. The alfalfa may have caused periodic fluctuations in the soil pH and gaseous CO_2 and accelerated the isotopic exchange as minerals dissolved and reprecipitated over time.

The values of REL20 in the unamended soils were intermediate and again did not vary with incubation time (Table II). In contrast to the other amendments, the REL20 values decreased with incubation time in the carbonate-amended soils (Table II). They were also significantly higher in the open jars than in the irradiated/sealed jars for this amendment. After 6 months, the REL20 was 95% in the open jars, indicating isotopic exchange with only a small part of the total soil inorganic 12C. The decrease in REL20 with time is as expected, indicating that isotopic mixing was progressing and that REL20 would approach 20% if the incubation were long enough. Even after 24 mos, REL20 was equal to or above 50%. It would be a very long time before isotopic exchange would be complete in this soil. The higher values for REL20 in the open jars are probably, as suggested above, related to the continuous diffusive loss of 14C our of the jars that essentially competed with isotopic exchange as a sink for 14C.

The acidified soil extractions and the peroxide titrations contained some organic 14C, in amounts that were not clearly linked to treatment. The presence of organic 14C was been confirmed by thermal oxidation of the soils (5).

CONCLUSIONS

Mechanisms exist to retain inorganic 14C in soils, even though most of it will be lost by gaseous evasion. Isotopic exchange may be accelerated by specific conditions, but was shown to be far from complete even after a 24-month incubation. The 14C retained in soils by isotopic exchange and by autotrophic fixation can be expected to reside for very long times, perhaps approaching the residence times of stable carbonates and organic matter in soil.

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Monday, February 27, 1995 Lunch Address
Address by Hon. Mike McCormack
Director, The Institute for Science and Society

THE IMPERATIVE SOCIETAL NEED FOR SCIENTIFIC LITERACY

Thank you, Dr. Post. Good afternoon Ladies and Gentlemen.
I'm certainly flattered to be invited to address this meeting of so many experts in the sciences and technologies related to the management of radioactive and mixed wastes. The work you are doing, and will do, will make a significant contribution, not only to environmental restoration, but also to building healthier attitudes among the people of this country with respect to future challenges involving nuclear power, food irradiation, the transportation, storage and processing of spent fuel, and the permanent storage of vitrified waste.
I congratulate Roy Post and all of you who have organized this conference, and all who are participating in it. Certainly the broad scope of research and technology you will review this week will leave little occasion for questions about handling radioactive or mixed wastes that won't have been thoroughly studied, and for which there will not be credible answers.
It is my sincere hope that we have reached a time when the lessons learned and the technologies and techniques for managing and cleaning up radioactive and mixed wastes, and for environmental restoration, can be implemented with aggressive programs to complete the various projects without further delays.
My comments today, however, do not relate directly to waste management or environmental restoration. Instead, I shall use this time to propose that you have an additional obligation to society beyond the excellent work you are already doing. I will discuss the need for enhancing the level of Science Literacy throughout society, but especially within our educational systems and our news media, and among our public officials. I have focused on this subject because I believe science literacy is essential to the development and growth of healthy attitudes among our fellow citizens in their consideration of subjects related to public health and safety, energy supply, industrial development, environmental protection, and adapting to a rapidly changing technological world; and indeed, to waste management.
Most scientists live within their own professions, and only a few venture far into the greater society -- in terms of public service, for instance. This is natural and traditional, and you are justified in claiming that scientists serve society well by their professional accomplishments. I submit, however, that we must all do even more -- for our professional respect, and indeed for our children and our nation.
During my twenty-four years' service as an elected official, including the decade of the 1970s in the U.S. congress, I frequently found myself working with colleagues who were uneducated about -- literally ignorant of -- the scientific or mathematical foundations required for the realistic consideration of policy issues involving such subjects as environmental protection, energy production, health and safety, industrial growth, risk assessment, and basic research, among others.
This inability to handle such issues rationally constitutes a dangerous malaise that is pandemic across our nation. It afflicts all too many public officials, members of the news media, and the entertainment industry. It has, most disturbingly, afflicted much of the educational community, where one would traditionally expect to find high standards of excellence and intellectual vitality.
For instance, a faculty member of a major university recently related to me an incident in a chemistry class for non-science majors. The students were asked for their reaction to the suggestion of having a nuclear power plant in their region of the state. The reaction was essentially unanimous and vigorously negative. They were then asked to define or describe radiation. Of a large class of university students, only three hands were raised, and these tentatively.
The message this incident brings to us is one that should be disturbing to all

thoughtful citizens. These students have been frightened into a state of irrationality without the slightest understanding of the feared subject, to say nothing of understanding who was responsible for manipulating their minds to create behavior inconsistent with their own best interest; inconsistent with the very concept of an institution of learning.

This is the most striking aspect of anti-nuclear and chemophobic attitudes. Many average, well- intentioned citizens -- including teachers, students, and even many professional men and women -- behave essentially as if they are members of a brainwashed cult, unwilling or unable to face the truth about a subject on which they have accepted misinformation, and have adopted attitudes and opinions without thinking, and which they will militantly defend without even trying to understand the simplest fundamentals about that subject; or who has poisoned their minds about it, or why.

This phenomenon provides a frightening demonstration of the fact that of all the ancient enemies of mankind, ignorance is the most menacing -- the most dangerous. It is the insidious source of fear and superstition -- the vehicles for the enslavement of human minds. Throughout the history of mankind, clever men and women have cynically exploited their fellow humans, perpetuating ignorance, generating fear and superstition, and profiting from the willingness of their exploited subjects to think and behave in a manner contrary to their own best interests -- contrary to their own health, safety and welfare.

Today, one can rarely pick up a newspaper or magazine that does not contain at least one article about health hazards presumably originating from some technology; or articles presuming to expose the hazards of "chemicals" -- almost always with the adjective "toxic". So it is, too, with anything connected with the words "electromagnetic fields", "nuclear", "radiation", or "radioactive". It becomes extremely difficult for any public official who wishes to make sense of societal issues involving these subjects. He or she may be immediately exposed to attacks from sensationalist elements of the news media, from gleeful demagogues running for public office, and from activist organizations pretending to protect the public -- or some aspect of the environment -- from industry, technology, or even scientific or medical research.

I encountered this phenomenon throughout my years of public service, and I was increasingly disturbed as the education community failed to provide the bulwark of intellectual integrity that I assumed was the hallmark of that profession. Even worse, I found that many members of the education community were actually perpetuating the mythology of this fear and encouraging hostility with respect to science and technology.

After leaving the Congress, I determined to try to help strengthen our education system, especially in science and mathematics, because I believe that a reasonable level of familiarity and comfort with these subjects (not to mention others) is essential for responsible citizenship and rational decision making on public issues involving science and technology. I have become ever more convinced that an understanding of at least a modest amount about the major scientific disciplines, a respect for the scientific method of problem solving, and an appreciation for science must be essential elements in the education of every child in this nation, starting in the primary grades. To accomplish this goal, all teachers must be at least scientifically and mathematically literate, unafraid of teaching science and mathematics, and inspired to excite their students about the fun of science and the promise of careers in science.

The Institute for Science and Society, of which I am director, was created to help accomplish these goals. Its mission is to enhance the level of Science Literacy throughout society; and within that mission, to provide non-threatening courses in Science Literacy for K-12 (primary, elementary and secondary) teachers. I am pleased to report that 390 K-12 teachers have completed our primary course in Science Literacy , "Stars, Bugs, Molecules and You". In addition, we are oversubscribed in special one-weekend, single discipline courses in "Science Literacy for Educators;" and this spring we will offer "Stars and Bugs" as part of the University of Washington's Master's Degree program in Education being offered at the University's satellite campus in Tacoma.

We are clearly enhancing the level of Science Literacy of the teachers who experience these courses, and changing their perspective about science, their place in the universe, and their obligation to their students. Teachers' comments after

completing the course include such statements as:
"I'll never be the same again."
"I'm a better person." "Now I care."
"I know now that I was cheated in college."
"Thank you for the wonderful intellectual kick in the pants."
"Every teacher should take it."
". . . now I'm aware of the consequences of scientific ignorance."
"Now I can teach the excitement of science."
These remarks are by teachers who are interested enough to take the course -- presently about one percent of the K-12 teachers in Washington State. This afternoon I will participate in the Teachers" Workshop under the chairmanship of Jim McCray. At that time I will discuss in detail our science literacy program.
Most teachers of grades one through six, except those specializing in some selected subject (art, music, P.E., etc.) are required to teach some science and mathematics as part of the students' daily studies. Most of these teachers have completed very few science or mathematics courses (some almost none), either in high school or the university; and this in spite of published requirements at most universities that some number of credits in laboratory sciences must be earned as a prerequisite for a teaching certificate. In addition, many of these teachers have received their training, and work within a culture where science illiteracy is the norm, and where fear of and hostility toward science and mathematics are prevalent.
This should be a matter of serious concern, and alarm bells should be ringing in the minds of all U.S. citizens, as we learn that most 14-year-old U.S. students lag far behind their contemporaries from all other developed countries in science and mathematics test scores. Ten-year-old students in the U.S. do much better when compared to other nations. This should cause serious questions, and the first one is "What happens to U.S. students -- or what doesn't happen -- between the ages of 10 and 14?"
A disturbing explanation for this phenomenon has been suggested -- one that appears to have been generally overlooked in some current campaigns to improve the quality of science education in U.S. schools. That relates to the stultifying impact that primary and elementary scientifically-illiterate teachers can hardly avoid having on their students.
Most children under 10 years of age are naturally "good scientists." They poke, probe, try to operate or take apart almost anything they encounter, and they ask endless "why" and "how" questions. Their teachers -- and their parents also -- are frequently intimidated or simply overwhelmed by questions in a subject in which they may be illiterate, and they may cut sincere kids off with criticism or rebuke. Thus, students are "turned off" or, as Carl Sagan has observed, "dumbed down". At the same time, a teacher may be forcing the kids to memorize material that neither teacher nor student understands, while the teacher "stays one day ahead" in the text. The result, all too frequently, is that the students are alienated from school or science, or both. Even when scientifically illiterate teachers think they are doing an adequate job of teaching, and are sincerely trying, they may be unintentionally turning off their students by their body language or facial expressions. This may be especially true for female and ethnic minority students. Thus, it should be no surprise that optional registration in science or mathematics courses at and beyond junior high school is so low. Many of these students are lost before they are contacted by a trained science teacher or by exciting special programs to improve the quality of science education. Again, this may be especially true for female and ethnic minority students.
It should be obvious that any program to upgrade the level of science education among U.S. students is doomed to failure if the primary and elementary teachers -- who have the attention of the young students most of the time -- are themselves scientifically illiterate, or are frightened of, or hostile to science and mathematics. The students will absorb the message of their role model, even if it is unintended.
This points directly to the problem of public understanding of subjects related to nuclear energy and waste management. If our fellow citizens of any age are frightened of and hostile to science itself, and to the scientific approach to problem solving, they will easily fall victims to misinformation and appeals to emotionalism about technologies which they do not understand and about which they may be suspicious. The goal of rational public consideration of issues involving

modern technology must be built on a campaign of scientific literacy for the majority of our citizens.

We have concluded that the most important undertaking in any effort to upgrade the quality of science education at any level must be first to provide attractive continuing education programs in Science Literacy for all K-12 teachers. Such programs must first focus on primary and elementary teachers, and should be accompanied by strong inducements that the courses be successfully completed. Such courses should -- and can -- be made exciting, fun, and rewarding; and should, above all, avoid the trap of frightening or threatening teachers with abstract and valueless material to memorize.

In addition, the schools of education throughout the country should include successful completion of a comprehensive curriculum in Science Literacy as a prerequisite for a teaching certificate; and these courses in Science Literacy should be made available to all non-science majors, rather than requiring completion of academic level courses in science disciplines.

I have proposed that a department of Science Literacy be created at Central Washington University, with which The Institute for Science and Society is affiliated. I have suggested a curriculum of 20 quarter credits in Science Literacy for undergraduates, and suggested that this be made a prerequisite for obtaining a teaching certificate. Under this proposal, Science Literacy classes would be taught by faculty from the various science disciplines. Thus the University would offer an academic curriculum in each science discipline as at present, and a program in science education for students who wish to qualify to teach science, and third, a comprehensive curriculum in Science Literacy. Our Science Literacy curriculum includes a brief overview of the major physical and biological sciences, an understanding of what science is, a course in mathematics for understanding science, one on societal issues involving science and technology, and one on the relationships of science and technology to industrial growth and economic stability.

I assume that some states and some schools of education are doing better than others in this matter, and you may wish to find out about yours. You should be aware, however, that there are land mines lying in wait for anyone recommending changes in any level of our educational systems. One is the resistance one would expect to proposing any change in the activities of any organization. This may be reinforced by the apprehensions of many members of the education community who themselves are scientifically illiterate and possibly frightened of, or hostile to including any study of science, however superficial, in their curriculum.

There will also be resistance among those who object on philosophical grounds to teaching any education course with "content" rather than simply in "methodology." There will be those who have difficulty understanding the distinction between Science Literacy and Science Education. There may also be resistance among some members of the science faculty at some institutions who may object to teaching Science Literacy to teachers because they consider it to be "watered- down science," and beneath their dignity. These obstacles, where they exist, must be cleared away if a comprehensive program in Science Literacy can be made a requirement for certification to teach.

There is yet another obstacle. It lies within some federal agencies that fund science education programs. It is born partially because of inertia, and partially because some of these agencies appear to have difficulty recognizing that all primary and elementary teachers are "science" teachers. I suggest that such federal agencies reconsider the need for Science Literacy training for all teachers, rather than primarily for those who teach science in middle and high schools; and that science education is essential for all students, starting with kindergarten.

I suggest that you who share the concerns I have expressed may wish to survey the level of Science Literacy among the teachers of your state, and the curriculum requirements at your state's schools of education. You may come to recognize your obligation to work to enhance the level of Science Literacy among your state's teachers, and within its schools of education. You may wish to enlist the participation of scientific societies in which you are a member. I consider this to be a critically important aspect of the societal obligation of all scientists -- but especially experts in the sciences related to radiation, because this is a raw nerve in the emotion-ridden minds of too many of our scientifically illiterate fellow citizens today.

Nor is your obligation limited to working with your education communities. Many

elements of the news media and entertainment industry are guilty of flagrant violations of their responsibilities to society. Their tendency to sensationalize real or imagined issues involving science and technology is causing terrible damage to our country, especially in the minds of the scientifically illiterate. It is the responsibility of every individual scientist and every scientific society to keep the pressure on the news media and the entertainment industry, insisting that they be responsible and accurate when dealing with any subject related to science and technology.

Most public officials -- especially state legislators and Members of Congress -- are scientifically illiterate, and thus functionally crippled with respect to their responsibility to legislate intelligently on societal issues involving subjects such as health, safety, energy, environmental protection, and others. There are countless stories, many of them true, about legislative bodies enacting bills that violated mathematical rules, changed mathematical constants, denied exponential growth, and attempted to repeal the Second Law of Thermodynamics. Members of the scientific community who are concerned that such incidents are possible need only look in the mirror to find the reason. In short, if you are frustrated with the apparent inability or unwillingness of your legislative body to treat issues involving science and technology rationally and with appropriate respect, you may wish to organize a non-partisan, or bi-partisan group of scientists to meet regularly with your legislators when they're at home. If this fails, you may wish to run for office.

In fairness, I should observe that most elected officials will welcome constructive assistance, especially from their constituents. They cannot help but respond to what they believe to be their constituents' wishes. If you and your scientific associates make it abundantly clear that you expect responsibility from your elected representatives in the consideration of issues involving science and technology, your requests will be heeded.

In summary, the obligation of all scientists is to accept their roles in society, and thus to accept the critical need for their involvement in all levels of education, public communication, entertainment, and government. It seems to me that each of us who has benefited from a good education has a positive obligation to give back to society something of ourselves -- something beyond just earning a living -- to help build a better world.

I frequently use a story about a man who fell off a cliff. As he fell, flailing his arms, he caught hold of a tree root protruding into space. There he hung, bouncing up and down. After yelling for help several times and getting no response, he shouted, "Is anyone up there? After a moment a voice seemed to come from space, and said, "I'm here." "Help me", yelled the man. "Have faith", the voice said, "and let go." The man looked down a thousand feet to jagged rocks, and up the sheer cliff for dozens of unclimbable feet, hesitated, and shouted, "Is anybody else up there?"

♀

The reality is that there isn't anyone else to help do what is necessary to improve the level of science literacy and rational consideration of public issues involving science and technology, except those of us who understand the issue and understand what must be done. I challenge each of you to accept that obligation to help overcome this problem. In addition to what you are doing, you must do more.

It may be that in the future you will be able to look back and know that because of your efforts, some of the major problems we face today have been corrected; and that this country is once again moving forward with a citizenry that is unafraid and vibrantly challenging the future. I hope this will happen. It's easily worth a lifetime commitment.

Session 07 -- Poster - LLW & General Waste Management
Co-chairs: Harold Sturm, WSRC;
R.D. Wojtasek, WHC
J.R. Cook, WSRC

7-1
INTEGRATION OF REMEDIATION STRATEGY WITH WASTE MANAGEMENT CAPABILITIES AND REGULATORY DRIVERS FOR RADIOACTIVE WASTE STORAGE TANKS AT THE OAK RIDGE NATIONAL LABORATORY*
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ABSTRACT
This paper addresses the plans and strategies for remediation of the Liquid Low-Level Waste (LLLW) system tanks that have been removed from service at the Oak Ridge National Laboratory (ORNL).
The Superfund Amendments and Reauthorization Act of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) requires a Federal Facility Agreement (FFA) for federal facilities placed on the National Priorities List. The Oak Ridge Reservation was placed on that list on December 21, 1989, and the agreement was signed in November 1991 by the U. S. Department of Energy Oak Ridge Operations Office (DOE-ORO), the EPA-Region IV, and the Tennessee Department of Environment and Conservation (TDEC). The effective date of the FFA is January 1, 1992. One requirement of the FFA is that LLLW tanks that are removed from service must be evaluated and remediated through the CERCLA process.
The Environmental Restoration Program intends to meet this requirement by using a "streamlined" approach for selected tanks. This approach will combine the CERCLA Site Investigation, Remedial Action, Feasibility Study, and Proposed Plan requirements into a single Interim Proposed Plan document. This streamlined approach is expected to reduce the time required to complete the regulatory process while attaining acceptable risk reduction in a cost-effective way.
OAK RIDGE NATIONAL LABORATORY BACKGROUND
Oak Ridge National Laboratory (ORNL) is a multidisciplinary research facility operated for the Department of Energy by Martin Marietta Energy Systems, Inc. ORNL began operation in 1943 as part of the Manhattan Project. The original mission of the laboratory was to develop a prototype graphite reactor and reprocess the reactor fuel for plutonium recovery. After World War II, the primary functions of ORNL were fuel reprocessing research; radioisotopes production and applications development; and nuclear reactor concepts development, testing, and operation. More recently, the laboratory has increased its role in biological, environmental, energy, and materials research. As a consequence of these multidisciplinary research activities, heterogeneous wastes, including liquid low-level radioactive, hazardous, and mixed wastes, have been generated in varying amounts over time.
Since its establishment, ORNL has operated numerous facilities that generate LLLW. LLLW originates from radioactive liquid discarded into sinks and drains in research and development laboratories and from facilities such as the Radiochemical Processing Pilot Plant, nuclear reactors, radioisotope production facilities, and the Process Waste Treatment Plant.
The LLLW system is a complex system with multiple facilities, users, and operators. The system is used for collection, neutralization, transfer, and concentration of aqueous radioactive waste solutions from generator facilities. Waste solutions are typically accumulated at source buildings, often in collection tanks located inside the buildings, and discharged to below-grade collection tanks that may receive wastes from several different source buildings. However, in many instances, LLLW is transferred from laboratory and hot cell drains through unvalved piping directly to underground collection tanks or to the central waste collection header. A network of below-grade piping interconnects the various system components. The collected LLLW is transferred to an evaporator, where it is concentrated. The concentrates are transferred to storage tanks.
Most of the LLLW system was installed more than 30 years ago. The original system, installed during the early 1940s, and its subsequent modifications were designed to minimize radiation exposure to LLLW system users and operators. The system includes features such as unvalved, gravity-drained transfer lines to prevent waste backup

into generator areas; shielded lines and tanks; and provisions for remote operations. As-built drawings for some of the older tank systems do not exist. Over the years, tank systems were removed from service as their integrity degraded or as programs were terminated. New tank systems installed during the past 10 to 15 years incorporate secondary containment and improved leak detection features. The LLLW system is thus a mix of singly and doubly contained tank systems. The portions of the system that have been removed from service consist almost exclusively of tanks without secondary containment.

OBJECTIVES

The objectives of the Environmental Restoration Program for the inactive LLLW tanks complement the objectives for a comprehensive environmental restoration of the Oak Ridge Reservation in keeping with the overall program mission.

ENVIRONMENTAL RESTORATION MISSION

Our mission is to remediate contaminated sites and contaminant releases to reduce current and future risk to human health and the environment. Early actions focused on reducing current or potential off-site risk are conducted in parallel with efforts to select, implement, and verify final remedies for contaminated sites. Remedial efforts are prioritized and innovative approaches are developed to support cost-effective risk reduction. All efforts are conducted with an emphasis on worker health and safety and with the goals of meeting regulatory requirements and the expectations of the public.

As stated in the FFA,

DOE shall remediate all tank system(s) removed from service. To the extent practicable, DOE shall remove or decontaminate or otherwise remediate all residues, contaminated containment system components (liners, etc.), contaminated soils, and structures and equipment associated with the tank system(s).

A primary objective of the Environmental Restoration Program is to remediate all LLLW tanks that have been removed from service to the extent practicable in accordance with CERCLA requirements. In addition to risk and risk reduction, applicable or relevant and appropriate requirements (ARARs) will be addressed in choosing a remediation alternative. Preference will be given to remedies that are highly reliable and provide long-term protection. Efforts will be directed toward permanently and significantly reducing the volume, toxicity, or mobility of hazardous substances, pollutants, and contaminants associated with the tank systems. Where indicated by operational or other restraints, interim measures short of full and complete remediation may be taken to maintain human health and ecological risks at acceptable levels until full remediation can be accomplished.

Environmental restoration requires decision making with available data containing uncertainties and traditional approaches can sometimes be very time consuming, expensive, and inefficient. Therefore, the environmental restoration process for LLLW tanks that have been removed from service will consider 1) employing innovative and technically sound approaches to tank remedial actions; 2) balancing short-term needs to protect health and the environment with long-term future use objectives for the Oak Ridge reservation; 3) remediating the inactive tank systems in a logical order that is integrated with other remedial actions; 4) attaining cost-effective risk reduction; 5) meeting environmental regulations; and 6) addressing the expectations and requirements of all stakeholders.

REMEDIATION STRATEGY

Background

The management strategy for timely and efficient remediation of all sites at ORNL is to divide the waste units into waste area groupings (WAGs), which are areas that are either geographically contiguous or hydrologically confined units. Within each WAG, one or more operable units (OUs) have been defined. These OUs are smaller, more manageable units chosen on the basis of contaminant pathways analysis, application of similar remediation technology, geographical consideration, assessment of early or time-phased action, and remediation efficiency or simplicity considerations. The designation of all OUs for the Oak Ridge Reservation is shown in the FFA.

Remediation Plan

As of December 1994, FFA Appendix F identified a total of 55 tanks that had been removed from service and designated these tanks as Category D. Eighteen tanks considered to pose the highest risk were assigned to WAG 1 OU 1, Gunite and Associated Tanks (GAAT), which will be remediated under a separate project. These 18 tanks will undergo the complete remedial action process including a remedial

investigation and feasibility study (RI/FS) leading to a proposed plan and a record of decision (ROD) as required by CERCLA. Treatability studies are currently being conducted on these tanks.

In parallel with the ongoing activities associated with remedial actions for the GAAT OU, a "streamlined" CERCLA process is being applied to the remaining 37 Category D tanks. These 37 tanks were preliminarily screened according to risk, remediation technology required, interferences with other piping and equipment, location, and available sludge removal techniques and storage requirements. On the basis of this preliminary screening, the tanks were assigned to one of five "batches" (I through V) for consideration of remedial action alternatives and these batches were tentatively scheduled for remedial actions. This prioritization will be further refined on the basis of results from the prioritization risk assessment and site investigation results.

For each batch of tanks, documentation will be prepared that incorporates all the requirements of the CERCLA remediation process leading to an expedited record of decision (ROD). The specific remediation alternative will be chosen on the basis of risk as described in EPA guidance manuals. This streamlined approach will combine the CERCLA Site Investigation, Remedial Investigation, Feasibility Study, and Proposed Plan requirements into a single Interim Proposed Plan document. The Interim Proposed Plan will clearly define the tank system components, both piping and containment (tank, vault, etc.), that will be included in the interim remedial action. The Interim Proposed Plan will also describe the means taken to safely secure the components not immediately remediated and will clearly indicate to which OUs these components will be assigned. Following the selection of a preferred remedial action alternative and its documentation in the ROD, remedial design and remedial actions will proceed. The goal will be to arrive at final remediation for each tank system, although in some cases interim actions may be indicated.

The Environmental Restoration Program recognizes that other tanks are scheduled to be removed from service as projects are completed to bring the active LLLW system into full FFA compliance. Active tanks whose removal from service is pending have also been tentatively assigned to batches. The same preliminary screening factors that were used for the initial batch assignment of the inactive tanks were applied to these currently active LLLW tanks. As they are removed from service and transferred to the Environmental Restoration Program, the tanks will be emptied and made to meet program acceptance criteria. As the program moves forward and more information becomes available, these tanks may be reassigned to other batches or they may be remediated as a separate batch or batches.

A team has been assembled to address remediation of the Batch I tanks. The team is a group of technical representatives from all involved organizations. The team, which meets regularly to coordinate and plan remedial action activities for the tanks that have been removed from service, provides a mechanism for integrated responses on remedial action issues to DOE, EPA, and TDEC.

The approach to remediation of each tank or batch of tanks can and should be viewed as a dynamic, flexible, customized process that must be adapted in response to the specific circumstances of individual tank and sites. Thus, the approach will be tailored to accommodate feedback on lessons learned from previous remediation actions and will not be a rigid step-by-step approach that must be conducted identically for every tank system.

Tank parameters have been examined and an initial ranking made to determine the first tanks or batches of tanks to undergo remediation through the CERCLA process as required by the FFA. Remedial actions for the Batch I tanks are scheduled to begin in FY 1995. All of the tanks in Batch I have previously been emptied and are thought to have no inflow.

Although the specific remedial action alternatives for a tank system, which will reduce the risk to acceptable levels, will be described in the Interim Proposed Plan, four general alternatives that are applicable to both direct-buried tanks and tanks located in vaults have been identified.

Alternative	Merits and limitations
Remove and dispose of tank and isolate piping.	Remediation of the tank will be complete. Adverse impacts on other OUs will be minimized. Risk reduction or other cost/benefit ratio must justify

	cost of removal and disposal. Removed tank may require treatment to meet appropriate waste acceptance criteria.
Leave tank in place and isolate piping.	To complete successful CERCLA remediation, risk posed by tank must be below EPA risk acceptability range. Future deterioration of tank could cause problems. Tank may require acceptance into the remedial action program of other OUs. Tank may require continued surveillance and maintenance.
Fill tank with stabilizing medium.	Stabilizing medium could minimize the accumulation of liquid in the tank. The tank must be cleaned prior to placing stabilizing medium. Any remaining contaminants could be leached from stabilizing medium if tank wall is breached. Added mass and volume of stabilizing medium might increase future excavation and waste disposal costs.
Leave tank as is.	If tank risk falls below EPA acceptable range, the tank can be removed from CERCLA consideration. The tank could remain under CERCLA, a surveillance and maintenance program, or both because of potential for inleakage and contamination.

Following successful remediation of the Batch I tanks, and using the lessons learned in these projects, as well as lessons from the GAAT OU project, the plan is to proceed to increasingly more complex or difficult tank or tank site remedial actions on the basis of knowledge and skills developed on these initial tank remedial action projects.

7-2

CHEMICAL MECHANISMS DISCUSSION - INEL: CESIUM IN GROUNDWATER INNOVATIVE TREATMENT

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ABSTRACT

The removal of cesium 137 from waste and groundwaters is a subject that has been approached by several investigators. The REFERENCE section provides some examples. Although there are approaches that have been shown to work there appears to be a limit of the applicability of these approaches when the generation of secondary waste is considered. This paper provides a possible new approach to the removal of low levels of Cesium 137 from water. The approach depends upon chemical oxidation, iron chemistries and bicarbonate additions under controlled pH parameters. The proposed mechanisms that could explain the limited data obtained to date, shows an ability to reduce the Cesium levels by about 93%.

BACKGROUND

ADTECHS Corporation was engaged in treating heaving metals and low levels of selected organics in a groundwater stream at a DOE site at the Idaho National Environmental Laboratories. During this work, an unanticipated analyte, Cesium 137, began to appear in low levels in the influent. Work was commenced on finding a suitable method of removing this Cesium from the stream. This goal is supported by the fact that Cesium-137 is found at many DOE sites and a practical solution to its removal would have additional benefits. As of this paper's writing, a fully approved methodology for Cesium removal and treatment has not been approved by the site prime contractor. It is anticipated that a process and site standard will be in place before the end of 1995.

The data provided in the Summary below, is based on pilot work performed at the site on the actual groundwater. At this time, full scale pilot treatment has not been permitted pending the decisions noted. However, it is felt that the results seen to date do warrant further study.

TESTING INFORMATION

The groundwater native pH was measured to be 8.88 +/- 0.2 at 20oC. The general levels of chemistry and type used is shown in TABLE I.

When the pH is raised into the 7.0 to 7.5 range with the base addition, an initial white tinged with umber precipitate is formed which when filtered turns mostly organe-umber. When air dried the filter cake becomes essentially white.

The light but large flock precipitate is readily filtered. Both a sand filter and a 1 micron cartridge filter work equally as well.

DATA SUMMARY

TABLE II includes a summary of the data taken on "before" and "after" treatment. On average, the samples tested were reduced by greater than 93% with some samples reduced to non-detect. Based on the standard deviation differences, a reduction of at least 90% seems apparent.

MECHANISMS DISCUSSION

(A) Conversion of monovalent Cesium into a Cesium Oxide (Cs2O) is accomplished by the addition of the oxidant. Also Cesium superoxide or Cesium trioxide may also form. The combination will produce a blend of colors of dark brown to dark orange/umber.

(B) The very low levels of Cesium requires a mechanism to remove it into a filterable cake. This step is accomplished by the judicious use of iron chemistries. FIGURE 1 shows that freshly precipitated Fe(OH)3 at various pHs will yield Hydroxo Iron complexes that range from Fe+3 to Fe(OH)2+1, Fe(OH)+2 , Fe(H2O)6+3 and others. Optimizing the iron chemistries in the pH range indicated in Fig. 1 assists in the formation of polymeric iron compounds which can engage the cesium oxides to form a co-coagulant/precipitant. While insufficient work has been done to optimize the conditions, it is known that polymerization and precipitation kinetics are effected by the concentration of the iron solution, the type, concentration and method of base addition; ionic strength; temperature and specific anion effects.

(C) The adjustment of the pH into the "optimal range" allows for the potential formation of carbonate or bicarbonate complexes that could be ensnared in the polymeric iron matrix. The appearance of orange-umber crystals when subjected to the air drying could be due to the conversion of cesium/iron oxide/hydroxide complexes into bicarbonate/carbonates which would be white. TABLE IV demonstrates the statistical experimental design proposed for evaluating the sensitivities and strength of the effects of each step in the chemical precipitation process. In this way, both first order and second order potential effects and strength of correlation to Cesium removal can be evaluated.

FIGURE 1

The equilibrium concentrations (Log C = Y) of selected hydroxo iron +3 complexes in contact with freshly precipitated Ferric hydroxide (Fe(OH)3 @ 25oC versus solution pHs were graphed above. The descriptive equations that relates the Log C to pH (X) are shown in TABLE III.

CONCLUSIONS

The usefulness of chemical precipitation for low levels of Cesium-137 appears to have merit. Preliminary bench work utilizing actual ground water has indicated a high percentage of removal is possible with a low generation of solid waste. The mechanisms for this removal are based around the ability of iron to form polymers and to engaged and entrap other oxides even when present at very low levels. Further work appears to be warranted.

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7-3

CENTRALIZED CEMENT SOLIDIFICATION SYSTEM FOR RADIOACTIVE WASTES

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ABSTRACT

A centralized cement solidification system has been developed to solidify various radioactive wastes such as liquid waste, spend ion exchange resin, incinerated ash and miscellaneous solid waste using a single solidifying facility and agent. This system is simple and highly reliable regarding the equipment and treatment process. Since it used high performance cement (HP-cement), waste loading can be raised and deterioration of waste forms after land burial is prevented. The HP-cement consists of slag cement, reinforcing carbon fiber, natural zeolite and lithium nitrate ($LiNO_3$). The fiber allows waste (spent resin) loading to be increased from 25 to 60kg-dry resin/200L drum. The zeolite, whose main constituent is clinoptilolite, reduces radiocesium leachability from the waste form to below 1/10. Lithium nitrate reacts as an inhibitor for corrosion of aluminum contained in the incinerated ash, which allows ash content in the waste form to be raised from 10 to 30wt%. This inhibitor is applicable to cement solidification of miscellaneous solid waste containing Al pipes and foil. Laboratory and full-scale experiments were performed to evaluate properties of the waste form and industrial applicability of this solidification system.

INTRODUCTION

The amount of radioactive wastes generated from nuclear power plants is decreasing annually, as advanced treatment processes have been developed. Recently a simple waste solidification system with low cost is highly desirable. We have developed a "centralized cement solidification system" in order to reduce the volume and cost of the solidifying facility. This is system allows several kinds of wastes to be solidified using a single solidifying facility and single agent. Our goals in developing the solidification system are:

1. reducing the number of solidified waste forms produced by increasing waste loading;
2. reducing release of radionuclides release into the environment;
3. obtaining a stable waste form for final land disposal; and
4. minimizing secondary waste generation.

Liquid waste, spent ion exchange resin, incinerated ash, and miscellaneous solid waste comprise the major fraction of low-and intermediate-level radioactive wastes generated in nuclear power plants. This research work was mainly aimed at developing a universal solidifying agent for all types of waste (High Performance cement, HP-cement) in order to realize the above four goals. Fundamental experiments were performed to optimize the composition of HP-cement, especially selection of additives for the cement. Simulated wastes were solidified, on a laboratory scale, using the HP-cement. Water resistance and leachability of waste forms were evaluated to predict their long-term stabilities after land burial. Furthermore, full-scale pilot plant tests, in which emphasis was laid on solidification of spent resin, were carried out to confirm an industrial applicability of the solidification system.

OUTLINE OF SOLIDIFICATION SYSTEM

A schematic of the centralized cement solidification system is shown in Fig. 1. Four features ensure the system is simple and highly reliable:

1. Several kinds of radioactive wastes can be solidified simply by mixing with cement paste using a single mixer.
2. Only one cement storage tank is necessary, because HP-cement is supplied as a ready-mixture containing carbon fibers (an reinforcing agent), clinoptilolite (an adsorbent for radiocesium), and lithium nitrate (an inhibitor for aluminum corrosion).
3. Wash drain generated from the mixer is reused as mixing water for the next batch. This significantly reduces the amount of secondary radioactive liquid waste.
4. Installation of a slide valve ensures the mixed paste of waste and cement is discharged without plugging the outlet.

In the case of spent resin, the solidification process consists of three steps. The

spent resin is first dewatered to a water content of about 50% using a centrifugal liquid/solid separator. The resin is transported into a mixer by use of a screw feeder. It is then mixed with HP-cement and water. Finally the mixed paste is fed into a steel drum and cured at room temperature after capping. When ash or liquid waste is solidified, the dewatering process is unnecessary. In the case of miscellaneous solid waste, the waste is put into a drum in advance. Then the open space in the drum is filled by the mixed paste of HP-cement and water. HP-cement can simplify the solidification process, because it needs neither pretreatments of wastes before solidification nor special equipment.

FEATURES OF HP-CEMENT

The main purposes and developed techniques, which solve the problems expected to be encountered during cement solidification and after land burial of the waste forms, are summarized in Table I. We concluded from the basic experiments described below that the HP-cement should contain slag cement, carbon fibers (a reinforcing agent), natural zeolite (an adsorbent of radiocesium), and lithium nitrate (an inhibitor for aluminum corrosion).

Solidification of spent resin

Water resistance: Main concerns in cement solidification of spent resin are water resistance of the waste form and leaching of radionuclides, especially Cs. Therefore, resin content in conventional waste forms has been controlled below 25kg-dry resin/200L drum (1). When resin content is higher, resin-containing waste forms tend to crack after water immersion due to resin swelling. Fundamental studies on the cracking mechanism led to development of a fiber reinforced cement which allows the resin content to be raised (2,3). The measured swelling pressure of solidified resin by cement was approximately 2.5MPa, which generated a maximal tensile stress of about 4MPa in the cement matrix. The tensile strength of fiber reinforced cement increased in proportion to fiber content. The hardened cement paste with 3vol% of carbon fiber (15 mmf 3mm long) had a tensile strength (about 5MPa) high enough to resist the resin swelling pressure. Figure 2 shows the relation ship between waste (mixed bead resin) loading and compressive strength of cementitious waste form, containing 3vol% of carbon fiber, before and after a 30-day water immersion test. The waste form reinforced by carbon fiber did not deteriorate during water immersion, even with as high a resin content as 60kg-dry resin/200L, while the waste form without fiber (con ventional cement) cracked in water and its strength decreased to about 0MPa within a few days.

Leachability: The distribution coefficients of cementitious waste form with and without mixed bead resin for the five main radionuclides are summarized in Table II. The waste forms with resin, except for Cs, showed much higher values than those without resin because of adsorption onto resin. Therefore Cs leachability from the resin-containing waste form is much higher than that of other radionuclides. The distribution coefficient of cement is relatively low for radiocesium (4), further the Cs selectivity of resin is lower than that for Ca in cement. Enhanced Cs retention ability in the waste form would be necessary for high waste loading. Although high pH and coexistence of Ca ions limits the possible choices of Cs adsorbent which can be added to cement, the fundamental experiments showed a natural zeolite, whose main constituent was clinoptilolite, had suitable adsorption ability for Cs. Figure 3 shows increases in Cs distribution coefficients of the hardened cement pastes with natural zeolite (clinoptilolite) and synthetic zeolites. Clinoptilolite was most effective additive for Cs adsorption among the tested zeolites because of its high selectivity for radiocesium. Cs leachabilities of resin-containing waste forms with 5wt% of clinoptilolite and without it are summarized in Table III according to the leaching rate and leachability index (5). The addition of clinoptilolite gave waste forms with a Cs leaching rate less than one-tenth that without adsorbent.

Solidification of incinerated ash and miscellaneous solid waste

When incinerated ash or miscellaneous solid waste (pipes, filters, etc.) is solidified with cement, the waste form often swells due to gas generation (6). Hydrogen gas is produced from alkaline corrosion of amphoteric metals contained in the wastes. We wanted to exclude pre-treatment or assortment of the wastes prior the solidification process. Alkaline corrosion experiments on Al metal were carried out to find an effective inhibitor which could be added in cement. We found that lithium ion helped to form an insoluble white film (about 5 mm thick) on the Al specimen which prevented hydrogen gas generation. The film was characterized by X-ray

diffraction analysis as the double salt of lithium aluminate, LiAl(OH)4Al(OH)3 xH2O. Then, lithium nitrate (LiNO3) was selected as the most effective inhibitor of Al corrosion, because the nitrate ion also showed an inhibition effect. Table IV shows that the addition of lithium nitrate (3wt% of slag cement) significantly reduced the rate of hydrogen gas generation from an aluminum plate placed in cement paste. Figure 4 shows the relationships between the ash content and the compressive strength of ash-containing cementitious waste forms. In the case of waste forms without lithium, their compressive strengths clearly decreased with addition of incinerated ash. They also swelled due to hydrogen gas generation and volume increased 20%. On the other hand, the strengths of waste forms with 3wt% added LiNO3 did not decrease notably until the ash content was as much as 30wt%. No volume expansion of the waste form was observed. We concluded this inhibitor can be applied to cement solidification of miscellaneous solid waste which contains Al pipes or foil.

The effect of lithium ion on aluminum corrosion was discussed based on the dissolution equilibrium of aluminum in alkaline solution. The equilibrium in lithium-free solution is as follows:

$$Al(OH)3 + OH- = Al(OH)4- \quad (1)$$

$$\log[Al(OH)4-] = \log K1 + pH - 14 \quad (2)$$

where the logarithmic equilibrium constant, logK1, is 0.18 for bayerite (7). In the case of disillusioning the presence of lithium ion, the amount of dissolved aluminum is limited by the deposition of Li-Al double salt as follows:

$$LiAl(OH)4\ Al(OH)3 + OH- = Li+ + 2Al(OH)4- \quad (3)$$

$$\log[Al(OH)4-] = 1/2\{\log K2 - \log[Li+] + pH - 14\}. \quad (4)$$

The logarithmic equilibrium constant, logK2, obtained in this study was 6.60. When pH is 13 (nearly the pH of cement paste), aluminum solubility in the alkaline solution is lowered below 1/100 by the addition of lithium ion (0.1M). Because the amount of hydrogen generation depends on the amount of dissolved aluminum, the addition of lithium salt reacts effectively as an inhibitor for gas generation.

PILOT PLANT TESTS

A demonstration pilot plant was constructed to confirm findings of the fundamental experiments. Its design specifications when handling spent resin were as follows.

Treatment capacity: 2drum/hour
Drum volume: 200L (net)
Waste loading: 60kg-dry resin (max)

Simulated spent resin was solidified into a 200L waste form by the following ratio,

Resin/water/cement = 100/60/140kg

where the resin was on a wet basis (water content = 43wt%). After curing for three months at room temperature, the package was cut and specific gravity was measured for core samples. Fairly good homogeneity was obtained for the actual size pack age. The package was immersed into a 1m3 water vessel and changes in its volume, weight and compressive strength were measured during more than three years.The strength was estimated from propagation velocity of ultrasonic waves in the waste form. Table V shows results after 3-year water immersion test. The weight gradually increased with time and became saturated at an increased value of 3.5wt% after immersion for about 100 days indicating water penetration into the waste form. Deterioration features such as cracking and swelling were not observed. The volume change was less than the experimental error of 0.1%, and compressive strength was more than 8.0MPa before and after the water immersion. These findings agreed with the results of fundamental experiments.

CONCLUSION

A centralized cement solidification system has been developed to solidify several kinds of radioactive wastes using a single solidifying facility and agent. It is simple and highly reliable regarding the equipment and treatment process. By using high performance cement (HP-cement), the waste loading can be raised and deterioration of waste forms after land burial can be prevented. The high performance cement consists of slag cement, reinforcing carbon fiber, natural zeolite and lithium nitrate (LiNO3). The fiber allows waste (spent resin) loading to be increased from 25 to 60kg-dry resin/200L drum. The zeolite (clinoptilolite) reduces radiocesium leachability from the waste form to below 1/10. Lithium nitrate reacts as an inhibitor for aluminum contained in the incinerated ash, which raises ash content in the waste form from 10 to 30wt%. This inhibitor is suitable for cement solidification of miscellaneous solid waste containing Al pipes and foil. A

full-scale pilot plant was constructed and 200L waste forms were prepared. The water resistance test was performed for more than three years and no deterioration was observed.

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7-4

CONCENTRATION OF IRON and COPPER STEAM GENERATOR CLEANING SOLVENTS

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ABSTRACT

Chemical cleaning is a key method for removing metal and/or mineral build up in power plant steam generators. Such cleaning compounds are tailored to remove specific metals such as copper and iron. The formulations differ in each case, but one formulation constant can exist: Each formulation contains ethylene diamine tetraacetic acid (EDTA), a metal binding (chelation) agent. The presence of EDTA requires that the spent cleaning solution be treated to destroy this chelant. This may be done through methods such as incineration (hot oxidation), wet air oxidation, and wet oxidation (boiling water). Regardless of the method of treatment, waste reduction is an important technique to be applied to EDTA-bearing aqueous wastes, such as steam generator cleaning solutions. This paper will outline some of the details surrounding the concentration of both a spent EDTA-Iron and an EDTA-Copper cleaning solvent. The effect of Concentration on pH and specific Gravity of the solvents will be reviewed as well as the limits of the concentration process.

Table I

IDEAL SODIUM EDTA SALT SOLUBILITY - pH RELATIONSHIP

EDTA may form four (4) sodium salts, mono through tetra. The average solubility of the sodium salts as the pH changes is demonstrated in the solubility equation below. The salt solubility\ pH relationship for the free EDTA acid in water is somewhat different. However, since free acid EDTA is present only in very low concentrations, if at all, in a spent steam generator cleaning solution, the Regression Equation below gives a better insight into the solubility characteristics of an ideal solution of heavy metal EDTA chelates (Fe, Cu). NaEDTA Solubility in Water @ 25oC = 1388 - 628 (pH) + 435 (ln pH)3 R2 = ~98, Standard Deviation = ~24 solubility units.

The pH range found in the spent steam generator cleaning solvents is typically between 8.3 and 9.5 prior to concentration. After concentration, the pH range found for the iron and copper cleaning solvents was from about 5.6 to 7.9. Apparently, due to factors such as variance in ionic strength and the presence of other chemicals such a trace peroxide and ammonium hydroxide as well as metal salts, the observed solubility of the EDTA chelates in the concentrates was higher than initially predicted and higher than might be anticipated by reviewing the data generated from the Ideal Salt Solubility equation above.

pH CHANGE IN IRON SOLVENT

In Fig. 1, the effect of the concentration step upon the spent iron solvent is demonstrated. The relationship shown is based upon averaging all of the batches. Each batch of cleaning solvent varied and the curve for the change in pH versus the concentration level was, therefore, not the same for each batch; but the data in the graph adequately presents the pH change effects observed.

The relationship of Concentration Factor (CF) and change in pH from the beginning of

the concentration step until the stopping point can be generally described by:
CF = -1.2 - 2.3 * pH Difference from Beginning to Stopping R2= .98
pH CHANGE IN COPPER SOLVENT
In Fig. 2, the variation in the copper concentrates at the end of the concentration step versus the initial waste cleaning solvent pH is compared. Again, as with the Iron solvent, this data represents an overview. The individual batches of copper solvent differed; but, on average, the relationship shown in Fig. 2 is a reasonable representation. The relationship of the Concentration Factor to the change in pH during concentration is expressed by the mathematical equation:
CF = ~ -125 + 15 * ln pH + 13 * pH R2= >0.95 [pH = difference in pH from beginning to stopping point]
SPECIFIC GRAVITY CHANGES- Fe SOLVENT
As the iron solvent is concentrated, the specific gravity changes. This can be used to determine when to stop the concentration process and, thereby, minimize the risk of salting out the concentrator. The information in Fig. 2 indicates how this process works.
The relationship of the iron Concentration Factor to the change in specific gravity is expressed by mathematical equation:
Natural Log of CF(Fe) = 10.7 + 17.9 * Sp. Gravity + 8.3 * (Sp. Gravity)2 -or-
CF(Fe) = e(10.7*Sp. Gravity) + 8.3 (Sp. Gravity)^2 R2 =>.95
Fig. 3
SPECIFIC GRAVITY CHANGES-Cu SOLVENT
As with the iron solvent, the Concentration Factor can be controlled through the use of changes in the specific gravity. Figure 4 shows this relationship. The mathematical expression that described this is:
CF = ~ -0.87 + 61 * Nat. log * (Sp. Gravity) R2 => 0.95
CONCLUSIONS
The concentration of iron and copper cleaning solvents based on EDTA chemistries is possible to accomplish with reasonable and useful efficiencies. The level of concentration before and after rising is shown below in Table II. The utilization of a large, "already in place 30 gallons per minute" evaporation system (upon which this data is based) created a much larger amount of clean-up water than would have been required had a specialized more compact evaporator been used. This reduced the total "net" Concentration Factors. For the Iron solvent this CF reduction was about 19% and for the copper, the final solvent concentrated, the percentage reduction was about 25%.
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7-5
SOLID RADWASTE FACILITY IMPROVEMENTS AT DIABLO CANYON POWER PLANT
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ABSTRACT
Diablo Canyon Power Plant (DCPP) is located in Central California on the Pacific coast. Pacific Gas and Electric Company (PG&E) owns and operates the two Westinghouse 1100 Mwe PWR units. The commercial operation dates for Units 1 and 2 were May 1985 and March, 1986, respectively.
In 1981 the NRC issued Generic Letter 81-38 which anticipated the need for utilities to store radwaste due to interruptions in the flow of radwaste to burial sites. At that time PG&E decided to make the investment and provide additional interim storage at Diablo Canyon. DCPP decided it was necessary to provide onsite storage for radwaste as an insurance policy in the event that burial site access was cut off. By 1983 the design was issued on a plan to extend the walls of the small existing storage facility. Although the initial design was issued in 1983, the design was

scrapped in favor of building a completely new building. The design for this new building was issued in 1985.

DESCRIPTION OF RADWASTE FACILITIES

The radwaste storage facilities at DCPP are composed of two buildings. The old storage building was built during the original construction. It was intended to provide storage for palletized drums of cement solidified boric acid and boxes of DAW. Figure 1 shows the layout of the old storage building. The old storage building is separated into six bays. Bay 1 was intended to be a truck bay, but is now used for storage of Radiation Protection equipment. Bay 2 is used to store highly radioactive drums containing cartridge filters, odd sized parts and valves. This bay also contains an air pallet, a spent filter storage cask, a processing shield and an area for sampling filters. Bay 3 an bay 4 were intended to provide storage for drums. Bay 5 originally contained tanks for accumulating oil. Bay 6 was intended for box storage.

It became clear in the early 1980's that the original facility was inadequate to meet the radwaste storage needs and an upgrade of the original facility was planned. This plan encompassed the demolition of the interior shield walls of the building and installing new cranes to enable high activity waste to be handled remotely. The largest size container that could fit in the low head room design was an 80 ft3 liner. New 5 ton bridge cranes were procured based upon this design. DAW storage was to be provided by a metal butler building erected on the roof of the old facility.

Reconsideration of this storage expansion design resulted in the construction of a new storage building. The old facility would be retained. The metal building would be added but serve as an on site laundry facility. The new building would utilize the 5 ton cranes already procured for storing liners of resin or filters and provide a vault for boxed DAW storage. The new building consisted of a control room, an HVAC fan room, a truck bay, two liner storage vaults each serviced by a separate bridge crane and a large DAW storage vault. The new building configuration is shown in Fig. 2.

A shielded rail car is provided to transport 80 ft3 liners from the solidification pad (east of the auxiliary building) into the truckbay. A closed circuit television (CCTV) system for remote operation of the rail car and each crane is provided. A remote liner inspection station and decontamination station is provided in the east liner vault. At the inspection station container weight and radiation and contamination surveys can be obtained remotely. Two CCTV cameras in this station and a turntable allow visual inspections of high activity liners to be performed remotely. The truckbay provides access for trailer mounted shipping casks. The shipping casks must be loaded using the east vault crane.

The south end of the new building is the location of the separate DAW storage vault. This vault has fire protection sprinklers and boxes can be stacked four high. All container handling in this vault is by forklift.

Construction of the building was started in 1985 in anticipation of disposal site closures at the end of 1986. With the extension of disposal site access, the bulk of the installation was not finished until 1987. Since the building was not needed until burial site access was lost, the startup and operational testing were given a low priority. In 1989 disposal site access was again at risk and an intensive testing program was initiated. These tests were completed and the facility was turned over to the plant. Neither crane had been certified by a state inspector, however, and disposal access was extended again. In 1992 and several attempts were made to certify the cranes. These attempts failed and resulted in physical damage to the cranes. In 1994 a team of individuals was assembled to provide operating cranes for certification and enable liners to be stored on site.

PROBLEMS WITH THE NEW BUILDING DESIGN

Soon after the physical building was erected problems with the design started to emerge.

Shipping Cask Access

The 80 ft3 liners were designed to fit into an 80 ft3 Type B shipping cask. These

casks lost NRC certification. The truckbay door opening is only eleven feet high. This opening will permit access of LSA greater than Type A casks but not the new Type B casks mounted on trailers. Various options were considered to resolve this including the enlarging the concrete door (too expensive), purchasing our own cask (expensive, but feasible if the cask could be rented out), purchase of a lowlift trailer or bare liner transfer from rail car to shipping cask outside the truckbay. No decision has been finalized since shipping is no longer an option and Type B cask shipping regulations are due to change again.

Remotely Operated Bridge Cranes

Two 5 ton bridge cranes traveled the length of the vaults and move liners to and from the vaults into casks or the rail car. Heco Pacific Corporation provided the crane and control room design and fabrication. Many problems evolved with the complicated design.

1. limit switches. In order to control the crane from a remote room, the crane was fitted with many limit switches. These limit switches provide three different speed zones and protection from banging liners into the walls or obstructions. Different crane speeds were necessary. Slow speeds are required to provide control while grappling an object. Fast speeds are needed to traverse long distances. The limit switches became a nightmare of complexity when modifications to the original design were needed. First, installation of the cranes in the new building added the twist of the inspection and decontamination stations for one of the cranes. The addition of a slow zone encompassing these stations created the need for more limit switches which made the design more complicated, prone to failure and much harder to trouble-shoot. During the startup testing in 1989 the hoist limit switches were replaced by programmable units to enable stacking of liners two high in the vaults.

2. festoon cable trolleys. The originally supplied bridge crane cable trolleys failed the 1989 startup testing. Since the cranes were originally procured for the low head room constraints of the old building, flat ribbon cable was not used. Round festoon cable was supplied and this cable was mounted in a coiled horizontal orientation. This design was an asset for truckbay shielding since no notch in a vault wall was required to permit the festoon cable to pass. This cable passed over the vault wall guided by a channel track. The channel track design included a curve section in the truckbay to stow the coiled cable when the crane was parked in the truckbay. Unfortunately, the cable trolley trucks would stick in the channel track. The entire cable track and trolley were replaced with a triangular track with external truck trolleys for each crane.

3. control cabinet. The crane control cabinets originally contained inverter boards. When the west vault crane was damaged in 1992, a fuse blew and damaged one of these boards. The boards were not readily available an the damaged one had to be sent off site for repair. When the east vault crane was damaged in 1993 and it was determined that the board had been damaged again, it was decided to replace the boards with digital inverters. These new inverters were installed in the first quarter of 1994.

4. Festoon cable. During the functional testing of the new crane inverter controls in 1994, the crane experienced control problems. The cause of the problems was traced to short circuits between conductors in the circular festoon cable. Friction between the individual conductor insulation sheathing developed as the cable coiled and uncoiled during bridge movement. Several short circuits were noted over the length of the cable. The festoon cable was replaced for both cranes with circular multi-conductor cable with smooth and hard insulation sheathing. The new cable eliminated the friction and short circuit problem.

5. liner grapple device. During the 1989 startup testing it was noted that improved indication of remote grappling would be helpful to crane operators. Installation of mechanical flags attached to upgraded grapple limit switch pins in 1994 enabled the existing CCTV system to provide the operator with this indication.

Crane Retrieval System

The original crane retrieval system was a manual ratchet on a wire tow rope. Although it was proven to work in 1989 it was very slow and labor intensive. A system that could be powered by a portable electric drill was requested. This modification was approved, installed and successfully tested in 1993.

High Rad Barriers

In 1991, NRC regulations on high rad barriers were enforced under a new interpretation. Since the building was not designed with this interpretation in

mind, modifications were required to bar and lock any potential access point. It was decided that the easiest items to lock were the truckbay shield door and personnel door. A keyed hand switch was added to the control console and a ventilation penetration 20 feet high in the truckbay wall was barred.

Building Roof

The roof of the new building was concrete with a built-up coating. In 1992 leaks in the roof of the DAW vault were noted. Inspection of the roof determined that cracks had developed in both the liner and DAW vault roof sections due to seasonal thermal cycling. These cracks were leak repaired in 1992. In 1993, the DAW vault roof leaked again through two leak repaired cracks. A new polyisobutylene (PIB) membrane roof was installed over both liner and DAW vaults in 1994. This roof has withstood the recent 500 year rainfall in California.

Ventilation Ducting

During planning for roof repair in 1994, it was noted that the exterior ventilation ducting had corroded. The damaged ducting was the fan suction section which was square duct. This section was replaced by round ducting in 1994 to eliminate water pooling and subsequent corrosion.

Inspection Station

The inspection station and decontamination station were supplied by Hydro Nuclear through a competitive bid. The inspection station was to have been supplied with a remote readout scale linked to the liner turn table. The startup testing in 1989 determined that no remote scale was provided. Due to an overzealous seismic review of the preliminary drawings, the vendor was ordered to remove the proposed scale. Modification of the station to retrofit a scale with the turn table proved to costly. A portable scale was procured that could be remotely grappled by the east vault crane and set on top of the turn table. Cables from the scale must be feed through a penetration in the inspection station shield wall.

MODIFICATIONS TO THE OLD BUILDING

Conversion of Bay 6 to a Clean Waste Sorting and Hazardous Waste Handling Area.

In order to implement radwaste minimization fully, a green is clean program was established from day one at the plant. An area was needed to accumulate and sort this potentially non contaminated trash. Hazardous waste regulations in California are very onerous. A location was needed to sample and survey liquid and solid hazardous consumable generated from both contaminated and non contaminated locations within the radiological controlled area (RCA) of the plant. A sort table and a clothes dryer were added to the bay to support clean trash handling. Two hooded tables and several drum hoods were added to the bay to support hazardous waste handling.

Mixed waste storage.

Bay 4 of the old building now serves as the mixed waste storage location on site. The tanks in Bay 5 have been removed and we petitioned State of California to approve mixed waste storage in this area. Plans are to perform a hazardous waste closure on Bay 4 and return it to radioactive storage.

Remote Handling Modifications to the Filter Storage Area.

One of the upgrades which proved to be beneficial from an ALARA point of view was completed in the Bay 2 of the Old Radwaste Building. Originally, the aisles used to store highly radioactive drums were serviced by a pendant controlled hoist on a monorail system. This, combined with a simple mechanical drum grab proved to be very inefficient, therefore, resulting in unacceptably high exposure to the workers.

A new design was implemented which provided remote actuation to the hoist through a retractable cable reel installed on top of the shield wall which feeds control wires inside one cable to the hoist, trolley, new camera and new electric drum grab. With this new system, the operator can control the grapple, the hoist and the trolley from a control console outside the shield wall. The system operation can be observed on a monitor installed on the control console. This new system saved approximately 3 Man Rem of exposure during the last refueling outage.

EVALUATIONS AND MODIFICATIONS NOT ADOPTED

Decontamination Station.

The New Radwaste Storage Building was furnished with a Decontamination Station. Next to the inspection area there is a completely enclosed and water tight decontamination chamber. The decontamination for small tools and equipment can be performed by a worker standing outside the chamber and using the protruding gloves and the decontamination gun supplied with the chamber. Next to the decontamination

chamber, behind a shield wall is the control console for the remote operation of the turntable and the decontamination chamber. This station was originally designed for use with freon as the cleaning solvent. Regulatory changes on ozone depletion made startup testing of the station in 1989 with freon imprudent.
Several vendors were contacted in late 1993 about converting the station to use a different solvent. Both CO2 and hot water solvents were examined. The cost of the CO2 unit was justified by the projected need to decontaminate a liner or two per year. Experience with hot water in other services was insufficient to insure that residue on the station walls would not create an airborne radiation area. At this time other substitute solvents are being investigated.
Liner Stacking.
During the startup testing in 1989 it was noted that depth perception should be improved prior to stacking liners in storage. During crane operator training, feedback will be requested from personnel on needed improvement, if any, to improve stacking. One potential change would be to covert a fixed scan camera on each crane to a pan and tilt type to provide depth perception. Another option now available with the digital inverters would be to add a remote readout at the control console for grapple height. Based upon operator feedback we will proceed with changes for stacking liners as required.
CONCLUSION AND RECOMMENDATIONS
Numerous changes were implemented to the radwaste storage buildings at DCPP. These changes have resulted in providing adequate storage to support continued plant operation now that access to a disposal site is lost. Dry active waste and mixed waste have been stored at DCPP without incident. The first liner, actually a canned poly HIC, was placed in storage by the east vault crane on January 27, 1995. Recommendations based upon this experience are that container size selection is very important. Large containers may be too large to be shipped in new casks certified under shipping rule changes while waste is in storage. Ensure that any cranes are state certified before accepting turnover of the building.
Initially, building of interim storage facilities seemed a straightforward project. Experience has shown that due to the many regulatory uncertainties this is a complicated and time consuming project. Examples listed in this paper detail problems with containers, remotely operated cranes, saltwater environment, changing technology and unexpected needs which all contributed to the complexity, expense and delays of the endeavor.

7-7
A SYSTEMS ENGINEERING ANALYSIS TO EXAMINE THE ECONOMIC IMPACT FOR TREATMENT OF TRITIATED WATER IN THE HANFORD KE-BASIN
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ABSTRACT
Federal and state agencies have established a Tri-Party Agreement (TPA) to address some key environmental issues faced at the Hanford Site. Under the TPA, the Department of Energy is currently under a consent order to reduce the tritium concentration in the spent fuel storage basin for KE-Reactor from 3.0 mCi/L to 0.3 mCi/L in the KE spent fuel storage basin, starting in 1996. The 100KE and 100KW Area fuel storage basins (K-Basins) at Hanford were built in the early 1950s to receive and provide temporary storage for irradiated fuel from the now shutdown KE and KW production reactors. In 1977, the KE-Basin began to leak at a rate of 13.5 gpm (51 L/min.), but decreased to 0.03 to 0.05 gpm (0.13 to 0.19 L/min.) by 1980. In 1993, the leak increased to a rate of 0.42 gpm (1.6 L/min.). This engineering analysis examines the relative costs to reduce the tritium concentration KE-Basin water using a polyphosphazene polymer membrane under development at Pacific Northwest Laboratory. The estimated cost of using the membrane to reduce the tritium concentration is compared to three no-treatment alternatives that include 1) disposing of the tritium-contaminated water directly to the Columbia River, 2) disposing of the contaminated water to the soil at the on-site Effluent Treatment Facility, and 3) disposing of the contaminated water by evaporation using solar evaporation ponds.

INTRODUCTION

The Department of Energy (DOE), other federal, tribal, and state agencies involved with Hanford Site are concerned about the KE-Basin water leakage and its possible effect on the environment. A Tri-Party Agreement (TPA) milestone has been established to remove all the spent fuel and sludge from the 100 Area fuel storage (KE and KW-Basins) in encapsulated form by December 31, 2002. In 1977, the KE-Basin began to leak to the ground at a rate of 13.5 gpm (51 L/min.), but decreased to 0.03 to 0.05 gpm (0.13 to 0.19 L/min.) by 1980. In 1993, the leak increased to a rate of 0.42 gpm (1.6 L/min.) Under the TPA, the DOE is currently under a consent order to reduce the tritium concentration in the KE-Basin to 0.3 mCi/L, starting treatment in 1996.

There are currently no commercially available methods to separate tritium from large volumes of water with relatively low tritium contamination levels. The Pacific Northwest Laboratory (PNL) is conducting a research effort to evaluate the use of membrane technology for separation of tritium from sources containing HTO as a contaminant.(1)

This engineering analysis examines the relative costs to reduce the tritium-concentration in the basin water using a deployment strategy that uses the polyphosphazene polymer membrane. The estimated cost to reduce the tritium concentration using the polyphosphazene membrane is compared to three no-treatment alternatives that include 1) disposing of the tritium-contaminated water directly to the Columbia River, 2) disposing of the contaminated water to the soil column after treatment at the on-site Effluent Treatment Facility, and 3) disposing of the contaminated water by evaporation using solar evaporation ponds.(2)

MEMBRANE DESCRIPTION

The results from the research conducted thus far indicate that the use of membranes to remove dilute tritium from water is achievable.(1) Aromatic polyphosphazenes were chosen as the polymeric material among membranes being investigated because they have been shown to possess excellent radiological, thermal and chemical stability as well as good tritium separation factors. The membrane system is similar to that of reverse osmosis and requires pressure from 10 to 48 psi (6.9 x 103 N/m2 to 3.3 x 104 N/m2). Research results have indicated that the polyphosphazene membrane within the system will provide up to 75% depletion of HTO after a single pass through the system. Additional research is being conducted to confirm and extend the usefulness of such a membrane separation procedure.

Based on an evaluation of the KE-basin treatment goals, and a knowledge of membrane system operations from research done at PNL, a membrane deployment concept has been identified that would reduce the tritium concentration in the KE-Basin. In this concept, water would be pumped from the KE-basin to a cascading membrane system made up of cocurrent multiple stages. The tritium-depleted stream from each stage would be recycled back to the basin. Concentrate would be used as the feed for the next stage. As the concentrate passes through each succeeding stage, the tritium concentration in the concentrate will increase. The total volume of concentrate leaving each stage in the conceptual scheme is set at 25% of the feed going into that stage. The tritium concentration increases in the concentrate stream to 220% of the feed to that stage and contains 55% of the tritium in the feed in 25% of the feed volume. When the concentrate leaves the final stage, it would then be solidified in cement and sent to an on-site DOE-operated low-level radioactive waste disposal facility.

To estimate the cost to reduce the tritium concentration in the KE-Basin using the membrane system, a mathematical model was developed to estimate the time required and waste volume generated during the different process scenarios. The model was based on the process configuration provided in Fig. 1. In this configuration, it is assumed that the tritium-depleted stream is continuously returned back to the KE-Basin, while the concentrated tritium stream is further processed to increase the tritium concentration and reduce the volume. A constant volume of water is maintained in the KE-Basin to ensure that any spent fuel elements that may be in the basin are not exposed to the atmosphere. Clean make-up water is added to the basin at the same rate that the concentrate stream is leaving the basin. Based on the model results, cost estimates were made for capital, operational requirements, and waste disposal.

MEMBRANE SYSTEM DESIGN CONCEPT ASSUMPTIONS

The design concept of the membrane system for removal of tritium and cost estimates

for the system are based upon the following assumptions:
1) Tritium concentration in the tritium-depleted stream is 60% of the concentration in
the feed stream entering each membrane stage for each pass.
2) Tritium-depleted stream to concentrate the stream flow ratios in each stage are
75%:25%.
3) The membrane system is configured in a cocurrent-flow series such that each
succeeding stage will increase the concentration of tritium in the brine stream (see
Fig. 1).
4) The system will operate with commercially available membrane technology.
5) The tritium-depleted stream is returned to the KE-Basin.
6) The starting material in the KE-Basin is 1 million gallons (3.8 million liters) of water.
7) The initial tritium concentration in the KE-Basin is 3.93 mCi/L.
8) The tritium concentration in the makeup water is 0.001 mCi/L.
9) No new tritium is formed or decays during the time the membrane system is operating.
A material balance was developed around the KE-Basin, pretreatment, and membrane processes to estimate the concentration of tritium in the tritium-depleted stream, concentrate stream, and KE-Basin at any time during the process.
A material balance may be estimated around any given stage:
Eq. (1)
where:
FRn = Feed rate entering the nth stage
Fpn = Tritium-depleted stream flow rate leaving the nth stage
Fbn = Concentrate stream flow rate leaving the nth stage
Based on the given assumptions for tritium removal from each stage, an equation to estimate the concentration leaving the nth stage may be derived:
Eq. (2)
where:
Cfn = Tritium concentration going into the nth stage
Cbn = Tritium concentration in brine leaving the nth stage
Co = Tritium concentration initially in the KE-Basin
To estimate the size of each stage, its feed rate must first be determined. Based on our assumptions, flow rates of the tritium-depleted stream to the concentrate stream will be 75:25, and the tritium-depleted streams from each stage will be returned to the K-Basin. Therefore, each stage receives the concentrate from the previous stage (i.e., FRn+1 = Fbn), and the feed rate at each succeeding stage is 25% of the feed rate of the stage up-stream of it (i.e., 0.25FRn = FRn+1, or 0.25Fbn = Fbn+1). An equation to estimate the flow rate from each stage can be determined by the following equation:
Eq. (3)
where:
F1 = feed rate entering first stage
Fb1 = tritium-depleted stream leaving the first stage
A differential balance was developed to indicate what is happening in the system at an instant of time. Based on the system configuration illustrated in Fig. 1 and the previous design equations, a differential balance was developed to estimate concentration of tritium in the KE-Basin at any time given initial tritium concentration and flow rate through the membrane system. This is shown below.
Eq. (4)
where:
n = number of membrane stages.
V = water volume in the KE-Basin, L.
vb = flow rate into the first membrane, L/min.
Ca = concentration of tritium in clean make-up water, mCi/L.
Cb = concentration of tritium in KE-basin water at t-time, mCi/L.
Co = initial tritium concentration in the KE-Basin.
t = time to reduce tritium concentration from Co to Cb, min.
Solving for t:
Eq. (5)
This equation estimates the time required to reduce the tritium concentration from

Co to Cb at a specified flow rate. Cost estimates can then be made based on the time required to operate the system and the volume of tritium separated, and the amount of waste resulting from the solidified tritium.

COST ASSUMPTIONS

The cost to reduce the tritium concentration in the KE-Basin using the membrane system was compared to other cost estimates made for three alternative disposition schemes for the water.(2) The membrane system concept and three alternatives considered were evaluated in terms of cost using the Equivalent Uniform Annualized Cost (EUAC) methodology that is identified in Refs. 2 and 3. These alternatives are 1) Columbia River discharge; 2) ground disposal after treatment at the Effluent Treatment Facility (ETF); and 3) solar evaporation at the 100K Area.

The KE-Basin currently has an existing pretreatment system to remove most of the dissolved materials and suspended solids and radionuclides (except tritium) from the KE-Basin prior to the any alternative action.(2) The cost for this pretreatment system was included in the total cost for each alternative. The cost for the pretreatment system is also included as part of the membrane system to provide a consistent cost analysis. The pretreatment system includes a sand filter, an ion exchange module (IXM), and an additional microfilter/ultrafilter and a polishing IXM that operates at 25 gpm (95 L/min.).

The costs for the alternatives were divided into capital costs and annual operating costs. The total purchased equipment cost for the pretreatment system is estimated to be $520,000. The estimates for the membrane system are based on the same cost factors and procedures used to estimate the cost estimates for the other alternatives. Additional costs include installation, testing, solidification, and disposal. The details for the development of the cost estimates for the other alternatives are provided by Hunacek and Gahir.(2) Cost estimates made for a membrane system were based on published information.(4,5,6,7)

To prepare the concentrate stream for final disposition, the stream is solidified into cement inside steel 208-liter containers. First, the concentrate stream is mixed with cement in a 208-liter container. Generally, 25 gallons (95 liters) is mixed with five 67-lb (30-kg) bags of cement and aggregate to obtain approximately 1 yd3 (765 L) of solidified material. The container is then transported 25 miles to the on-site disposal facility. The 1994 cost for on-site disposal of low-level radioactive waste is $49.96/ft3 ($1.76/L). The tritium concentration in the containers will not exceed the maximum concentration of 5 x 109mCi/m3 allowed at the disposal facility.

The estimated EUAC methodology estimate for disposing of the tritium-contaminated water to the Columbia River, disposing of the tritiated water to the soil at the Effluent Treatment Facility, and disposing of the tritiated water by evaporation using solar evaporation ponds was $2.25 million, $3.26 million, and $3.05 million, respectively. The cost estimates made for the membrane system range from $79 million to $2 million, depending on the number of stages used and variations in operating and feed flow rates.

The EUAC for the membrane system was calculated using a 7% time value of money and a project life of six years. The EUAC methodology estimates provides the annual operating and maintenance costs for the treatment system, including the costs associated with capital equipment investment.

The following assumptions were used to estimate the cost for the membrane system:

1) The maximum concentration of the solidified tritium does not exceed the waste acceptance criteria of 5 x 109 mCi/L.(2)
2) The 95 L/min. pretreatment system used in this analysis is the same one used in Hunacek.(2)
3) The concentrate stream is solidified using cement for final disposition.
4) The solidified tritium is transported 25 miles (38 km) to the on-site disposal facility.

The waste generated will be disposed at the Hanford Facility as DOE-generated low-level radioactive waste.

RESULTS

Three scenarios were considered to examine the costs for operating the membrane under different circumstances. The first scenario compared the cost of a 25-gpm (95-L/min.) system with a different number of membrane stages to the three alternatives identified in Hunachek and Gahir.(2) The second scenario examined the time effects with the system operated at 5-gpm (21.4 L/min.). The third scenario

examined the requirements needed to treat the tritium-contaminated water in one year for a different number of membranes stages. A brief summary of results are provided in Table I.

The first scenario assumed a constant feed water flow rate of 25 gpm (95 L/min.) that is consistent with the flow rate of the other alternatives. These three alternatives include discharge of the tritiated water to the Columbia River, ground disposal after further treatment at the Effluent Treatment Facility (ETF), and solar evaporation at the 100K Area seen in Fig. 2. The most expensive system was the one-stage membrane system. This system could reduce the tritium concentration to 0.3 mCi/L in 0.4 years with an EUAC cost of $79 million. For this option, the volume of solidified waste generated is estimated to be 4.4 million liters. Most of the cost is attributed the disposal of the solidified tritium waste. When the tritium concentration is increased in the concentrate stream, the disposal cost decreases significantly, but at the expense of treatment time. For example, to operate a 10-stage membrane system with a flow rate of 25 gpm (95 L/min.) into the first membrane, the EUAC cost is $2 million, 40 times less than the one-stage membrane system. The volume of waste generated was estimated to be 63,738 liters. But the process time needed to reduce the tritium concentration is 77 years, 190 times longer than a 1-stage membrane system. The optimum number of stages required ranges from 4 to 6 stages, with a cost from $9.4 million to $3.7 million, depending on the time specified. The cost to operate the membrane system for a different number of stages decreases asymptotically due to the increase in labor required to operate the system for a longer period of time.

The second scenario assumed that the flow rate into the first stage was 5 gpm (21.4 L/min.). The reason for this scenario was to examine the effect that the flow rate had on the membrane system seen in Fig. 3. The cost to reduce the tritium concentration in the KE-Basin was similar to the 25 gpm (95 L/min.) for the corresponding number of stages. For one- and ten-stage systems, the cost estimates were $79 million and $2.1 million, respectively. The time required to operate these systems increased significantly, 1.6 years for a one-stage system and up to 342 years for a ten-stage system. The volume of waste generated is the same for the corresponding number of stages when comparing the 25 gpm (95 L/min.) membrane system to the 5 gpm (19 L/min.) membrane system, but the process time required to generate the solidified waste increased in direct proportion to the decrease in flow rate. The cost to operate the membrane system for a different number of stages decreases asymptotically due to the increase in labor required to operate the system for a longer period of time.

The third scenario assumed a one-year processing time to reduce KE-Basin tritium concentration from 3.93 mCi/L down to 0.3 mCi/L. The economic analysis indicated that the cost ranged from $79 million to $6.5 million, depending on the number of membranes required to remove the tritium from the KE-Basin water as seen in Fig. 4. When a one-stage membrane is used to reduce the tritium concentration in one year, a flow rate of 9 gpm (34 L/min.) is required. The volume of solidified waste generated is approximately 4,400 m3. The capital and operation cost for a 1-membrane system is estimated to be $1.7 million. The disposal cost for the resultant solidified waste is estimated to be $77 million.

When a ten-stage membrane system was considered, a 1,900 gpm (7,200 L/min.) unit was required to reduce the tritium concentration to 0.3 mCi/L. The volume of solidified waste generated is estimated to be 0.4 m3. The capital and operating cost for a 10-stage membrane system is $25 million. The disposal cost for the solidified waste based on the estimated waste volume is $64 thousand.

The optimum cost for a one-year treatment to reduce the tritium concentration is estimated to be $6.6 million, and uses and 6-membranes with a flow rate of 177 gpm (670 L/min.). The estimated volume of solidified waste generated from this option is 3.9 m3. The capital and operational cost estimated for this system is $5 million. The disposal cost for the solidified material is $1.5 million.

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7-8

APPLICATIONS OF FIBER-REINFORCED CONCRETE CONTAINERS IN FRANCE AND IN THE USA

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ABSTRACT

Commercial generators, of low-level radioactive wastes in 31 states, are faced with the dilemma of unavailability of commercial low-level radioactive waste disposal facilities for some undefined period of time. Many commercial generators have developed contingency plans for on-site storage of generated waste until such time that the developing compact disposal facilities become available. Requirements for containers to be used for extended periods of on-site low-level radioactive waste storage include durability, ruggedness, shielding capability, handling flexibility and acceptability at the compact disposal facilities. Chem-Nuclear Systems, Inc. (CNSI) has developed a container, which meets the stated requirements, for use as an on-site storage container and as a final disposal container. This versatile container, known as the Multi-Use Container (MUC), is also well suited for waste processing operations and over-the-road transport.

The MUC consists of a fiber-reinforced concrete outer shell with a polyethylene inner container. The utilization of fiber -reinforced concrete as a low-level radioactive waste disposal container material was developed in France by COGEMA in the late 1980s. The use of polyethylene as a low-level radioactive waste disposal container material began in the United States in 1981. Combining the two materials provides technical and operational advantages that give the MUC superior versatility. This paper provides details on the design and operational applications of the MUC. Information on the certification of the MUC as a high integrity container is also presented.

INTRODUCTION

Chem-Nuclear Systems, Inc. (CNSI) recognized in early 1991 the potential for interruption of access to disposal facilities for many of the commercial low-level radioactive waste generators. Many commercial generators had arrived at the same conclusion, and were developing contingency plans to store generated waste on-site until such time as disposal facilities were available. In response to this, CNSI embarked on a program to develop a container that could be used by generators for on-site storage and as the ultimate disposal container.

In the late 1980s, CNSI was selected as developer and operating contractor for three of the compact disposal facilities. These included the Southeastern Compact in North Carolina, the Appalachian Compact in Pennsylvania, and the Central Midwest Compact in Illinois. Each of these disposal facilities requires that low-level radioactive waste be placed in a concrete overpack for disposal. Polyethylene containers received at CNSI's Barnwell disposal facility are also placed in concrete overpacks prior to final disposal. With this in mind, attention was focused on development of a durable and efficient concrete storage/disposal container.

In France in the late 1980s, a multiple year research effort by COGEMA, a world leader in the nuclear industry, culminated in the development of fiber-reinforced

concrete containers for containment and disposal of low and medium level wastes. These containers were certified as satisfying all French safety requirements relating to waste immobilization and disposal by ANDRA, the French national radioactive waste management agency (1). SOGEFIBRE, a subsidiary of Societe Generale pour les Techniques Nouvelles (SGN), member of the COGEMA group, began fabricating the containers on a production scale in July 1990 utilizing a dedicated production facility in Valognes, France. The containers were supplied to the French Atomic Energy Commission (CEA), COGEMA, ANDRA and others to store and dispose of low and medium level waste.

In June 1991, SOGEFIBRE made a presentation on the fiber-reinforced concrete technology to CNSI. The fiber-reinforced concrete containers developed in France fit well with the CNSI objective of developing a concrete container. However, given the differences in waste processing techniques and regulations between the U.S. and France, it was recognized that an additional element would be needed in the container design.

CNSI has been a leader in the development and utilization of polyethylene containers for use with low-level radioactive waste processing and disposal since early 1981. Polyethylene containers are well known for providing a highly corrosion resistant, virtually impermeable waste container. The polyethylene containers are familiar to the commercial generators of LLW and are simple to manufacture. The concern with polyethylene has been the structural effect of long-term sustained load. The combination of an external concrete shell to handle compressive structural loadings with an internal polyethylene container circumvents this concern. It was clear that a container composed of a fiber-reinforced concrete structural shell with a polyethylene inner container would provide a highly durable and versatile container which could be used not only for storage and disposal, but also for waste processing activities currently practiced by the generators and over-the-road transport for shipments containing Type A quantities of waste.

Moreover, CNSI was confident that such a container would be certified by the U.S. Nuclear Regulatory Commission as a high integrity container under the provisions of 10 CFR 61 and the Branch Technical Position on Waste Form. Subsequently, SOGEFIBRE sub-licensed the fiber-reinforced concrete container technology to CNSI and a joint effort was undertaken to develop a Multi-Use High Integrity Container. This paper will discuss the MUC design and its operational applications.

CONTAINER DESCRIPTION

The container is composed of a rectangular prismatic concrete shell and an inner polyethylene container. The inner polyethylene container may have the shape of a right circular cylinder or rectangular prism. The inner polyethylene container provides a
highly corrosion resistant, virtually impermeable container surrounding the contained waste. The fiber-reinforced concrete outer shell provides a high compressive strength, highly durable, and radiation-shielding structural shell. This combination of fiber-reinforced concrete structural shell and inner polyethylene container takes optimum advantage of the capabilities of both materials. The fiber- reinforced concrete shell provides the required mechanical strength, and is designed to provide durability with respect to the disposal environment as well as radiation shielding. The inner polyethylene container provides a highly effective corrosion barrier between the contained waste and the concrete shell, and since it is essentially impermeable, it also serves as an effective barrier to radionuclide migration.

This combination of materials also combined the expertise of both SOGEFIBRE and CNSI. By the end of 1994, SOGEFIBRE had produced around 30,000 fiber-reinforced concrete containers for disposal of low and medium waste in France. CNSI has provided over 7,500 polyethylene containers for disposal of low-level wastes in the United States.

The design of the MUC has benefitted from this combination of experience and knowledge gained in container production and utilization.

Figure 1 defines the MUC in its disposal configuration and demonstrates the inner polyethylene container and the fiber-reinforced concrete structural shell. Table I provides dimensions, weights and volumes of the MUC family of containers.

The inner polyethylene container is designed to be closed by means of an induction welding process which heat seals the lid to the vessel body. This feature allows the inner polyethylene container to be sealed in a remote operation.The fiber-reinforced

concrete shell is closed with a precast concrete lid for storage and transport, Fig. 2. This lid is removed, and a cast-in-place fiber-reinforced concrete lid is placed for final disposal.
Handling of the MUC is accomplished by utilizing grooves cast into the container bottom to permit handling with forklift tines, or by using lifting attachments in the topside of the container walls to permit handling with overhead cranes.The inner polyethylene container is equipped with a passive filter vent to allow venting of any gases generated in the waste, but no release of solid material. No vent is required in the fiber-reinforced concrete shell.The polyethylene container can also be equipped with an optional sample tube and seal plug. This sample tube permits sampling and inspection of the contained waste until such time as the concrete disposal lid is placed.Additional shielding can be added to the container in the interstitial space between the polyethylene inner container and the concrete structural shell. This shielding can consist of conventional concrete, high density concrete or iron/steel, and is selected to optimize shielding characteristics and economics. The size of the polyethylene inner container is adjusted to accommodate the shielding thickness required.

OPERATIONAL APPLICATIONS

The MUC is a highly versatile container well suited to the processing, storage, transport and disposal of low-level radioactive waste (Table II).

Processing Applications

Several MUC processing configurations are shown in Fig. 3. The MUC may be used to process a wide range of low-level radioactive wastes. Current dry and wet waste processing techniques may be used in conjunction with the MUC similar to current processing in conventional steel or polyethylene containers. Disposal of filters or irradiated hardware in the MUC is also similar to disposal in currently used containers.
Waste can be processed directly in the inner polyethylene container with the inner polyethylene container in place inside the fiber-reinforced concrete shell. In the event the MUC cannot be positioned to allow direct waste processing due to weight or size limitations, the MUC can be supplied with a removable cylindrical inner polyethylene container that can be loaded in a separate area and then rejoined with the fiber -reinforced concrete shell. This processing configuration would also be applicable for higher activity waste loadings when waste loading in the container was projected to require over-the-road transport in a Type B radioactive waste shipping cask. In this case,the cylindrical inner polyethylene container will be shipped separately to the disposal facility in a Type B radwaste shipping cask, and placed into a fiber-reinforced concrete shell at off-load at the disposal facility. The inner polyethylene container, cylindrical or rectangular, can be supplied with a full opening top lid for bulky waste or a smaller manway size opening for liquid waste processing.

Storage Applications

The MUC is an excellent storage container for extended on-site storage applications. The fiber-reinforced concrete structural shell has been evaluated for exposure to environmental conditions, such as precipitation, sunlight and potential accident conditions such as fire or dropping, and demonstrated to be a highly durable structure (2). Polyethylene containers have also been previously evaluated and shown to be suitable for safe storage of low-level radioactive wastes for extended periods of time (3).
The proven durability of the MUC, coupled with the shielding capability, make it an ideal storage container. Coupling this with the pending U.S. NRC certification as a high integrity container, ensures that the MUC used for storage will be acceptable for disposal at a licensed disposal facility without repackaging.

Transport Applications

Over-the-road shipment using the MUC provides both economic advantage and a high degree of operational simplicity. There are two options for use of the MUC for over-the-road transport. As noted previously with waste quantities requiring shipment in an approved Type B package, the cylindrical inner polyethylene container must be shipped separately in a U.S. NRC certified radioactive waste shipping cask. The inner polyethylene container will then be placed in a fiber-reinforced concrete structural shell at cask off-load at the disposal facility. All other waste can be shipped in the MUCin its transport configuration shown in Fig. 4.
This configuration consists of a removable energy absorbing impact limiter and

impact limiter and MUC tie-down system. The MUC and its transportation configuration have been evaluated and tested to verify conformance to Titles 10 and 49 of the Code of Federal Regulations. This evaluation demonstrated acceptability as a Type A packaging (Specification 7A container) in accordance with the U.S. Department of Transportation regulations.Successful full-scale drop testing of the MUC was performed to verify the MUC integrity under worst case accident scenarios.

Disposal Applications

The MUC disposal configuration is shown in Fig. 1. For disposal, the precast concrete transport/storage lid is removed.Any space between the inner polyethylene inner container and fiber-reinforced concrete wall is filled with a cementitious material. A fiber-reinforced concrete lid is then cast in place to close the MUC. The MUC is designed for shallow land burial to a depth of 25 feet and for engineered facility disposal with MUC stack heights up to 30 feet. As previously noted, the pending U.S. NRC certification as a high integrity container will ensure MUC acceptance at licensed disposal facilities, and provides certification of the fact that the MUC will remain structurally stable in the disposal environment for a period of at least 300 years.

HIGH INTEGRITY CONTAINER CERTIFICATION

A key element in the development of the MUC is obtaining U.S. NRC certification of the MUC as a high integrity container under the provisions of 10 CFR 61 and the Branch Technical Position on Waste Form. CNSI and SOGEFIBRE worked together to integrate the fiber-reinforced concrete methodology, test data, and industrial test results,together with the MUC design and high integrity container regulatory requirements specified in 10 CFR 61.56(b) (1) and the U.S. NRC Branch Technical Position on WasteForm (Revision 1). Included in this effort was full-scale prototype container testing which was conducted in the United States and in France by a joint CNSI/SOGEFIBRE team.

CNSI submitted the MUC Topical Report, "Multi-Use container High Integrity Container Licensing Topical Report, Docket # WM-107", to the U.S. NRC on July 23,1992. After thorough evaluation and review, U.S. NRC certification of the MUCas a high integrity container, is expected in the first quarter of 1995. Following NRC certification of the MUC as a HIC, it is CNSI's intention to use this technology to obtain approval for a cylindrical version of the MUC as well as container sizes in addition to those described in Table I.

CONCLUSION

The MUC is a versatile container well suited to commercial low-level radioactive waste generators' requirements. The MUC is designed to function as a process container, storage container, over-the-road transport container, and as a disposal container. The technology incorporated into the MUC includes the experience of COGEMA in developing fiber-reinforced concrete containers used for low and medium level waste disposal in France, the fiber-reinforced concrete container manufacturing experience of SOGEFIBRE and CNSI experience with polyethylene containers and low-level waste management services to the U.S. commercial nuclear industry.

The MUC accommodates low-level radioactive waste processing techniques currently in use and incorporates features which make it a simple easy-to-use container. Its design, incorporating high durability materials and self-shielding capability, makes it an excellent storage container. U.S. NRC certification as a high integrity container ensures MUC acceptability at licensed disposal facilities without the need for repackaging the contained wastes.

Use of fiber-reinforced concrete containers in France, their impending use in the United States, and projected use in eastern Europe demonstrate that the fiber-reinforced concrete container technology is becoming a major technology for low-level radio active waste disposal.

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RADIOLYTIC HYDROGEN PRODUCTION DURING THE LONG TERM STORAGE OF SPENT ORGANIC ION EXCHANGE RESINS

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ABSTRACT

Removal of radionuclides from contaminated process waters requires the extensive use of organic ion exchange resins at the Savannah River Site (SRS) and at Hanford. During storage of the spent resin, radiolysis from the radionuclides sorbed on the resin produces hydrogen and can create a flammable mixture in the storage container. This report presents results of radiolysis studies involving typical organic anion, cation, and mixed anion/cation resins used at SRS. These results include measurements of hydrogen production, water-soluble material formation, and changes in mass, volume, and moisture content due to exposure of the resins to gamma radiation. Hydrogen production rates, expressed in terms of G values or the number of molecules produced per 100 eV of energy absorbed by the resin, are used to estimate a minimum time to reach a flammable mixture in an actual SRS reactor moderator resin storage container. The G(H2) values determined for the anion resin, the cation resin, and 1:1 volume mixtures of the two, are in the range of 0.1-0.3 molecules/100 eV. Oxygen depletion was observed for all the resins irradiated in air-sealed systems. We have also examined the effect of liquid slurry type and the effect of varying water content on the radiolytic hydrogen production using a Savannah River Technology Center (SRTC) developed resorcinol-formaldehyde polycondensation type cation exchange resin. Total radiolytic gas production is highest for resorcinol resin-acid slurries and radiolytic hydrogen evolution is highest for resorcinol resin-water slurries. For resins irradiated with various amounts of water present, maximum hydrogen evolution occurs for the resin containing highest water content (~97 wt% water), with G(H2) = 0.7 molecules/ 100 eV. The measured hydrogen yield decreases linearly to a minimum of G(H2) = 0.1 molecules/100 eV for a pre-dried (~0 wt% water) resin.

INTRODUCTION

Organic ion exchange resins are used at Savannah River Site (SRS) and Hanford as well as many commercial nuclear installations to remove radionuclides from process waters that have become contaminated.(1-4) Contamination of reactor moderator water can occur during reactor operation or the water in basins used to store spent fuel such as the K-basin at Hanford can become contaminated. After these resins have been used to the extent that they are no longer efficient for removing the radionuclides, the resins need to be stored and eventually placed in permanent disposal. During this storage, radiolysis from the radionuclides on the resin produces hydrogen (5,6) and can create a flammable mixture in the storage container.(7,8) Recent reviews stressing the importance of understanding the effects of radiation on resins used in the nuclear industry have been published. (9,10) This report presents results of radiolysis studies designed to provide an estimate of the rate of hydrogen production from two commonly used resins so that the time to reach a flammable mixture (4% hydrogen in air) in the storage container can be estimated. We also include in this report further studies on our previously reported investigation of the radiolytic stability of a resorcinol-formaldehyde ion exchange resin.(11,12) The resorcinol-formaldehyde polycondensation-type cation exchange resin is currently being investigated for potential cesium removal applications at Hanford and the Savannah River Site (SRS).(11-13)

EXPERIMENTAL

Ion Exchange Resins

Two commercial resins that are used at SRS for moderator water decontamination were irradiated with Co-60 gamma rays. One resin was an anion exchange resin (Amberlite IRA-400-OH) and the other a cation exchange resin (Amberlite IR-120-H).(1,2) Both resins had a styrene-divinyl copolymer as the organic matrix. The anion resin contained trimethyl quaternary ammonia groups and the cation resin contained sulfonate groups. These resins are identical to those used at Hanford except the anion resin at Hanford contains ammonium groups rather than the trimethyl ammonium groups. The two resins were irradiated separately and then irradiated as a mixture. The mixed bed resin was a 5:1 anion to cation equivalent mixture formed from mixing

equal volumes of each resin. The resins are usually used in the mixed bed form because this form simultaneously removes cationic and anionic radionuclides from the contaminated water. The resorcinol-formaldehyde resins used in this study were of the potassium (K) form that results from the condensation reaction between resorcinol and formaldehyde in KOH.(11)

Irradiation Methods and Systems

The ion exchange resins were irradiated with Co-60 gamma rays and the rate of gas formation measured as a function of radiation dose. The anion and cation resins were each irradiated as received, i.e. neither resin was contacted with water prior to radiolysis. The mixed bed resin (damp) was irradiated in the swollen form (~50% water) in which they will be stored in the reactor areas at SRS. The mixed resin sample was prepared by mixing equal volumes of the anion and cation resins, rinsing with ASTM Type-I water and vacuum filtering immediately before irradiation. ASTM Type-I water is a highly purified form of water with resistivity > 18 Mohm.cm. A mixed bed resin (dry), which was not contacted with water prior to radiolysis (~35% water), was also studied. Moisture contents of the resins were determined from heating ~ 1 gram samples in triplicate at 60oC in a convection oven until no further mass change, typically for about 5 days. Higher drying temperatures were avoided due to possible thermal decomposition of the resins. (2)

Gas evolution studies involving the resorcinol-formaldehyde resins were carried out using slurries of 0.5 gram K-form resin to 1.0 ml volume of 101-AW Hanford simulant, ASTM water, and 0.5M nitric acid. (The 101-AW Hanford simulant has the composition of the caustic supernate in the 101-AW tank at Hanford. See Table I of reference 11 for its composition). K-form resin/ASTM water slurries containing varying resin to water ratios were also irradiated to investigate the effect of the resin to water ratio on hydrogen production.

Irradiations were performed using a Co-60 gamma ray source submerged beneath 25 ft. of water. All resins were irradiated in 45 ml stainless steel air-sealed vessels. Prior to sealing the systems, no attempts were made to purge air from the slurry or the void volume of the system. Two identical samples were simultaneously irradiated to provide a duplicate data set for each system studied. The vessels were connected via 4 ft. of 1/8 in. O.D. stainless steel tubing and 24 ft. of 1/8 inch O.D. nylon pressure tubing to pressure transducers outside the radiation field. The pressure transducers were equipped with digital readouts. The pressure was measured as a function of radiation dose. The dose rate was 6.6-6.9 E+05 rads/hr based on the standard potassium iodide (KI) dosimeter solution. (14,15) The temperature during the irradiations was ~ 30oC due to gamma heating. Gases were collected after the irradiation in evacuated 1 L glass bulbs that had been evacuated immediately prior to gas collection using a mechanical pump. Total collection of system gases in this manner enabled the analysis of a homogeneous gas sample. Prior gas sampling techniques involved typically 5 successive 5-10 cc aliquots that were sampled ~ 4 ft. above the radiolysis vessels using gas chromatography syringes. (11)

All the SRS reactor moderator resins were rinsed with 50 ml of ASTM Type-I water before and after irradiation. The resulting leachates were analyzed to compare the difference in water soluble products leached from both the unirradiated and irradiated resins. The 50 ml rinses were analyzed for pH, ion content (anion scan and NH4+), and both total inorganic carbon (TIC) and total organic carbon (TOC). The anion and cation rinsed resins were not irradiated in this study. The anion and cation resins were rinsed only to provide leachates for comparison to leachates derived from irradiated as received anion and cation resins.

Analytical Methods

Gas compositions were determined by gas chromatography (GC) using a previously described Varian Model 3400 gas chromatograph.11 This GC instrument was equipped with both a thermal conductivity detector and a flame ionization detector. The columns used were a Chromosorb-101 column and a molecular sieve 13X column with argon as the carrier gas. Various standard gases (Scott Specialty Gases) consisting of H2, O2, N2O, CO, CO2, and CH4 in the range of 0.1 to 20 vol% mixed with N2 were used for calibration. Trimethylamine was detected using gas chromatography / mass spectroscopy (GC/MS) instrumentation. The GC/MS consisted of a Hewlett Packard 5090 series II GC and a Hewlett Packard model 5971 quadrupole mass spectrometer. The concentrations of various anions and the ammonium cation were determined from the SRS reactor moderator resin rinses using a Dionex ion chromatography (IC) instrument. Anions were separated using a Dionex AS4/AG4 column with an equimolar

(0.002M) NaHCO3/Na2CO3 eluent and conductivity detection. Cations were separated using a Dionex CS3/CG3 column with a 0.034M aqueous HCl eluent and conductivity detection. Total inorganic and organic carbon measurements were made on the 50 ml rinses of the SRS reactor moderator resins using a previously described O. I. Co. Total Carbon analyzer.11

RESULTS AND DISCUSSION

KI Dosimetry

The standard KI dosimetry system is based on measurement of hydrogen and oxygen produced from the radiolysis of aerated 0.1mM aqueous KI solutions in the dose range of 5E+06 to 1E+10 rads. (14,15) This dosimeter was selected for use in the present study to provide 1) a dosimetry calibration system that was identical to the irradiation configuration used in all of the resin and resin/slurry irradiation experiments and 2) an accurate measure of the hydrogen yields for all the resin and resin/slurry systems studied. Figure 1 shows the average pressure changes produced when two identical aqueous 0.1mM KI solutions were simultaneously irradiated in air-sealed vessels. For the aqueous KI dosimetry system (see upper curve in Fig. 1) the pressure starts to linearly increase after a short time period during which the solution becomes saturated with the two radiolytically produced gases, hydrogen and oxygen. Table I shows the gas compositions determined for the KI dosimetry system. Rates of gas production were calculated in terms of G values or molecules produced per 100 eV of energy absorbed by the system. These G values for the KI dosimeter are shown in Table II. Also shown in Table II is the experimentally determined dose rate that is calculated from the total number of moles of gas evolved during the radiation time. (See equation #17 of reference 15). The average value of 6.93 E+05 rads/hr found for the KI system is comparable to the value of 6.87 E+05 rads/hr that was previously determined from the same irradiation source using thin film nylon dosimeters.16 As shown in Table II, our experimentally determined G values for hydrogen and oxygen production are in good agreement with the published values of Hart and Gordon.14

Gas Production From Radiolysis of SRS Moderator Resins

Figure 1 shows the average pressure changes produced when the SRS reactor moderator resins were irradiated in air-sealed vessels (see lower four curves). Two identical resin samples were irradiated simultaneously for each of the four resin systems shown. In all the air-sealed resin systems, the pressures initially decreased rapidly, followed by approximately linear increases over time. Both the anion and the mixed resin (damp) systems were pressurized at the end of the irradiation to ~ 2.0 psia above the starting pressures of 1 atmosphere, or ~ 14.67 psia. The mixed resin (dry) system final pressures were ~0.5 psia above the starting pressure and the cation system final pressures showed no significant change from the starting pressures. The initial pressure decreases observed are due to oxygen depletion from the air initially present in the sealed systems. Oxygen depletion from the irradiation of organic resins (5,6) inorganic zeolites, (17) and concrete (18) in sealed systems has been reported. Table III shows the gas compositions determined for the SRS reactor moderator resins. Hydrogen was produced in all the resin experiments. Small amounts of methane, carbon monoxide, and carbon dioxide were also produced. With the anion resin, we have also detected trimethylamine using GC/MS techniques. The irradiated anion resins had strong ammonia-like odors as well. Rates of gas production were calculated in terms of G values or molecules produced per 100 eV of energy absorbed by the system. These G values for the resins are shown in Table IV. For the mixed bed (damp), anion, mixed bed (dry), and cation resins, the G values for hydrogen are 0.29, 0.26, 0.13, and 0.12 molecules/100 eV, respectively. The precision of these results is 10% or better.

Radiation Effects on SRS Moderator Resin Properties

Radiation induced changes in mass, volume, and moisture content for the irradiated SRS reactor moderator resins are shown in Table V. Masses were obtained by weighing the radiolysis vessels containing the resin before and after irradiation. Volume changes were determined from measurements of the height of the resin columns contained in the radiolysis vessels before and after irradiation. As described above, the resin moisture contents were determined from heating ~ 1 gram samples in triplicate at 60C in a convection oven until no further mass change. The anion resin shows the largest changes in mass, volume, and moisture content. All irradiated resins decreased in volume. The mixed resin (damp) was the only resin irradiated in which an increase in moisture content was not observed.

Water-Soluble Products Leached From SRS Moderator Resins
The SRS reactor moderator resins were each rinsed with a single portion of 50 ml of ASTM Type-I water before and after irradiation. The resulting leachates were analyzed for soluble products leached from both the unirradiated and irradiated resins. Radiolytic production of dissolved ion and carbon in the irradiated resin rinses was calculated in terms of G values or molecules of dissolved species produced per 100 eV of energy absorbed by the 26-31 g samples irradiated (resin and some H2O). These G values for the water-soluble ion and carbon components are shown in Table VI.
The ion concentrations in the unirradiated and irradiated rinses from the mixed resins (damp) were all < 8E-05M. There were small amounts of organic carbon detected (3E-04M for unirradiated rinses and 9E-04M for irradiated rinses) and very little inorganic carbon detected (<1E-04M for both unirradiated and irradiated resin rinses). Similar results were found for the mixed resin (dry) rinses. These results indicate that water-soluble products released from the irradiated mixed resins are insignificant relative to water-soluble products released from the individual irradiated anion or cation resins discussed below.
The ion concentrations in the unirradiated and irradiated rinses from the anion resins were all insignificant at concentrations < 8E-05M, except for the nitrite and ammonium ions which were detected in the irradiated resin rinses at concentrations of 2E-04M and 6E-04M, respectively. These two ions were not detected in the unirradiated anion resin rinses. Three prominent chromatographic peaks at larger retention times than the ammonium ion chromatographic peak appeared in the ion chromatograph of the irradiated anion resin leachate. Comparison of these peaks with the elution pattern of substituted amines indicates that the components are most likely due to methyl-, dimethyl-, and trimethylamine dissolved in the leachate. A very large increase in the soluble organic carbon content of the anion resin rinses due to radiation was measured. The soluble carbon G values shown in Table VI were calculated using the formula weight of carbon = 12. The large amounts of dissolved organic carbon, ~ 0.2M, in the irradiated anion resin leachates are consistent with the IC data discussed above, i.e. the presence of dissolved substituted amines. Similar production of water-soluble aliphatic amines has been reported by Hall and Streat from the destruction of strong-base functional groups during the radiation-induced decomposition of aqueous slurries of organic anion exchange resins.19 The irradiated anion resin leachate also contained strong ammonia-like odors similar to the odors from the actual irradiated anion resin. The unirradiated anion resin leachate was odorless.
The ion concentrations in the unirradiated and irradiated rinses from the cation resins were all insignificant at concentrations < 8E-05M, except for the formate and sulfate ions. Formate and sulfate ions were detected in the irradiated resin rinses at concentrations of 9E-04M and 5E-02M, respectively. Both formate and sulfate ion concentrations were 3E-04M in the unirradiated cation resin rinses. Similar sulfate production from radiolytic degradation of organic cation exchange resin has been reported. (1) Large amounts of dissolved organic carbon in both the unirradiated cation resin leachate (0.019M) and the irradiated cation resin leachate (0.017M) were detected. Both the unirradiated and the irradiated cation resin leachates were noticeably yellow in color. We suggest that this color is due to soluble organic components leached from the unirradiated cation resin and soluble organic degradation products leached from the irradiated cation resin. All other leachates from the mixed resin and anion resin were colorless.
Calculation of the Shortest Time to Generate a Flammable H2 Air Mixture
During storage of the spent reactor moderator resins, H2 from radiolysis can form a flammable gas mixture in the vapor space above the resins if this space is not ventilated. Calculations involving radiolytic hydrogen generation in sealed radioactive waste containers have been reported. (7,8) From the equations given below, (7) it is evident that the times necessary to create flammable mixtures in sealed storage containers are directly proportional to the free volume in the container and inversely proportional to both the amount of resin present and the dose rate to the resin from the radionuclides sorbed on it.
Eq. (1)
with:
Partial pressure H2 for flame = 0.59 psi (based on LFL of 4% H2 in air)
Eq. (2)

with:

G(H2) = # molecules of H2 produced per 100 ev absorbed
N = Avogadro's number, 6.02E+23
dE/dt = total radiation dose rate, ev/hr
R = the gas constant, 0.043 psi ft3/mole oK
T = temperature, oK
V = volume in gas phase or free volume, ft3

Spent ion exchange resins from SRS reactor moderator purification activities are stored at SRS in ~40 ft3 stainless steel containers. Each container of dimensions ~4 ft height by ~3 ft diameter contains ~30 ft3 of resin and ~10 ft3 of free volume. Radiation measurements on the outer surface of these containers containing actual spent mixed resin indicate dose rates in the range of 0.1 to 18 r/hr at surface contact. The deionizer resins in the containers had been drained a small amount and were in the deuterated form resulting from the processing of reactor moderator water in SRS reactors. The radiation is due predominately to C14, H3, Sr90, and Cs137, all of which have half-lives
> 10 yrs, that are sorbed on the spent mixed resins.20 Using the maximum measured dose rate of 18 r/hr it has been estimated that the 30 ft3 of spent resin could have a maximum of ~4 W of activity . The total radiation dose rate for the 30 ft3 of spent resin is then calculated to be (4W x 2.25E+22 ev/hrW) = 9E+22 ev/hr. Using the measured G value for hydrogen production for the damp mixed bed resin (47 wt% water) of 0.3 molecules per 100 ev and a free volume of 10 ft3, the minimum time to reach a flammable mixture of hydrogen in air inside a sealed container at a temperature of 25oC is calculated from equation 1 above to be ~1,026 days or ~2.8 years.

Resorcinol-Formaldehyde Resins

Potassium form resorcinol-formaldehyde resins were irradiated in 101-AW Hanford simulant, ASTM water, and 0.5M nitric acid using 0.5 gram to 1.0 ml slurries. The average pressure differences produced when two identical slurries of resin were simultaneously irradiated in sealed vessels are shown in Fig. 2. Compositions of the gases produced in each system are given in Table VII. The gas compositions shown in Table VII of the present study were determined from the total collection of the gases present in each irradiated system. Table VIII shows the G values for the various gases produced. The G values shown for the resin in 101-A Hanford simulant slurries have been corrected for the dose absorbed by the resin-slurry, which has an experimentally determined density of 1.3 g/ml. Hydrogen is the predominant gas component produced for both the resin-101-AW Hanford simulant and the resin-water slurries. CO2, H2, and N2O are the main gas components determined for the resin-0.5M HNO3 slurries. It does appear that nitrogen may indeed be depleted in the resin/acid system. These gas compositions and G values presented in this study are to be compared to previous gas composition and G value results (see Tables V and VI of reference 11) that were determined from averages involving the collection of typically five successive 5 cc increments taken from a single sample using gas chromatography syringes. We suggest that while our present results are not significantly different from those previously reported for the same three resorcinol resin-slurry systems, both the accuracy and precision (typically 10% or better) with which the present results were determined is much better due to the revised gas sampling methods.

Gas Production from Different Resorcinol Resin/Water Ratios

Potassium form resorcinol-formaldehyde resins were irradiated in ASTM water at varying resin to water ratios. Additional systems consisting of 1) resin stored under ambient conditions (~20 wt% water), 2) dried resin (~0 wt% water), and 3) ASTM water containing no resin were also irradiated in this study. The change in the average system pressures as a function of absorbed dose for each set of two identical samples irradiated simultaneously are shown in Fig. 3. The systems containing slurry-like mixtures, i.e. the 1, 7, and 13.5 gram resin to 27 ml water ratios, show a linear increase of total pressure with absorbed dose (see upper three curves in Fig. 3). The two systems containing 1) 17 grams of dry resin (0 wt% water) and 2) 27 ml of ASTM water with no resin present, showed very little pressure change over the ~15 Mrad dose. The two highest resin fraction systems, the 18 gram resin / 27 ml water and the 22 gram moist resin (~20 wt% water), showed initial decreases in total system pressure indicative of oxygen depletion, followed by a gradual increase in total pressure. The measured G values for H2 from slurries containing different mass fractions of resin are shown in Table IX. The mass fraction of resin

is the approximate fraction of radiation absorbed by the resin in the resin/water slurries. These resin mass fractions were calculated using a measured starting resin moisture content of (20.80.1%). The moisture contents were determined from heating triplicate samples of the resin in a convection oven at 98oC until no further mass change was observed, typically for ~4 days.
Figure 4 shows the G(H2) values from Table IX plotted as a function of the mass fraction of resin present. With no resin present, only a small finite amount of hydrogen is produced in the radiolysis of a sealed system containing pure water.(21) The plot of G(H2) vs. mass fraction of resin in Fig. 4 indicates a sharp increase in G(H2) due to a small mass fraction (0.03 wt% resin) of resin present. The hydrogen yield then decreases linearly as the mass fraction of resin is increased. Similar results involving the decrease of hydrogen evolution with decreasing water content (or increasing resin fraction) have been reported for sulfonic acid type organic cation exchange resins.10 For all the gas compositions in which resin is present, H2 is the predominant gas produced with traces of CO, CO2, and CH4 also present.

CONCLUSIONS

Based on the data in this study we conclude the following:
1. More soluble products are leached from either irradiated anion resin (ammonia and substituted amines) or irradiated cation resin (sulfate) than are leached from irradiated mixed resin.
2. Irradiation of the anion resin or the mixed resin (damp) produces larger hydrogen yields and larger total system pressures than irradiation of either the cation resin or the mixed resin (dry).
3. The time to reach a flammable hydrogen-air mixture for storage of the mixed resin (damp), with G(H2) = 0.29 molecules/100 eV, in an actual sealed SRS spent resin storage container is calculated to be > 1,000 days.
4. Total gas production is highest for the resorcinol resin/0.5M HNO3 slurry and the hydrogen yield is highest for the resorcinol/ASTM Type-I water slurry.
5. Radiolytic hydrogen production decreases with increasing resin mass fraction for resorcinol resin/ASTM Type-I water mixtures.

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7-10

LLW SOLIDIFICTION WITH HIGH VOLUME REDUCTION BY INORGANIC BINDER

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ABSTRACT

Nuclear power stations, spent fuel reprocessing facilities, and radio-isotope (RI) laboratories generate wastes that have low-level radioactivity. The use of an alkali activated slag binder has been studied for the solidification and stabilization of these wastes.
The slag effectively formed waste products having good physical properties with high waste loading. Moreover, the results of this study suggest slag has the ability to become a common inorganic binder for the solidification of various radioactive wastes.
This paper also describes the fixation of radionuclides by the slag binder.

INTRODUCTION

Volume reduction of radioactive wastes and the stabilization of radionuclides are important items from the viewpoint of waste management. Currently, cementization, bituminization, plastic solidification(1) and vitrification(2) are the solidification technologies employed for the immobilization of radioactive wastes in nuclear fuel cycle facilities.
The main sources of radioactive wastes in reprocessing plants for spent fuels and nuclear power stations are low-level liquid wastes (sodium nitrate, sodium borate, boric acid, sodium sulfate, spent ion-exchange resins, incinerator ash, etc.) and dry wastes (polyethylene sheets, rubber paraphernalia, concrete debris, metal pipes, valves, etc.). By employing suitable solidification processes, these various wastes can effectively be treated and stabilized.

However, a simplified radwaste process system is needed whereby various wastes will be efficiently embedded and stabilized using a common binder solidification process. In this study, alkali activated blast furnace slag was selected as a candidate common binder because it is inorganic and is a low-cost material. Such slag is a by-product of the ferrous metal industry and is composed entirely of inorganic constituents. Several studies of alkali activation of slags and applications have been reported.(3,4) However, a detailed study addressing the use of alkali activated slag to solidify radioactive wastes has not previously been performed. This study investigated the application of slag for the solidification of various radioactive wastes. The results indicate that this technique offers high volume reduction and efficient immobilization of radioactivity.

CHARACTERISTICS OF ALKALI ACTIVATED SLAG

Hardening

A normal Japanese industrial slag was used in this study. Its composition was 39.0% SiO_2, 12.5% Al_2O_3, 6.0% MgO, 38.8% CaO, and 0.5% Fe_2O_3. Its density was 2.9 Kg/dm3. Alkalies were used to activate and harden the slag and a dispersant was added to improve the homogenization of the mix.

For normal hardening, the weight ratio of water to binder (W/B), where the binder represents the slag, dispersant, and additives, was in the range of 0.2. Hardening was promoted by increasing the curing temperature to 60-90C for about 24 hours. The effects of various chemical compounds on the hardening was investigated. The results suggested there was no retardation of hardening when a few percent by weight of any of the following chemicals were present: sodium nitrite, sodium carbonate, sodium dihydrogen phosphate, EDTA-2Na or hydrazine.

Properties

The measured compressive strengths of the hardened slag are 80 MPa with a 14-day cure at ambient temperature and 155 Mpa with a 24-hour cure at 90C. The measured bending strength of the hardened slag varies from 20-70 MPa. These properties are excellent, much better than most conventional cement binders.

SOLIDIFICATION EXPERIMENTS

This experimental study used several simulated radioactive wastes to examine waste solidification by the alkali activated slag. Pulverized sodium nitratewithout water of crystallization were prepared. Powder and bead type of mixtures of cation and anion exchange resins were dewatered to provide water contents in range of 45-60%. The specimen mixture was formed by combining the binder, simulated wastes, and water and mixing in a 3-dm3 agitator for 24 hours. The specimens were then formed by placing the mixture in molds followed by curing at ambient temperature or 60C for 24 hours.

RESULTS OF SOLIDIFICATION-SODIUM NITRATE

A large amount of low-level liquid waste containing sodium nitrate is generated in reprocessing facilities using the PUREX process. Typically, the bituminization process is used to embed the waste. Cement solidification is not commonly used because waste with a high loading of nitrate salts retards the hydration of cement. Recently, a process has been examined to drastically reduce the waste stream sources of sodium nitrate.(5)

However, this paper reports the solidification of powdered $NaNO_3$ by activated slag.

Compressive Strength and Swelling in Water Immersion

Table I shows the compressive strengths of slag products containing 8-60wt% sodium nitrate salts. These specimens were cured at ambient temperature. The strengths increased with curing times, even in the product with 60% waste loading.

To study the swelling of the product, specimens were immersed in ambient temperature water for 21 days. The weight of the specimens decreased slightly but no swelling was evident. These results were consistent for all amounts of nitrate salt loading waste.

The activated slag binder formed high-loading waste $NaNO_3$ products that had excellent physical properties.

Leachability of NO_3^-

The leachability of NO_3^- from sodium nitrate waste in the product is an important item in the evaluation of disposal safety because of the discharge limit of NO_3^- to the environment. The specimens used for the leaching test conformed to the Standard Leach Test procedure (SLT) specified by the American National Standard ANSI/ANS. 16.1-1986.

Figure 1 illustrates the relation between the cumulative fractions leached and the

leaching times for samples of products with 20, 40, and 60% nitrate salt. The testing time was about one month. From these data, the effective diffusivity, D (cm2/s), was calculated and the Leachability Index, LIX, of NO3- was obtained. As shown in Table I, the LIX of NO3- was about 7.5, which indicates that discharge of the nitrate anion is effectively controlled and limited, even when the waste loading of sodium nitrate is high.

Leachability of Tc-99

Liquid NaNO3 waste often contains Tc-99. Conventional cement binders are unable to effectively immobilize Tc-99. The leachability of Tc-99 from activated slag was examined by SLTs performed on specimens containing sodium nitrate and a radioactive Tc-99 tracer. The specimens had a weight loading of sodium nitrate of 40% and were cured at 60C for 24 hours. The test duration was 50 days.

In Table I above, the LIX of Tc-99 was 11. Thus, very low leachability was measured in a product with a high loading of NaNO3. As discussed later, reducing action by the slag (a low Eh oxidation- reduction voltage in the activated slag appeared) causes to the Tc-99 immobilization.

SPENT ION-EXCHANGE RESINS

In general, ion-exchange resins processed with conventional cement binders are difficult to solidify and to form into products having good properties. Therefore, advanced pretreatment methods(6) and the use of fiber cement(7) have been investigated.

The solidification of ion-exchange resins by activated slag was studied. The simulated wastes, mixtures of cation and anion resins, both bead and powdered type resins, were mixed and embedded in slag. The properties of products are presented in Table II. The mixture ratios of cation/anion resin were 1:1 and 2:1 for bead and powdered, respectively. Following 8 weeks of water immersion, the products exhibited no swelling and only negligible weight increases. The other characteristics were excellent. Photographs of samples taken after 4 and 8 weeks of immersion are shown in Fig. 2.

The study revealed products which can be formed without pretreatment that contain 50-60 Kg of wastes (dry based) per 200-dm3 drum. The limitation for the amount of loading waste depends mainly on the viscosity of the activated slag paste and the water content of the resin.

FIXATION OF RADIONUCLIDES

As described above, the alkali activated slag provided a high LIX for Tc-99 leachability in products containing NaNO3. To confirm the fixation of other radionuclides by the activated slag, the sorption capability has been quantitatively investigated. Radionuclides of Co, Ni, Cs, Sr, Am, and Tc were selected. The alkali activated slag was pulverized and added to an aqueous solution containing a tracer of each radionuclide. After standing for 7 days at ambient temperature, the slag powder was separated from the liquid by a filter. The radioactivity in the separated liquid was measured by a Ge-detector or a liquid scintillation counter.

The Kd value, a quantitative capability of sorption for a radionuclide, is defined as:

Kd (dm3/kg) = ([Initial radioactivity] - [residual in solution after 7 days])
/ [residual in solution] ([solution volume] / [slag weight])

Table III summarizes Kd values for each radionuclide. To compare the effect of activation of the slag, raw slag without activation was also used. For all radionuclides, the alkali activation provided more immobilization than raw slag. The alkali activated slag has high Kd values for all radionuclides except Tc-99. The Kd values of Cs and Sr are higher than those for conventional cement binders while the Kd values of Co, Ni, and Am are of the same order as for conventional cement.

The Kd value of Tc-99 was not satisfactory. But, a high Kd value was obtained with product formation under oxygen free conditions. In this condition, the Eh voltage measured was low for the alkali activated slag solution. It is shown that the activated slag initially has a reducing action. It is also suggested that the low leachability of Tc in the NaNO3 product, described above, is related to the slag's reducing action.

CONCLUSIONS

The application of alkali activated slag to the solidification of various radioactive wastes was studied. The activated slag embedded and formed a product with up to 60% loading of sodium nitrate that has low leachability of Tc-99 (LIX 11)

and no swelling. Ion-exchange resins also were effectively solidified by activated slag.
The sorption of radionuclides by the activated slag showed higher Kd values for Cs, Sr, Tc than for those associated with conventional cement binders.
This study suggests activated slag has the ability to be a common binder for solidification of various wastes and can provide a high reduction of waste volume.
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7-11
PROGRAM FOR FREE RELEASE OF ZONE 3 LIKELY CLEAN SOLID WASTE AND RECYCLABLES AT ONTARIO HYDRO'S DARLINGTON NUCLEAR GENERATING STATION (DNGS)
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ABSTRACT
An innovative solid waste management program developed at the Darlington Nuclear Generating Station has resulted in $1 million annual savings in avoided low level radioactive waste (LLW) processing and storage costs. The process is supported by a trained and dedicated staff and utilizes multiple checks and barriers to ensure complete segregation and thorough monitoring of waste prior to its release as clean. The program has been reviewed by the Atomic Energy Control Board (AECB), external auditors and interested members of the public. It was commissioned with $150,000 in equipment costs and no additional labour costs.
BACKGROUND
Darlington is a CANDU (Canada, Deuterium, Uranium) design, 4 unit station with a total generating capacity of 3680 MWe. The first Unit was placed in service January 1991 and the last in April 1993. It is important to note that at Ontario Hydro NGS's, contamination control is structured around the division of the Station into zones numbered 1 through 3. A clarification of the "Zone" concept follows:
Zone 1- a clean zone that is kept free of contamination. The Administration building and Powerhouse lunchroom are Zone 1 areas.
Zone 2- normally maintained free of contamination but has the potential for cross contamination from areas with radioactive systems.
Zone 3- contains radioactive systems and work areas. There are numerous Zone 3 areas widely spread throughout the four DNGS units.
ZONE 3 WASTE MANAGEMENT PROGRAM
Traditionally, DNGS has treated all solid waste produced in Zone 3 areas of the plant as contaminated. This resulted in large volumes of materials being shipped to the Ontario Hydro Bruce Nuclear Power Development (BNPD) Radioactive Waste Storage Site, located on the shores of Lake Huron 400 km from Darlington. It was recognized that a significant volume of waste collected in Zone 3 areas was potentially free of contamination. Therefore, DNGS and Scientific Ecology Group (SEG) developed a unique system to specifically address the reclassification of waste materials collected in Zone 3 areas as free of contamination. Figure 1 illustrates Low Level Radioactive Waste (LLW) volumes before and after implementation of the program.
Program Objectives

The main objectives of the Zone 3 Likely Clean Waste Management Program are:

To prevent the inappropriate release of detectable levels of activity to the municipal landfill by individually hand frisking each item for potential contamination using sensitive instrumentation and monitoring techniques that improve the probability of detection.

To prevent the inappropriate classification of clean (inactive) materials as contaminated (active) which can result in large volumes of clean materials being sent for storage to the BNPD RadWaste Site.

To minimize waste disposal at the Public landfill and comply with the Metro Toronto landfill ban criteria which prohibits the deposition of wood, cardboard, scrap metal or paper at the Public landfill.

Source Term

The first action prior to implementation of the program was to conduct a source term analysis to identify radionuclides that could potentially be present in the waste stream. Results of the analysis were used to ensure appropriate selection and calibration of instrumentation as well as define which areas of the Station should be excluded from the collection of materials eligible for monitoring for unconditional release. Routine sampling results are forwarded by the chemistry department, and a periodic characterization reviews will be conducted, to monitor for any changes to the source term. CANDU reactors contain large quantities of deuterium in the form of heavy water in the moderator and coolant which when bombarded by neutrons produces tritium. Tritium is the dominant component in some waste streams.

Waste Collection/Segregation

The original waste collection system in Zone 3 consisted of a three can set for segregation of incinerable, compactable and non-processable waste. All waste collected was treated as active. Waste collection stations were modified to provide for initial segregation of three new waste categories in Zone 3:

Likely Clean,

Presumed Active, and

Excluded Area.

For this reason, careful selection, surveillance and maintenance of waste stations is essential to the programs success. It is important to note that the "Likely Clean" waste stream may be directed to offsite landfill or recycling markets only if it is monitored and confirmed to be free of contamination.

Different coloured waste containers and bags are used to aid in identification of the waste streams during any stage of the process. Station staff are trained to ensure that only waste for which they know the radiological history, and believe to be free of contamination, is placed in the "Likely Clean" cans. Excluded area waste stations are few in number and are restricted to fuelling machine maintenance areas, hot particle areas, decontamination facilities and tritiated rooms. This waste requires a minimum of rehandling and is packed directly for shipment to BNPD RadWaste site.

Control of all 540 waste stations has been achieved by barcoding each station location and maintaining a master floor plan and database of their positions. Each bag of waste is barcoded and scanned during collection to register its point of origin, waste stream type and unique identification number.

Facility Layout/Equipment

A large room previously used for storage of active waste shipping packages was converted into the "Likely Clean" processing area. This room has a very low background, typically .05 mSv/hr or approximately 40 cpm on a pancake type frisker, and is ideal for monitoring waste for unconditional release. A second room was dedicated for handling "Presumed Active" and "Excluded Area" waste with provision for future sorting of "Presumed Active" waste up to 10 mSv/hr per bag.

The "Likely Clean" processing area was fitted with the following instrumentation and equipment:

Tritium monitor

Two waste bag monitors with barcode reader and data docking station

SEG ventilated sorting tables each with an NE Technology CM7A and traditional pancake contamination meter

Continuous air monitor

In the "Presumed Active" and "Excluded Area" waste processing area, a computer with barcode reader and data docking station was provided to track the active waste

stream until the third bag monitor is completed. A modified version of the bag monitor program is used to provide a consistent method of data collection, presentation and management. The active sorting table has been fitted with two HEPA ventilation systems to provide extra safety when monitoring the "Presumed Active" waste stream.

Barcoding System

A comprehensive barcoding and database management program has been added to the waste collection and bag monitor systems. The waste bag monitor database is accessible from the station computer network to allow daily review by the responsible system engineer. Each bag of waste processed is tracked as to its point of origin; waste classification; date and time of collection; tritium content; date and time of monitoring; weight; and total and specific activity.

This data system provides a method to:

Track waste volumes generated for each waste stream type at each Unit. This creates accountability among the Units, which function as separate businesses, and promotes further waste minimization activities. Accurate data allows each Unit to be charged for the waste it actually produces.

Identify the source of unexpected active waste and aid in investigation of the cause and resolution of the problem.

Determine waste volumes produced for a specific job by selecting waste stations associated with a particular work activity or outage.

Comply with the AECB requirement to investigate, and take action to prevent reoccurrences, of instances of contaminated Zone 3 "Likely Clean" waste exceeding specific limits for dose rate and tritium concentration.

Monitor program effectiveness and identify areas of the station where additional measures are required to further reduce waste production.

Continued improvements are being observed as the station staff become educated about the importance of waste minimization and their responsibilities associated with the waste management program. Figure 2 illustrates the effectiveness of the front end segregation step in the process.

Technical Surveillance/Quality Assurance

Technical Basis documents and instrumentation operation and calibration procedures have been developed to document instrument use in an unconditional release program. Instrumentation maintenance staff have been trained in maintenance and calibration of all associated instrumentation and are available to support the program on an as needed basis. Daily routine sheets are completed by waste processing staff to document instrument operational requirements specified by the unconditional release program. All documents are filed as permanent Quality Assurance records.

All materials determined to be clean, after monitoring at the ventilated sorting table, are rebagged into recyclable or landfill categories and directed to a second waste bag monitor for an independent QA check. These materials are barcoded to indicate their point of origin as the final step in the sorting process and are scanned into the QA bag monitor database prior to monitoring. If a bag of waste or recyclable fails this check, it is redirected back to the sorting table for remonitoring.

Waste Handling staff perform a variety of duties in the waste handling area and rotate their responsibilities periodically. The hand frisking of bag contents normally does not exceed 2 hrs per individual per day (Human Factors).

The database for waste bag tracking serves as an ongoing surveillance system to ensure any errors in the process are detected and taken care of in a timely manner. This database also provides useful information for report generation and further performance enhancement of the program.

Training of Station Staff

Formal training packages and examinations were developed and delivered to waste collection and processing staff to document their qualification for participating in a free release program. Only trained and qualified staff are permitted to collect or process waste, and operate any of the associated instrumentation. General traffic to the Waste Handling Facility is also restricted. A thorough understanding of all aspects of the process is essential to ensure that radioactive materials are not inadvertently deposited in a clean area. The Likely Clean Waste Sorting Room is secured and only accessible to trained and qualified staff.

A generic waste awareness seminar has been prepared and delivered to all Station staff to reinforce the correct use of the waste collection system. Tours of the

waste handling facility have also been provided to staff so that they may gain an appreciation of the importance of placing waste in the correct disposal containers. Ongoing training during radiation protection requalification courses and monthly safety meetings ensure that Station staff are kept informed. This method has provided us with excellent feedback on the program and further methods of improvement.

The success of this program can be attributed to the excellent cooperation and teamwork of many work groups within the Station.

INSTRUMENTATION SPECIFICS

Waste Bag Monitor

Waste bag monitors are used to monitor and track Zone 2 "Clean" and Zone 3 "Likely Clean" waste. The monitor uses seven large area plastic scintillation detectors, five for the cavity walls and one for each half of the front door assembly. The detection capability is 270 Bq/bag for Cs-137 with a 15 second count at typical background levels in the Likely Clean Waste Sorting Room. The monitor's Minimum Detectable Activity (MDA) is calculated and stored by the system software, based on user configurable parameters, each time a background update is completed. Efficiencies are determined with a calibration jig that simulates the typical density of a bag of waste and centres the source in the monitors cavity.

Each monitor includes a barcode reader to scan bag number and tritium labels and a docking station to download waste collection data into the monitor's computer. When a bag is monitored, the weight, tritium content and total and specific activity are appended to the collection data and stored in the monitor's database. Also stored is the current MDA value and whether an alarm initiated. One monitor is configured to track "Likely Clean" waste with an alarm setpoint that determines a bag's eligibility for sorting. A second monitor tracks waste that has been unconditionally released and is used as a QA check with the alarm set near the instruments MDA.

SEG Ventilated Sorting Table

"Likely Clean" waste is monitored and segregated on an SEG ventilated sorting table constructed from stainless steel and Lexan and customized for DNGS. The table includes shelves and mounting brackets for a HEPA vacuum and monitoring instrumentation. Three containers are provided for sorting waste which is confirmed to be free of contamination with an additional container for segregating active waste on the opposite side of the table. A 2000 cfm HEPA ventilation system draws air across the table through a prefilter mounted on a rear wall.

NE Technology CM7A

The CM7A is an alarming contamination meter capable of simultaneously detecting a & b/g events using a 100 cm2 gas proportional detector and pulse height discrimination. Additional features include:

- Background subtraction for the b/g channel
- High background alarm
- Excellent sensitivity
- Separate audible tones for a & b/g events
- Unit fault alarms

The CM7A uses a fixed counting time of 1 second allowing the instruments sensitivity to be calculated at a 3 cm/sec scanning speed. Typical MDA values for Cs-137 and Th-230 are provided in Fig. 3 (not provided). Problems associated with monitoring for a contamination are addressed by procedure and during training.

Table I

The high alarm and high background alarm setpoints are calculated and set based on a maximum allowable background of .1 mSv/hr. Typical background levels in the waste handling facility do not exceed .05 mSv/hr which decreases the false alarm rate to approximately one every hour. When not in use for surveys, the CM7A's function as sensitive area monitors providing an early indication of an increase in background levels. The instrument is source checked at the beginning and end of each shift with sources approximately equal in strength to the alarm setpoints of both channels. Source checks are documented and archived as QA records.

Traditional Contamination Meter (Ludlum 177)

A pancake probe is used when the larger size of the CM7A probe is impractical. For free release of materials, the user is instructed to scan the article listening for an increase in audible count rate. If an increase is detected, the probe is held stationary until the reading stabilizes. Any article with a reading greater than the minimum detectable count rate (MDCR) of the instrument for the current background is

considered contaminated. An MDCR versus background count rate chart is provided in the operating procedure and at the ventilated sorting table.

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TRACE ORGANIC REMOVAL BY PHOTOCHEMICAL OXIDATION

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ABSTRACT
Photochemical oxidation methods can be used for the destruction of dissolved organic contaminants in most process effluent streams, including those originating from the nuclear power sector. Evaporators can be used to separate organic contaminants from the aqueous phase if they are nonvolatile, but a large volume of secondary waste (concentrate) is produced, and the technology is capital-intensive.
This paper describes two different types of photochemical oxidation technologies used to destroy trace organics in wastewater containing oil and grease. The experimental results show that all of the EPA 624/625 (volatile/extractable organic) contaminants present are effectively removed. The first photochemical oxidation unit evaluated used a combination of ozone gas and ultraviolet (UV) light at the 254 nm wavelength. The effluent contained some organic acids, which are apparently not easily converted to carbon dioxide by the process. The presence of interfering carbonate ions (1,2) at concentrations of up to 1000 mg/L favoured operating the unit with water adjusted to a pH of between 6 to 8.
The other photochemical oxidation reactor investigated was a three-phase system employing a TiO_2 catalyst (with gaseous or liquid oxidant) and ultraviolet light, also at a wavelength of 254 nm. Liquid waste solutions containing oil and grease in concentrations of up to 150 mg/L were effectively treated, and there was no increase in pressure drop across the reactor. Two chemical oxidants were investigated (in addition to the dissolved oxygen in the water), including hydrogen peroxide and compressed air. It was necessary to employ hydrogen peroxide as an oxidant to achieve the method detection levels of the analytical equipment. The catalytic reactor was the more efficient based upon conversion of dissolved organic carbon to CO_2.

INTRODUCTION

Liquid waste streams at nuclear generating stations can become contaminated from a variety of different sources which include: organics released from lubricated pump seals, from hydraulic hose breaks, and organic-based liquids that get discharged into the sumps from hydraulically-operated valves and pistons. Heavy water can often become contaminated with various organic compounds, including oil and grease, and would be appropriate for treatment with the technologies described in this paper. About 2500 m^3 per year (0.66 million US gallons) of Chalk River Laboratories (CRL) waste, are volume reduced by a combination of continuous crossflow microfiltration, spiral wound reverse osmosis, and tubular reverse osmosis membrane technologies (3). The concentrate produced from the membrane systems is evaporated while simultaneously adding a bitumen emulsion in a thin-film evaporator to immobilize the radioactivity (Fig. 1). The distillate produced carries trace amounts of organics, including a variety of aromatics such as phenolics, naphthalenes, and substituted benzene derivatives.

CHEMICAL OXIDATION

Chemical oxidation is a process during which electrons are removed from a substance to increase its oxidation state. Because oxidation processes tend to be energy intensive and therefore costly, their breadth of application is limited. This is particularly the case for wastewater treatment. Because oxidation processes are fundamentally stoichiometric, proportionally greater amounts of oxidant are required for high levels of organic concentration (5).

Although the effluent quality from the evaporator meets all discharge criteria for radioactivity, there is some carryover of organics in the distillate. The source of these organics is primarily the distillation of lighter components in the emulsified bitumen from the evaporator used for immobilization of the aqueous waste concentrate (TFE-1). Chemical analyses conducted at CRL have shown that about 95% of the organics present in the condensate stream are volatilized from the bitumen, while the other 5% originating in the evaporator feed are entrained as liquid droplets with the vapor (4).

Comprehensive analyses of the distillate samples taken during the operation of the thin-film evaporator identified organics in the two phases which included an insoluble lighter phase and a dissolved soluble heavier phase (4). The insoluble phase represents the bulk of the organic present in the effluent stream. It contains greater than 96% of the total organic compounds found in the samples. The insoluble layer is composed mainly of aliphatic compounds (69%) with carbon chains ranging from 9 to 17 carbons. The remainder of the organics are aromatics.

A qualitative guide to the reactivity of selected organic compounds to oxidation is as follows (5):

High reactivity: phenols, aldehydes and aromatic amines;
Medium reactivity: alcohols, alkyl substituted aromatics, aliphatic ketones;
Low reactivity: halogenated hydrocarbons, saturated aliphatic compounds, and benzene.

EVALUATION OF TECHNOLOGIES

Previous efforts showed that charcoal absorption is effective for organic removal from the distillate. However, this technology generates a relatively large amount of secondary waste, which makes the process non-attractive from a materials handling and disposal perspective.

More recent work showed that pretreatment of the distillate stream with an oil coalescer removed greater than 96% of the oil and grease and saturated aliphatic compounds. Dissolved organics were extracted into the oil-rich lighter phase, which was periodically ejected from the coalescer using the principle of conductivity difference between the phases.

Two photochemical technologies that were investigated for organic removal downstream of the oil coalescer included: a UV-ozone reactor to oxidize soluble organics (primarily aromatics), and a titanium dioxide catalytic reactor. The UV/ozone batch reactor system was compared with the catalytic continuous reactor system, based upon the removal efficiency of trace organics (EPA 624/625 series), and percent conversion of organic carbon to carbon dioxide. Comparison between different processing conditions was made somewhat difficult due to the variability of the feed stock itself.

TITANIUM DIOXIDE CATALYTIC REACTOR

Each vessel of the reactor is comprised of an outer stainless-steel jacket, with an internal photocatalytic matrix, and a lamp that emits low-intensity ultra-violet (UV) light (at 254 nm wavelength), mounted coaxially within the jacket. Around the lamp is a sleeve of fiberglass mesh coated with TiO_2 to form the catalyst matrix. The vessels of the catalytic reactor are 1.6 m in length and 0.05 m in diameter. The reactor was set-up so that the four sets of three vessels (comprising one module-row) were in series with each other. Sample ports are located at the effluent point after each set of three vessels. The discharge pressure from the pump is set at about 585 kPa, to overcome the pressure drop across the unit. The unit occupies 1.65 m x 0.50 m x 0.80 m.

Low to intermediate-level radioactively-contaminated wastewater was passed through the reactor without any buildup of radiation field on the vessels. Based on this observation it is thought that the reactor could be used to treat organically-contaminated upgrader feed water.

The process was operated in both the single-pass and recycle modes to evaluate the effect of contact time on organics destruction efficiency. A variety of different experimental conditions were investigated including: the type and concentration of oxidant injected with the feed, pH of the feed, and the contact time. Oxidants that were tested included dissolved oxygen (already present in the water), hydrogen peroxide (at three different concentrations), compressed oxygen (at 550 kPa), and compressed air (at 550 kPa). The experiments were conducted at different pH conditions to determine if alkalinity had any significant impact on the oxidation rate.

UV/OZONE OXIDATION REACTOR

The UV/ozone reactor required distillate to be mixed with ozone in the presence of UV light (at 254 nm) in a reaction chamber. The dimensions of the reactor are about 0.97 m in height and 0.35 m in diameter. Ultraviolet lights extend into the reactor to about 0.1 m from the base. The three UV lights are immersed within the reactor at equidistant intervals along its top. There is a resident volume of about 45 L inside the reactor at all times.

The ozone generator received clean, dry production air from the air preparation unit and the feed air compressor system. This system delivered ozone gas to a maximum of 2% by weight (or 10 mg/L by concentration) in the carrier air stream. The reactor system was operated batchwise for these experiments. The feed tank was filled with about 100 L of process water, and then adjusted to the appropriate pH with mineral acid or sodium hydroxide. Water was continuously metered and removed from the reaction chamber at 22 L/min.

REMOVAL OF ORGANIC CARBON WITH CATALYTIC REACTOR

Since the application envisaged here was for treatment of wastewater produced from a variety of sources (3), there was considerable variation in the feed concentrations to the reactor. In some cases contaminants were below method detection limits (MDL) in the feed water. Hence, all figures shown in this paper have been normalised to reflect conversion efficiencies.

The pH of the wastewater feed was lowered from about 8 down to 4 to remove the carbonate and bicarbonate ions in the water (as carbon dioxide), which were both known to be present at a concentration of about 650 mg/L in the feed. The presence of bicarbonate is known to interfere in the oxidation of the organic contaminants, since there is an oxidation equilibrium with carbonate, and hence, it acts as a scavenger for the available oxidants (1). The reaction of hydroxyl ion (the active oxidant), OH, with bicarbonate to produce carbonate radical anion, (CO3-) , occurs with a rate constant of 1.5 x 107 M-1s-1 (Eq. 1). This carbonate then reacts with hydrogen peroxide via Eq. (2).

OH+ HCO3- H2O + [CO3-] (1)

[CO3-] + H2O2 HCO3- + [HO2] (2)

REMOVAL OF VOLATILE ORGANIC CARBON IN CATALYTIC REACTOR

Volatile organic carbon (VOC) refers to those organic contaminants included in the EPA 624 series. In most cases there do not exist specific discharge guidelines for these contaminants in Canada (6). However, the Ontario Provincial Water Quality Objectives (PWQOs) were used to evaluate how efficiently the water had been treated. Where there are no available PWQOs, the MDL of the analytical equipment was used to evaluate the effectiveness of the reactor.

Figure 2a,b,c,d shows the removal efficiency of the catalytic oxidation reactor for various volatile organics contaminants at a process pH of 4, and with a 3 minute contact time in the reactor. The first bar represents the feed concentration, the second bar after one module-row, the third bar after two module-rows,while the last bar represents the concentration after four module-rows. Toluene can be effectively reduced from 2 mg/L to 0.1 mg/L (which is within the PWQO), while m-xylene decreases from 10.7 mg/L to 3.38 mg/L, which is above the PWQO of 1 mg/L.

Figure 2b shows the results for an experiment in which dissolved oxygen was also used in the presence of the catalyst at a pH of 4. In comparison with the experiment in Fig. 2a, the liquid was passed from a drum into the reactor at a flowrate of 2 L/min for a period of 2 hours. The total contact time in the reactor was 18 minutes. The effluent from the reactor was recirculated back to the reactor. For each contaminant the second bar represents the fraction remaining after seven minutes of contact time in the reactor. The third bar represents the concentration left after 18 minutes in the reactor. In Fig. 2b the m-xylene can be reduced from an initial feed concentration of 5.9 mg/L to a final effluent concentration of 1.21 mg/L, representing a removal efficiency of about 80%.

Figure 2c illustrates the results of an experiment conducted with compressed air as an oxidant at a gauge pressure of 585 kPa. Each bar represents the passage of the liquid through an additional module-row. The water was continuously passed through the reactor at a flowrate of 2.7 L/min, and was not recirculated back to the feed drum. These results indicate that compressed air may be an effective oxidant for the removal of volatile organic contaminants from wastewaters.

Figure 2d shows the results of an experiment carried out with the hydrogen peroxide used as an oxidant. The flowrate of hydrogen peroxide into the reactor was

maintained at 1 L/h, and its concentration in the feedwater was 500 mg/L. The flowrate of the waste feedwater was maintained at 3 L/min, and the water was not recirculated through the reactor. Each successive bar represents the concentration after another module-row. The concentrations of toluene and m-xylene were below the method detection limits (MDL) of 0.1 mg/L, and hence the effluent quality met the existing PWQOs. Only o-xylene was above the MDL after once-through treatment with hydrogen peroxide.

Other experiments were conducted at higher hydrogen peroxide concentrations of 1 000 mg/L and 1 500 mg/L. However, these experiments showed that there was no further conversion of the organic contaminants. Hence, there was no significant advantage of using higher peroxide concentrations for the contaminant concentrations present in this feed stream. Considering that there may be several hundred mg/L of dissolved organic carbon in the feedwater, the 500 mg/L of hydrogen peroxide does not seem excessive.

REMOVAL OF EXTRACTABLE ORGANIC CARBON WITH TITANIUM DIOXIDE REACTOR

A variety of extractable organics (EPA 625 series of contaminants) were present in the distillate wastewater stream of which the most abundant included: phenol, naphthalene, 1-methyl- naphthalene, 2-methyl- naphthalene, and biphenyl. The concentration of total phenolics must be less than 20 mg/L to meet Federal Discharge criteria (6). There are no guidelines for the other contaminants. For phenol the PWQO is 1 mg/L; the PWQO is 2 mg/L for 1- and 2-methyl naphthalenes, and it is 0.2 mg/L for biphenyl.

Figure 3a,b,c,d shows the removal of EPA 625 contaminants without the benefit of any additional oxidant and the water was not recirculated. The contact time in the reactor was about 3 minutes. Each successive bar in the histogram represents the passage of the fluid after passage through another module-row. The concentrations of the contaminants are reduced marginally after treatment by the reactor, but are significantly above the PWQO guidelines. Only in the case of naphthalene is the concentration reduced to the MDL. About 50% removal of the contaminants is achieved on average in practice. The conditions employed were too mild for the effective removal of these organics.

Figure 3b shows the removal efficiency with an overall contact time of 18 minutes. For each contaminant the second bar represents the fraction remaining after seven minutes of contact time, while the third bar represents the concentration left after 18 minutes in the reactor. Destruction efficiencies in excess of 90% were observed for all contaminants. For the cases of the methyl-substituted naphthalene derivatives the final effluent value is at the MDL. For phenol, its concentration could not be reduced below 13.6 mg/L, indicating that more severe oxidizing conditions (an increase of contact time or a stronger oxidant) are required to remove this contaminant.

Figure 3c shows the removal of the extractable contaminants with the compressed air oxidant at an applied pressure of 550 kPa in a single pass through the reactor. For 'bis-phthalate', (PWQO = 0.6 mg/L), the final effluent concentration of 2.2 mg/L is only marginally above the PWQO. It is apparent that the catalytic reactor is capable of removing organic contaminants to very low values (usually to the MDL) in the presence of compressed air, even when challenged by significantly high concentrations of contaminants in the feedwater.

The removal of extractables with two minutes of contact time and hydrogen peroxide as an oxidant (at a flowrate of 1 L/h, and a concentration of 500 mg/L), is shown in Fig. 3d. The PWQOs for phenol, the methyl naphthalenes, and biphenyl can all easily be achieved. Greater than 95% of the feed concentration of the various contaminants are removed. Both naphthalene and biphenyl are oxidized to other products after the first module-row; for the other contaminants, including the methyl naphthalene derivatives, there is a gradual reduction of the contaminant across all four stages.

Evans (7) has noted that the end-product of the oxidation of naphthalene is phthalic acid, and it is improbable that further oxidation to carbon dioxide would take place. Gaul et al. (8) have found that the products of the oxidation of 1-methyl- and 2-methyl- naphthalene compounds include: 2-acetylbenzaldehyde, and (E) and (Z)-3-(2-acetylphenyl)-2-butenal, and (Z)-3-(2-acetylphenyl) propenal.

Although it was initially thought that the titanium dioxide catalyst might foul in the presence of large concentrations of oil and grease present in the feed stream, this was not observed. The catalytic reactor did not lose any throughput in spite of

oil and grease concentrations exceeding 150 mg/L, and no increase of pressure drop across the reactor was noted. This would suggest that an organics-laden waste stream containing significant concentrations of oil and grease can be treated. In all cases the oil and grease in the effluent from the reactor was reduced to less than 3 mg/L after the first module-row, which is in compliance with Federal regulations (6). The effluent from the reactor was clear, which was visual indication that the colour associated with organic material had been removed.

UV/OZONE REACTOR PERFORMANCE FOR ORGANICS REMOVAL

Although theory suggests that ozone in the presence of UV irradiation should be able to oxidize organic material to final products like carbon dioxide and water, ozone actually oxidizes an organic impurity until a refractory compound is formed that is stable with respect to ozone (9).

The evidence is overwhelming that the hydroxyl radical (OH) is the key intermediate when ozone decomposes in water. Rate constants for the reaction of OH with organic substances are commonly in the range of 109 to 1010 mol-1s-1 (1). In the course of mineralization of any organic contaminant, oxidation will logically involve a series of intermediates of progressively higher oxygen to carbon ratios on the way to carbon dioxide. Such intermediates make the conversion process multicomponent even if a single component exists in the feed. Hence, the total oxygen demand to mineralize the organics to carbon dioxide is usually in excess of stoichiometric requirements in a multicomponent system.

A significant concentration of straight chain aliphatic compounds were present in the distillate feed to the reactor. Typically the removal efficiencies of these species (from open characterization GC-MS analysis), averaged between 50 to 70% with the UV-ozone reactor. Higher carbon chains were relatively more difficult to remove from solution. It has been confirmed that the primary end-product of the batch UV-ozone oxidation are organic acids. These organic acids can be further oxidized to carbon dioxide if the pH of the liquid is maintained between 6 to 8.

Figure 4ab,c,d shows that all organic contaminants could effectively be reduced to the MDLs after a five minute contact time. It was necessary to maintain the pH below 8 to ensure that the organics were oxidized, and to minimize the scavenging of oxidants by bicarbonate (Eq. 1). For those organic contaminants having secondary and tertiary carbon-hydrogen bonds, such as straight and branched-chain hydrocarbons, the following chemical reactions are possible. Aldehydes are converted to carboxylic acids, primary and secondary alcohols to carboxylic acids and/or ketones, ethers to alcohols and esters, and hydrocarbons to alcohols and ketones (7).

REMOVAL OF EXTRACTABLE ORGANIC CONTAMINANTS WITH UV/OZONE REACTOR

Various extractable organic contaminants (EPA 625) were present in the distillate feed stream to the reactor. Figure 4b shows experimental results for which the ozone reactor was challenged with a highly organically-contaminated feed water, with an ozone inlet concentration of 10 mg/L and an initial feed pH of 6. All contaminants are reduced by at least 90% after 10 minutes of contact time with ozone. In spite of oil and grease concentrations approaching a few hundred mg/L in the feedwater, the effectiveness of the ultraviolet lights within the reactor did not deteriorate with processing time. No decrease of organic removal efficiency was observed from replicate experiments with visibly oily wastes.

Figure 4c shows the removal efficiencies of the UV/ozone reactor for various straight chain aliphatic organics ranging from tridecane (13 C atoms) to dodecane (20 C atoms). The results are presented for a pH of 6 and an ozone concentration of 10 mg/L. No discharge guidelines exist for these organic constituents but they can be effectively reduced in 5 minutes of contact time to within 20% of their initial values. After 10 minutes of contact time, less than 5% of the initial levels of these organics remain in solution, and the concentrations shown are at the MDLs.

CONCLUSIONS

The UV/ozone batch reactor, while effective in removing most priority contaminants (including phenolics to below 20 mg/L), was most effective at an operating pH of between 6 to 8. It was not as effective at removing oil and grease as the titanium catalytic reactor and the primary end-products of the oxidation were organic acids. The process was operated batchwise, which put it at a disadvantage with the catalytic reactor, which operated in a continuous fashion.

The titanium dioxide catalytic reactor was capable of reducing both EPA 624 (volatile) and EPA 625 (extractable) priority contaminants to below the method detection limits of the GC/MS analytical equipment. Phenolics were effectively

reduced to well below the Canadian Federal Discharge limit of 20 mg/L, and oil and grease was reduced to well below the 15 mg/L imposed guideline for Federal Establishments (6). Other aromatic compounds, including naphthalene and methyl-substituted naphthalene derivatives, were effectively reduced. Dissolved oxygen was sufficient to remove the colour associated with organics from the wastewater, even though not all of the organic carbon was removed. A concentration of 500 mg/L of hydrogen peroxide was found to be sufficient for the removal of the organics present in the evaporator distillate stream.

The dissolved organic carbon was not all converted to carbon dioxide in the catalytic reactor; some intermediate oxidation products were formed, which included some organic acids. Visible concentrations of oil and grease did not foul the catalyst or reduce the throughput of the system.

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NITRIC-PHOSPHORIC ACID OXIDATION OF SOLID AND LIQUID ORGANIC MATERIALS

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ABSTRACT

Nitric-phosphoric acid oxidation has been developed specifically to address issues that face the Savannah River Site, other defense-related facilities, private industry, and small-volume generators such as university and medical laboratories. Initially tested to destroy and decontaminate SRS solid, Pu-contaminated job-control waste (a heterogeneous mixture of plastics, cellulose, rubber, resins, metals, etc.), the technology has also exhibited potential for remediating hazardous and mixed-hazardous waste forms. The process is unique to Savannah River and offers a valuable alternative to other oxidation processes that require extreme temperatures and/or elevated pressures.

To address the broad categories of waste, many different organic compounds which represent a cross-section of the waste that must be treated have been successfully oxidized. Materials that have been quantitatively oxidized at atmospheric pressure below 180oC include neoprene, cellulose, EDTA, tributylphosphate, and nitromethane. More stable compounds such as benzoic acid, polyethylene, oils, and resins have been completely decomposed below 200oC and 10 psig.

The process uses dilute nitric acid in a concentrated phosphoric acid media as the main oxidant for the organic compounds. Phosphoric acid allow nitric acid to be retained in solution well above its normal boiling point. The reaction forms NOx vapors which can be reoxidized and recycled using air and water. The addition of 0.001M Pd(II) reduces CO generation to near 1% of the released carbon gases. The advantages of this process are that it is straightforward, uses relatively inexpensive reagents, operates at relatively low temperature and pressure, and produces final solutions which are compatible with stainless steel equipment. For organic wastes, all carbon, hydrogen, and nitrogen are converted to gaseous products. The process catalysts are either not consumed or can be regenerated. If interfaced with an acid recovery system which converts NOx back to nitric acid, the net oxidizer would be oxygen from air.

INTRODUCTION

The purpose of this program has been to demonstrate a nitric-phosphoric acid destruction technology which can treat a heterogeneous waste stream. This technology is being developed to convert hazardous liquid and solid organics to inorganic gases and salts while simultaneously performing a surface decontamination of the noncombustible items (the decontamination effects of this process are not discussed here). Plutonium-contaminated waste is an issue because of its stringent storage, shipping and disposal requirements.

The process envisioned will be configured to handle a wide range of solid Pu-contaminated waste of which 60% is combustible. The process will oxidize the combustibles (a mixture of 14% cellulose, 3% rubber, 64% plastics, 9% absorbed oil, 4% resins and sludges, and 6% miscellaneous organics) without requiring separation from the noncombustible portion. The system is being developed to operate below 200oC at moderate pressures (0-15 psig).

OXIDATION CHEMISTRY

The technology being developed is unique to SRTC and is the subject of WSRC invention disclosures. The process identified by SRTC is a wet-chemical process for completely oxidizing organic materials at moderate temperatures and pressures using common inorganic acids, nitric and phosphoric. It differs from other comparable technologies in that it does not require the use of extreme processing conditions or RCRA-listed hazardous metals to decompose the waste.

The process uses dilute nitric acid in a concentrated phosphoric acid media as the main oxidant for the organic compounds. Phosphoric acid allows oxidation at temperatures up to 200oC and is relatively non-corrosive on 304-L stainless steel near room temperature. (1) A simple process that uses oxygen from air or another readily available cheap oxidant as the net oxidizer would be relatively inexpensive per unit of waste consumed.

A wide range of organic compounds have been completely oxidized in varying concentrations of nitric acid, including nitromethane, cellulose, tartaric acid, tributylphosphate, EDTA, neoprene, benzoic acid, polyvinlychloride, and polyethylene. Trace Pd^{+2} converts most of the CO in the released carbon gases to CO_2. (1) The oxidation is usually complete in one to three hours for most organic materials. Aliphatic compounds, because of their relative stability, oxidize slower than non-aliphatic materials.

Although oxidation of most organic compounds by nitric acid is energetically favorable, the reaction rate is slow due to the stability of the carbon-hydrogen bond. (2,3) The oxidation of organic compounds is initiated by dissolved NO_2 and NO in solution and propagated by the production of organic radicals. For many types of oxygenated organic compounds the attack by NO_2 can be first order. (1) For aliphatic compounds, higher concentrations of NO_2 and NO are needed to obtain comparable oxidation rates. The initiating reactions form organic radicals which may then readily react with nitric and nitrous acids or NO_2. (1)

Once carbon-chain substitutions begin, hydrogen-carbon bonds on carbon atoms which are also bonded to oxygen are weakened. (1) This allows much quicker hydrogen abstraction and further oxidation. As the organic molecules gain more oxygen atoms the organic molecules become increasingly soluble in the nitric-phosphoric acid solution. Once in solution, the molecules are quickly oxidized to CO_2, CO, and water. If the original organic compound contains chlorine, hydrochloric acid will also be formed.

Acid recycle will be a key part of the process. As the reaction progresses, NO and NO_2 are released from solution and nitric acid is depleted. As a result, the NO and

NO2 need to be recovered as nitric acid in order to maintain the initiation of reactions. At the same time, HCl gas from the oxidation of chlorinated compounds must be allowed to pass through to be recovered in a subsequent step. Acid recovery units for converting NOx to nitric acid are a commercially available technology.

EXPERIMENTAL SETUP

All oxidation experiments were conducted using some variation of the following. A glass reaction vessel is set up with a mixture of nitric and phosphoric acids; palladium catalyst is also added to help convert CO to CO2. The custom-made glassware uses Teflon fittings and Viton o-rings to create gas seals. Pressure was measured using a Marshalltown Manufacturing compound gauge which measures from 30"Hg - 30 psig. The system temperature is monitored using a laboratory thermometer in some cases and a Luxtron Model 750 Fluoroptic temperature probe in others. Sulfamic acid (EM Science) removed NO2 gases from the reaction vessel off-gas stream so they did not interfere with downstream carbon dioxide measurements. Downstream of the sulfamic acid is drierite (W. A. Hammond Drierite Co.) which absorbs moisture from the gas stream. Ascarite-II (Thomas Scientific) absorbs CO2 and then is weighed to determine how much carbon dioxide has been released from the reaction. Weights are taken using a calibrated Mettler AE200 balance which is accurate to 0.0001g. Polyethylene samples were taken from Nalgene bottles. Trimsol is a product of Master Chemical Company.

OXIDATION OF DIFFERENT COMPOUNDS

Early experiments demonstrated the application of nitric-phophoric oxidation for various organic materials, and the importance of adding trace Pd catalyst to reduce CO generation. (1) These tests show that in 0.05-0.1M HNO3 in H3PO4 at 120-160oC and atmospheric pressure that many types of compounds can be quantitatively destroyed. Compounds completely oxidized, within experimental error, include cellulose, EDTA, tributylphosphate (TBP), nitromethane, and neoprene. Addition of 0.001M Pd(II) reduced the percent CO of the total CO and CO2 released by a factor of 15-25. For example, cellulose released 20% CO in the absence of Pd(II) and only 0.9% when Pd(II) was present; for TBP, 43% CO evolved without palladium compared to 1.3% with Pd(II), and the percentage of CO released during nitromethane oxidation was reduced from 60% down to 2.3%. Using similar oxidation conditions, it was found that the destruction of polyethylene was slow and incomplete.

In separate tests, high density polyethylene has been quantitatively oxidized to CO2 and CO using typical microwave sample digestion conditions (150-160oC, elevated pressures) in 1M HNO3/13.9M H3PO4 (Table I). (4) Addition of 0.001M Pd(II) to solution reduces CO production by a factor of three; higher concentrations will probably reduce the CO fraction even further. Recent work has shown that the primary advantage to sealed-vessel microwave dissolutions is the ability to obtain both high temperatures and acid concentrations at the same time. The rapid oxidation of aliphatic plastics such as polyethylene and PVC depends on obtaining higher temperatures and acid concentrations. Experimental work in this area is discussed later.

As discussed earlier, oxygenated compounds are more easily decomposed than aliphatic compounds, requiring one NO2. to cleave a bond instead of three. This is clearly depicted in Fig. 1 which shows the oxidation, under different conditions, of the primary compounds in the target feed stream. Each sample of PVC, polyethylene, cellulose, and benzoic acid contained comparable levels of carbon. Cellulose oxidizes fastest because it is already partially oxygenated. The differences between PVC and polyethylene stem from different dissolution characteristics. Polyethylene dissolves faster than PVC and, therefore, experiences faster reaction rates due to the increased surface areas of its dissolved state. The nearly constant oxidation rate of PVC may be attributed to its slow dissolution.

The oxidation of benzoic acid, an aromatic compound, is relatively fast at atmospheric pressure; the high oxidation rate is not surprising because benzoic acid readily dissolves in the process liquid. Oxidation rates of ion exchange resins, a component of the target waste stream which contains styrene, are expected to be comparable to plastics and not benzoic acid. However, the benzoic acid oxidation does demonstrate that aromatic compounds will be completely oxidized once they dissolve.

PARAMETRIC STUDIES

As is evident from Fig. 1 and the fact that plastics account for about 40% of SRS solid transuranic (TRU) waste, the rate-limiting step in the process is oxidation of

plastics. Because of this, a parametric study was run with a water-soluble oil (Trimsol) to better understand the effects of temperature and acid concentration on soluble aliphatic compounds. Trimsol simulates plastics once they dissolve; it is also the same oil used in machining operations throughout the DOE Complex, particularly Rocky Flats.

Although it is known that reaction rate increases with both temperature and acid concentration, our experiments quantify those effects. Figure 2 shows the importance of temperature. Calculated oxidation rates for runs at 120, 135, 150, and 165oC, respectively, are 0.7, 1.8, 3.3, and 5.5 mL/(hr.liter of solution). It is interesting that the data suggests that the reactions at 120, 135, and 150oC probably do not go to completion, which is in general agreement with what was reported by Seminov. (2) Follow-up studies revealed the role of nitric acid concentration at 165oC. The corresponding oxidation rates for 0.1, 0.5, and 1.0M nitric acid in concentrated phosphoric acid are 1.4, 5.0, and 5.7 mL/(hr.liter of solution).

A subsequent test was run to determine which parameter plays a greater role, temperature or acid concentration. In this experiment, starting solutions were made at 155, 170, and 185oC which had the maximum soluble nitric acid concentration; solubility is a function of temperature. The respective nitric acid concentrations in concentrated phosphoric acid (determined by ion chromatography measuring the nitrate ion) were 0.148, 0.0645, and 0.0195 g/mL. The data has been plotted in Fig. 3.

The initial oxidation rate for each experiment is approximately 35 mL/(hr.liter of solution). The results were unexpected as each reaction exhibits essentially the same initial reaction rate independent of the acid concentration-temperature combination. However, the oxidation characteristics of more stable compounds is apparent as the reaction at 155oC stops before completion and the reaction at 170oC achieves complete oxidation at a much slower rate than 185oC. The slowing of the reaction at 185oC may be a function of acid depletion, but this has not been determined experimentally.

OXIDATION OF PLASTICS

The earliest experiments in the oxidation of plastics used sealed vessels and microwave heating. This approach was taken because microwave vessels offered high-pressure capabilities, and it was originally thought that microwaves might play a role in the oxidation. Making precise statements based on the early tests is difficult due to the inability to monitor temperature or pressure. Nonetheless, experiments run using microwaves and the same digestion conditions provided insight into the effects of nitric acid concentration on polyethylene dissolution (Table II). This set of data reflects quite well what was observed in comparable experiments using Trimsol. A later microwave test during a vendor demonstration suggests that the temperature for the tests represented in Table II was approximately 150-160oC and 15 psig (205.1 kPa).

The results observed with microwaves fostered tests using conventional heating to oxidize polyethylene. Runs at 175 and 190oC, 0-5 psig (101.5-136.0 kPa), and the maximum soluble concentration of nitric acid yielded the results of the bottom two curves in Fig. 4. Once again, as was observed with Trimsol in Fig. 3, the initial reaction rates are almost identical. It was also observed, consistent with Fig. 3, that the test at the higher temperature showed better oxidation of the more-stable, long-chain intermediates. The temperature effect for plastics is not as pronounced as that for the oil because the plastic is a solid and has a more limited surface area in contact with solution. Tests using the same reaction temperature and acid concentration while varying the system pressure seemed to have little effect on oxidation rates.

Because the parameter of pressure by itself had no observable impact on plastic oxidation, it was determined that the primary value of pressurized systems (as in the microwave) is that they permit both higher reaction temperatures and acid concentrations instead of forcing the selection of one or the other. The effect of having both higher temperatures and acid concentrations is clearly shown in the top curve for polyethylene in Fig. 4. Calculations using the graph yield oxidation rates at 175, 190, and 205oC of 0.036, 0.034, and 0.107 g/cm2.hr, respectively.

Based on this experiment, it is believed that optimum conditions for easily-oxidized or short- chain compounds such as cellulose involve lower temperatures and higher acid concentrations due to rapid reactions and corresponding rapid acid depletion.

Conversely, the optimum system for plastics, where the oxidation of long-chain intermediates is more important, will seek a safe way to maximize temperature and nitric acid concentration using elevated pressures. However, moderate oxidation rates can still be achieved using slightly lower temperatures and atmospheric pressure.

MATERIALS OF CONSTRUCTION

The fact that nitric-phosphoric acid attacks stable organic materials at elevated temperatures makes it likely that it will also aggressively corrode many metals. Prior corrosion data and information in the literature on the corrosivity of phosphoric acid and nitric acid provided hope that high-alloy metals would be satisfactory for construction of main processing equipment. (5) This, however, is not as substantial an issue as it first appears. Since the system will operate near atmospheric pressure, glass-lined and Teflon-lined vessels are suitable materials for the oxidation vessels. The use of glass-lined and Teflon-lined vessels helps keeps the capital cost of equipment and replacements low. All other equipment which handles lower-temperature processing can be constructed from less expensive materials such as 304L or 316L stainless steel or Hastelloy C-276.

SUMMARY

Nitric-phosphoric acid-air oxidation has been developed to address the treatment of heterogeneous solid waste. This technology aims to destroy or decontaminate Pu-contaminated job control waste, a heterogeneous mixture of plastics, cellulosics, rubber materials, and noncombustibles. Since the issue of contaminated organics is not unique to SRS Solid Waste Management, the goals of this program are also consistent with other issues at Savannah River, DOE facilities, DoD installations, commercial nuclear operations, hazardous waste generators in private industry, and small-volume generators such as university and medical laboratories. Based on current data, the technology has also exhibited potential for remediating hazardous liquids and solids.

To address this broad category of waste, many organic compounds have been quantitatively oxidized in nitric-phosphoric acid. These compounds represent a cross-section of waste that must be treated, and contain most types of chemical bonds to be encountered. Elevating the temperature to 200oC and the pressure to 15 psig significantly enhances oxidation rates, particularly for plastics, resins, and solid aromatic compounds. The use of even higher temperatures and pressures could eliminate the need to use a phosphoric acid medium, but is currently perceived as undesirable because of the impact on scale-up and safety.

The process is nearly ready for testing with larger-scale equipment (20-50 liters) using organic feeds with little or no metals. Additional developments in the areas of volatile organic compounds and mixed aqueous-organic streams, although not applicable to Solid Waste Management, could prove to be beneficial to SRS as well as other generators of hazardous or contaminated organic materials.

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7-15

HEAP LEACH FOR URANIUM-CONTAMINATED SOIL REMEDIATION: COUPLED EXPERIMENT AND MODELING FOR PROCESS OPTIMIZATION

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ABSTRACT
Heap leach of the Fernald Environmental Management Project (FEMP) uranium-contaminated soils has consistently resulted in removal efficiencies of about 85 percent in the laboratory by use of sodium-carbonate-bicarbonate leach solutions in column experiments. Preliminary optimization has been achieved by variation of leach solution concentration, carbonate-bicarbonate stoichiometric ratio, pH, and application rate. To improve hydraulic characteristics of the FEMP soils, a variety of soil preparation techniques were employed that included agglomeration with cement and/or water, and by addition of various amounts of sand. About 15 percent of the total uranium in the soil has so far not been amenable to carbonate heap leach.
In order to better understand this phenomenon and optimize efforts to increase the uranium removal efficiencies, a number of geochemical models have been investigated. The initial model suggests two distinct, and kinetically controlled, desorption processes are occurring during the leach process. The model suggests that longer carbonate-bicarbonate solution residence times in the heap results in lower leachate solution to soil ratios for equivalent uranium removal. The model does not indicate that uranium is removed from the FEMP soils by either colloidal material or by dissolution of solid uranium phases. Intimate coupling of experiments, detailed analytical characterization and modeling has provided the understanding necessary to apply results to scaled up environmental activities.
INTRODUCTION
The site of the Fernald Environmental Management Project (FEMP) is located about 18 miles northwest of Cincinnati, Ohio. The site was formerly known as the Feed Material Production Center (FMPC) and operated from 1953 to 1989 as a production site for uranium compounds for use in defense facilities. As a result of operations at the FMPC two to four million metric tons of uranium-contaminated soil at the FEMP warrant remediation.
Carbonate heap leach has been successfully employed for leach of uranium from low-grade uranium ore. (1) Heap leach of FEMP uranium-contaminated soils has consistently resulted in the removal of about 85 percent of the total uranium by use of sodium-carbonate-bicarbonate leach solutions in column experiments conducted in the laboratory. (2) The reader is referred to Turney, et al (3), for details of those experiments.
Preliminary optimization has been achieved by variation of leach solution concentration, carbonate/bicarbonate stoichiometric ratio, pH, and application rate. To improve hydraulic characteristics of the FEMP soils, a variety of soil preparation techniques were employed that included agglomeration with cement and/or water, and by addition of various amounts of sand. (4) About 15 percent of the total uranium in the soil has so far not been amenable to carbonate heap leach.
In the uranium mining and milling industry low grade uranium ore has been heaped up on impermeable pads and then leached with sodium carbonate/bicarbonate ($Na_2CO_3/NaHCO_3$) solutions. It has been proposed that uranium-contaminated soils can be heaped up and leached in the same manner with sodium carbonate/bicarbonate solutions. The experiments that were modeled for the purpose of this paper were FEMP soils contaminated with uranium as a result of air emissions at the FMPC. The soils have been characterized as having an initial total uranium concentration of about 1396 mg U per kg soil. The major uranium phases in the FEMP soil are autunite ($Ca(UO_2)_2(PO_4)_2 \cdot 8\text{-}12H_2O$ - 55 percent), uranium metaphosphate (UO_2HPO_3 - 34 percent), and uraninite (UO_2 - 5 percent). (5) Other species present in the FEMP soil likely include schoepite (UO_3H_2O) (6).
DISCUSSION OF AQUEOUS GEOCHEMICAL MODELING
Desorption Modeling
In order to better understand the leach phenomenon and optimize efforts to increase the uranium removal efficiencies, geochemical models have been defined and investigated. A desorption model developed at the Los Alamos National Laboratory (LANL) that assists in analysis of actinide geochemical systems was used to aid in the interpretation of the data collected from the FEMP soil carbonate leach tests. Use of this model calls for comparing the concentrations of the uranium in the leach solution vs. the concentration of the uranium in the soil at specific time steps. When plotting the concentration of the uranium in the effluent solution (on the

abscissa in mg U per L effluent) against the uranium concentration of the solid (on the ordinate in mg U per kg FEMP soil), two relationships emerge. Figure 1 and Fig. 2 show data from an experiment that employed a 9 cm diameter, 150 cm tall column containing 10 kg of FEMP soil that had 1 kg sand added to improve hydrological characteristics fo the leach process. Two distinct trends are readily observed in the Figs. These trends are not to be confused by the darkened data points in Fig. 1 that show the dissolution process at the beginning of the column experiment.
Fig. 1. Plot of concentration of uranium in effluent vs. FEMP soil showing first desorptive trend.
Fig. 2. Plot of concentration of uranium in effluent vs. FEMP soil showing second desorptive trend.
The first trend is shown in Fig. 1 by a line having a slope of -0.25 L effluent per kg FEMP soil (indicating a desorption phenomena) and a y-intercept of 464 mg U per kg FEMP soil. The second trend is shown on Fig. Y where the slope is -2.1 L effluent/kg FEMP soil and a y intercept of 332 mg U per kg FEMP soil. While the slopes and intercepts vary slightly from experiment to experiment, virtually all the experimental uranium leach data produces similar trends (column experiments ranged in size from 80 g to 210 kg of soil).
The first trend coincides with the time during which most of the uranium is leached from the soil near the beginning of the leach process. The slope indicates that if this trend were to predominant for the entire leach experiment that it would require 0.25 L of bicarbonate solution per kg of FEMP soil to leach the soil to a final concentration of 464 mg U per kg soil. This would account for about 67 percent of the total uranium in the FEMP soil.
The second trend occurs as uranium dissolution decreases slightly over a long period of time at the end of the leach process. The slope of this trend indicates a requirement of 2.1 L of bicarbonate solution per kg of FEMP soil to achieve a final concentration of 332 mg U per kg soil. This accounts for dissolution of 9 percent of the total uranium in the soil for a total predicted removal of 76 percent of the uranium in the soil.
The desorption model suggests that two distinct, and kinetically controlled, desorption processes are occurring during the carbonate leach process. It is not readily apparent if the two trends are occurring simultaneously, but it is presumed that if that is the case the first uranium desorption trend is masking the second trend. The model suggests that lengthened carbonate/bicarbonate solution residence times in the heap results in lower leachate solution/soil ratios for equivalent uranium removal. The model does not indicate that uranium is removed from the FEMP soils by either colloidal material or by dissolution of solid uranium phases.
Predictive Geochemical Modeling
A software package for geochemical modeling of aqueous systems, EQ3/6, has also been employed to provide understanding necessary to apply results of column experiments to scaled up environmental activities. The major components of EQ3/6 (7) include a speciation-solubility code that deals with the concepts of thermodynamic equilibrium, thermodynamic disequilibrium and reaction kinetics; a reaction path code which models water/rock interaction or fluid mixing in either a pure reaction progress mode or a time mode, a data file preprocessor, a supporting software library, and several supporting thermodynamic data files that contain both standard state and activity coefficient-related data.
The majority of the data files support the use of the Davies or B-dot equations for the activity coefficients, the other files support the use of Pitzer's equations. The code uses a hybrid Newton-Raphson technique to make thermodynamic calculations. Each data file corresponds to a general formalism for treating the activity coefficients of the aqueous species and contains the relevant activity coefficient data as well as standard state thermodynamic data. The data files used with EQ3/6 are; 1) the "com" (for composite) data file is specific to the extended DeBye-Hckel formalism and is a product of Lawrence Livermore National Laboratory, 2) the "alts" data file is based on the "com" data set but includes a data set on actinides, 3) the "sup" data file, a data base based on Johnson, Oelkers and Helgeson's (8) software package SUPCRT92, 4) the "nea" data file is based entirely on Grenthe et al (9) and a product of the Data Bank of Nuclear Energy Agency of the European Community, (5 the "hmw" data file based on Harvie, Moller and Weare (10), and (6 the Chemval ("chv") data file where Chemval is a code-comparison project involving various countries conducting nuclear waste disposal studies (11).

The results of a number of EQ3/6 runs is shown in TABLE I. Starting values of pH, uranium, sodium and carbonate/bicarbonate concentrations of the leach solution and effluent were contained in the input files. Several different data files, described above, were used to compare and validate the different output. While only one data point is examined in TABLE I, other runs indicated the same relationships between output file data.

TABLE I. EQ3/6 output for a single data point with several different data bases.

The EQ3/6 output shows that in every case more than 99 percent of the uranium in solution is in the uranyl tricarbonate complex ($UO_2(CO_3)_3^{4-}$). Except for the "chv" data file output, where uranium concentration must be input as uranyl (UO_2^{2+}) instead of $UO_2(CO_3)_3^{4-}$, the activity of UO_2^{2+} ~ 10-20 and U(IV) ~ 10-45. The activity of uranyl dicarbonate ($UO_2(CO_3)_2^{2-}$) is always two order of magnitude less than $UO_2(CO_3)_3^{4-}$.

The saturation states of a number of selected minerals (shown as log Q/K) indicate that all of the minerals shown are undersaturated by at least one order of magnitude. Autunite is undersaturated by more than 38 orders of magnitude. Schoepite is undersaturated by approximately seven orders of magnitude and uraninite by about 8 orders of magnitude.

It can be drawn from the EQ3/6 runs that the carbonate leach solution is not near close to saturation with any of the major uranium minerals that are present in the FEMP soil. It would seem that the first part of the desorptive process is not effected by kinetics, but rather by accessibility of the uranium in the soil by the carbonate leach solution. The desorptive trend shown in Fig. 1 occurs where the carbonate rich leach solution can easily access the uranium. The point of intersection between the trends shown in Figs. 1 and 2 is where the desorption process becomes kinetically controlled.

Post-leach analysis of the leached soil in the columns shows that while there is a reduction of uranium concentration at all locations within the column, there is a greater uranium concentration reduction at the top of the column rather than at the bottom of the column. Because there is not a solubility problem, i.e. precipitation of previously solubilized uranium complexes as shown by EQ3/6, it is concluded that kinetics is governing the dissolution of uranium at the bottom of the soil column.

CONCLUSIONS

Efficiency of remediation of uranium-contaminated soils by sodium-carbonate-bicarbonate leach is dependent largely on accessibility of the uranium in the soil to the leach solution, and to a lesser degree, kinetics. Modeling shows two desorptive trends occurring during the leach process; the first an uninhibited dissolution of the uranium, the second trend an indication of kinetic control of the dissolution process. A software package for geochemical modeling of aqueous systems further demonstrates that the leach solutions are not close to saturation with respect to any of the uranium minerals or complexes that are expected to be found in the uranium-contaminated soil. This further indicates sufficient leach solution is available for uranium complexation within the soil column, but during the second desorptive trend the uranium is either not physically available for complexation, and kinetics dominates the desorption process.

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WASTE MINIMIZATION OPPORTUNITIES AT A LOW-LEVEL RADIOACTIVE WASTE TREATMENT FACILITY

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ABSTRACT

SEG is committed to reducing the generation of waste at its processing facility, at the source, to recycle and reuse as many materials as possible, and to volume reduce any material that is designated as waste. SEG's objective is to eliminate as much waste as possible going to burial whether it is commercial, DOE, or its own secondary waste streams.

INTRODUCTION

Environmental regulations in the 1970s and 1980s were promulgated to rectify the historical pollution that contaminated the air, water, and soil. These regulations were implemented at great cost to society and industry, as illustrated by the billions of dollars that have been spent on environmental restoration.
Regulatory initiatives in the 1990s, however, reflect a forward-thinking approach to prevent future corrective actions through the initial avoidance of pollution. Avoidance or minimization of pollution at the source, prior to release to the environment, is the objective of these regulations.
SEG understands the impact of its own waste generation and the importance of a sound waste minimization program. With the rising costs of using disposal sites, coupled with other compelling economic and environmental reasons to support pollution prevention, SEG has implemented a program to avoid, reduce, or reuse the waste by-products of its processing operations.
SEG realizes there are other benefits to waste reduction. There is less liability, improved worker health, potential cost savings, and a positive environmental image presented to the company employees and to the community.

WASTE MINIMIZATION ORGANIZATION

SEG understands that management commitment is the cornerstone of a successful minimization program. And that commitment helps ensure that all levels of personnel understand the impacts of waste generation. Some of the effects of needless secondary waste generation are:

- Less profit because of high disposal costs for unnecessary waste production

Problems with regulatory compliance issues.
SEG upper management has expressed its waste minimization objectives through written program directives and allocation of resources dedicated to meeting those objectives.
WASTE MINIMIZATION OBJECTIVES
To minimize all wastes and prevent pollution, the plant enforces the following objectives:
Accept only wastes that can be treated and disposed of in a timely fashion, including customer and secondary wastes
Implement waste minimization procedures to reduce the generation of waste at the source
Sort and segregate wastes to achieve the optimum volume reduction and efficiency in our treatment systems
Recycle and reuse as many materials as possible
When secondary waste generation is unavoidable, our objective is to volume reduce this material to the maximum extent possible, again reducing the economic and environmental impact to the plant. These principal operational objectives will be managed with the necessary Quality Assurance and Environmental, Safety, and Health commitment to protect the worker, the public, and the environment.
WASTE MINIMIZATION ACTIVITIES AND TECHNOLOGIES
The theory underlying waste minimization is that it is far more logical to produce less waste rather than to develop extensive treatment schemes to ensure that waste is managed in an environmentally sound manner. The following are activities and technologies implemented at the SEG facilities to help minimize waste generation.
Waste Stream Segregation
When waste is generated, proper handling and containerization is essential.
If waste is suspect then it will go through the Green Is Clean process to determine whether or not it is contaminated. SEG's Green Is Clean program uses a low-Level Waste Analysis (LLWA) System employing HPGe gamma spectroscopy to verify that wastes meet free release requirements. Only waste deemed potentially clean is accepted for processing in this program. Waste is screened by a metal detector and pre-screened for contamination. All bags passing SEG's pre-screening are placed into cardboard drums for monitoring by the LLWA system. Waste with no detectable activity is shredded as clean waste and disposed of as SEG secondary waste in an industrial landfill. Samples of the shredded waste are analyzed using SEG's laboratory HPGe system to monitor 10 CFR 61 characterization. Wastes rejected from the Green is Clean process are treated as radioactive material and processed using one of SEG's other waste processing/volume reduction methods.
An active training program is in place to help employees realize the importance of waste minimization and implementation of waste minimization incentives.
In various operations, steps have been taken to eliminate waste generation or volume reduce the waste generated during the operations. The following are the specific operations and the implementation steps for minimization:
Incineration
Before incineration, waste is sorted so that incineration will be more efficient, will protect the incinerator from materials not utilized efficiently, and meet licensing standards.
Only certifiable incinerables are accepted from generators so that all materials will be incinerated efficiently to achieve a volume reduction factor of at least 200:1.
Waste manifest is carefully checked so that SEG does not accept waste that is non-processable.
Waste acceptance criteria is strictly communicated and enforced.
Waste processing procedures are analyzed to see if production can be improved. This is done by changing the feed composition or the flow rate into the incinerator. This is especially applicable when adding oil and water to the incinerator.
SEG installed combustion chamber burners to improve carbon burnout and to reduce the volume of hearth ash waste and ensure that it is non-hazardous.
SEG now burns waste oil instead of propane in the primary and secondary incinerator chamber burners to maintain proper operating temperatures.
SEG has further volume reduced the end product by installing a vitrification unit in the end of the incineration process. Ash can either be vitrified or ash can be compacted before sending to burial.

If ash is determined not to pass TCLP, instead of storing this waste, it is solidified so that it can be sent for disposal.

Scrubber blowdown solution is evaporated, dewatered, dried, or supercompacted instead of solidified to elimate volume increases in the waste stream.

When bags of incinerator additive are empty, they are surveyed through the "green is clean" program to be certain that they are not radioactive, and then appropriately disposed of. If they are found to be radioactive, then they are compacted or incinerated. If they are found not to be radioactive, then they are sent for independent verification and sent to the sanitary landfill.

Metal Recycling

SEG segregates and feeds materials into the metal melt so that the end product will certain to be recyclable.

SEG compacts the slag coming from the metal recycling process to volume reduce the secondary waste.

Decontamination

Decontaminate materials and tools as a first approach rather than declaring them a waste.

Other types of minimization techniques

Incinerate disposable protective clothing.

Minimize the number and size of contaminated areas at SEG.

Supercompact nonhazardous bag filter, prefilters, and HEPAs.

Substitute nonhazardous materials for cleaning and degreasing as well as eliminating the potential generation of mixed waste.

All possible waste sources are identified and controls, procedures, and training are instituted to ensure that contamination control is followed by minimization of spills. Spill response kits will be kept immediately adjacent to work areas to allow rapid response to a spill. Good housekeeping techniques are maintained so there is a clean local working environment.

Good housekeeping and storage management practices are always being upgraded to be certain that uncontaminated materials do not come in contact with contaminated. Inventory control of any new hazardous materials will be rigidly controlled to not allow any mixed waste generation.

SEG carefully considers material handling to ensure the potential danger of drum damage or incidents with drums is considered in all handling operation. First, SEG engineers identify the proper equipment to be used for safe handling. For example, examination will be performed to ensure that drums are in suitable condition for lifting prior to actually making the lifts. Positive lifting means will be employed to ensure potential loss of containment. Experienced operators will be used and continually trained to minimize potential problems associated with unfamiliarity with equipment of SEG detailed written procedures.

Any spills or leaks are promptly reported and cleanup actions initiated immediately by trained personnel. Measures are taken so that, in the case of an accident, no contamination of groundwater will occur.

EDUCATION AND TRAINING

The following are activities related to educating and training individuals at SEG related to waste minimization.

A pre-job briefing or checklist is completed before entering a contaminated area to do specific work so that unnecessary tools and materials do not enter the contamination zone. Workers are trained to perform special tasks with the use of mock-ups, classroom training, or personal instruction before entering the contamination zones.

Efforts are made to ensure workers understand their assignments before entering a contamination area, making them better prepared with the proper equipment and tools for the job.

Containments are used whenever possible to prevent the spread of contamination. Containments reduce labor associated with decontamination after repairs.

Only those materials that are known not to be RCRA regulated are ordered so that the generation of mixed waste is reduced.

Intensive training is provided to each health physics technician on proper waste management techniques.

Procedures are carefully written and followed so there is little chance for a spill or contamination of materials.

Training is provided to all radiation workers on proper techniques for waste

handling, segregation, decontamination, and minimization.

Environmental regulations are updated regularly to update procedures for handling and processing of waste.

Recycling Opportunities

The following are activities implemented at SEG to recycle materials instead of designating them as waste:

Use contaminated water in the incinerator instead of fresh water. A new water system has been implemented.

Recover melt metal from the sorting process in the metal recycling operation. Decontaminate materials and reuse instead of designating them as waste.

Substitute sulfur polymer cement (a recycled product) for an organic epoxy for the microencapsulation of hazardous baghouse flyash.

Involvement of all Employees

SEG has a bonus program that controls a portion of the amount of all SEG quarterly bonuses based upon the amount of waste that is generated. These goals and allocated amounts for specific tasks are identified to the employees and each is made aware of the importance of the goal to the individual's bonus. In addition, training efforts and good procedural steps, consistent to the operating requirements, are provided for each employee to follow. In addition, each employee is trained and qualified to perform certain tasks. Only those employees who have completed necessary training, education, or experience levels will be permitted to perform such tasks.

CONCLUSION

SEG personnel are committed to finding and implementing ways to reduce waste generation. This waste minimization program is changing with the introduction of new progressive waste minimization activities that are implemented regularly at SEG.

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COST-EFFECTIVENESS ANALYSIS OF THE SEAMIST MEMBRANE SYSTEM TECHNOLOGY

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ABSTRACT

SEAMIST is a new technology that consists of an airtight membrane liner that is pneumatically emplaced inside the borehole. The positive air pressure inside the liner maintains the integrity of the borehole structure. Sampling ports with attached tubing, absorbent collectors, or various in situ measuring devices can be fabricated into the liner and used for monitoring volatile organic compounds (VOCs), semivolatile organic compounds (SVOCs), pesticides, herbicides, polynuclear aromatic hydrocarbons, polychlorinated biphenyls, or radioactive substances. In addition, small instruments can be guided through the lined borehole and measurements taken inside at specified intervals.

The purpose of this study is to analyze the cost and performance effectiveness of this new technology. To do so, we constructed four hypothetical scenarios in which utilization of the SEAMIST system can address various needs of the Department of Energy's environmental remediation program. Two of the scenarios involve vertical boreholes (or vertical instrument configurations) and two involve horizontal boreholes (or horizontal instrument configurations). The four scenarios jointly address contamination by VOCs, SVOCs, various water-soluble toxic substances, and low-level radioactive waste. One of the scenarios involves towing an instrument through a borehole and taking measurements of moisture levels in the surrounding soil.

INTRODUCTION

The purpose of this report is to investigate and describe on the cost and performance effectiveness of the SEAMIST membrane and instrumentation emplacement technology by analyzing how it compares to existing methods designed to accomplish the same task or set of tasks. In this analysis, we attempt to describe SEAMIST in a sufficient number of different scenarios in order to adequately demonstrate the breadth of its capabilities. Each scenario emphasizes a different function or addresses a different need of the environmental remediation objectives of the Department of Energy. Because of the diversity of capabilities of this new technology, each scenario must use a different conventional technology as a baseline of comparison.

SEAMIST is an innovative technology with many attributes:

- SEAMIST is an innovative technology that can facilitate measurements of soil-borne contaminants in horizontal and vertical boreholes
- SEAMIST consists of an airtight membrane that is pneumatically emplaced inside the borehole along with any attached sampling or measuring equipment, e.g., sampling ports, absorbent collectors, in situ sensors
- SEAMIST can be used to facilitate characterization and monitoring for VOCs, SVOCs, pesticides, herbicides, PAHs, PCBs, radioactive substances, metals, and other soil- or water-borne contaminants
- SEAMIST can also be used as a platform from which to tow in situ instruments such as cameras, neutron logging tools, and sensors through the borehole to obtain real-time data
- SEAMIST can be installed permanently with grout, semi-permanently with sand, or on a non-permanent basis by using positive air pressure
- SEAMIST can be a substitute for conventional borehole casing, but can also perform some functions that have no simple baseline of comparison, e.g., it can be used in conjunction with absorbent wicking pads to obtain samples of pore fluid contaminants on a recurring basis
- The magnitude of the cost savings possible from using SEAMIST instead of conventional methods increases as the depth of the contamination increases and increases as the variety of contaminants at a site increases

ANALYSIS AND CAVEATS

This analysis to determine the cost effectiveness of using the innovative SEAMIST technology is performed within the context of five scenarios. Each scenario highlights a different characteristic or need for some realistic set of site conditions that the DOE may encounter. Scenario 1 consists of a deep VOC contaminant plume (about 100 ft) which must be characterized and then monitored. Scenario 2 also involves a VOC contaminant plume; however, this plume is very shallow. Scenario 3 involves contaminants which are not volatile, but which exist in the pore fluids of the soil (see bullets). Scenario 4 involves taking moisture measurements to detect leakage under a low-level radioactive land disposal pit. Finally, Scenario 5 represents a combination of the requirements of Scenarios 1 and 3; its purpose is to demonstrate that there is synergism and economies of scope possible which result in additional cost savings over and above those of the separate scenarios.

Successful use of the SEAMIST technology requires that the geology of the site be sufficiently stable so that the borehole does not collapse before the membrane is emplaced. Also, the borehole surface must not be so rocky or sharp that it will tear the membrane.

COST DRIVERS

The cost drivers for both the new and the selected baseline technologies depend on the specific scenario. For the deep VOCs of Scenario 1, the SEAMIST system consists of vapor monitoring ports fabricated into the membrane. The baseline in Scenario 1 was chosen to be vapor monitoring ports constructed in conventional PVC casing. Both systems have tubing that carries the local VOC vapors to the surface for sampling and analysis. The cost drivers for Scenario 1 are the cost of the membrane for SEAMIST and the cost of the labor to construct the port system for the baseline.

Scenario 2 deals with shallow VOC contamination. The SEAMIST technology in Scenario 2 requires horizontal boreholes, whereas the baseline in this scenario consists of shallow, implanted soil vapor probes. Again, the cost of the fabricated membrane is a major cost to the new technology; however, that cost is far exceeded by the cost to drill the required horizontal boreholes. In this scenario, the new technology is not more cost effective; it is far more economical for shallow VOC contamination to use a system of soil vapor probes. It is worth noting, however, that for situations in which the contamination is not accessible from directly above the surface (e.g., across waterways or under buildings) horizontal boreholes and SEAMIST may be the only alternative.

Scenario 3 involves obtaining pore fluid from the vadose zone to analyze for water-borne contaminants such as nonvolatile organic compounds, PCBs, PAHs, pesticides, metals, and even tritium. The SEAMIST setup consists of absorbent wicking pads attached to the side of the membrane which are in direct contact with the borehole soil. The baseline of comparison is a series of vertically stacked, analogously placed, pressure-vacuum suction lysimeters. It is important to note that in this scenario, sampling for contaminants differs between the two alternatives. The cost drivers in this scenario for the SEAMIST setup are the membrane and the

cost of drilling the boreholes. For the baseline, the principal costs are the lysimeters, the lysimeter installation, tubing, and in particular, the cost to sample the lysimeters for pore fluid.

Scenario 4 involves the use of a neutron logging tool to obtain measurements of soil moisture under a low-level radioactive land disposal pit. Four horizontal boreholes are drilled underneath the land disposal pit. A SEAMIST liner is everted in the borehole as the neutron logging tool is simultaneously towed through it while taking neutron attenuation measurements at 5-ft intervals. The baseline consists of permanently casing the boreholes with aluminum and similarly towing the logging tool using a pulley system. The cost of the conventional aluminum casing is far more than that of the SEAMIST liner.

The final scenario represents a combination of the sampling requirements of Scenarios 1 and 3. The purpose of Scenario 5 is to demonstrate the economies of scope that can be achieved with SEAMIST that are not possible with most conventional technologies. The combination of vapor sampling ports and absorbent wicking pads in one membrane produces additional savings over the sum of the two separate SEAMIST system costs. When the two sets of requirements are combined, the separate sampling systems can be combined into one membrane. Both sampling systems can not be combined for the baseline. This integration capability is by far the most advantageous cost and performance advantage of SEAMIST.

Cost Savings

The cost effectiveness of using this new technology was calculated for each scenario. The cost data used to calculate the cost effectiveness were based on actual costs or stated prices from vendors or on a combination of both. Therefore, no uncertainties were considered for these scenarios. The results are summarized in Table I. For detailed information regarding cost estimates, see Henriksen and Booth, 1993.

Overall, SEAMIST was shown to be more cost effective the deeper the contamination and the greater the variety of contaminant substances. SEAMIST can often be configured to perform tasks that require two different conventional technologies. This introduces economies of scope which can result in significant cost savings. SEAMIST is very easy to remove and to dispose of after use. This is in contrast to conventional casing, which can only be abandoned after costly procedures.

ONGOING DEVELOPMENTS

New, stronger fabric materials are continually being developed for use as SEAMIST membranes. Additional innovative uses for SEAMIST include functioning as a conduit liner or straddle packer. SEAMIST has also been used in obtaining gas permeability measurements, for fracture flow mapping, and to measure brine flow.

CONCLUSIONS

SEAMIST can save from 16% to 74% of the cost of using conventional technologies, depending on the application

SEAMIST can sometimes perform tasks for which there is no conventional analog

In contrast to expensive cased borehole abandonment procedures, discontinuing use of SEAMIST may consist of removing the sand or disconnecting the air flow and then backfilling (see Table II).

REFERENCES

A.D. Henriksen and S.R. Booth, "Cost-Effectiveness Analysis of the SEAMIST Membrane System Technology," Los Alamos National Laboratory document LA-UR-93-3750, October, 1993.

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REALIZATION OF COST AVOIDANCE TECHNIQUES THROUGH THE CONTROLLED AREA TRASH SEGREGATION

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ABSTRACT

The "Green-is-Clean" program was initiated by Waste Programs Management (WPM). This program was an effort to segregate office trash generated in the radiologically Controlled Area so that it could be disposed of as sanitary rather than as low-level radiologically contaminated waste. The controls for demonstrating "no rad added" for this waste are that the office paper that comes into the Controlled Area as clean

and is never stored or used in any area other than the office areas. The authorized offices are routinely monitored (hand held instruments and smears) by Radiological Assessment for contamination and personnel handing this paper go through the personal contamination monitors prior to leaving the Controlled Area.
This project began with a pilot program in April, 1993 in which Radiological Assessment radiologically surveyed 100% of all Controlled Area office trash collected in order to verify that the administrative controls were sufficient to demonstrate "no rad added" and allow release of this waste to a sanitary landfill. Controlled Area office personnel were not informed of the pilot program to provide a worst-case scenario. Based on the results of these surveys, a 10% random sampling plan was developed in August, 1993, as a continuing overcheck of the program. Currently, Porters collect trash generated from approved Controlled Area offices and package it in green translucent trash bags. These bags are placed next to the contaminated trash dumpsters where transportation collects them twice weekly. A Radiological Control Technician performs a visual inspection for prohibited items and surveys the outer surfaces of each trash bag prior to being released as clean to a sanitary landfill. The contents of 10% of each collection is surveyed 100% to provide confirmatory sampling results. A procedure, SOP 20-C-617, Collection and Disposal of Controlled Area Office Trash, along with a contingency plan for incidents when a contaminated item is detected has been developed. These documents provide guidance for collecting, packaging, surveying, transporting and disposing of non-contaminated Controlled Area office trash to a sanitary landfill.

INTRODUCTION

The Fernald Environmental Management Project (FEMP)is a Department of Energy (DOE) owned facility that produced high-quality uranium metals for military defense for nearly 40 years. Fernald suspended production of uranium metals in 1989 and formally ended production in 1991. The Fernald mission has changed from one of production to environmental restoration. Even though production days have ceased, the perception still exists that everything that enters the Radiologically Controlled Areas at the Fernald are automatically deemed Low Level Waste (LLW) and must be disposed of as such unless demonstrated otherwise through analysis or administrative controls. This perception has caused an enormous amount of non-contaminated waste to be disposed of as LLW at a substantial cost differential. Clean waste is prohibited from disposal at the Nevada Test Site (NTS), and its appearance within the compacted bales of contaminated trash is a potential violation of the (NVO-325) Waste Acceptance Criteria (WAC). In reviewing NTS audit results and potential cost savings, FERMCO initiated a program to segregate Controlled Area office trash from the contaminated trash waste stream.

TECHNICAL BASIS FOR FEMP UNRESTRICTED RELEASE

DOE Orders 5400.5, "Radiological Protection of the Public and Environment", and 5820.2A, "Radioactive Waste Management", establish requirements and guidelines for managing radiologically contaminated waste at DOE sites. Order 5820.2A stresses waste minimization and states "...Low Level Waste (LLW) shall be managed on a systematic basis using the most appropriate combination of waste generation, reduction, segregation, treatment and disposal practices..." Order 5400.5 establishes numerical limits and qualitative guidelines for determining the effectiveness of segregation efforts. The requirements of these orders have been incorporated into FERMCO procedures and are being used on a daily basis for the Controlled Area Trash Segregation Program to demonstrate "no rad added" through strict administrative controls. This allows for disposal or recycle of significant volumes for clean trash from FEMP Radiologically Controlled Areas.

IMPLEMENTATION OF THE CONTROLLED AREA TRASH SEGREGATION PROGRAM

The "Controlled Area Trash Segregation Program" was initiated by FERMCO in 1993. This program was developed in an effort to segregate office trash generated in the Radiologically Controlled Area in order to dispose of it as sanitary rather than as LLW. This program utilizes a combination of administrative controls and process knowledge to ensure the material meets the required release criteria prior to leaving the FEMP site. The administrative controls are threefold. First, Porter personnel responsible for collecting the Controlled Area office Wash and Radiological Control Technician personnel responsible for radiologically surveying the trash receive specific training in accordance to the approved procedure. Secondly, Radiologically Controlled Area office personnel receive informal program overviews describing the methodology used to segregate the office trash along with a

review of the list of prohibited items. Thirdly, green tinted Wash bag liners are placed in each Controlled Area office trash can to indicate to office personnel and the Porter workforce of the approved areas in the Controlled Area Trash Segregation Program.
Process knowledge is utilized to demonstrate "no rad added" for this waste. It is understood that office paper that comes into the Radiologically Controlled Area is noncontaminated and is never stored or utilized in any area other than the office areas. The authorized offices are routinely radiologically monitored (hand held instruments and smears) by Radiological Control Technician for contamination. Personnel handling this paper must pass through the personal contamination monitors prior to leaving the Controlled Area.
The "Controlled Area Trash Segregation Program" began with a pilot program in April, 1993. Radiological Control Technicians radiologically surveyed 100% of all Controlled Area office trash collected in order to verify that process knowledge and administrative controls were sufficient to demonstrate 'no tad added" and allow release of this waste to a sanitary landfill. Controlled Area office personnel were not informed of the pilot program to provide a worst-case scenario. Based on the results of these surveys, an acceptable sampling plan that meets established acceptable criteria was established.
The objective of a sampling plan is to obtain a representative sample from a population and use the results of this sample to make assumptions about the rest of the lot with a specified level of confidence. In order to identify an appropriate sampling plan, assumptions must be made. This includes determining an acceptable fraction defective and a desired probability such that a lot exceeding the acceptable fraction defective will be rejected. Statistics associated with the unrestricted release of material generally employ a 95% confidence level that a contaminated sample will be rejected. The same confidence level is used for this sampling plan.
An appropriate sample size was chosen to meet the acceptance criteria. A reasonable lot size was chosen that considers the practical aspects of lot holding times and segregation/storage requirements of the lot while awaiting sample results. The number of trash hags collected in a week is considered the lot from which the sample size is chosen. The actual number of items per bag varies considerably based on the bag's constituents. Past data has shown that this can vary anywhere from approximately 300 to 1000 items ber bag. Because of this volume, a random sample is collected and radiologically surveyed per week to ensure a minimum number is met to satisfy the 95% confidence level. The rest of the week's pickup is held until the sample population has been radiologically surveyed. If radiological contamination is detected in the sample, then the rest of the population associated with that sample is disposed of as radioactive waste. Further steps are taken to identify the cause of the finding and corrective action implemented to prevent recurrence prior to reinstating the area of origin into the program. A contingency plan was developed to address these actions.
A procedure has been issued to provide instructions to the Porter workforce, Site Transportation, and Radiological Control Technicians. The procedure includes steps for collecting, packaging, radiologically surveying, transporting and disposing of non-contaminated Controlled Area office, breakroom and restroom trash. Based on the areas collected during the pilot program, Radiological Compliance issued an approved listing of Controlled Area offices, breakrooms and restrooms. The trash cans located in these approved areas contain a green trash can liner. This administrative control provides an indication to office area occupants and Porter personnel of the approved areas in the Controlled Area Trash Segregation Program. The Porter workforce collect trash generated from these specific areas, package it in green tinted translucent trash bags and indicate the building or area of generation directly on each bag with a permanent marker. These bags are placed in specific control points identified throughout the Controlled Area. Site Transportation collects the green bags twice weekly. A Radiological Control Technician accompanies Transportation on the collections and performs a visual inspection for prohibited items, radiologically surveys the outer surfaces of each trash bag and randomly selects the number of bags to satisfy the sampling plan requirements. The contents of these bags are radiologically surveyed to provide confirmatory sampling results. If the contamination level meets the limits of DOE Order 5400.5, the lot is disposed of as sanitary waste in a sanitary landfill.

Since April 1993, the cost savings associated with this program has amounted to $140,000. The volume of trash segregated as non-radiological waste has amounted to nearly 30,000 cubic feet. By segregating the non-radiological waste, the FEMP not only establishes a firm commitment to send only LLW to the Nevada Test Site, but results in substantial cost avoidance.

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A PROBABILISTIC ANALYSIS OF DECOMMISSIONING COSTS

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ABSTRACT

The wide range of cost estimates associated with facility decommissioning has raised concerns among utilities, regulators, and communities regarding the financial adequacy of nuclear decommissioning trust funds. Evidence from the Nuclear Regulatory Commission and TLG Engineering cost codes suggests there is disagreement in assessing decommissioning costs. Low-level waste (LLW) disposal costs have been identified as the most important and uncertain component of the total decommissioning costs. Cost estimates from a variety of industry experts and stakeholders ware assessed to determine a reasonably accurate distribution of anticipated LLW disposal costs and associated cost drivers. This new cost estimation procedure improves the traditional point estimate methodology by providing trust fund managers and decision-makers with a distribution of total decommissioning costs. This study utilizes risk analysis and industry survey data to assess a range of decommissioning costs for a 3400 megawatt thermal pressurized water reactor. Results indicate that LLW disposal costs are highly uncertain, generating a total decommissioning cost range of $380 to $775 million (1993 US$). This cost range was generated using the new estimation procedure and can lead to a more effectively managed trust fund and safer decommissioning operations.

INTRODUCTION

Utility preparation and execution of safe and cost-effective nuclear power plant decommissioning must include sound engineering and improved cost estimating techniques. Since 1978, the Nuclear Regulatory Commission (NRC) has published several documents intended to provide utilities with guidance and requirements regarding decommissioning nuclear facilities and accruing the necessary funds for decommissioning. One such requirement is that utilities must develop and submit a decommissioning plan for the eventual shut-down of their nuclear power plants; this entails a discussion of the decommissioning cost estimate and its methodology, the decommissioning project scope, and the financial instruments and methodology for collecting decommissioning funds.

The minimal level of decommissioning funding, as outlined by NRC guidance, applies to the removal of all radioactivity contaminated material and termination of the utilities' license. Aside from the uncertainty of contaminated material removal, utilities may be faced with collecting additional funds to remove or retrofit other structures and return the site to greenfield conditions, in some instances. This has lead to varying decommissioning cost estimates and a debate regarding the impact of inadequate nuclear decommissioning trusts. The debate centers on two issues, uncertainty and inadequacy with regard to costs and trust funds.

Provided the NRC's scope of decommissioning, the primary source of uncertainty is the cost of disposing low-level radioactive waste. These costs are thought to be driven by current waste regulations, future regulatory uncertainty, disposal facility operations and capacity, and siting complications. A recent 1993 estimate states that total low-level waste disposal costs account for 20 to 50 percent of total decommissioning costs, and burial accounts for 50 to 90 percent of disposal. Table I examines other cost relationships.

Although current estimates show low-level waste disposal (composed of packaging, transportation, and burial) as the dominant cost category, changes in work scope may reveal that staff labor or spent fuel dispositioning are greeter. In either case, implications of an inadequate nuclear decommissioning trust are intergenerational inequities and potential safety concerns from financially distressed or defunct utilities.

Supplying utilities and regulators with an understanding of what drives low-level

waste costs, provides an ability to more accurately estimate nuclear decommissioning costs, thereby enhancing a utility's decisions and benefiting the rate-paying community. The objectives of this study are to develop a questionnaire that assesses future low-level waste disposal costs, assess a range of disposal costs from industry experts, modify a decommissioning cost estimating code to incorporate a reasonable range of disposal costs, and generate a distribution of total decommissioning costs for a Westinghouse designed pressurized water reactor (PWR) and facility. This methodology is applicable to all pressurized water reactors and associated facilities if site-specific data are included.

LITERATURE REVIEW

Technology, Safety, and Cost of Decommissioning a Reference Pressurized Water Reactor Power Station - NUREG/CR-0130 was the NRC's initial reevaluation policy document that focused on power plant decommissioning. This report, released in 1978, assisted the NRC and public utilities by evaluating the technical aspects of decommissioning a power reactor and describing three prescribed decommissioning strategies: Decon, Safestor, and Entomb.

NUREG/CR-0130 is particularly relevant to this report since it addresses the impact of increased shallow-land burial costs. A comparison of empirical evidence from NUREG/CR-0130 and NUREG/CR-5884, Revised Analysis of Decommissioning for the Reference Pressurized Water Reactor Power Station (1993), reveals a significant change in the weight and sensitivity of overall disposal costs and the total decommissioning cost estimate to burial costs. Since NUREG/CR-0130, the NRC and independent firms have published updated cost estimates and frameworks for estimating costs. Several of the publications attribute rising cost estimates to a lack of decommissioning experience, the multitude of reactor and facility designs, variations in labor estimates for working in radioactive areas, geographical variation, escalating burial charges, and increasing energy rates. In turn, some critics recommend higher contingencies to offset future energy, labor, and burial cost escalation.

The Atomic Industrial Forum (AIF) published Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates (3) in 1986; this cost estimating framework is the basis of this study because of its availability and widespread acceptance among utilities. The AIF framework emphasizes breaking the complete decommissioning project into hundreds of discrete tasks under the general project elements of 1) planning and engineering, 2) site preparation, 3) decommissioning operations and license termination, and 4) site restoration. This approach allows for easy manipulation, the use of risk analysis - when used in conjunction with computerized spreadsheets, and provides readily defensible cost estimates.

With regard to trust funds, Regulatory Guide 1.159 outlines two collection methods and several financial assurance instruments associated with trust fund investments. The accrual methods, external and internal, determine how decommissioning funds are managed. The external method uses investment fund managers and bankers to manage collected funds; the internal method allows the utility to manage collected funds through infrastructure and/or market investments while exercising fiduciary responsibilities throughout. The rationale for external trusts is supported by the benefits of independent money management (i.e., investment by bankers rather than investment by utilities), greater fund growth and inspection, and creditor protection. In general, external trust fund management is expected to better serve the ratepayer's interests than internal trust management (4). Many of the documents cited throughout this paper have contributed to improving cost estimation, but have not developed quantitative methods for addressing the uncertainty and risk associated with any part or the whole decommissioning cost estimating framework. The issue of burial cost uncertainty arises ultimately from pending waste disposal release criteria since waste volumes will be a function of the release criteria. This may explain the increase in disposal costs relative to total costs by 93 percent, when comparing 1978 and 1993 NRC decommissioning cost studies. The significant increase suggests that modeling cost uncertainty, identifying the components of waste disposal costs, and applying the resulting intuition to a thorough cost estimating methodology will supply utilities and public utility commissions with an effective list of options for improved and responsive decision-making. Therefore, this report focuses on the uncertainty of future burial charges by assessing a reasonable range of low-level waste disposal fees from industry, yielding a range of decommissioning cost when input to a site-specific

costing model.
INDUSTRY QUESTIONNAIRE AND COST DRIVERS
This study examines the impact of risk, in the form of varying low-level waste costs, on a pressurized water reactor decommissioning cost estimate. Using data from a Westinghouse designed 3400 megawatt-thermal PWR, TLG's proprietary Decommissioning Cost Exposures and Radwaste code (DECCER) and an ExcelTM add-in program @RISKTM, an array of burial and final decommissioning cost estimates will be generated. This study utilizes TLG's site-specific method because of its detailed nature, adaptability, and acceptance by utilities. A questionnaire, see Fig. 1, regarding future low-level waste costs was provided to many industry experts. Wuestionnaire resonses associated with base burial costs were incorporated into building-block approach documented by Atomic Industrial Forum (1986). The TLG cost sode specifies tim cost categories related to decontamination, waste removal, waste burial, and other costs, providing a single point estimate. By incorprating a representative range of full-burdened low-level waste costs with TLG's cost code, a probabalistic reange of total decommissioning costs will be generated.
Twenty-five qualified candidates were canvassed to participate in this survey. Participants were "qualified" based on two criteria: 1) the individual is highly knowledgeable of the nuclear decommissioning dilemma and 2) the individual or their organization typically responds to Nuclear Regulatory Commission decommissioning rules and policy. The first criterion sought candidates with published papers on aspects of decommissioning and radioactive waste management, extensive industry experience, a demonstrated interest in promoting the environment, and/or accepted decommissioning guidance or policy. The second criterion sought a broad and balanced perspective of future low-level waste costs. Public records were examined to identify nuclear industry policy respondents and the following groups were solicited for survey participation: Sixteen individuals accepted the questionnaire, and seven, with varying industry backgrounds, responded. Each participant was asked to reply with a minimum, most-likely, and maximum response -providing a range of current year class A LLW disposal costs. academia, industry consultants. environmental/consumer advocate groups, LLW compact organizations, public utility commissions. government agencies pertaining to nuclear waste issues, utilities, and environmental/waste management companies (underlines indicate responding groups).
The five survey questions were divided among two cost accounting categories, base and overhead costs. The base burial charge cost, referred to as base cost, includes the following: LLW facility operation costs (regional compact fees, maintenance and closure costs), political uncertainty costs, and additional regional compact fees for unaffiliated disposers (questions 1, 3, and 4). The overhead cost includes fees for handling oversized disposal containers and municipality taxes (questions 2 and 5). Individual cost curve distributions were constructed using simulation analysis and Latin Hypercube sampling. Cost curves, from the seven constructed distributions, were interpreted by plotting simulation results and sketching a representative base and overhead curve. The aggregated or full burdened cost curve data were input to the TLG decommissioning cost code to produce the PWR decommissioning cost range. The aggregated, full burdened cost results uncover a minimum, maximum, and mean value of $197, $1230, and $702 per cubic foot. This implies that future class A low-level waste burial costs may vary between $197 and $1230 per cubic foot, in 1993 dollars. Table II summarizes the characteristics of each cost curve.
Cost Drivers
Based on questionnaire responses and sensitivity analysis of decommissioning costs, several factors tend to increase LLW costs. The first three factors relate to questionnaire base cost components such as LLW facility operating costs, political and administrative uncertainties, and compact fees. These three base cost factors represent 93 percent of the full burdened class A LLW costs. Another cost element disposers will be subjected to is a site closure fee. This fee is analogous to a mine reclamation cost and is an area likely to see increases during the next five to fifteen years. The final two factors affecting waste costs are disposal facility site availability and waste volume. Facility availability remains uncertain as evidenced by siting complications in California, Texas, and the Central Compact. Waste volumes will be driven by regulations, reactor housekeeping, and waste minimization efforts. This suggests waste volumes may or may not explicitly increase costs.
THE PWR DECOMMISSIONING COST ESTIMATE

This study examines the impact of low-level waste disposal cost uncertainty on a pressurized water reactor decommissioning cost estimate and assumes a Decon decommissioning strategy with dry spent nuclear fuel storage conditions. To include waste disposal cost uncertainty, the TLG cost code was modified to include a range, rather than waste disposal cost point estimate, equipping utilities and regulators with a complete picture of disposal and total decommissioning cost ranges.

LLW end Decommissioning Cost Estimate Ranges

To determine the risk low-level waste disposal imposes on a decommissioning cost estimate, probabilistic analysis was employed using TLG's cost code and a LLW disposal cost range based on questionnaire data. DECCER's burial cost component was supplemented by US Ecology low-level waste disposal costs, with class A, B, and C waste volumes provided by TLG. Based on the LLW volumes assessed by TLG, DECCER class A LLW disposal base, overhead, and full burdened costs were measured and substituted with questionnaire and economic data. An example of questionnaire based full burdened waste class costs is supplied in Table III.

Simulation analysis provided the following minimum, expected, and maximum low-level waste disposal values shown in Table IV. Incorporating the LLW disposal ranges with DECCER's other nine cost categories generated a range of total decommissioning costs. The expected PWR decommissioning cost range is illustrated in Table IV. The range of total decommissioning costs, $382 to $775 million, indicates future total costs are highly uncertain due to LLW disposal costs. This confirms earlier suspicions that LLW burial and disposal costs are the single most uncertain variable affecting decommissioning project costs. Figure 2 conveys the extreme variation and outcome differences among the three cost ranges, given in cumulative function form; the questionnaire based range dominates nearly all model outcomes, as expected.

SUMMARY AND CONCLUSIONS

Several decommissioning cost estimating methodologies were reviewed for their thoroughness, adaptability, and consistency with Nuclear Regulatory Commission guidance. Evidence from these reviews confirms that various cost estimating approaches taken by utilities, regulators, and contractors produce a wide range of total cost estimates. This is a concern to utilities, ratepayers, and public utility commissions because total costs are the basis for collecting adequate nuclear decommissioning trust funds. Varying estimates are the result of uncertain decontamination technologies, exposure limits, waste release criteria, and disposal costs. The one complication, or concern, common to all estimates that poses the greatest uncertainty is future low-level waste burial costs. The burial cost is driven primarily by disposal facility capacity and decommissioning waste volumes. Data reported in this study confirms this waste disposal dilemma and suggests approaches to manage the uncertainty. Most notably is the USA of risk analysis to bound and explain the cost drivers affecting future disposal cost increases. A more complete understanding of the disposal dilemma will provide improved methods to package, ship, and bury LLW and enhance present strategies for accumulating and managing nuclear decommissioning trust funds. Low-level waste burial costs account for 20 to 50 percent of the total decommissioning cost estimate, and waste disposal costs account for 50 to 90 percent of the low-level waste burial costs. Clearly, waste disposal is the primary cost component of LLW burial costs and contributes significantly to decommissioning project costs. Evidence of low-level waste's importance and sensitivity to total costs prompted this study and its objectives to define a reasonable range of future low-level waste disposal costs and strategies to counteract or decrease the disposal cost's eventual influence on total costs.

The TLG cost code, DECCER, was reviewed and determined to be best suited for this decommissioning cost study. The DECCER methodology is well documented, it can be easily adapted for risk analysis, its results are accepted by the Nuclear Regulatory Commission, and it is utilized by most utilities. To supplement the DECCER disposal cost category and provide a range of decommissioning costs, a questionnaire was developed to identify future low-level waste disposal costs. The range of LLW disposal costs was constructed using questionnaire responses from industry experts and simulation analysis. Construction of a representative cost range represents a key development for cost estimators and trust fund managers by incorporating insight from industry, government, and academia to assist with bounding the low-level radioactive waste disposal dilemma. Seven participants from academia, government, and industry provided their assessment of low-level waste disposal facility operational costs, special surcharges, compact fees imposed on out-of-compact

disposers, other costs, and municipal taxes. The sum of the five questionnaire costs ranged from a low of $83 per cubic foot, to a high of $4,838 per cubic foot, suggesting that disposal costs are highly uncertain and the cost components driving those costs are misunderstood. This span of costs was interpreted to define a reasonable and representative range of base and overhead costs; the full burdened cost range is characterized by the following parameters: minimum, $197; mean, $702; maximum, $1,230 per cubic foot.

The questionnaire cost range was supplied to the TLG cost code, DECCER, to generate a PWR decommissioning cost estimate range. Reactor equipment and inventories are modeled after a Westinghouse designed pressurized water reactor, 3400 megawatts-thermal. PWR decommissioning costs vary between $382 and $775 million if future class A LLW disposal costs varied between $197 and $1,230 par cubic foot, indicating that LLW disposal cost uncertainty greatly impacts decommissioning and should be the focal point of future resources. The range of possible decommissioning costs decreases to less than $200 million when limiting the questionnaire cost distribution to 80 percent of the total PWR cost distribution. With respect to trust funds, the impact of collecting various trust fund levels based on questionnaire LLW cost data, Fig. 3 is provided. Figure 3 relates that if one decided to collect $400 million to decommission this study's reference PWR, the chances of decommissioning costs exceeding the trust fund are 97 percent. A more conservative individual may choose to collect $600 million, this would reduce the chance of decommissioning costs exceeding trust funds levels to just 26 percent. Risk analysis assisted with establishing these decommissioning project cost limits and identifying the true magnitude of cost variability. To assist with diffusing decommissioning complications, risk analysis and decision analysis techniques can be merged with traditional cost estimating methodologies, as performed by the natural resources and manufacturing industries, to provide public utility commissions and trust fund managers with a better picture of future decommissioning costs (5), (6). Industry knowledge, use, and acceptance of risk analysis and decision analysis techniques will lead to a clearer understanding and definition of the issues, cost drivers, and relationships affecting plant operations and decommissioning. This will provide support for innovative regulatory approaches, trust fund management methods, and decommissioning strategies.

Several equitable and efficient approaches must be developed to manage the disposal dilemma and understand the application of risk analysis. Approaches such as improved forecasting tools, regulator education, and community involvement will assist with decreasing future low-level waste disposal uncertainty. Forecasting tools must incorporate historical trends, new cost developments, and facility capacity constraints and availability dates to be effective. Utilities and contractors must introduce regulators to risk analysis theory and application, and emphasize its value, benefits, and limitations. Finally, host communities and contractors may adopt the siting strategy implemented by the Appalachian States Compact and Chem-Nuclear to minimize disposal facility siting and construction delays. These approaches allow utilities to effectively plan for reactor shut down and decommissioning, rather than reacting with crisis management measures. Regulators will benefit by understanding the utilities operating, shut down, and decommissioning strategy and bases, while assuring proper management and collection of adequate nuclear decommissioning trust funds. Finally, communities will benefit from equitable energy rates, safe decommissioning, and prudent disposal operations.

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WASTE MINIMIZATION MEASUREMENT AND PROGRESS REPORTING AT THE SAVANNAH RIVER SITE*

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ABSTRACT

Westinghouse Savannah River Company is implementing productivity improvement concepts into the Waste Minimization Program by focusing on the positive initiatives taken to reduce waste generation at the Savannah River Site. Previous performance measures, based only on waste generation rates, proved to be an ineffective metric for measuring performance and promoting continuous improvements within the Program. Impacts of mission changes and non-routine operations impeded development of baseline waste generation rates and often negated waste generation trending reports. A system was developed to quantify, document and track innovative activities that impact waste volume and radioactivity/toxicity reductions. This system coupled with Management-driven waste disposal avoidance goals is proving to be a powerful tool to promote waste minimization awareness and the implementation of waste reduction initiatives. Measurement of waste not generated, in addition to waste generated, increases the credibility of the Waste Minimization Program, improves sharing of success stories, and supports development of regulatory and management reports.

INTRODUCTION

The Savannah River Site (SRS) is a key Department of Energy (DOE) facility, focusing on national security work; environmental and waste management activities; and economic development and technology transfer initiatives. Owned by the DOE and operated under contract by the Westinghouse Savannah River Company (WSRC), the complex covers 310 square miles near Aiken, South Carolina, bordering the Savannah River.

Waste Minimization and Pollution Prevention (WMin/PP) at the Savannah River Site is a continuous improvement process. Recall the basic elements of the cyclical continuous improvement model: Baseline process; Identify and implement improvements; Define performance expectations; Measure and evaluate performance; Repeat the cycle. Obviously, the continuous improvement model breaks down if the measures and metrics chosen do not relate to performance goals. This paper addresses improvements the SRS has made in aligning measures with program WMIN/PP performance goals.

TRADITIONAL PERFORMANCE MEASURES

The performance objectives of the SRS WMin/PP program are to eliminate unnecessary waste generation and maximize the life of SRS waste storage and disposal facilities through cost-effective source reduction, recycling, and volume reduction initiatives. The traditional measure used at SRS, and across industry, to gauge WMin/PP progress is the volume of waste generated and placed in Treatment/Storage/Disposal (T/S/D) facilities. There are several reasons for the widespread use of this measure:

Convenience - The systems to track and measure waste generation were in place long before waste minimization became a focused initiative. Why develop new measures when we have one already?

Credibility - Waste volume is not a subjective measure. The reliability of waste volume data is rarely subject to dispute and cannot be manipulated.

SRS WMin/PP program planners have concluded that the total waste volume, when used exclusively as a WMin/PP measure, is a poor barometer of program effectiveness. As stated, our WMin/PP performance objective is to eliminate waste, yet our only measure is the waste we fail to eliminate. Bowling provides an analogy; the performance objective is to knock down as many pins as possible and the performance measure is the number of pins knocked down. If we approached bowling in the same manner as waste minimization, the metric would be the number of pins left standing, i.e., a measurement of failure.

There are instances where measuring the amount of waste generated can be an effective indicator of WMin/PP progress. Specifically, when baselines permit accurate predictions of waste generation. You're familiar with the widget example where waste per widget can be calculated and used as a baseline. Waste minimization success can be concluded by measuring waste per widget, before and after waste minimization initiatives. In the utility industry, waste per outage can be an effective measure. This can be further illustrated with the bowling example. Even if we measured the pins left standing, we could draw conclusions about how well our performance was because we have a fixed baseline of 300 pins per game.

NEW PERFORMANCE MEASURES NEEDED

The nature of waste generating activities at the SRS, and across the DOE Complex, is changing. Production missions have all but been replaced with environmental remediation and D&D missions. Yesterday's production facility has become today's waste. Volumes of this legacy waste, and secondary waste generated during cleanup, are predicted to spiral upwards. Budget uncertainty, land use issues, and evolving regulations make predictions of waste volumes, even on a short-term basis, extremely difficult. Utilizing total waste volume as a measure of WMin/PP effectiveness is fruitless; the measure bears little, if any, relationship to the process being measured. In the bowling analogy (last time, promise), our situation resembles a modified game where the number of pins available each frame or game changes unpredictably and we choose to measure our performance by counting pins we fail to knock down.

It should be intuitively obvious that if the objective of WMin/PP is to eliminate waste and pollution, then we should measure the amount of waste that is eliminated (WMin/PP success) versus the amount of waste generated (WMin/PP failure). In concept this is straightforward; the challenge is establishment of a data collection mechanism to support this measure. The SRS has deployed such a system to measure and quantify the impacts of WMin/PP initiatives.

IMPROVED MEASURE

A Pollution Prevention Activity Form (PPAF) (1) was developed to capture and quantify the impacts resulting from discrete WMin/PP actions. The Form is straightforward and requires little effort to complete. A copy of the PPAF, with instructions, is provided as Attachment 1 a, b, c. The intent of the PPAF is to document innovative pollution prevention activities being implemented at SRS. The PPAF helps fulfill regulatory and DOE reporting requirements for waste reduction activities, facilitates pollution prevention technology transfer, and provides a credible basis for WMin/PP cost savings calculations.

Use of the PPAF is being institutionalized at the SRS by mandating completion of the form, by the initiator of the WMin/PP action, in a Company level procedure. Instructions for Form completion are available and key personnel have been trained on proper completion. Further, the DOE has defined WSRC Cost Plus Award Fee (CPAF) goals for waste avoidance with the requirement that avoidance be documented on PPAFs. Since mid 1993, over 90 PPAFs have been submitted and verified to be accurate. Over 4770 cubic meters of solid low level radioactive waste avoidance has been documented representing a savings of $2M in waste disposal costs alone. In addition, almost 60,000 kg of hazardous and low level mixed waste was avoided due to WMin/PP actions. Use of the PPAF at SRS has been successful due to this bottom-up (procedures and training) and top-down (company Award Fee goals) approach.

To illustrate the value of collecting waste avoidance data, refer to Fig. 1 which is a measure of total SRS solid waste (sanitary, hazardous, radioactive). The reductions in waste are dramatic, however, going back to the previous discussion: are these reductions due to WMin/PP or are they simply a result of mission downturns?

Figure 2 shows the same waste generation data as Fig. 1, plus an overlay of waste avoidance data extracted from PPAFs. This improved WMin/PP program measure clearly shows the impact WMin/PP has had on reduced waste generation rates.

CONCLUSION

The traditional WMin/PP performance measure of waste volumes cannot be used as the sole measure of WMin/PP program effectiveness. Measuring waste eliminated due to WMin/PP actions is a superior performance indicator. This tool facilitates cost benefit analyses, helps justify additional investment in WMin/PP, and strengthens program credibility.

ACKNOWLEDGMENT

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WASTE MINIMIZATION APPLICATIONS FOR CAPACITY SHORTFALL REDUCTION IN PUERTO RICO
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ABSTRACT
Section 104 (c)(9) of CERCLA requires all states to submit a Capacity Assurance Plan (CAP) projecting hazardous waste generation and management capacity for the next 20 years. Puerto Rico's CAP identified several hazardous waste management categories, such as metals recovery and incineration of sludges/solids, for which capacity shortfalls were predicted. CDM Federal to evaluate possible options for reducing the projected shortfalls. One of the mechanisms studied was waste minimization.
The results of CDM Federal's study on the appropriateness and feasibility of using waste minimization to reduce hazardous waste management capacity shortfalls in the Commonwealth of Puerto Rico is presented below. This study is unusual in that it considers hazardous waste generation and management activities for an entire island, and evaluates the overall effectiveness of implementing waste minimization programs in various key industries to reduce the capacity shortfalls, rather than concentrating on waste minimization at a single facility.
CDM Federal obtained and evaluated hazardous waste generation and management data from several sources, including the Puerto Rico Hazardous Waste Manifest Database; the Toxics Release Inventory; and the Biennial Reporting System. One of CDM Federal's most important findings was that over 95% of the hazardous waste managed in commercial HWMFs is generated by just 26 facilities in seven different industries in Puerto Rico. This finding is significant because it indicates that waste minimization efforts implemented by a small number of facilities could substantially affect waste generation patterns for the entire island. The benefits of targeting only a limited number of industries, rather than imposing an "across-the-board" requirement that all industries implement waste minimization programs, are obvious, especially when considering the limited information and technical resources available to many Puerto Rican industries and regulatory agencies.
The seven industries generating the majority of the hazardous waste in Puerto Rico include: plastics and synthetic resins; medicinal compounds and botanical products; pharmaceutical preparations; petroleum refining; plating and polishing; printed circuit boards; and surgical appliances and prosthetic devices. CDM Federal evaluated potential waste minimization techniques for these industries and calculated that the various hazardous waste streams generated by these industries could be potentially reduced by 15 to 80%, thereby significantly reducing the predicted capacity shortfalls.
The results of this study suggest that the implementation of aggressive waste minimization programs within the identified industries has significant potential to reduce the predicted capacity shortfalls in Puerto Rico. By focusing waste minimization efforts on the identified industries rather than attempting to impose them on all industries, the Puerto Rico regulatory agencies can make the most efficient use of their limited resources. In addition, with little or no capital expenditure, nearly all facilities can implement better operating procedures to reduce unnecessary waste and to improve the efficiency and economy of their manufacturing processes.
BACKGROUND
Section 104 (c)(9) of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), as amended, requires all states and territories to assure

that adequate hazardous waste management capacity for waste generated within the state will be available during the next 20 years. Capacity assurance plans (CAPs) submitted to the U.S. Environmental Protection Agency (EPA) form the basis for the assurance of capacity in any contract or cooperative agreement for remedial actions between the state and the EPA Administrator. The EPA Administrator determines whether or not the capacity assurances provided in the CAP are adequate.

The Commonwealth of Puerto Rico submitted its CAP to EPA on February 15, 1992. The CAP submission provided waste generation and management analyses for the base year 1987 and analyses for the projection years 1989, 1995, and 2009. The data in the CAP was based on the Biennial Reports and the National Treatment, Storage and Disposal Facility (TSDF) Survey. The CAP predicts that there will be capacity shortfalls at commercial facilities for several waste management categories in each of the three projection years. The major commercial capacity shortfalls (i.e., greater than 400 tons) occurring by the year 2009 are as follows (see Table I).

Capacity shortfalls are also indicated for other waste management categories, such as sludge treatment and aqueous organic/inorganic treatment. However, these shortfalls are minimal in comparison to the five categories listed above.

APPROACH

CDM Federal investigated several options for reducing these shortfalls including source reduction/waste minimization activities. Our primary focus was to identify those waste minimization activities which would have the greatest impact on reducing the projected capacity shortfalls. To do this, the waste streams whose reduction would cause the greatest reduction in the projected capacity shortfalls were first identified. For example, metals recovery is projected to have an annual capacity shortfall of 7,835 tons in the future. In order to most effectively reduce this shortfall, it was necessary to identify those waste streams that are generally treated by metals recovery. Having accomplished this, manifest databases, the Toxics Release Inventory (TRI) and the Biennial Reporting System (BRS) were used to identify the industries producing the target wastes in substantial quantities (i.e. > 1 ton/year) for off-site treatment. With the critical waste streams and industries identified, CDM Federal proceeded to evaluate appropriate waste minimization techniques.

CDM Federal determined that over 95% of the hazardous waste generated in Puerto Rico and treated off-site is generated by 20 facilities which fall into seven groups of industries as classified by Standard Industrial Classification (SIC) codes. Waste minimization potential was evaluated primarily for those seven industries (see Table II).

FINDINGS

Following are summaries of several of the waste minimization techniques identified by CDM Federal for the industries listed in Table II.

Innovative techniques for allowing reuse of scrap plastic are currently being tested by various plastics manufacturers. Foam plastics may be dissolve in a solvent, thereby reducing the volume of waste relative to the typical low density, high volume solid form of foam plastic. Even if dissolved waste plastic is not reprocessed, dissolution of solid foam plastic will reduce the total volume for disposal. Scrap plastics composed of mixtures of different materials may be depolymerized to their original monomers for potential reuse. Ideally, the dissolved solution would be suitable for subsequent reprocessing of the plastic. However, this technique has not yet been perfected.

The pharmaceutical industry is a critical one, as the majority of hazardous waste generators in Puerto Rico fall into this category. Unfortunately, this industry also has unique barriers to source reduction and waste minimization efforts. Our study discovered that there is a reduced opportunity for technology transfer, due to the proprietary nature of the industry, and the product specific processes utilized. Regulatory constraints and concern over trace impurities, prevent the implementation of activities such as solvent recovery and reuse, despite the fact that solvent use and spent solvent waste generation is very high for this industry. High costs (in time and resources) associated with obtaining approval from the U.S. Food and Drug Administration (FDA) for process changes, material substitutions, and other modifications which might impact product quality and safety represented another hurdle.

Even with these barriers, however, some waste minimization may still be achieved through the application of one or more of following methods (see Table III).

1) Minimization of laboratory and R&D wastes, by implementing formal waste collection and segregation systems, and discouraging the collection of solvents and other wastes in non reusable containers which must then be disposed as hazardous waste. 2) Process and equipment modifications when approved, that will allow more efficient utilization of raw materials, resulting in less waste generation. 3) Material substitution, when approved, such as the substitution of degradable solvents or nonhalogenated solvents for halogenated solvents, or the substitution of aqueous systems where possible. 4) On-site small scale recycling of solvents using a reflux apparatus that will produce clean solvents for reuse and distillation bottoms for disposal. 5) Segregation of wastes streams, particularly halogenated and non-halogenated solvents, to allow increased on-site treatment and/or disposal of wastes, and less need for off-site treatment or disposal. 6) Installation of solvent recovery systems (such as distillation or extraction units) if approved. If concerns about product purity are barriers to use of recovered solvents, these solvents may be appropriate to use for cleaning equipment that does not come directly into contact with products, such as storage tanks and equipment exteriors.

A large portion of solvent wastes generated is composed of solvent-contaminated water from equipment and floor washwaters. This volume may be substantially reduced through application of one or more of the following methods. 1) Minimizing spills, thereby reducing the frequency of floor washings necessary. 2) Optimizing automated equipment cleaning cycles to use the minimal amount of water necessary. 3) Using the wastewater generated from the final rinse of one cleaning cycle as the pre-rinse on the next cycle. 4) Using low volume, high efficiency cleaning (e.g., spray heads rather than hoses).

Petroleum refineries produce large quantities of sludge generated from the API separators. The volume of sludge can be significantly reduced through dewatering the sludge. The products will be soil and a heavy-oil-and water mixture. The heavy-oil-and-water mixture can be passed through a condenser and separator, with a portion of the oil returned to the facility for reuse and the water sent to the wastewater treatment plant. If the soil contains heavy metals, it may be treated through soil washing. The products of soil washing will be heavy metal-containing water and sludge. The sludge will be dewatered, resulting in soil, which may be disposed of, and wastewater. The heavy metal-containing water must be treated in a wastewater treatment plant, resulting in clean water for reuse and hazardous waste sludge containing heavy metals. The volume of this sludge will be far smaller, with a much higher concentration of heavy metals, than initially generated.

Facilities utilizing electroplating or other metal finishing processes are generally major producers of metal-containing wastewater and sludges, as well as acids and chlorinated solvents. Minimization of wastewater treatment sludges from electroplating operations can be accomplished through the application of one or more of the following methods: 1) Installation of ion exchange columns to absorb metals from the wastewater stream prior to discharge to the wastewater treatment plant. 2) Utilization of electrolysis or electrowinning for the recovery of metals from plating baths, resulting in purification of the waste stream prior to discharge or reconstitution of the plating bath for additional use. 3) More efficient use of settling agents in wastewater treatment and sludge thickening operations, since up to 85% of the F006 sludge generated in some facilities has been found to be as a result of the use of sludge settling agents such as calcium sulfate. 4) Reducing impurities which shorten the life of plating baths.

Studies have shown that up to 50% of the waste streams treated by electroplating facilities as metal-containing wastes may actually be from processes unrelated to plating. Segregation of waste streams can reduce the amount of wastewater treated unnecessarily, and thereby reduce the volume of sludges produced. As an alternative to disposal, sludges and other metal-rich wastes may be transferred to other facilities that will use the wastes as they are, or reuse them after treatment, thereby reducing the waste stream by up to 100%.

Plating or rinse bathwaters are often disposed of on a routine schedule, that may exceed required frequency and thereby produce a greater volume of metal-containing wastewater and F006 sludges than necessary. Implementation of small scale bath testing would allow operators to determine when bathwaters have reached their safe loading capacity, and eliminate unnecessary discharges.

Circuit board manufacturers also produce large quantities of F003 and F006 wastes. F003 solvent wastes may be reduced through implementation of alternative degreasing

methods such as aqueous solutions coupled with ultrasonics, and process modifications to use "clean" fabrication techniques, thereby reducing or eliminating the use of degreasing solvents.

Metal-containing wastewaters generated during electroplating processes may be treated through ion-exchange or reverse osmosis units. These treatments purify the wastewater by absorbing the metal ions until the units are saturated. When the units are saturated with metal ions, they are regenerated by washing out the ions with a regenerating solution. The resulting metal-rich regenerating solution can then be reused as a plating solution. This process reduces the volume of metal-containing wastewater discharged to the wastewater treatment plant, and consequently the volume of metal-containing sludges produced. Depending on the specific industrial processes and materials used, this method can result in a 40 to 70% reduction of metal-containing sludges.

Contaminated wastewater is produced by most industries. Depending on the specific characteristics of the wastewater, contaminants may be removed for disposal or for reuse through distillation, evaporation/condensation, ion exchange, reverse osmosis, and hyperfiltration using membranes. These methods may remove varying amounts of the contaminants from the wastewater, producing clean water for reuse or discharge, and contaminant-rich solutions or sludges which may be reused or disposed of. The percentage waste reduction possible for these methods varies extensively from method to method.

Plant management practices can have significant impact on waste reduction. Such practices include employee training programs to ensure that personnel involved in operating equipment or handling wastes are capable of detecting chemical releases, safely handling hazardous materials, and understanding regulatory requirements. Management incentives can encourage innovative ideas from knowledgeable employees which result in improved waste reduction or recycling. Closer supervision and documentation of process procedures can prevent unnecessary spills, material losses and production of out-of specification products, and will help reduce waste generation during maintenance or emergency shutdowns. Production scheduling can increase the efficiency of raw material use and minimize equipment cleaning operations.

Substantial quantities of hazardous waste are generated through the overstocking of inventory. Automated inventory control has been proven to produce significant decreases in these wastes. In addition, using materials on a first-in/first-out basis minimizes waste composed of expired chemicals. Proper storage and handling of raw materials will minimize spills and leaks.

Storage tanks and process vessels must be regularly inspected for physical integrity, and overflow alarms and secondary containment measures installed, to reduce the potential for spills and leaks. Preventive maintenance of equipment, including cleaning, making minor adjustments, lubricating, testing, measuring, and replacing minor parts can substantially reduce the volume of waste generated from equipment failure or mechanical breakdown.

Many hazardous waste streams are actually composed of two or more waste streams. Waste stream segregation, including the separation of hazardous and nonhazardous materials, chlorinated and nonchlorinated solvents, and liquid and solid waste can reduce waste volumes, simplify disposal, and facilitate recovery and recycling.

Waste exchanges involve the transfer of a waste to another company for use as is, or for reuse after treatment. Metals and solvents are the wastes that are most frequently recycled through waste exchange, due to their high recovery value. Other wastes that may be recycled through waste exchanges include inorganic chemicals, organic chemicals, and metal sludges. Since the treatment of solvent wastes and metal-containing wastewaters and sludges places a significant demand on four of the five target waste management categories, waste exchanges may provide a feasible alternative to treatment in Puerto Rico.

Taking into account the waste minimization techniques proposed the potential percentage reductions of many of the waste streams produced by the seven target industries were estimated. Due to the extremely variable and complex nature of some of the processes and materials used by some of the industries it is not possible to estimate percentage reductions for all waste streams for all industries. The percentage reductions possible are listed in Table IV, for the wastes produced by each industry.

CONCLUSIONS

The Puerto Rico CAP predicted that there would be insufficient hazardous waste management capacity over the next 20 years in metals recovery, incineration, stabilization, landfilling, and other treatment (primarily wastewater treatment). Waste minimization has significant potential to reduce the predicted capacity shortfalls in PR. CDM Federal has determined that over 95% of the hazardous waste generated in Puerto Rico is produced by just twenty facilities in seven industries. Therefore, the implementation of aggressive waste minimization programs at these facilities could significantly reduce the capacity shortfalls predicted for Puerto Rico as a whole. Spent solvents, contaminated wastewater and metal-containing sludges are the primary wastes that should be targeted for minimization. Process-related minimization techniques include installation of recovery systems for solvents and metals from wastewater; sludge reduction through dewatering, and thermal desorption; and decreased use of sludge settling agents. Non process or industry specific minimization techniques include waste-stream segregation and better operating practices. Nearly all facilities can implement better operating procedures with little or no capital expenditure, to reduce unnecessary waste and to improve the efficiency and economy of the manufacturing processes.

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LOW-LEVEL WASTE DISPOSAL FACILITY SAFETY ANALYSIS EXPERIENCE: ESTABLISHING RADIONUCLIDE INVENTORIES

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ABSTRACT

This paper describes work performed to estimate radionuclide inventories (Material at Risk) for use in developing upgraded safety documentation for the Solid Waste Management Facility (SWMF) at the Department of Energy's Savannah River Site (SRS). The general approach was to: 1) identify the best available information for the current inventory of each subfacility (or the maximum credible inventory for process-type subfacilities, such as the compactor); 2) adjust the current inventory for uncertainty allowances, biases, and projections of future additions; and 3) convert the information to a form readily usable for safety assessments. Since waste disposal began over 40 years ago and the last extended period of high-power reactor operation was in 1988, radioactive decay is significant for several important nuclides (e.g., tritium, most fission products, and most neutron activation products). Much of the available data on material deposited in the SWMF pre-dates development of current nuclear industry quality assurance requirements. Since the inventory estimates were partly based on this early data, the final inventory values were qualified for use in safety assessments by an interdisciplinary team review process, conducted in accordance with SRS procedures (in addition to the normal detailed technical peer review).

INTRODUCTION

This paper describes work performed to estimate radionuclide inventories (Material at Risk) for use in developing upgraded safety documentation for the Solid Waste Management Facility (SWMF) at the Department of Energy's Savannah River Site (SRS). The SWMF includes disposal and storage facilities for low-level radioactive waste

generated at SRS and other government-owned locations. The SWMF is primarily composed of "burial grounds," where waste has been disposed of in trenches since 1953, concrete vaults, scheduled to be used for disposal starting in 1994, and above-ground paved "pads," where transuranic (TRU) waste is stored in containers awaiting future disposal at the Waste Isolation Pilot Plant (WIPP) in New Mexico. The SWMF also includes areas for above-ground storage of contaminated used equipment, a waste compactor, tanks for storage of slightly contaminated reprocessing solvents, a facility to certify waste prior to shipment to WIPP, and buildings for storage of containerized mixed and hazardous waste.

The SRS is an approximately circular region approximately 30 km (18 miles) in diameter and approximately 800 square kilometers (300 square miles) in area. Most of the areas (or "subfacilities") making up the SWMF are located near the center of the site.

DESCRIPTION

The objective of the work was to produce radionuclide inventories (including projections for future material) that are conservatively bounding for safety assessments (but not grossly overconservative so as to artificially cause predicted accident consequences to exceed acceptance limits) without excessive effort or cost. Projecting future inventories was crucial because the inventories of waste storage/disposal facilities tend to increase; several of the areas making up the SWMF have the physical capacity to operate for several more years or can be expanded.

Inventories were needed for several different types of safety assessments. Some types of assessments need total subfacility (or subfacility-segment) inventories. Others need the inventory in the containers that can be affected by a single postulated credible event. Several different types of inventories were developed for each subfacility to meet these needs. For example, the TRU inventories included evaluation-basis inventories for single containers of several different types.

The general approach was to: 1) identify the best available information for the current inventory of each subfacility (or the maximum credible inventory for process-type subfacilities, such as the compactor); 2) adjust the current inventory for uncertainty allowances, biases, and projections of future additions; and 3) convert the information to a form readily usable for safety assessments. For most of the subfacilities, this readily-usable form consisted of lists of the radioactivity (in curie units) for 27 key radionuclides, plus two pseudonuclides, "other beta-gamma emitters" and "other alpha emitters," to cover less important nuclides not on the standard list.

Difficulties encountered included dealing with: 1) the "expandable" nature of a waste storage/disposal facility; 2) the many different areas within the SWMF and the different types of waste in each area; 3) the number of waste streams coming to the facility (which may change as the SRS mission shifts to decontamination and decommissioning); 4) lack of detail in data in the existing computer-based inventory tracking system; and 5) the need to estimate inventories for material already in the facility that was assayed, packaged, and emplaced under procedures and standards that have evolved over a 40-year period (some older procedures allowed higher single-container inventories than are allowed today).

Since waste disposal began over 40 years ago and the last extended period of high-power reactor operation was in 1988, radioactive decay is significant for several important nuclides (e.g., tritium, most fission products, and most neutron activation products). The estimates of radionuclide inventories took credit for decay where data was readily available and the decay was expected to result in significant inventory reductions.

Available data for some material does not specify its precise isotopic makeup (some enriched uranium at SRS has relatively large amounts of U-234 and U-236) and, for short-half-life material, the isotopic mix changes while the waste is stored. The estimation techniques were conservatively based on assumed isotopic mixes that maximized the calculated radiation doses from postulated releases of the material. In some instances, the assumed isotopic fractions summed to more than 100%. For example, the (14.1-year half-life) Pu-241 content of plutonium was counted as both parent (beta-emitting) Pu-241 and as daughter (alpha-emitting) Am-241 to ensure calculated doses will be bounded for material of any age.

RESULTS

The results of this work, which consist of several multi-column (and in some cases,

multi-page) tables of nuclide activity values, were documented in an Engineering Calculation prepared in accordance with SRS procedures.
Much of the available data on material deposited in the SWMF pre-dates development of current nuclear industry quality assurance requirements. Since the inventory estimates were partly based on this early data, the final inventory values were qualified for use in safety assessments by an interdisciplinary team review process, conducted in accordance with SRS procedures (in addition to the normal detailed technical peer review).
Projected future inventories are very sensitive to initial assumptions, such as the scheduled operational status of certain subfacilities, continued use of current waste packaging procedures, and continuation of restrictions on bringing specific types of waste into certain areas. These assumptions were documented in detail in the Engineering Calculation so that, in the future, they can be linked to administrative controls to protect the safety basis.
ACKNOWLEDGMENT
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MEETING THE CHALLENGES OF REMOTE HANDLED LOW LEVEL WASTE DISPOSAL AT THE IDAHO NATIONAL ENGINEERING LABORATORY RADIOACTIVE WASTE MANAGEMENT COMPLEX
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ABSTRACT
This paper describes an original approach to providing safe disposal of Remote Handled Low Level Waste (RH-LLW) that is required to meet the Performance Assessment (PA) objectives for the Radioactive Waste Management Complex (RWMC). The system known as the RH-LLW Concrete High Density Polyethylene Vault Liner System allows a cost effective means of complying with the U.S. Department of Energy (DOE) requirements. The design approach is modular in nature, allowing additional disposal capacity to be added on a demand basis, thereby eliminating significant upfront costs associated with large facilities built on estimated needs over many years. The concept and design approach reduces the financial and technical risks associated with large construction projects. Additionally, upgrades to future installations and modifications to new disposal liners put into service based on changing regulations and technologies are possible. The versatile design of the RH-LLW Liner System offers additional benefits including potential future retrieval, a minimum 100 year design life, and continuous below ground protection from ground water intrusion and direct contact with the surrounding environment.
BACKGROUND
The RH-LLW Lined Vault design was driven by the need for disposal of high curie concentrations of Tritium (H-3) generated at the Test Reactors Area (TRA) Advanced Test Reactor (ATR) and its subsequent effect on the PA objectives for disposal of specific isotopes at the Idaho National Engineering Laboratory (INEL) RWMC. Requirements for site specific PA objectives are promulgated by DOE-HQ Order 5820.2A, and the effect of radiation disposal on the environment for future generations. To meet the criteria of DOE-HQ Order 5820.2A, PA requirements, and the management of Low Level Radioactive Waste, Chapter III, a screening process is required for all isotopes planned for disposal. Due to the significant quantities of H-3, Co-60, and Ni-63 present in past and future TRA/ATR waste streams, a screening procedure was performed to validate the significant dose contributor for the ground water pathway. To complete this, the Source Term and Hydrologic Transport Models were used in the assessment. Based on the screening assessment, it

was determined that disposal of current inventories of H-3 would exceed the committed effective dose equivalents for the all pathways analysis of 25 mrem/y criteria for the time period considered in the RWMC PA. The U.S. Department of Energy, Idaho Operations Office (DOE-ID) requires strict compliance and conformance for the disposal of Low Level Waste (LLW), and that all waste (isotope specific) is properly characterized prior to disposal. Proper disposal at the RWMC requires every "new" waste stream be evaluated to the same rigorous administrative and technical process, not only for the absence of hazardous materials, but also to determine what effect the transport of radioisotopes have on the environment and the potential uptake for the inadvertent intruder after the institutional control period. Proper collection and disposal of all LLW at the INEL, requires the operating contractor to be in verbatim compliance with all appropriate DOE, Nuclear Regulatory Commission (NRC) regulations, the Environmental Protection Agency (EPA), and 10 CFR 61.55, Tables 1 and 2, Isotope Classification Criteria. The document, Joint NRC-EPA Guidance on a Conceptual Design Approach for Nuclear Facilities, describes an acceptable facility based upon a double liner and leachate collection system with a specific design life, based on hazards. It allows generators and treatment storage and disposal facilities to develop alternative designs based upon their specific needs. These alternative designs must meet the technical and regulatory criteria as set forth by federal and state agencies, and meet the PA objectives for that facility. The RWMC considers the PA to be a Safety Analysis Report (SAR) equivalent level document and that a defensible evaluation be a prerequisite to disposal. The challenges at the INEL were based on the limited area which is available for the disposal of high radiation, short lived low level radioactive wastes and the need for cost effective solutions. The lack of double leachate collection under the designed geographic area, adequate ground water monitoring wells, ground water sampling, and individual container degradations due to soil chemistry inspections placed a greater emphasis on efficiency and challenged conventional methods of disposal. The need for the waste to be remotely handled and protection to personnel and the environment added to the design challenge. The authors have developed a design approach based on meeting the site specific challenges.

DESIGN FEATURES OF THE RH-LLW CONCRETE LINED VAULT SYSTEM

A conceptual design is provided in Fig. 1 which shows a cutaway view of the single piece unit assembly. Figure 2 shows the modular assemblies nested in a plan view "honeycomb" with side view construction. Figure 3 shows the HDPE vault liner detail and sectional view, and Fig. 4 shows the HDPE vault liner lifting and squaring device.

DESCRIPTION

The design is based on primary containment via an inner, high density polyethylene cylindrical liner which will accommodate 55 gallon containers, inserts, or unpackaged waste. This inner liner is free standing and is contained by an outer concrete liner providing structural integrity, secondary containment, and additional leachate collection. The outer concrete liners are designed in a hexagonal pattern which allows a self locking configuration and modular design. Additional vaults can be added to the honeycomb design on an as needed basis, thus lowering construction and facility operating costs. Additional benefits gained by the unique design and self locking construction, are zero subsidence between vaults, no maintenance in the future, and it provides a minimum of wasted space between vaults. This provides maximum disposal potential at the RWMC, or any disposal facility that requires segregation of waste types or lacks the necessary soil depth space for subsurface disposal. The enhanced disposal concept allows close proximity of waste types without the problems associated with migration or high radiation.

DESIGN VERSATILITY/DESIGN SPECIFIC BENEFIT

Although the existing methodology is to install this system below grade at the RWMC, the versatility of the design would allow installation on or above existing grade, or any area where below ground installation or disposal is not an option or permitted, or is inadvisable due to ground water rock formations, existing waste, or where a potential for future monitoring may be required. The design also allows for retrieval when technologies become available for reprocessing or where separation and segregation is required in the future.

MODULAR CONCRETE VAULT DESIGN

The reinforced concrete vault consists of a reinforced base, center section, top flange, and reinforced shield plug (see Fig. 1). To best utilize the available

space at the RWMC and match existing disposal packages, it was determined the project would be best served by using standard, off the shelf pipe diameters which could be manufactured locally, without the need for special tooling. The construction and design choice consists of 20 foot 54 inch inside diameter right vertical cylinder constructed in a two piece design configuration. The lower, or base section, is designed with a hexagonal six sided design which will allow for proper indexing and grid layout at placement (see Fig. 2, "Plan View"). The final disposal package would reside in the high rad center and top section, which is constructed of a male straight wall to female pipe flange, with a female upper flange that nests the shield plug (see Fig. 2, "Side View". The base and center section assemblies are grouted together with an elastomeric sealing material that is specially designed to prevent the intrusion of ground water or runoff from entering the vault disposal cavity. Prior to mating the shield plug with the vault body, the annular spaces between the vault center sections are filled with fine granular sand to lock the vault bodies in place. This also provides additional shielding between the vault shields and completes the primary boundary and support structure for the shield plug and secondary High Density Polyethylene (HDPE) inner liner. The functional and operational requirements (F&OR) for the primary concrete vault system was originally determined to be 50 years (minimum). To meet the challenge and additional requirements for this project, evaluations were required by both engineering and computer modeling to further extend the design life from 50 to 100 years minimum.

HDPE INNER LINER DESIGN AND CONSTRUCTION

The HDPE inner liner is designed as a stand alone primary containment for the waste package. The HDPE liner is constructed in three pieces; vault body, bottom plate, and macro-encapsulation lid plate (see Fig. 3). The vault body is constructed as a single piece continuously formed straight wall pipe or tube 19 feet 4 inches long with a side wall thickness of 1.625 inch minimum. The bottom plate is 1.00 inch flat that is thermo-welded to the vault body prior to installation.

The lid plate is the same construction and size as the bottom, with one exception: The lid section is equipped with a dual nichrome flat ribbon wire that is extruded in high density polyethylene. The extruded ribbon is approximately .5 inches wide by .375 inches thick with the dual nichrome flat wire evenly spaced in the center of the ribbon. This concept is a "first" in sealing and closure technology for remote handled high rad waste. The ribbons are thermally welded to the lid plate prior to placement on the vault body. The dual ribbons are equipped with 12 to 16 foot pigtails that will allow space and distance shielding for the technician performing the vault lid installation and sealing operations. Due to the remote location of the technician, and no direct contact with the loaded vault, personnel exposures to direct ionizing radiation can and will be minimized to as low as reasonable achievable (ALARA) limits. The nichrome flat ribbons are activated by a specially designed encapsulation control unit. The encapsulation control unit energizes the ribbon, thus forming a remote heat activated seal with the lid and vault body. Again, the remote activation encapsulation seal can be accomplished with zero radiation exposure to workers and provides a solid encapsulation bond between the lid plate and vault body. The encapsulation seal eliminates the possibility of material "creep" that would be common to standard sealing technologies, i.e., threaded or mechanical sealed flanges with elastomeric or chemical grouts. If future entry into the loaded vault is required, the encapsulation seals can be re-energized and the lid plate can be removed. Due to the nature of the heat seal, this would potentially ruin the integrity of the sealing surfaces which would require a new lid plate and surface preparation on the vault body (upper sealing surface).

After activation of the encapsulation seal, the control unit is disconnected and the pigtails are dropped onto the upper vault lid with the lid rigging. The design of the HDPE and concrete vault interface provides an 8 inch annular space between the bottom of the shield plug and the top of the HDPE liner thereby eliminating contact of the shield plug with the vault liner and providing a secondary storage area for rigging and encapsulation control wiring.

Please Note: The lid is equipped with a dual wire system that provides a backup to the primary system if a failure or a short circuit in the primary system were to occur. Testing was performed on the single wire system without a failure, but due to the potential of re-entry exposure contamination, it was determined that a backup system would provide an additional level of protection and safety. Activation of

the backup system is not required, but will be performed to increase the lid to vault body sealing area.
To prevent the potential health hazard of direct high ionizing radiation out the top of the vault system, and to provide a safe working platform for operations personnel and technicians at the RWMC, the RH-concrete HDPE lined vault system has been designed with a reinforced concrete shield plug approximately four foot thick. The shield plug design employs the same hexagonal shape and interlocking capability as the concrete base. The plug design also incorporates recessed coil bolt lifting eyes that can be installed to remove the plug from an empty vault and removed. It can then be plugged to prevent damage to the lifting eyes and while providing a smooth working platform for equipment and personnel. Upon completion of the loading and HDPE sealing operation, the concrete shield plug would be replaced with the spaces between the vault plugs grouted to prevent rain and snow runoff from seeping between the shield plugs. A third level of protection might employ the use of non permeable HDPE flat liners over the top of the filled vaults with a soil berm and vegetation cover to complete the closure process. Due to the uncertainty of future programs and the specific curie contents from the different INEL facilities expected to use this disposal system, it was determined that the most versatile design would incorporate a plug design and shielding equivalent to provide for the worst case scenario.
CONCLUSION
The RH-LLW concrete disposal vault offers a variety of effective and efficient design alternatives that will allow the design to be tailored to a variety of situations, needs, or specific package sizes and types and will offer an innovative solution to the management and disposal of rad and low level mixed waste. In addition to this project, efforts are in place to fabricate a prototype package overpack design that will employ a similar closure device and macroencapsulation feature that meets the definition and intent of macroencapsulation for mixed wastes per EPA and greater confinement criteria per NRC and DOE requirements.
Upgrades to the basic vault design can be accomplished with little or no impact to the facility. In addition to isotopes and highly contaminated material, the advantages of secondary and tertiary containment allows a strong management methodology. This dispels the negative public perception that the DOE and its operating contractors have a radioactive nuclear waste management mentality of "business as usual" mode and of not being proactive in their approach to the long term future management of highly radioactive wastes and the safety to the public and the environment. It is the commitment and policy of the DOE and LITCO to provide a level of confidence to the public and its shareholders that confirms our shared concern over the nuclear and waste management issues facing the United States today.
Alternative Study
Attachment 1
Cost Analysis: In determining the most cost effective alternatives, a comparison study was conducted between the following:
No Action
Upgrading the existing facility
Installing the RH-LLW Concrete High Density Polyethylene Vault Liner System.
Study on 1:
No Action.
This would allow the facility to remain "as is" with no change or upgrades.
Conclusion
DOE-ID and LITCO found the no action alternative to be inconsistent with the national policy and commitment on the environment, future public exposure, and INEL workers health and safety. Container degradation due to soil chemistry, leaks, and spread of radioactive contamination would not be known until the level of contamination spread was far beyond effective remediation. Radio isotopic spread to the Snake River Aquifer would pose and unacceptable risk to the stakeholder down gradient of the INEL. The cost of non-conformance, at a minimum, would close the RWMC due to violation of the RWMC PA, not to mention the multiple per-violation fines from State and Federal regulators. The cost of the no action alternative would far exceed the cost of alternatives 2 or 3.
Advantages: None
Disadvantages: Many
Cost of Non-conformance: Unknown at this time

Study on 2:
Upgrading the existing facility.
This would require installation of a complete monitoring well system down gradient from the proposed disposal location. To meet existing regulatory requirements, a new facility would also be required to develop a ground water sampling and analysis plan with clean-up contingency plans, and installation of a double lined leachate collection system under the disposal area.
Conclusion
DOE-ID and LITCO would have to relocate the existing facility to a new location due to the subterranean nature of disposal. Determining "need" for a new facility would require detailed solid waste projections, isotopic characterization, and long term planning. Additional costs include preparing an environmental assessment (EA), and an environmental impact statement (EIS). Costs to construct a new disposal facility, less the EA and EIS, would exceed approximately $100 million. This is a low estimate considering the changes in regulatory requirements and locating the facility potentially off the Snake River Plain Aquifer which might not be possible at the INEL. In addition, the above costs do not address waste or environmental inspections or the appropriate National Environmental Policy Act (NEPA) documentation required for a new facility. These costs would be highly speculative and would far exceed the cost of construction.
Advantages: A new facility would meet current EPA, State and Federal regulations for disposal of CH and RH waste. The facility disposal capability could be expanded if adequate leachate collection was provided in the upfront planning.
Disadvantages: Unknown costs of permitting, future regulations, and "need" present an unacceptable risk to the INEL.
Total Cost: Specifically unknown, estimated in excess of $100 million.
Study on 3:
Installing the RH-LL Concrete Vault HDPE Liner System.
This would require minor modification to the existing facility, the addition of underground power lines or remote electrical generating capability, remanufacture or retrofitting cask bearing and pads, and the purchase of slip-in liners with remote encapsulation seal control units.
Conclusion
The RWMC found Alternative 3 to provide the most benefits of the alternatives examined. Minor modification to an existing facility precludes the need for additional evaluation to the NEPA requirements and eliminate the need for the EA and EIS permitting process. Associated facility modifications and transport cask discharge systems upgrades are as follows:
Manufacture of concrete vault and HDPE vault liners
Remanufacture cask discharge systems shield plates
Design internal loading channel for cask to vault interface
Design and procurement of special manufacturing and handling equipment to build and install HDPE vault liner system
Develop Detailed Operating Procedures (DOPs) for vault loading and closure process
Purchase H-3 monitoring instrumentation and equipment trailer.
COST BREAKDOWN AND SETUP COSTS FOR SIXTEEN
HDPE VAULT LINER UNITS:
HDPE vault body blank 19 feet 4 inches long x 52.5 inches
in diameter. Unit cost 6.5K each x 16 units needed. 104.0K
HDPE lid and bottom blank 1 inch x 52.5 inch diameter.
Unit cost (set) 1.2K per vault x 16 sets needed.
19.2K
HDPE encapsulation control unit. Single time purchase.
Unit cost 8.5K x 2 units (primary and backup) needed. 17.0K
Inset seal groove cutter with centering device (single time purchase) 9.5K
HDPE vault liner welding fixture for bottom plate installation (single time purchase).
6.5K
HDPE vault liner squaring fixture to maintain vault body roundness
during welding operations (single time purchase). 3.5K
HDPE lid blank inset seal groove lid preparation with encapsulation
seal installation. 1.4K each x 16 units needed.
22.4K

HDPE bottom plate installation with spark integrity test.
Per unit cost 1.4K x 16 units needed. 22.4K
HDPE vault liner installation in RH-LLW concrete vaults. 40.0K
Per unit cost 2.5K x 16 unit installations. TOTAL 244.5K
Minus single time purchases and set up costs.
-36.5K
208.0K

NOTE: The RWMC currently has 100 each RH-LLW concrete vaults installed with outyear projections of 400 total vaults identified (1998 projections). The costs associated with design, procurement and installation are approximately 20.0K per unit, or 100 units installed is equal to 2.0M total. We feel the per unit costs will be significantly reduced on outyear procurements due to the elimination of the upfront design and analysis costs associated with the start up of this project.

ADVANTAGES

Meets all existing disposal and packaging regulations without building a new facility. Provides container corrosion protection, remote, modular design, multiple boundary protection, and meets the DOE/LITCO ALARA goal of no exposure to personnel. Additional vaults may be added to the system as the need arises.

DISADVANTAGES

Only one prototype has been purchased for the RWMC. This will only provide disposal for four TRA/ATR inserts of RH-Low Level canal waste. Additional vaults can be added but would require the validation process be successfully completed. Manufacturing and setup costs are extremely high for single unit purchases.

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SOURCETERM: AN INTERACTIVE LOW-LEVEL RADIOACTIVE WASTE DATABASE

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ABSTRACT

SOURCETERM is a comprehensive computer program that combines an interactive low-level radioactive waste (LLRW) database with waste accumulation and radioactive decay algorithms, enabling planners to forecast the quantities and characteristics of LLRW requiring management and disposal. The tabular results from SOURCETERM are direct inputs to disposal facility siting, design, and long term Performance Assessment efforts. Although a variety of independent software programs have been available in the past to manage source term data, project waste generation, and evaluate radioactive decay, SOURCETERM is unique in its combination of these functions into a single comprehensive planning tool.

SOURCETERM provides LLRW planners with the following database management and modeling capabilities:

Managing the LLRW source term database, including the characteristics, generation rates, and accumulation of LLRW expected during the operational life of a LLRW disposal facility. Specific functions include browsing, adding, editing, deleting, sorting, and displaying the data. SOURCETERM currently allows the user to manage the following waste characteristics for each waste stream in the database: waste stream, waste class, generator, total volume (m3 or ft3), total activity (Ci), radionuclide inventory, solidification agent (if applicable), physical/chemical characteristics, container type, container volume (m3 or ft3), container weight (kg or lb), container on-contact radiation levels, number of containers, waste volume (m3 or ft3), waste weight (kg or lb), waste density (kg/m3 or lb/ft3), packaging efficiency, annual generation rate, and applicable deposition years.

Modeling the accumulation of waste in a disposal facility according to user-defined scenarios. A waste generation scenario includes selected waste streams (characterized in the database), generation rates, starting points, and ending points over time. The model incorporates both accumulation functions and the effects

of radionuclide decay on the radionuclide inventory (and in-growth of daughter products) over any user-specified period of time.

Producing a range of user-selected reports that include 1) listings of the source term input database, including waste characteristics and generation rates, 2) volume and activity projections for any user-specified period of time, 3) radionuclide inventories for any user-specified period of time, and 4) cross-tabulations of waste characteristics with waste quantities, including both volumes and radionuclide activities.

SourceTerm was developed specifically for the Microsoft Windows operating system using the C++ programming language. All data and modeling results can be viewed on-screen, and the information can be exported in tabular form to other programs, such as spreadsheets, databases, and word processors.

INTRODUCTION

SOURCETERM is a comprehensive computer program that combines an interactive low-level radioactive waste (LLRW) database with waste accumulation and radioactive decay algorithms, enabling planners to forecast the quantities and characteristics of LLRW requiring management and disposal. The tabular results from SOURCETERM are direct inputs to disposal facility siting, design, and long term Performance Assessment efforts. This paper presents an overview of the software, it's capabilities, and sample results for a typical LLRW disposal facility.

DATABASE MANAGEMENT

There are two databases used by SOURCETERM to model a LLRW disposal facility: a database of waste stream characterizations and a database of radionuclide decay information (half-lives, daughter products, etc.). The database management functions used to enter and maintain both of these databases are typical and cover both working with files and the individual records that they contain. Although the native, binary file formats used by the program are unique to SOURCETERM, the data can be imported from and exported to dBase compatible files.

The primary data entity for SOURCETERM is the waste stream. A waste stream characterization includes both qualitative and quantitative information and is stored in a single file. This characterization includes a waste category, an identification of the generator of the waste, the waste class (A, B, or C), the annual volume and activity generated, a range of applicable years, multipliers for minimum, expected, and maximum cases, and a breakdown of activity by radionuclide. Also, there can be multiple container types for each waste stream. These containers are characterized by the container type (B-25 box, 55-gallon drum, etc.), volume, weight, packaging efficiency, number of containers generated, and the physical/chemical properties of the contents of the container. For each container type (and for each waste stream), there can be multiple on-contact radiation levels, expressed as a fraction of the containers at some radiation level.

WASTE ACCUMULATION ALGORITHM

The waste accumulation algorithm is relatively simple: it implements behavior much like that of an interest bearing savings account. Waste is deposited at regular intervals (one year, end of period) and radionuclide decay is calculated on the remaining balance. The complexity of the waste accumulation function lies in managing the waste quantities (total volume, total activity, and activity by radionuclide) for each of the waste characteristics of interest (waste class, waste category, waste generator, waste stream, and radiation levels by waste class and container type).

The user creates a waste accumulation "scenario" by specifying a scenario title, a beginning year, and an ending year. When the scenario is generated, the accumulation function models the total waste inventory in the disposal facility over the range of years specified in the scenario. For each year in the scenario, the existing radionuclide inventory is decayed for one year, and then the entire database of waste stream characterizations is scanned to identify those waste streams that apply to that year. The contents of each applicable waste stream is added to the accumulating inventory. With respect to radionuclide decay, the waste is placed in the disposal facility at the end of the year, so that there is no decay calculated for the waste stream on the year that it is deposited.

For each scenario, the following information is accumulated:

The user-specified title of the scenario (e.g. "60-Year Scenario"),

the user-specified beginning year (e.g. "1995"), the user-specified ending year (e.g. "2054"),

the total volume of waste accumulated for each case, where a "case" is defined as the minimum, expected, and maximum amount anticipated,
the total decayed activity accumulated for each case,
a list of decayed activity totals by radionuclide for each case,
a list of undecayed activity totals by radionuclide for each case,
a list of waste quantities by waste class (e.g. Class A, B, or C) for each case, where a waste quantity
includes total volume, total decayed activity, and
a list of decayed activity totals by radionuclide,
a list of waste quantities by waste category (e.g. boiling water reactor, etc.) for each case, a list of waste quantities by waste generator (e.g. utility, medical institution, etc.) for each case
a list of waste quantities by waste stream (e.g. resins, irradiated reactor components, etc.) for each case,
a list of annual inventory totals (total volume, total decayed activity, and decayed activity by radionuclide) for each case, and
a list of radiation levels (waste volume at that level) by waste class and container type for each case.

A similar waste accumulation function can be utilized for a single waste stream. In this case, the total volume, total decayed activity, and total decayed activity by radionuclide for each case (minimum, expected, maximum) is accumulated over a specified range of years for the selected waste stream. This function can be used to assess the incremental impact of a single waste stream over the life of the disposal facility.

RADIONUCLIDE DECAY ALGORITHM

The radionuclide decay algorithm is necessarily complex, given the nature of radioactive decay. A common occurrence in radioactive decay is a parent radionuclide decaying to a daughter that is also radioactive. Also, it is not uncommon for a parent radionuclide to have two possible daughter products, each of which may themselves be radioactive. Consequently, it is possible to set up a radioactive decay chain where each node can have zero, one, or two daughter nodes, depending on the characteristics of the radionuclides involved.

The source of these radionuclide decay characteristics is a database of radionuclides, their half-lives, daughter products, and branching probabilities. This information is in the public domain and is available from a number of sources. One prominent source is the RADIOACTIVE DECAY DATA TABLES by David C. Kocher, DOE/TIC-11026, available through NTIS. The radionuclide decay database distributed with SOURCETERM (the file "nuclide. dk") is the same as that used to generate Kocher's decay data tables, published in 1981.

A generalized form of the radioactive decay equation is required to calculate the activity remaining after an elapsed time for the parent radionuclide and any daughter products. Although the Bateman equation is a generalized decay equation commonly used for straight decay chains, it does not incorporate the effects of multiple daughter products from a single parent radionuclide. It was therefore necessary to derive a similar equation that would account for multiple branching daughter products. The resulting generalized radioactive decay equation used in SOURCETERM is:

Equation A

As an example, for the third radionuclide in a branching decay chain (the parent is the first radionuclide) the above expression evaluates to:

Equation B

This decay algorithm is used heavily by the waste accumulation functions to determine the impact of radionuclide decay on the accumulating inventory. It is not uncommon for a typical 60-year scenario to require millions of decay calculations. Consequently, for waste accumulation purposes, an activity of less than one pico-curie is considered to be zero. This greatly reduces the number of radionuclide decay calculations required (almost a 50% reduction) and improves the processing performance without affecting the numerical significance of the radionuclide inventories.

A single radionuclide can be decayed by selecting a radionuclide from the decay database and specifying an initial activity and elapsed time. The default values are one curie and one half-life. This function can be used both for validating and verifying the operation of the decay algorithm and as a general radionuclide decay

calculator.

MODELING RESULTS

SOURCETERM enables planners to forecast the quantities and characteristics of LLRW requiring management and disposal. The modeling results include:

an annual waste inventory comprised of total volume, total activity, and activity by radionuclide for each of the minimum, expected, and maximum cases;

a total waste inventory by waste characteristic (class, category, generator, and stream);

a total radionuclide inventory, including a comparison of the decayed activity with an undecayed activity for each radionuclide to show the effect of incorporating radioactive decay into the model; and

the volume of waste within a specified surface radiation level range for each waste class and each container type in the disposal facility.

These results are direct inputs to disposal facility siring, design, and long term Performance Assessment efforts. They can be exported in tabular form for use in other programs, such as spreadsheets, databases, and word processors. Results can be saved on disk in either ASCII text or dBase file formats, and they can be copied and pasted into other Windows applications through the Windows Clipboard.

A few results for a typical 60-year scenario are presented in the following figures. The impact of radioactive decay on the accumulating inventory is apparent in Fig. 1, which illustrates the total activity over time. Prominent events are identified in the scenario, such as when nuclear power plants begin decommissioning. Figure 2 shows a breakdown of the prominent radionuclides at the point of closure of the facility. Figure 3 shows a distribution of the accumulated inventory at closure by waste class, both on a volume basis and an activity basis. As expected, the Class C waste contains the most radioactivity in the smallest volume. Figure 4 shows the distribution of waste, both on a volume basis and an activity basis, at closure by waste category. The general categories used include non-utility waste streams, utility routine waste streams, and utility decommissioning waste streams.

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DEVELOPMENT OF AN EDUCATIONAL PARTNERSHIP FOR ENHANCEMENT OF A COMPUTER RISK ASSESSMENT MODEL

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ABSTRACT

The Multimedia Environmental Pollutant Assessment System (MEPAS) is a computer program which evaluates exposure pathways for chemical and radioactive releases according to their potential human health impacts. MEPAS simulates the exposure pathways through standard source-to-receptor transport principles using a multimedia approach (air, groundwater, overland flow, soil, surface water) in conjunction with specific chemical exposure considerations. This model was originally developed by Pacific Northwest Laboratory (PNL) to prioritize environmental concerns at potentially contaminated U.S. Department of Energy (DOE) sites. Currently MEPAS is being used to evaluate a range of environmental problems which are not restricted to DOE sites. A partnership was developed between PNL and Mesa State College during 1991. This partnership involves the use of undergraduate students, faculty, and PNL personnel to complete enhancements to MEPAS. This has led to major refinements to

the original MEPAS shell for DOE in a very cost-effective manner. PNL was awarded a 1993 Federal Laboratory Consortium Award and Mesa State College was awarded an Environmental Restoration and Waste Management Distinguished Faculty Award from DOE in 1993 as a result of this collaboration. The college has benefited through the use of MEPAS within laboratories and through the applied experience gained by the students. Development of this partnership will be presented with the goal of allowing other DOE facilities to replicate this program. It is specifically recommended that DOE establish funded programs which support this type of a relationship on an on-going basis. Additionally, specific enhancements to MEPAS will be presented through computer display of the program.

INTRODUCTION

Multimedia Pollutant Assessment System Computer Program

The Multimedia Environmental Pollutant Assessment System (MEPAS) is a human health risk computational model (Multimedia in this context refers to multiple environmental transport and exposure media.) (1,2) MEPAS takes the nontraditional approach of integrating all major exposure pathways into a single public health computational tool. MEPAS employs an integrated, physics-based approach that couples source, contaminant release, migration and fate for environmental media (groundwater, surface water, air), with exposure routes (inhalation, ingestion, dermal contact, external dose) and risk/health consequences for radiological and non-radiological carcinogens and non-carcinogens (Fig. 1).

Throughout MEPAS' development and subsequent application, PNL has subjected the methodology to thorough external evaluation. Extensive review by the scientific community has fostered and confirmed the validity and usability of MEPAS. (3) Some of the evaluations and applications of MEPAS are:

Table I

MEPAS' versatility and sound technical basis make it applicable for a wide variety of risk estimation applications, which has resulted in its extensive use by PNL, DOE, universities, state governments and private firms for a variety of risk estimation problems.

Mesa State College

Mesa State College (MSC), located within the western slope of Colorado, provides liberal arts and sciences undergraduate education to approximately 4500 students annually. An Associate of Applied Science degree entitled Environmental Restoration Engineering Technology (ERET) was implemented the fall of 1990. This program was intended to address the immediate need of creating technologists who could efficiently resolve the complicated environmental problems which our industrialized nation has created. The ERET degree was initiated by strong interest from the Grand Junction Projects Department of Energy (GJPO DOE) Office and its prime contractor, RUST Geotech.

It was recognized during the development and implementation of the A.A.S. degree that there was an equally critical need for a baccalaureate degree in this general area of study. A Bachelor of Science degree in Environmental Restoration and Waste Management (ERWM) was initiated for the 1993-94 academic year. The ERWM program is an interdisciplinary based degree which fosters an understanding of natural science, mathematics, and communication skills coupled with environmental courses which integrate the support courses in an applied manner. The enrollments within the two degrees were approximately 130 students the fall of 1994.

This relationship with the GJPO DOE office facilitated establishment of several other industry and governmental partnerships. The result has been a quality educational program which is responsive to work-force needs.

EDUCATIONAL PARTNERSHIP ADDRESSES SPECIFIC DOE NEEDS

PNL/MSC Addressing Cooperative Objectives

MEPAS 1.0 was created to estimate risk at DOE site to enable DOE to prioritize the cleanup problems. The tool's most important feature is that it was to apply to many installations and many different sites. The approach chosen by the MEPAS team was unique in the sense that they started with the data that was commonly available at the installation and then created the models to use the data. To make the models general enough for wide spread use semi-analytical approaches were chosen over finite element, this is because the data required for a finite element model is very site specific. The structure of the package was designed (Fig. 1) and experts in specific fields were used to write the actual models.

The MEPAS 2.3 user interface was written to ease the burden of entering data into

text files that are read by the MEPAS models directly. (12) MEPAS 2.3 UI was written in Clipper for dBase III. This interface did ease the burden of data entry but had many drawbacks. Specifically the interface was limited to the MS-DOS operating system and the database structure was spread across many files making sharing of MEPAS data and input difficult if not impossible.

In 1990 MSC began working on enhancements of environmental software to be used for educational purposes on equipment donated from NCR/AT&T. Coincidently Dr. Karl Topper, an environmental professor at MSC, had worked with Dr. Gene Whelan during their doctorate studies. Dr. Whelan was one of the primary developers of the RAPS/MEPAS methodology. This relationship led to an agreement whereby students at MSC would work on programming enhancements of MEPAS and use of MEPAS for educational purposes while PNL would have the rights to the newly developed software. Initially the college funded several students to begin this work. However, this arrangement was unsatisfactory due to limited funds. PNL arranged for summer appointments through the DOE Student Research Associate Program administered by the Northwest College and University Association for Science (NORCUS: This program is now administered by the Associated Western Universities (AWU)). Further funding augmentation was obtained through a DOE Environmental Restoration and Waste Management Distinguished Faculty Award presented to Dr. Topper in 1993. Unfortunately, this two-year award was shortened to one year due to elimination of this valuable program by DOE.

Specific MEPAS Enhancements

Since the MEPAS models were developed as stand-alone models not all of the MEPAS package needed to be modified, only the user interface. During the summer of 1992 PNL assigned five MSC students to tackle the task of the MEPAS UI conversion. At the beginning of the appointment, the students spent a week with the staff at PNL designing a new MEPAS user interface. The remainder of the work was completed at MSC under the direction of Dr. Topper and in coordination with PNL personnel. The new interface used pull-down menus and dialog box methods which reflected commonly available computer software. In addition the design optimized the data handling so that all of the user input would be in one file, making it trivial to share data with other MEPAS users. The new and improved MEPAS UI made converting the units of input values easier for the software developers as well. The new MEPAS package was completed during the summer and was named MEPAS 3.0.

In 1993 Oak Ridge used MEPAS 3.0 to do a risk analysis of the Savannah River Site and validated the usefulness of this undergraduate student product. As a result of this collaboration PNL received a more user-friendly (Fig. 2), more flexible software package and decided to employ one of the students as a staff member. For this joint effort, the MEPAS team received a Federal Laboratory Consortium award for Technology Transfer in 1993.

The new data handling for MEPAS 3.0 made many new products possible. One task that is often difficult for environmental models is a sensitivity or uncertainty analysis of the models. (13) Both require the user to change multiple parameters many times to see the effect of the input changes. The sensitivity analyses allows the modeler to evaluate the degree of accuracy required within the input parameters. MEPAS 3.0's new data handling facilitated development of the sensitivity/uncertainty analysis routine because all that need be modified is the one input file generated by the new MEPAS 3.0.

Four MSC students were selected to start work on developing a MEPAS sensitivity shell during the summer of 1993. The result of this work has been a stand-alone package that can do sensitivity/ uncertainty analysis on the MEPAS models as well as on other models which conform to the data format that MEPAS 3.0 uses. This package used some of the same parts as the MEPAS 3.0 interface so that a user feels like they are running one application.

The MEPAS Sensitivity/Uncertainty Analysis Module is now maintained by PNL staff with the aid of the original MSC developers. It has been used to compute the uncertainty in the Hanford Remedial Action Environmental Impact Statement as well as a poster session on ecological risk estimates, which used a model other than the MEPAS models.

Computational models often produce and analyze large amounts of numerical data which makes it difficult to visualize. This problem, along with the need to produce useful results to decision makers, has now focused the attention of the MSC and PNL partnership on the process of producing report ready charts and tables from MEPAS

3.0 results. This task required the designers to understand MEPAS and be able to work with the same tools that have been used in the past. Once again this task of design and implementation of a Graphics Display user interface for MEPAS was passed to MSC.

During the spring of 1994 three MSC students began working on the Graphics Display for MEPAS. This is the most ambitious task that has been taken on by MSC because it requires the students to develop parts of a much larger application. The MEPAS Graphics Display is not entirely completed yet but significant accomplishments have been realized. MEPAS is no longer a single program of models, it is a collection of tools that allow the user to do a thorough job of estimating risk for contaminants from waste sites. MEPAS allows the user to calculate, graph and assigned a probability distribution to all parts of the calculation (Fig. 3). The majority of the tools in the toolbox have been produced by the MSC and PNL partnership in a very cost-effective manner.

LESSONS LEARNED AND PROJECT BENEFITS

This relationship between PNL and MSC has resulted in a truly symbiotic partnership in which each organization has realized significant benefits. We hope that other DOE facilities and contractors will seek similar partnerships. There are many reasons why there may be some initial reluctance to initiate such activities, but the benefits have far outweighed any inconveniences. The following highlight a few of these benefits:

Efficient, effective development of enhancements to MEPAS which has resulted in greatly improved usability and flexibility.

Using the undergraduate students with faculty and PNL supervision has allowed development and implementation of ideas which might never otherwise be accomplished because of the enthusiastic expertise the students bring to the project along with the extremely low cost of this approach.

Enhancement of the technology transfer mission of DOE through MEPAS improvements and educational use.

Enhancement of the undergraduate student experience through working on a "real" project.

National recognition for the on-going benefits of this project.

This project has also had some constraints which had to be addressed. However, each one of these constraints were readily overcome with sufficient attention. The first touchy issue was protection of rights to the software developed within the project. Yes, the lawyers on both sides had to become involved to help produce a legally acceptable document. The next major issue has been project funding. This has been addressed through the use of student research appointments within the AWU infrastructure and by the DOE research faculty award. Using this approach requires careful attention to deadlines in terms of submitted paperwork. Given that the AWU funding only addresses direct student wages, other means of funding must be sought for this to be a sustainable project. We have signed an MOU which administratively supports the idea that MSC and PNL may seek joint proposals to help seek such augmentation funds in a more stable manner. This is an area which we will be actively seeking as the efficiency of this program is unmatched within the DOE system.

Another point in terms of "lessons learned" is that there needs to be a strong commitment and responsive communication between the partners. We have used a combination of computer communi cations (Internet) to send computer files/messages and telephone conferences for group discussions. However, there is a need to directly fund the supervising staff so that they have the dedicated time to be responsive as needs arise. This leads back to the previously discussed need for on-going funding which is dedicated to this type of program. Thus, it is our recommendation that DOE and its contractors re-evaluate some of their funding priorities to include development of similar partnerships. There needs to be competitive programs which specifically target undergraduate colleges to develop similar projects. The costs are extremely insignificant relative to the benefits.

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7-30
ESTABLISHMENT OF THE ENVIRONMENTAL APPROVAL COUNCIL AT THE WIPP IN AN EFFORT TO MAXIMIZE WASTE MINIMIZATION
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ABSTRACT
The Waste Isolation Pilot Plant (WIPP) is a research and development facility located in Southeastern New Mexico, designed to demonstrate the safe disposal of defense-related transuranic (TRU) waste in bedded salt 655 meters (2,150 feet) underground. Even though the main purpose of the WIPP is to act as a disposal facility, there are still hazardous wastes being generated on site. To keep site generated waste at an absolute minimum, the Westinghouse Waste Isolation Division (WID) established the Environmental Approval Council.
The Environmental Approval Council at the WIPP is comprised of individuals having knowledge of the different types of waste generated on site and the various regulations. Examples of the areas represented on the board include Resource Conservation and Recovery Act, transportation, waste minimization, operations,

safety, fire protection and property management. The intent of the council is to have all applicable organizations present that can identify alternative methods to decrease waste generation.
The board reviews all site documents that are associated with waste generation activities. When a waste generating activity is identified, a Waste Stream Approval Form is initiated. The board meets twice a week to review waste streams and determine the ultimate fate of the waste. The packaging, characterization, substitutions, recycling possibilities, and the disposal facility are determined before the waste generating activity begins. Also, the Waste Stream Approval Form will be used as a record to document every waste stream on site.
This process enables the WIPP to reduce waste generation, increase recycling activities, and maintain control over waste streams at the site. Due to the establishment of the Environmental Approval Council and its processes, the decrease in cost associated with waste disposal is significant.
This paper describes the benefits of a waste management system and the process for establishing such a system. It highlights the cost avoidances that can occur and the increased site control over generated waste.

INTRODUCTION

The Department of Energy (DOE) Waste Isolation Pilot Plant is a mined geologic repository, located in Southeastern New Mexico, designed to demonstrate the safe disposal of defense related transuranic (TRU) waste in bedded salt 2,150 feet underground.
Even though the disposal stage has not started yet, there is still hazardous waste being generated on site. The Environmental Approval Council (EAC) was established to address the need of documenting and managing the waste streams on site. This need was met by incorporating the EAC approval into the review process already in place. Through this incorporation, the EAC approval only extended the original process; therefore, an additional burden was not put on the customer. Improving customer satisfaction and soliciting pertinent information for managing waste streams were the drivers for establishing the council.
The council met some resistance in the beginning stages but gained support based on the idea that it would not only facilitate the approval process but also eliminate the confusion over the required sign-offs. The need to eliminate this confusion was accomplished even though the underlying focus continued to be awareness and control over waste generation. In the end, this approval process has proven to be a useful tool in documenting waste streams and maximizing waste minimization.

BACKGROUND

The need for more control over waste-generating activities arose when the hazardous waste being shipped for disposal was examined. It was apparent that the organizations on site generating the waste were not conscious of the process of waste disposal and associated costs. A system of examining waste before generation was needed.
Every activity that was performed on site, whether it was routine maintenance or a major construction project, was approved by definite areas of expertise within the Environment, Safety, Health and Regulatory Compliance (ESH&RC) Department. It was often difficult for customers of ESH&RC to obtain all required signatures in one trip to the department. In addition to this problem, the ESH&RC representatives approving each process were not sharing information with each other, creating the duplication of effort in some instances. Also, minor projects were required to obtain the same levels of approval as large projects, therefore, creating long turnaround time. This system obviously needed to be refined.
The EAC was established to meet the needs of the customer and incorporate the needs of the department. This council meets twice a week to review all work being done on site to approve and document the waste streams being generated. This allows the waste generation information to be collected early in the process, therefore, giving more time to characterize, sample, or identify recycling/reuse opportunities. The establishment of the EAC provided a focal point for other departments needing waste stream approval and allowed the council members to meet and discuss the various issues of each activity. This eliminated the duplication of effort by the individuals approving each process.

PROCESS

The Environmental Approval Council is managed by the Council Facilitator (CF) and the Deputy Facilitator (DF). Each project that is to begin on site is routed through

the CF, usually in the form of Engineering Change Orders (ECOs) or Plant Work Requests (PWRs). At this time, the CF initiates the Waste Stream Approval Form. This form documents the different areas that will be affected and the waste streams that will be generated. The CF logs the project onto the tracking database and completes as much of the Waste Stream Approval Form as possible. At this time, each project is rated according to size and importance and is split into one of the following categories: 1 - minor, every day activity; 2 - emergency situation; 3 - large, substantial activity.

For minor activities (1), the waste stream is documented and the Waste Stream Approval Form is routed to obtain the correct signature approval in a timely manner. Each activity is documented so that if the same project is encountered again, it will have instant approval because of the record of approval from the prior activity.

Emergency situations (2) are similar to minor activities. Since there is not enough time to hold up the approval process for concurrence at the bi-weekly meeting, the activity is documented and approval is obtained quickly. These projects are still reviewed for waste characterization and waste minimization opportunities, but source reduction cannot be evaluated because of the quick turnaround required.

Large, substantial projects (3) are logged into the database and the Waste Stream Approval Form is started. These forms are collected by the DF and summarized in a listing that is distributed the day before each of the bi-weekly meetings. This allows the council members adequate time to review the upcoming projects prior to the actual meeting to examine past problems or new methods to reduce the amount of waste generated. If the council member(s) does not have any problems with the project, an approval can be granted prior to the meeting thereby eliminating the need for that council member to attend.

The EAC meetings are held on Wednesday and Friday mornings. At the EAC meeting the proposed project is presented and discussed. If there are any concerns from the involved parties, they are discussed in the meeting. The person cognizant of the waste generation activity is in attendance to answer any questions that arise. If there are changes to the process that are decided in the EAC meeting, they will be documented. The EAC has the authority to block an activity until the process has been approved by the council. The approval signatures are documented, and the work order is released at that time. The maximum turnaround time for large, substantial projects is 2 1/2 working days.

BENEFITS

The establishment of the EAC at the WIPP has given the ESH&RC Department better control over waste streams. The EAC allows better management of waste being generated on site by reviewing processes prior to generation to incorporate source reduction techniques. It also allows advance notice to effectively characterize, sample, or recycle/reuse the waste streams generated. By documenting the activities in a database, the ESH&RC Department has up-to-date information to facilitate the process for preparing various environmental reports that must be submitted regarding waste generation amounts.

Another benefit from the EAC is being able to provide the site with a more thorough and timely ESH&RC review. The EAC has also increased awareness of the role of ESH&RC and of waste disposal at WIPP.

CONCLUSION

By incorporating the needs of the department and our customers, the EAC has brought about the necessary control on waste streams, while enhancing the departmental review process that was previously in place. The marriage between the need of the customer and the need of the department eased the incorporation of the waste stream approval process. This program has proven to be a creative and effective tool in managing the waste streams at the WIPP site.

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WASTE MANAGEMENT AUTHORITY AND THE WASTE STREAM APPROVAL PROCESS

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ABSTRACT

Early in the spring of 1993 a program was developed at the Idaho National Engineering Laboratory to promote excellence in the area of waste management. The intent of the program was to formalize the idea of prohibiting waste generation

without sufficient preplanning and approval. In support of this program, a Waste Management Authority (WMA) was initiated at the DOE's Idaho Chemical Processing Plant (ICPP). Development of the WMA has provided an effective method of implementing waste management technologies and techniques to improve waste management operations.

The WMA is comprised of technical representatives from various organizations and disciplines at the ICPP. Together as a team, the WMA has a wide variety of responsibilities in the area of waste management, including minimizing waste, identifying new regulatory issues, ensuring that appropriate treatment technology development/requirements are addressed, reducing waste management cost through planning, communicating lessons learned, improved reporting accuracy, and enhancing communication between facility operations, waste management, and environmental personnel.

The purpose of the WMA is to review and approve/disapprove mixed, hazardous, radioactive, and liquid industrial waste streams prior to their generation. It's primary focus is to ensure that wastes are identified, treated, stored, and disposed in compliance with all applicable regulations. The program is also designed to ensure that waste streams are eliminated/minimized, or recycled/reused when possible.

Prior to waste generation, generators are required to complete a waste profile which identifies the waste generating activity; planned waste composition, pollution prevention/waste minimization activities, sample and analysis data, sampling requirements, and treatment, storage, and disposal options. This information is then submitted to the WMA and a interactive review meeting is conducted with the waste generators. This gives everyone involved an opportunity discuss any issues and resolve comments.

Once a waste stream is approved, the waste profile information is entered into a waste management database. Much of the data gathered is used to support various other reporting databases. Additionally , information in the database is used to evaluate any newly proposed waste streams. This enables the WMA to identify previously approved technologies and techniques, thus eliminating any redundancies in sampling, treatment, etc.

The WMA concept and the waste stream approval process can be used at any facility and was piloted at the ICPP with the intent of being used at other sites in the DOE complex.

BACKGROUND

Industry, in general terms, is a generator of numerous waste types which require proper handling and subsequent disposal. Wastes must be generated in many instances to carry out company objectives. Strategically linked to this waste generation process, however, are company waste minimization objectives, process efficiency goals, cost savings objectives, waste predictions/tracking, safety concerns, planning, compliance and reporting requirements for various environmental agencies. All facets of these activities require integration and coordination to successfully accomplish them while maintaining cost effective compliance.

Traditional environmental organizations at generator facilities manage waste handling, disposal and compliance. However, these organizations are not necessarily responsible for waste minimization or cost savings of process related functions. Although conceptually not a new or unique way to account for waste, environmental organizations can provide these and other beneficial functions through planning, forethought and a matrix to evaluate and maintain data. Such is the case with the WMA.

Lockheed Idaho Technologies Company (LITCO) recently refined the basic concepts behind such an organization and developed a multi-departmental Waste Management Authority (WMA). Development of the WMA formalized a long-standing policy of prohibiting waste generation without sufficient preplanning and approval. In support of this policy and to improve its effective implementation, the WMA program was developed with the vision of establishing a more comprehensive approach to waste management.

The WMA is based on benchmarked information from NASA and a number of Government Owned/Contractor Operated (GOCO) sites, with the underlying theory that anything that comes into an industrial site must be disposed of, or be discharged in one form or another as a product, solid or hazardous waste, or emission. In the case at the Idaho Chemical Processing Plant (ICPP) and other GOCOs, this also includes

radioactive and mixed waste. Therefore, the final form, regardless of its destiny, should be qualified and quantified to the extent possible during the planning stages to make certain that company objectives are met while ensuring compliance throughout the material life cycle. Of course this is not always possible, for example, in the case of remediation waste, but can be accommodated within any planning forum.

WMA IMPLEMENTATION

In the spring of 1993, work began on the pilot program that resulted in the development of a team of topical experts from throughout the ICPP entitled the Waste Management Authority. This multi-departmental team is comprised of technical experts representing research & development analytical chemistry environmental compliance, facility operations, and waste management organizations. The first six months of the program was devoted to the development of company procedures and forms and establishing a six month implementation schedule. Because of the number and complexity of specific waste streams at the ICPP, it was determined that a "phased in" approach would be necessary to start the review process. Initially, the WMA evaluated mixed waste followed by hazardous, radioactive, and liquid industrial waste streams that were currently being generated. Then they began looking at newly proposed waste streams prior to their generation. The complete process from the development of procedures to full implementation of the WMA took approximately twelve months. During that time, the WMA reviewed and approved 150 individual waste streams. Subsequently, annual reviews and approvals are performed for all of the waste streams.

WMA ELEMENTS

The overall philosophy of the WMA is that waste generators provide information about their waste stream prior to generation so that a critical review can be made as to the appropriateness of generating the waste. The primary function of the WMA is to ensure that these waste streams are controlled, minimized, and handled in compliance with applicable laws, rules and regulations. In order to accomplish this function a comprehensive review by the WMA is performed on various information provided by the generator. A waste profile of the generation process is submitted that includes the following fundamental elements: 1) Waste generating process description, 2) waste minimization activity or pollution prevention opportunity assessment, 3) hazardous materials used in the process, 4) physical and chemical characteristics, 5) anticipated volume, 6) hazardous and/or radioactive characteristics, 7) waste characterization information including analytical data or process knowledge, 8) waste disposal destination, 9) waste handling and packaging information, and 10) applicable treatment capability. By providing detailed information in the planning stages of a waste generation activity, the WMA is able to assess all of the options available and make recommendations that LITCO's environmental objectives are met.

LESSONS LEARNED

Since implementation of the WMA, communication between generators and the environmental/ waste management infrastructure has improved significantly. The information required for various reports, i.e. Idaho Hazardous Waste Quarterly Report, Radioactive Waste Management Information System, Federal Facilitities Compliance Act, etc., has greatly improved because the identification and quantification of current and future waste streams is gathered and reviewed in a consistent manner. Operational safety has improved in the area of chemical compatibility of waste treated at the ICPP or sent off-site for treatment and disposal. Through the waste stream approval process new waste streams have been identified that require additional treatment technologies and these waste streams have been presented to management in a more timely manner as a result of the WMA. Enhancements to technology transfer for waste minimization,
process development, and various other critical aspects to waste management have all been centralized in a database for consistent reporting to EPA, State, DOE and other regulatory agencies that inspect the ICPP.

THE FUTURE

Goals of any environmental program such as the WMA follow the obvious; to track waste, maintain on-site compliance, ensure off-site approvals are in place for disposal, maintain accurate records for required reporting and many more regulatory aspects. But many of the varied waste streams that exist in the DOE complex make achieving these goals very difficult. In addition there is a growing trend at the state and federal levels toward increased regulations of solid and hazardous waste. As such, LITCO's pilot program, the WMA, has demonstrated that it is an excellent

vehicle for keeping up with the constant changes in world of waste management. By establishing the WMA, LITCO has developed a unique approach to managing waste that keeps pace with their vision of environmental leadership. The success of the WMA pilot program has led LITCO to expand the program to other facilities at the Idaho National Engineering Laboratory (INEL) under the direction of LITCO and to share the concept with other GOCOs within the DOE complex.

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COMPUTATIONAL FRAMEWORK AND DATABASE FOR ACCIDENT ANALYSES OF WASTE MANAGEMENT ALTERNATIVES IN THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAM*

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ABSTRACT

A computational framework and integrated PC-based database have been developed by Argonne National Laboratory (ANL) to assess atmospheric releases from facility accidents in support of the U.S. Department of Energy Environmental Management Programmatic Environmental Impact Statement (EM PEIS). One objective of the EM PEIS project is to provide an automated capability for performing the large number of calculations required in the accident analysis. These calculations are required for evaluating the relative risk of the many combinations of process technologies, facility selections, and site consolidation strategies in the EM PEIS alternatives for each waste type.

The computational framework developed accesses several relational databases that contain information used to determine the risk-dominant accidents for each waste stream on the basis of throughput volume, radionuclide characteristics, treatment process, and other data needed to calculate releases. This paper describes the computer framework and supportive database used to conduct accident analyses and assess potential atmospheric releases that may affect on-site workers and off-site members of the public.

INTRODUCTION

A PC-based computer code was used to develop source terms for operational accidents and for externally initiated accidents such as airplane impacts or natural phenomena. Both radiological and chemical source terms can be evaluated; however, the following description pertains only to calculation of radiological source terms. A radiological source term is defined as the amount (in curies) of each radionuclide released during an accident, where release is assumed to be instantaneous. The code uses unit-risk factors, defined to be the consequence associated with a unit release of a radionuclide from a given facility and site, obtained from Oak Ridge National Laboratory (ORNL) to screen accident sequences for risk dominance. It allows risk-based comparisons of EM PEIS strategies for consolidating the retrieval, storage, and treatment of wastes at DOE sites throughout the country. The database incorporated into the code includes a radionuclide inventory, waste generation rate, physical and chemical characteristics of the waste streams, associated treatment process throughput, amount of waste at various components of the treatment facility that can be affected in an accident, accident frequencies, and other pertinent parameters needed to calculate chemical and radiological releases (1,2). This paper describes the computational framework and supportive database used to conduct accident analyses and assess potential atmospheric releases that may affect on-site workers and off-site members of the public.

METHODOLOGY AND COMPUTATIONAL IMPLEMENTATION

Methodology and Modules Description

This section describes the methodology and computational framework for the facility accident analysis for the EM PEIS. Figure 1 provides an overview of the accident analysis system and its interaction with other portions of the EM PEIS project. As shown in Fig. 1, the accident analysis system performs three major tasks: 1) prescreening for risk-dominant sites, facilities, and process options; 2) development and frequency estimation of accident sequences; and 3) development of

radiological source term information for accident sequences. Implementation of this analysis included selection and development of the risk-dominant accident sequences (determined by considering both the likelihood and severity of plausible accidents) and generation of the source term output.

The radiological source term associated with each accident is the product of five factors that vary for each radionuclide within the inventory affected by the accident.

Source Term = MAR DF ARF RF LPF (1)

where

MAR = material at risk,
DF = damage fraction,
ARF = airborne release fraction,
RF = respirable fraction, and
LPF = leak path factor.

MAR is the total inventory of waste in a facility with the potential of being impacted, and the DF is that portion of MAR involved in a particular accident sequence. MAR and DF are functions of the process, the facility, and the accident initiator. The variable ARF refers to the fraction of the potentially available radionuclide inventory rendered airborne as a result of the accident. ARF is a function of the original composition of the waste, the physical properties of the radionuclides, and the accident mechanisms. The respirable fraction of the airborne release is RF. The product of ARF and RF, referred to as the respirable airborne release fraction or RARF, therefore, represents that portion of the radionuclide inventory that is released and respirable. The leak path factor (LPF) is the fraction of the airborne inventory that passes through the containment barriers and filters to escape into the atmosphere. The LPF is a function of the physical form of the nuclide being released and the integrity of the containment systems. The source terms were then combined with site-specific unit-risk factors developed by ORNL and the frequency of occurrence estimates for the accident sequences to establish the reference risk-dominant accidents for more rigorous evaluation. Overall accident health effects will be calculated by ORNL.

Several integrated modules have been developed to determine the risk-dominant accidents for each waste stream on the basis of throughput volumes and radionuclide characteristics. Three classes of accidents are considered in the accident analysis for each waste type: 1) general handling accidents, 2) accidents at storage facilities, and 3) accidents involving treatment processes and facilities. The calculation of MAR depends on the class of accident; the MAR for a general handling accident is given by (assuming that the waste is contained in a 55-gal drum with a volume of 0.2 m3) the following equation:

MARgeneral handling = CONCi 0.2 (2)

where

CONCi = concentration of radionuclide "i" in treatment stream (Ci/m3).

The calculation for a treatment facility took the following form:

MARtreatment facility = TR CONCi TAO (3)

where

TR = treatment throughput rate (m3/yr), and
TAO = average residence time of MAR in treatment facility (yr).

The treatment throughput rate and radionuclide concentration were calculated by the WASTE_MGMT computational model (1) and are a function of DOE site, treatment technology, and alternative siting configuration. The values of TAO were developed as a function of treatment technology (where incineration is an example of a treatment technology) from available safety documentation. The calculation of MAR for a storage facility was estimated by the following equation:

MARstorage facility = CONCi SC (4)

where

SC = generic storage facility size (m3), which was generally taken to be 2,000 m3 (equivalent to about 10,000 drums).

An event-tree approach was used to calculate the probability associated with particular accident initiators and event sequences for each site. Generic event trees were used for the analysis, although the model includes an option allowing the user to specify an alternative tree to reflect more precise site-specific data.

All the main components of the accident analysis methodology have been programmed into a menu-driven FoxPro 2.5 database system capable of processing any of five

waste streams (HLLW, TRUW, LLW, LLMW, and environmental restoration [ER]) under the various alternatives. The programming challenge was to develop a user-friendly, PC-based system capable of processing large amounts of data with the flexibility to accommodate various alternatives, waste streams, and site-specific information. To make the data more manageable, tasks were divided among several different modules. The accident analysis modules are as follows:

Event Tree develops accident tree and assigns probability to each event sequence.

Material at Risk calculates MAR from waste management output and is a function of waste stream, physical form, treatability category, radionuclide inventory, and treatment throughput.

Risk Calculation links MAR, damage rate, and accident probability information with atmospheric release and unit-risk factors to obtain consequences for screening risk-dominant accidents.

Risk-Dominant Sequences identifies the highest risk accident initiators and event sequence for each site.

Source Term Output automatically creates source term files for ORNL based on site risk-dominant sequences.

Figure 2 shows the databases and data flow for the accident analysis methodology. The output from the WASTE_MGMT computational model was used as the accident analysis input. To predict the volumes and radionuclide concentrations of the MAR for the various cases, a data preprocessing methodology was applied. The methodology outlined in the previous section is generally applicable to all radiologically contaminated waste streams analyzed within the EM PEIS: HLW, TRUW, LLW, and LLMW. The MAR is then linked with the appropriate unit-risk conversion factor, which is a function of site, radionuclide, and type of receptor.

A spectrum of accident scenarios that could occur during treatment, storage, or handling activities with the chosen waste stream were developed on the basis of the waste's physical characteristics and type of activity. The product of the MAR and the unit dose conversion factor was then combined with the ARF, RF, and LPF, which are a function of release class, accident category, and radionuclide. Preliminary health effects estimates of the accident consequence were derived by including the DF for the various accident sequences. The conditional probabilities of the individual accident sequences were combined with the initiator probability to become the annual sequence probability. The risk for each accident sequence and site was simply calculated from the consequence and annual sequence probability. Determination of the risk-dominant sequence resulted in the source term generation for the associated releases. These source terms were transmitted to ORNL for the final health effects calculations.

Input Files

The accident analysis model requires a number of databases to analyze a case: the MAR file, a file containing the accident initiator frequencies, an accident scenario file, and a series of databases containing information on radionuclide, release, and dispersion characteristics. Figure 3 shows abbreviated layouts of the various database files. The shaded areas in Fig. 3 indicate key fields that, when combined, would be unique to each record and are used to link or retrieve associated data in the other files.

Fig. 3

SAMPLE ACCIDENT ANALYSIS CALCULATION

To demonstrate the interaction of the modules, a sample calculation (Table I and Table II) is presented below along with a chart of the data flows (Fig. 2). The example shown involves a general handling accident involving one 55-gal drum containing ER-generated LLW. General handling accidents are expected to dominate the radiological and chemical risk to the work force because of the close proximity of the workers to any release. This class of accidents includes drops and spill of a container or punctures by a forklift. Tables I and II provide example output files generated for a LLW case that includes ER-generated waste. Table I is part of a data file that presents the calculation of the product of MAR, ARF, LPF, and the unit-risk conversion factor; estimation of the consequence would require the further multiplication of this product with the DF. Determination of the sequence probabilities for the different accident scenarios is shown in Table II as a function of waste- processing activity and accident release class. The risk for a given accident sequence is the product of the consequence and the annual sequence

probability.
FUTURE DEVELOPMENT
Although much has been accomplished in terms of automating the accident the analysis, additional work is ongoing in optimizing the computational routine and automating graphics capabilities to further facilitate the evaluation of waste management alternatives across waste streams.
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7-35
DEVELOPMENT OF A DECISION SOFTWARE PROGRAM FOR RADIOLOGICAL EVALUATION DECISION INPUT (REDI)
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ABSTRACT
In December 1992, the Department of Energy (DOE) implemented the DOE Radiological Control Manual (1) (RCM). Westinghouse Idaho Nuclear Company, Inc. (WINCO) submitted an implementation plan showing how compliance with the manual would be achieved. This implementation plan was approved by DOE in November 1992. Although WINCO had already been working under a similar Westinghouse RCM, the DOE RCM (1) brought some new and challenging requirements. One such requirement was that of having procedure writers and job planners create the radiological input in work control procedures. Until this time, that information was being provided by radiological engineering or a radiation safety representative. As a result of this requirement, The Idaho National Engineering Laboratory developed the Radiological Evaluation Decision Input (REDI) program.
INTRODUCTION
During a March 1993 Defense Nuclear Facility Safety Board (DNFSB) visit to the Idaho Chemical Processing Plant (ICPP), WINCO was questioned about how the expertise of other Westinghouse organizations was being utilized in the development of programs required by the DOE RCM (1). Although some examples were given, WINCO felt that more involvement on a corporate-wide level could provide great savings throughout the Westinghouse organization. Soon after that visit, a corporate-wide committee was organized to develop a program for the development of Radiological Work Packages.
WesTIP Team
Because the DNFSB had questioned the use of Westinghouse expertise from other sites in RCM (1) implementation, WINCO officials began to review items which were to be implemented in the near future for the possibility of corporate involvement. The Radiological Work Package process was a very good possibility, and was eventually chosen for corporate committee review.
WINCO sponsored the committee known as the Westinghouse Technologies to Improve Processes (WesTIP) Combined Team Review. The goal of the committee was to reduce the costs and time associated with development of Radiological Work Packages while improving quality and consistency. WINCO, West Valley Nuclear Services (WVNS), Westinghouse Savannah River Company (WSRC), and Westinghouse Hanford Company (WHC) chose to participate in the project, and selected members with appropriate expertise for the team. This committee then met in Pittsburgh, Pennsylvania to begin learning the WesTIP Process and begin applying it to the creation of radiological work packages.
The first step in the WesTIP Process was to define the current process being used. To do this, the committee members outlined the process being used by their

individual facilities and then the processes were combined to get an overview of the basic process being used within the Westinghouse Complex. Problems associated with each step were discussed, and those which were deemed to be significant were then listed on the flow chart model of the process.

Once the overview was completed, each step in the process was reviewed to see how much time it took and how much it cost to complete. Costs for individual steps ranged from $25 to $1,625 and time spent ranged from 0.1 day to 5 days per step. Those steps which cost the most or took the most time were then highlighted on the flow chart and reviewed to see where reductions could be made.

Findings revealed that it was taking an average of 37 days and costing an average of $6,875 to complete one radiological work package. Using the WesTIP technique, the committee developed a plan which would allow a radiological work package to be completed in 7 days and cost $2,480. This established a time reduction of 30 days and a cost savings of $4,395 per work package (Fig. 1). Because of the number of work packages created in a year, the committee determined that significant yearly savings would be realized through the use of the program they proposed. A key point to implementation of the new program was the need for a risk-based decision tree process for creating and completing radiological work packages.

The committee discovered that gathering data from various resources was the largest consumer of both time and money. Much of the data gathered was standard information which was recreated every time a radiological work package was needed. This collection of information was spread over several different steps in the original process.

With the decision made that the goal of the group would be to reduce time and costs associated with gathering information and writing work packages, the work of improving the process could begin. Several changes to the process were proposed, including creating "points of contact" who would function as area experts and creating a decision tree program which would eliminate recreation of information for every work package.

The committee decided that approximately 80 percent of the information recreated every time a radiological work package was developed could be placed into a decision tree software, thus eliminating the need for recreation of that information. This information included Radiological Control Manual (1) requirements, Federal Regulations, common work standards and policies, and local procedural requirements.

Development of REDI

Once the WINCO representatives returned from Pittsburgh, they began to explore hardware/software resources and availability. They made the determination that DClass, a commercially available software, would be used to develop their decision tree. This determination was based upon applicability, adaptability, and cost. Because DClass was already being used by WINCO personnel who would be doing the programming, a significant cost savings was realized.

The first step in development of the REDI program was to produce a risk-based decision tree which could be input into DClass. To accomplish this, a number of radiological packages were reviewed to determine the questions that must be answered in order to create a valid work package. The answers to these questions were not the same in all cases, and were dependent upon the specifics of the work to be completed. With this in mind, multiple choices were designed to cover all probable answers to a question. These choices correspond with precise output devised to be used for final work package details.

Prior to the development of the REDI program, planners and procedure writers would normally solicit information from a number of resources. With the use of the REDI program, the information could be automatically compiled by simply answering area-specific questions. By answering such questions, the requirements could be narrowed for the specific job to be performed. The program was to be designed to automatically create radiological input containing the appropriate guidelines. This input would then be available for use in creating radiological work package portions of procedures.

The REDI program asks the user multiple choice questions. Based upon the answers to those questions, appropriate information is placed into a radiological input package (Fig. 2).

Features of REDI

One of the main features of the REDI program is the capability it offers for electronic review and approval of radiological work packages. REDI is installed on a

network with access by all individuals in the review and approval process. When the package is ready for review, it is electronically transferred. When the appropriate personnel have reviewed the package, they enter a password which allows the package to be sent back to the originator with comments or approval. If changes are made to the package, prior approvals are voided and the package is retransmitted for approvals. Any forms associated with the work evolution such as radiological work permits can also be attached and forwarded electronically.
The approved package is returned to the document originator who then places the package in the work control procedure for use in the field. The REDI cycle is not yet complete; however. Post-job critique information is
entered as "lessons learned" before the work package can be closed out and considered complete.
All packages are stored electronically. Each package has its own unique file name which allows retrieval for review or modification at any time. For future referencing, packages can be retrieved using the file name assigned to the package. The information contained in this package can be used for future work packages, either in its entirety or by removing pertinent sections. "Lessons learned" will be of great value when planning for future work. Users can refer to previously developed radiological work packages to retrieve pertinent information for the development of the current work package.
CONCLUSION
Development of REDI is an ongoing process. It is currently being used at WINCO, and will be released to other Westinghouse GOCOs when the initial testing phase is complete. It will be customized for each facility, and will likely be released for other DOE facilities shortly after delivery to Westinghouse.
Effective October 1, 1994, WINCO is no longer a contractor at the ICPP. Lockheed Idaho Technologies Company (LITCO) has assumed the contract for the entire Idaho National Engineering Laboratory (INEL), including the ICPP. All future development of the REDI program will be under the auspices of LITCO.
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INCINERATION OF LOW LEVEL RADIOACTIVE VEGETATION FOR WASTE VOLUME REDUCTION
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ABSTRACT
The purpose behind DOE's changing mission at the Savannah River Site (SRS) includes increasing activities for Waste Management and Environmental Restoration. At SRS there are a number of Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) locations that are contaminated with radioactive materials, support dense vegetation, and are targeted for remediation. Two such locations have been studied for non-time critical removal actions under the National Contingency Plan (NCP). Both of these sites support about 23 plant species. Surveys of the vegetation show that radiation emanates mainly from vines, shrubs, and trees and range from 20,000-200,000 disintegration / minute (d/m) beta gamma.
Planning for the removal and disposal of low-level radioactive vegetation was done with two principal goals: to process contaminated vegetation for optimum volume reduction and waste minimization; and for the protection of human health and environment. Four alternatives were identified as candidates for vegetation removal and disposal; chipping the vegetation and packing in carbon steel boxes (lined with synthetic commercial liners) and disposal at the Solid Waste Disposal Facility at SRS, composting the vegetation, burning the vegetation in the field, and incinerating the vegetation. One alternative, incineration, was considered a viable choice for waste volume reduction, safe handling, and the protection of the environment and human health. Advantages and disadvantages of all four alternatives have been evaluated.
For waste volume reduction and the ultimate disposal of radioactive vegetation, incineration is the preferred method. Advantages of incineration are that volume

reduction is achieved and low-level radioactive waste is stabilized. For incineration and final disposal, vegetation will be chipped and packed in cardboard boxes and discharged to the rotary kiln of the incinerator. The slow rotation and long resident time in the kiln will ensure complete combustion of the vegetative material. The ash from the incinerator will be solidified with cement and rendered immobile. Solidified ash will control the potential contaminant pathways for the protection of human health and environment.

INTRODUCTION

At the Savannah River Site (SRS) there are a number of sites that have been contaminated with radioactive materials. Two such sites have been placed in the Resource Conservation and Recovery Act (RCRA) Facility Investigations/CERCLA Remedial Investigations (RFI/RI) Program. These sites are to be closed. The first part of closure implementation is to remove radioactive vegetation from the project sites.

The objective of this study was to develop planning strategies for radioactive vegetation removal, environmentally safe disposal, and to achieve maximum waste volume reduction. Three alternatives for vegetation removal and four options for vegetation processing were considered during the study. For this study, an analysis was performed on the type and volume of vegetation to be removed, level of radioactivity present in the vegetation, and options for vegetation removal and disposal.

LEVEL OF RADIOACTIVITY IN VEGETATION

The Health Protection Department (HP) at SRS has been periodically recording the rates and concentration of radioactivity in vegetation since the 1970's.1 The range recorded by HP is contained in Table I. A probe survey conducted on vegetation showed that the radiation levels ranged from 20,000-40,000 d/m beta/gamma and came mainly from vines and shrubs. Radiological survey observations made of a sweet gum tree showed a radiation level of 200,000 d/m beta/gamma, the trunk of a tree showed a radiation level of 40-60 mR/hr at 5 cm above the ground. The tree under observation was about 12-13 cm in diameter, and broken limbs from the tree showed a radiation level of 5 mR/hr.

TYPE OF VEGETATION

Both project sites support dense vegetation. About 23 types of trees, shrubs, and grasses were observed growing within the boundaries of the project sites.2 Trees are of mixed heights and girth; many appear to have reached a height of about 12 m with a breast-height diameter of about 30-36 cm. Additionally, there are a number of young trees that have attained breast height The majority of shrubs are in the range of about 1-2 m in height.

VOLUME OF VEGETATION

The volume of fresh vegetation to be removed is presented in Table II. This table shows that the estimated vegetation to be ranges from 375-495 m3.2

VEGETATION REMOVAL ACTIONS

According to the Federal Facility Agreement (FFA)3, the two project sites under study must be assessed and remediated. Removal alternatives for these sites should be consistent with nine CERCLA remedial evaluation criteria.4 Closure options for these sites are to be environmentally sound. To establish a range of potential consequences and funding requirements for the chosen removal alternative of these two sites, the following three alternatives were examined for vegetation removal:

- no action (no removal of vegetation)
- cutting of vegetation and no removal
- cutting of all vegetation and removal

Alternative 1 - No Action

For this alternative, the vegetation at the project sites would not be cut or removed. The vegetation would continue to grow and multiply, and presumably continue to absorb radionuclides from the soil. Radioactive substances would continue to migrate along contaminant pathways to potential receptors. The potential impact to human health and the environment would continue unabated with the possible exposure of wildlife and the surrounding ecosystem from contaminated vegetation.

Exposure of radioactive vegetation to the public's health could occur as a result of transportation via several pathways. The potential contaminant pathways of concern are atmospheric, groundwater, surface water, and fugitive dry vegetative material. The potential contaminant exposure routes are ingestion, inhalation, and dermal contact. As a result of this alternative, the contaminant pathways and exposure

routes would remain active and continue to cause adverse impacts on the environment.
Alternative 2 - Cutting of Vegetation and No Removal
This alternative involves cutting down the vegetation and leaving it on the ground. As a result, there will be no removal of the contaminated vegetation. This action will continue to impact the human health, the environment, and wildlife. Long-term remedial actions (close the site and construct a cap system) could be hindered because of interference from the large volume of dead vegetation lying on the ground. Contaminant pathways will remain active and unobstructed and that would cause the possible migration of radioactive substances off the site.
Dead vegetation lying at the ground, within the project boundaries, would biodegrade. As a result, biodegraded vegetation would reintroduce the radionuclides to the environment, either through wind erosion of vegetative materials or with surface water run-off (until the site is closed and a cover system installed). This could have a significant detrimental impact on the ecology of water bodies located in the near vicinity of the project sites.
The integrity and performance of the closure cap/cover system installed without removing vegetation could suffer adversely. Over a period of time, unremoved vegetation would undergo volume reduction due to biodegradation and could cause a serious threat to the performance of the cover system including localized subsidence, settlements, or complete failure of the cover system. Failure of the closure cover system would eventually reintroduce contamination to the environment.
Alternative 3 - Cutting of All Vegetation and Removal
This alternative processes the vegetation and removes it entirely from the project sites. This alternative will reduce and limit the likelihood of human and wildlife exposure to radioactive substances, as the contaminated medium would be physically removed, and alter and control the potential contaminated pathways. Removing the vegetation should also make the implementation of long-term remedial actions, cap construction, feasible.
PLANNING FOR VEGETATION REMOVAL, WASTE REDUCTION, AND DISPOSAL
The design analysis required consideration of each of the following removal and vegetation processing management practices; burying the removed vegetation at the Solid Waste Disposal Facility, open burning of vegetation, composting of vegetative mass, and incineration of vegetation.
Handling of Removed Radioactive Vegetation
Large trees, shrubs, small saplings, and other ground cover removed with the aid of mechanical equipment must be chipped and placed in 0.54 m cubed (21 inches cubed) cardboard boxes for incineration, and/or lined carbon steel boxes (2.8 m3 volume [96 cu ft.]), for storage at the Solid Waste Disposal Facility. This practice will ensure safe storage and efficient handling of removed radioactive vegetation until its final destination for disposal is established.
Vegetation Processing for Waste Reduction and Disposal
The following four options were considered for processing radioactive contaminated vegetation for final disposal:
Disposal at the Solid Waste Disposal Facility: Remove all chipped vegetation from the project sites in lined- carbon steel boxes (approximately 130170 boxes) and transport them to the Solid Waste Disposal Facility at SRS. The disadvantage of this option is that considerable space will be required at the Solid Waste Disposal Facility for the storage of about 130170 boxes. Additionally, the vegetative mass over the period of time will decompose and reduce in volume, thus creating additional void space in boxes.
Composting the Vegetation: Composting the vegetation from the project sites could be done under controlled conditions. However, the composted vegetation would not be suitable for soil amelioration as it will still be contaminated with radioactive substances, and subsequently contribute contamination to the environment. Therefore, composting at SRS is not recommended because of its foreseeable detrimental impacts to the environment.
Open Burning of the Vegetation: Burning of radiological contaminated vegetation is unacceptable due to the high potential for an uncontrolled release of radioactive substances to the environment. Because of environmental consequences, this option for vegetation processing is not recommended.
Incineration of Vegetation: Based on a cost analysis, it is still to be decided whether incineration of vegetation is done either at a commercially licensed incinerator or at the Consolidated Incineration Facility (CIF) at SRS. The CIF,

located at SRS, is currently under construction with operations scheduled in February 1996. The facility is designed to treat mixed wastes containing both hazardous and radioactive contaminants. The actual waste streams designated for treatment using CIF are currently being evaluated under the National Environmental Policy Act (NEPA).5 If the CIF is selected as the acceptable option for treatment of non-hazardous, low-level combustible waste, then the CIF can be used to treat radioactive vegetation. The advantages of incineration are a substantial reduction of vegetation volume, and better control of contaminant pathways leading to the environment.

Incineration Process: The chipped vegetation from the project area will be packed in 0.54 m cubed (21 inches cubed) disposal cardboard boxes. These boxes are discharged into the rotary kiln incinerator by a ram-feeder assembly. The kiln operating temperature ranges from 760-870oC. The slow rotation of the kiln promotes thorough combustion of the waste material. Ash drops from the discharge end of the kiln into a water-filled ash collection tank. This ash is scooped from the collection tank by a backhoe and dumped into a 55 gallons (208 l) drum until it is partially filled. Cement is then added and mixed with the ash to form a stabilized waste matrix to meet RCRA's Land Disposal Restriction (LDR) treatment standards.6 Solidified ash drums (cured drums) are placed in onsite-concrete vaults for final disposal.

CONCLUSION

Three vegetation removal alternatives for these project sites were analyzed. The only feasible alternative for vegetation management was to remove all the vegetative growth from the project sites. The benefits of this alternative are that the contaminant pathways will be controlled and threats to human health and environment would be significantly reduced. Additionally, the implementation of this alternative will create conditions that are conducive to long-term integrity of the closure cap system as part of the final remedial action.

Four options for vegetation disposal processing from these two project sites were considered. Two options, open burning and composing were not recommended because they can pose serious risks to human health and the environment, and these two options will continue to contribute to the contaminant pathways. Analysis performed for volume reduction and controlling the contaminant pathways indicate that incineration of radioactive vegetation would be the best management option for waste volume reduction and safety of the public and environment. This option would control the potential contaminant pathways and protect human health and the environment. Additionally the final disposal of solidified ashes will require less space for burial as compared to disposal of chipped vegetation at the Solid Waste Disposal Facility. However, if the incinerator is not permitted to burn radioactive vegetation, then the second best option for vegetation disposal would be to place chipped vegetation in carbon-steel boxes and bury them at the Solid Waste Disposal Facility at SRS.

ACKNOWLEDGMENT

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MAGNETIC SWING ADSORPTION PROCESS FOR ACTINIDE AND HEAVY METAL REMOVAL FROM WASTE WATER

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ABSTRACT
A magnetic swing adsorption (MSA) process is proposed, which utilizes a column of magnetic adsorbent surrounded by a magnetic field to effect separation of certain metal species from solution. In this process, an electromagnet is simply energized for adsorption and shut-off for elution. During the adsorption mode, an effluent stream is produced that is depleted of certain metal species, whereas during the elution mode, an effluent stream is produced that is enriched in these same metal species. Both the magnetic field strength and relative flow rates during adsorption and elution control the degree of enrichment of the metal species. Magnetic adsorbents in various physical forms can be used in this MSA process for the removal of a variety of metal ions from waste water. A synergistic effect between a magnetic adsorbent material and an external magnetic field is thought to be responsible for the enhanced removal of metals from waste water. Results presented for the removal of plutonium and americium using a magnetic resin clearly demonstrate this synergy. These results also point the way for the development of a variety of new external field-enhanced separation processes.
INTRODUCTION
Adsorption swing processes have utilized a variety of physical properties, such as pressure, temperature, concentration, and electric potential to effect separation between various species in fluid streams. Pressure and temperature, as swing variables, have received the most attention in recent scientific literature; and both pressure swing adsorption (PSA) and thermal swing adsorption (TSA) have been commercialized to perform a variety of gas separations and purifications (1-3). Moreover, hybrid processes utilizing, e.g., both pressure and temperature (4) or concentration and temperature (5), are also being developed, as are processes utilizing potential swing adsorption (6,7). However, to our knowledge, the coupling of magnetic properties with adsorption properties has not been utilized to effect separation. To fill this gap we propose a magnetic swing adsorption (MSA) process. The closest technology to MSA of which we are aware is based on suspending adsorbent particles in a magnetically stabilized fluidized bed (8-10). This fixed-bed configuration has allowed for the development of a two-dimensional, cross flow chromatograph (8); the focusing of biochemicals under an imposed pH gradient (9); and easy processing of viscous and particulate containing feeds (10). However, none of these studies have considered exploiting the effects of the magnetic field on the adsorptive properties of the adsorbent. This paper presents a demonstration of MSA, using a magnetic adsorbent material in a fixed-bed (column) mode surrounded by an electromagnet, for the removal of certain metal ions from waste water. Experimental results are presented for the removal of actinides (plutonium and americium) from waste water using this resin and MSA. Also demonstrated is a synergistic effect between the magnetic resin and the external magnetic field, manifested as a significantly enhanced adsorption capacity. Further, a comparison between conventional ferrite-based processes (where no magnetic field is applied) and this ferrite-based MSA process is also presented, along with a description of a conceptual MSA process. Overall, the results presented in this paper illustrate the potential of MSA processes, and they point the way for the development of a variety of new, field-enhanced separation processes.
BACKGROUND
Magnetic Separation Processes
Magnetic separation and high gradient magnetic separation (HGMS) processes are used extensively in the processing of minerals (11,12), and more recently for water treatment and environmental remediation (13-15). We would like to emphasize, however, that this new MSA process differs significantly from these common magnetic

separation and HGMS processes. Conventional processes use, for example, a fine stainless steel wool to form a magnetic matrix within a flow field of a solution containing mineral particles to be separated. Ferromagnetic or paramagnetic particles in this solution move in the direction of an increasing magnetic gradient, and are attracted to and stick to this energized stainless steel mesh. Diamagnetic materials, on the other hand, move in the direction of a decreasing magnetic gradient, and are essentially repelled from the mesh. This phenomenon has resulted in the development and commercialization of numerous magnetic separation and HGMS processes, which have the ability to remove particles from solution. Therefore, in order for these processes to remove soluble metal species from solution, they must utilize precipitating or flocculating agents to first effect formation of the particles. In contrast, our MSA process has the ability to remove soluble metal species directly from solution as well as colloidal and particulate matter.

Our MSA process is unique because the highly porous adsorbent material not only acts as a magnetic matrix, but also contributes an adsorptive component to the system. We believe that this adsorptive component allows for the removal of soluble metal species from solution, and that the coupling between the magnetic field and the magnetic adsorbent material accounts for the enhanced adsorption capacity observed in our experiments (see below). As a result, precipitating or flocculating agents are not needed in MSA processes. This allows for the direct treatment of a variety of aqueous streams containing soluble metal species, and for the subsequent recovery of the soluble metal species from a concentrated eluent solution. Moreover, because of the highly porous nature of some magnetic adsorbent materials, high magnetic field strengths may not be necessary to significantly enhance the adsorption capacity of a magnetic adsorbent material, as demonstrated below.

Magnetic Adsorbent Materials

Recently there has been increasing interest in developing new magnetic adsorbent materials (10,16). This interest stems from their use in magnetically stabilized fluidized beds (8-10), or in their recovery from a batch adsorption process (10,16). In both situations, however, the adsorption component was not coupled with the magnetic component. This is also true of traditional magnetic materials, which have been used simply as adsorbents to separate a wide range of substances, such as dissolved metal species, particulate matter, and organic and biological materials; however, they have been used almost exclusively for treating metal-laden waste waters (17-20). For the most part, these magnetic materials have utilized the well-known adsorptive properties of iron-based materials for various metal species. For example, ferrites, even in the absence of an external magnetic field, make excellent adsorbents for the separation of hazardous metals (cadmium, lead, mercury, etc.) and actinide elements (americium, plutonium and uranium) from waste water (17-21).

Ferrites are a class of mixed valence iron oxide compounds having the crystal structure of spinal, $MgAl_2O_4$. Iron atoms in iron ferrite ($FeOFe_2O_3$), or magnetite, can be replaced by many other metal ions without seriously altering the spinal structure. They are also ferromagnetic crystalline materials which are soluble only in strong acid. Ferrites can be applied to the treatment of waste water in various physical forms. Natural magnetite needs activation to have the same capacity as freshly prepared ferrite. Kochen and Navratil (22) have found that magnetite ore can be effectively activated to increase the surface area and adsorptive characteristics of the material; they have also discovered a synergistic effect in using supported ferrites, in a fixed-bed (column) mode with an external magnetic field, for removal and subsequent concentration of plutonium and americium from waste water.

EXPERIMENTAL

Resin Preparation

MPE resin beads were synthesized with activated ferrite particles attached to the outer bead surfaces (23,24). This was performed using a two-step procedure and reagent-grade materials (22). MPE resin synthesized according to this procedure contained 28 wt. % magnetite (i.e., 280 mg of magnetite per gram of dry resin). In the supported resin form, the magnetite freely adsorbed actinide ions of interest, while the interstices between the beads promoted good flow-through properties in the fixed-bed mode. Note that non-magnetic, polyamine-epichlorohydrin resin (PE) was also synthesized following this same procedure. However, in this case, magnetite was not used in the resin synthesis. This report contains results based on both types of resins, MPE and PE.

Column Preparation and Operation

The upper portion of a glass, chromatographic column (19 mm id. x 25 cm) was packed with a small plug of glass wool and a 10 cm plug of No. 431 stainless steel wool and placed (with the bottom portion of the stainless steel wool) between the pole faces of an electromagnet. The bottom and top of the column were fitted with a stop-cock and rubber stopper with exit tube, respectively. Tygon tubing was attached to both ends of the column.

Next, the MPE resin was activated and pumped (up-flow at 10 mL/min via a peristaltic pump) into the glass column while a field strength of 0.5 Tesla (5000 Gauss) was applied to the electromagnet; the resin was retained in the column by the stainless steel and glass wool plugs. The column was then ready for use.

For a typical experiment with simulated waste water (de-ionized water spiked with americium and/or plutonium ions, and prepared fresh before each experiment), the magnet was energized, which created a magnetic field strength of approximately 0.3 Tesla (3000 Gauss). The solution containing the actinide ions was then pumped through the column (up-flow at a specified flow rate, see below), and the effluent was radiometrically monitored by alpha (Pu) and gamma (Am) counting effluent fractions (22); the results were plotted as a function of time (plotted in this work as the volume of effluent that was passed through the column) until the actinide ions began to break through the column. For an experiment to demonstrate MSA, while passing solution containing plutonium ions through the column, the electromagnet was energized for a short period of time (to allow a few liters of effluent to emerge from the column) and then de-energized for a short period of time (again, to allow a few liters of effluent to emerge from the column). Further experimental details are described elsewhere (22), and the results of these experiments are given below.

RESULTS AND DISCUSSION

Breakthrough Experiments with MPE and PE Resin

Twenty grams of MPE resin (particle size range of 0.063 to 0.125 mm) were evaluated for actinide removal from water at pH 12.0 (column mode). The breakthrough curves for plutonium and americium are shown in Fig. 1. A total of 341 L of actinide solution (1.28x10-4 g/L plutonium-239 and 3.57x10-7 g/L americium-241) were processed through the MPE resin bed (up-flow) at 3.5 mL/min/cm2. The results showed that 325 L of actinide solution were lowered to 2.77x10-8 g/L plutonium and 7.17x10-10 g/L americium before the plutonium activity in the effluent started to increase. This translates to 4.16x10-2 grams of plutonium and 1.16x10-4 grams of americium that were sorbed from 325 L of water. Thus, 8550 L of actinide contaminated water could be treated with one liter of MPE resin.

For comparative purposes, the PE resin (no magnetite) was also evaluated for actinide removal from water at pH 12.0 (column mode). Breakthrough curves for both the PE and MPE resins are shown in Fig. 2. Note that the PE and MPE columns were prepared and operated in an identical fashion. Three liters of actinide solution (9.42x10-5 g/L plutonium) were pumped (up-flow) through 20 grams of the resin (particle size range of 0.063 to 0.125 mm) at 3.5 mL/min/cm2. The results showed that 2.75 L of actinide solution were lowered to 5.16x10-8 g/L plutonium before the activity in the column effluent started to increase. Although the minimum concentration observed for plutonium was comparable to the MPE resin, breakthrough occurred two orders of magnitude sooner. This represents treating only 72 L of contaminated water with one liter of PE resin, a significantly reduced capacity compared to the MPE resin. These results showed that the non-magnetic PE resin and the stainless steel wool contributed very little to the significant adsorption capacity exhibited by the MPE resin. The results also allude to the synergistic effect that apparently occurs between a magnetic field and a porous, magnetic adsorbent material.

A comparison can also be made with conventional, non-magnetic batch processes. Traditionally, ferrites have been used effectively for actinide removal, but usually in a batch mode in the absence of a magnetic field. For example, two grams of magnetite (when used in a batch operation with no magnetic field) are required to treat one liter of waste water and lower the plutonium concentration from 10-4 to 10-8 g/L (20). Scaling these results to our current work indicated that 650 grams of magnetite would be required to lower plutonium from 10-4 to 10-8 g/L in 325 L of water. However, in a magnetic field, only 20 grams of the MPE resin (containing 28 wt. % magnetite) were required to achieve similar results. Clearly, water decontamination can be improved significantly using magnetic adsorbents in the

presence of a magnetic field, as compared with traditional batch decontamination methods employing ordinary ferrites with no external magnetic field. This marked improvement is also illustrated below with the demonstration of the proposed MSA process.

Effect of Magnetic Field

To show the effect of the magnetic field on plutonium removal using the MPE resin, the electromagnet was energized for a short period of time and then de-energized for a short period of time, while passing a few liters of plutonium solution through the column. The conditions for this experiment were similar to those used previously with the MPE resin. Samples of the column effluent were also filtered through Whatman No. 41 (20 to 25 m) filter paper and analyzed to check the complete removal of the resin particles. The effluent profile is displayed in Fig. 3. Clearly, the external magnetic field, coupled with the magnetic MPE resin, resulted in an apparent synergistic effect that was responsible for high efficiency removal of actinide ions from waste water. While the magnet was energized, the plutonium concentration in the effluent decreased rapidly by two and one half orders of magnitude during the time 3.5 L of solution were passed through the column. When the magnetic field was turned off, the plutonium concentration began to increase rapidly during the time 1.5 L of solution were passed through the column, generating a nearly symmetric profile about the time when the field was turned off. In fact, the desorption kinetics appeared to be faster than the adsorption kinetics, possibly indicating the enrichment effect even at identical feed and elution flow rates and in the same flow direction (see below). Overall, this experiment demonstrated the cyclic, reversible nature of magnetic field-enhanced adsorption processes, and the possibility of achieving fairly rapid adsorption and desorption kinetics (mass transfer) with MSA processes.

In qualitative terms and based on simple magnetic (25) and adsorption (26) theory, we suggest an explanation of the apparent synergistic effect between a magnetic field and a magnetic adsorbent material, manifested as a marked increase in adsorption capacity for certain paramagnetic metal ions. The magnetic susceptibility of a paramagnetic substance generally decreases with increasing temperature. Furthermore, at room temperature, the atoms of most substances are in a state of thermal agitation. This thermal agitation, in the form of random fluctuations in the atoms, opposes alignment of the atomic moments in a magnetic field. This is, in fact, why most paramagnetic substances are influenced only weakly by a magnetic field. However, when an atom, e.g., a paramagnetic metal ion in solution, approaches a solid surface and becomes adsorbed (temporarily or not), energy is released (adsorption is a spontaneous, exothermic process). Some of this energy should be manifested as a reduction in the thermal agitation of the adsorbed species. Further a decrease in thermal agitation due to adsorption should have an effect that is similar to a decrease in temperature. As a result, the magnetic susceptibility of a substance should increase in the presence of a magnetic field and a magnetic adsorbent material, giving rise to a synergistic effect on the adsorption capacity. If this is indeed the case, it can be anticipated that the effect of the magnetic field would be greater for more strongly adsorbed substances. Experimental and theoretical research is ongoing to find quantitative evidence to support this supposition.

Magnetic Swing Adsorption (MSA) Process

A conceptual, two-bed MSA process, designed to continuously remove and recover metals from solution, is illustrated in Fig. 4. This figure shows the magnetic field and associated column step sequences, and the bed loading profiles during adsorption and elution. It is worth noting that there are many similarities between a MSA process designed for the simultaneous removal and recovery of metals from solution, and a simple PSA process designed for the simultaneous removal and recovery of solvent vapors from air (27). With this in mind, a MSA process could be operated as follows.

Two identical beds (Columns I and II) would operate in tandem in a cyclic fashion. While one bed is magnetically energized and in the adsorption mode, the other bed would be de-energized and in the desorption or elution mode. During the elution mode, the void spaces within the column would be flushed by directing a certain fraction of the raffinate from the energized bed through the de-energized bed as purge; this purge flow necessarily being counter current to the feed flow. Ideally, a decrease in the counter current purge flow rate relative to the feed should

generate an extract having a higher metals concentration relative to the feed. There would also be no particular reason to saturate the entire column during adsorption or to completely regenerate the column during elution, as is typically done in conventional waste water treatment processes. In this way, the beds would approach cyclic steady-state operation (similar to PSA systems); and the bed loading profiles would cycle in time, as indicated in Fig. 4. At cyclic steady-state, two streams would be produced continuously: raffinate depleted of the unwanted metals and extract enriched with these metals. Assuming no breakthrough, 100% recovery of the metals would be achieved. Moreover, the certain fraction of the bed that would not be utilized specifically for adsorption would be used to contain the mass transfer zone and act as a guard against breakthrough (refer to the bed loading profiles in Fig. 4). This would ensure a low concentration of metals in the raffinate during the adsorption step; yet enough of the bed would be used to provide sufficient capacity to temporarily contain the metals and subsequently produce a concentrated extract during the counter current elution step. The degree of
enrichment of the metal species could be controlled both by the magnetic field strength and by the purge to feed ratio, i.e., the relative flow rates during elution and adsorption, respectively.

CONCLUSIONS

We have demonstrated an apparent synergistic effect between a polymeric adsorbent containing activated magnetite and an external magnetic field. This synergistic effect yielded an enhanced adsorption capacity for actinide removal from water, as compared to using conventional ferrite-based processes. Based on the well known metal adsorptive properties of various ferrites, these kinds of magnetic adsorbent materials can be used to remove a wide variety of heavy metals (Pb, Hg, Cd, etc.) and actinides (Th, U, Pu, Am, etc.) from a wide variety of aqueous waste solutions, but with a significantly enhanced adsorption capacity as explained above. Furthermore, a variety of inexpensive supports, such as sand, glass, etc., could be utilized in place of the polymer resin (22).

In addition, we have shown that magnetic adsorbents can be regenerated simply by turning the magnetic field off. This provides a mechanism not only for reuse of the magnetic adsorbent material, but also for recovery of the heavy metals and actinides from a concentrated stream. This is accomplished using a proposed magnetic swing adsorption process, where an electromagnet can be energized for adsorption and shut-off for elution. The degree of enrichment of the metal species can be controlled both by the magnetic field strength and by the relative flow rates during adsorption and elution.

Overall, we feel that MSA processes have the potential to make a significant impact on the development of new technologies for the removal of soluble metal species from aqueous solutions. We are presently exploring other applications of MSA, including the removal of certain heavy metals from waste water; and we are developing new magnetic adsorbent materials. Clearly, the results from this study and from our future studies are pointing the way for the development of a variety of new, field-enhanced separation processes.

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FIGURE CAPTIONS

Figure 1. Breakthrough curve for plutonium removal and americium removal with MPE resin in a magnetic field of 0.3 T.

Figure 2. Breakthrough curves for plutonium removal with MPE and PE resins in a magnetic field of 0.3 T.

Figure 3. Breakthrough curve demonstrating a magnetic effect on plutonium removal with MPE resin and illustrating the proposed MSA process.

Figure 4. Magnetic field and column step sequences, and bed loading profiles during adsorption and elution for a conceptual, two-bed MSA process. Notation: t = time, tcyc = cycle time, z = axial position in bed, q = bed loading, qs = saturated bed loading at feed concentration.

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HOME RADON MONITOR MODELED AFTER THE COMMON SMOKE DETECTOR

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ABSTRACT

The Environmental Protection Agency (EPA) has declared that five million or so of the nation's 80 million homes may have indoor radon levels that pose an unacceptably high risk of lung cancer to occupants. They estimate that four times as many people die from radon-induced lung cancers as from fires in the home. Therefore the EPA has recommended that all homes be tested and that action be taken to reduce the radon concentration in homes that test above the 4 pCi/L level. The push to have homeowners voluntarily test for elevated radon levels has been only marginally successful. A reliable, inexpensive, and accurate in-home radon monitor designed along the same general lines as a home smoke detector might overcome much of the public reluctance to test homes for radon. Such a Home Radon Monitor (HRM) is under development at Los Alamos National Laboratory. To be acceptable to the public, HRMs should have the following characteristics in common with smoke detectors: low cost, small size, ease of installation and use, low maintenance, and high performance. Recent advances in Long-Range Alpha Detection technology are being used in the design of an HRM that should meet or exceed all these characteristics. A proof-of-principle HRM detector prototype has been constructed and results from tests of this prototype will be presented.

INTRODUCTION

The Surgeon General has warned that radon is the second leading cause of lung cancer (1). Radon is estimated to cause between 7,000 and 30,000 lung cancer deaths in the United States each year. Between 1,000 and 4,500 of these deaths could be avoided if the five million homes with elevated radon levels could be identified and mitigated to levels below the 4 pCi/L action level recommended by the Environmental Protection Agency (EPA) (2). For example, in New Mexico, it is estimated that about 8 to 10% of the state's 632,000 homes have been tested, with about 25% showing radon levels above the 4 pCi/L level. The average cost for short-term testing (2 to 90 days) is about $150 and for long-term testing (longer than 90 days) the average is about $35 per home (3). Long-term tests generally provide a better measurement of the year-round radon exposure from living in a particular house and represent the best measurement method on which to base remediation decisions, especially for homes with radon concentrations near the action level. The average cost per home to mitigate high radon levels is estimated to be about $1,200 so the total remediation cost to US homeowners and landlords could reach as high as $6 billion. With such large expenditures on the part of the home-owning public being recommended by the EPA, it is important that the decision to mitigate be based on the most reliable and accurate measurements available.

A reliable, inexpensive, and accurate in-home radon monitor designed along the same general lines as a home smoke detector might overcome much of the public reluctance to testing for high radon levels. This detector, the Home Radon Monitor (HRM), would measure radon concentration levels at intervals of one hour or less. These measurements would be stored in onboard nonvolatile memory so that the integrated long-term radon exposure could be extracted, compared to a threshold exposure level, and an alarm activated if the threshold was exceeded. The radon concentration history would be maintained for at least one year and could be read through a communications port for detailed analysis of exposure patterns. To be acceptable to the public, such a detector would need to have the following characteristics in common with smoke detectors:

a. Low cost HRMs should retail for less than $50 each.
b. Small size HRMs should be no bigger than a wall clock.
c. Ease of installation HRMs should hang on the wall like smoke detectors.
d. Low maintenance HRMs should require, at most, a yearly battery replacement.
e. Ease of use HRMs should have a high-radon alert light or audible alarm.
f. High performance HRMs need high accuracy and stability to preclude false alarms.

Recent advances in Long-Range Alpha Detection (LRAD) technology (4), under development at Los Alamos National Laboratory, are being used in the design of an HRM prototype that should meet or exceed all the requirements listed above. A

proof-of-principle HRM prototype has been built to addresses the size, maintenance, and performance objectives. To provide a significant improvement over existing radon measurement devices, this proof-of-principle prototype should have an operating range of at least 0.1 to 100 pCi/L and a 1-hr measurement accuracy of 0.1 pCi/L at the remediation action level of 4 pCi/L recommended by the EPA.

LONG-RANGE ALPHA DETECTION TECHNOLOGY

Monitors based on LRAD technology measure alpha contamination by detecting the ionization that alpha particles create in air (5). The energy lost by an alpha particle in ionizing an air molecule is approximately 35 eV, so a typical 5.5-MeV alpha decay from 222Rn will produce approximately 157,000 ions. Each ionized electron will quickly attach to another air molecule creating an ion pair consisting of two charged molecules. These ion pairs can be transported to an electrode where the current produced can be measured. Airflow LRADs use air currents to transport ions to the collection electrode. Detectors of this type are useful for monitoring irregularly shaped or metal objects (6). Electrostatic LRADs use an electric field to attract ions onto the electrode. Detectors of this type have been successfully used for monitoring flat surfaces such as concrete walls and floors or for soil surface contamination measurements (7). Alpha decay rates can be measured in two distinct modes. If the alpha decay rate is low enough to preclude pileup, current pulses produced by individual alpha decays can be counted. This is called the "single event" measurement mode. If the alpha decay rate is too high to resolve individual current pulses, the integrated current at the ion collection electrode is measured. This is called the "integrated current" mode. LRADs are unique in that the ionization produced from alpha decays in air is measured directly, without the gas gain, special gases, thin windows or other characteristics typical of ionization detectors.

To measure radon gas, air samples are drawn into a detection volume through electrostatic and particulate filters via fans or through diffusion. The electrostatic filter prevents the 107 ions/cm3 that are generally present in air from entering the detection volume. The particulate filter insures that radon progeny attached to airborne particles do not enter into and decay in the detection volume, creating ions that would bias the measurement. Ions produced and collected within a detection volume can be attributed to one of the following sources.

Material Contamination: The material from which an LRAD is constructed will contain some alpha contamination. When contamination located on the inside surface of the detection volume decays, ions will be produced and the measured current will increase. The background signal that can be attributed to material contamination is typically less than 0.02 femtoamperes (fA) of integrated currenta or 0.05 counts/minute per liter of detector volume.

Penetrating Radiation: External sources of penetrating radiation, such as cosmic rays, will produce ionization that will add to the overall current measured. However, a minimum ionizing cosmic ray will only produce approximately 1000 ion pairs per liter of detection volume. Therefore, the integrated current due to cosmic rays is less than 0.1 fA per liter of detector volume. Penetrating radiation will not produce large enough current pulses to contribute to the single-event count rate.

Radon Gas: Because radon is a noble gas, it is unaffected by particulate or electrostatic filtering and will enter into the detection volume. Some radon will decay and produce ionization in this volume. The ionization produced by these decays is proportional to the concentration of radon. A radon concentration of 1 pCi/L will produce from 1 to 2 fA of integrated current, depending on detector design and geometry, and 2.2 counts/minute per liter of detector volume.

Radon Progeny: Radon progeny from radon decays inside the detector can plate out on inside surfaces or be swept out of the detector by the airflow. In airflow LRADs most of the progeny are swept out of the detector, while in electrostatic LRADs most of the progeny become attached to the inside walls. In-growth from 222Rn daughters reaches equilibrium in about three hours. Because radon daughters 218Po and 214Po are also alpha emitters, electrostatic LRADs at equilibrium will see three alphas for each 222Rn atom that decays within the detection volume.

HOME RADON MONITOR PROTOTYPE

An HRM prototype detector has been constructed from a set of nesting stainless steel camping pans. The simplicity of design and construction inherent in LRAD-technology-based detectors is readily seen in the HRM schematic diagram and

photograph shown in Fig. 1. Sample air enters the 1.5-L detection volume via diffusion through an electrostatic/particulate filterb attached to the lid. Filtering is required to prevent radon progeny attached to particulates, such as dust motes, cigarette smoke, or automobile exhaust, from entering into and possibly decaying in the detection volume. The HV pin is held at high-voltage (typically 300 VDC) relative to the outer pan. The Teflon standoffs are designed to minimize leakage currents between the inner and outer pans and between the inner pan and the high-voltage pin. A current path exists from the inner pan to the outer pan through the signal BNC connector and an electrometer. An electrostatic field of about 40 V/cm exists within the detection volume. Radon gas decaying within this volume will produce ions which are attracted to the HV pin, the lid, and the inner pan. The charge that collects on the inner pan is grounded to the outer pan through the electrometer. The current pulses produced by collecting the ions from individual alpha decays can be counted or the total integrated current measured.

The current generated in an LRAD-technology-based detector from the ionization produced by 1 pCi of radon gas is less than 1 fA. Measurement of such small currents requires a very sensitive electrometer coupled to a detector designed and constructed to eliminate leakage currents. Figure 2 shows a block diagram of an electrometer specifically designed for LRADs. The current to voltage input stage, A1, of this electrometer uses an ultra-low-input-current operational amplifier in the standard current to voltage configuration. This amplifier maintains the "-" input at the same potential as the "+" input by means of the closed loop feedback action. Because the "+" input is tied to ground or zero voltage, the "-" input is forced to this same potential and functions as a virtual ground. The current flowing into this stage, Iin, flows through the feedback resistor, Rf, which produces an output voltage, Eout, equal to -(Iin.Rf), therefore for an input current of 1 fA, and an Rf of 100 GW, Eout would be -100 V. The input conversion gain is selectable providing -100 V out for either a 1 fA or 10 fA input current. The current to voltage input stage, A1, is followed by a gain stage, A2. This stage provides a buffered output and has a selectable gain of 1 or 10. A1 also drives A3, which provides an integrated version of the signal. The time constant of this integrator is selectable for either 5 or 10 seconds. The electrometer includes a gain potentiometer for calibration and an offset potentiometer for nulling the circuit. The circuit uses very little power and can operate for several years on a set of batteries.

The conversion of femtoampere currents seems relatively straightforward, but several effects can cause errors in the measurement. First, leakage currents at the critical node, the "-" amplifier input, must be kept extremely low if one is trying to resolve femtoampere currents. Also the amplifier's bias current, i.e., the current that flows into the amplifier's inputs, must be very low. Although bias current errors can be canceled out by providing an offset adjustment, bias currents vary with temperature. There are other errors, such as the temperature coefficient of resistance and the voltage coefficient of resistance, which cause the conversion gain to change as a function of temperature or voltage. Cancellation techniques have been incorporated in the design of this electrometer to minimize these effects.

The HRM prototype detector was tested using sources of 222Rn and 220Rn gas constructed by enclosing naturally occurring uranium and thorium ores in bottles. Gas from these bottles was injected into the HRM prototype detector to study speed of response, radon daughter in-growth, and single event pileup. Data acquisition and analysis were done using a personal computera. Electrometer outputs were acquired at 3 to 5 readings/second, and the results were displayed on the computer screen as a strip chart and recorded to disk. The raw data were averaged into 1-minute bins and, along with the single event count, the results were displayed and recorded to disk for later analysis.

RESULTS

The current output of the HRM prototype detector operating in a 1 pCi/L radon environment over a typical 1-minute time interval is shown in Fig. 3. The current pulses at 20 and 40 seconds are due to ions from individual alpha decays. The average integrated charge per single event from the 5.5-MeV decays of 222Rn in the HRM prototype detector is about 2.4 x 10-14 Coulombs. This value corresponds to an ion collection efficiency of 95%. In a 40-V/cm electrostatic field, all the ions from a single alpha decay are collected within a 3-second time interval. These single event current pulses can be easily distinguished from the background current

if the frequency of decays is low enough to preclude significant pile-up. For the HRM prototype detector, single event pile-up starts to significantly affect the precision of radon-level measurement at about the 15 pCi/L level.
Several LRAD prototype detectors have been calibrated in the Radon/Radon Daughter Environmental Chamber operated by RUST Geotech at the DOE's Grand Junction Projects Office in Colorado (8). These detectors were operated in the integrated current mode in a variety of configurations. Figure 4 shows detector response to radon concentrations ranging from 0.06 to 825 pCi/L. For each detector at each point, a 1-hr data set was taken after radioactive equilibrium was reached. The arithmetic means of these 1-hr data sets, minus a constant background due to leakage currents and cosmic rays, are plotted against the mean radon concentration as determined by RUST Geotech personnel. As can be seen from this plot, LRADs operating in integrated current mode respond linearly, with R2s of 0.999 or better, over the entire 4-decade radon concentration test range.
By combining the single event counting mode with the integrated current mode, the HRM prototype detector spans an operating range from zero to 800 pCi/L or higher. At the low end, precision is limited by single event counting statistics. For the 1.5-L HRM prototype detector, this leads to a 1-hour measurement precision of 0.1 pCi/L at a radon concentration of 4 pCi/L, assuming material contamination of 0.05 counts/minute and that half of the 218Po and 214Po radon daughters will decay into the detection volume instead of into the detector walls. A larger detection volume would improve measurement precision at the expense of lowering the changeover point from the single event counting mode to the integrated current mode.
Long-term stability of an LRAD-technology-based HRM operating in single event mode is assured due to the digital nature of the measurement. Calibration in the single event mode is similarly simplified. There is a large detection region, from about 1 to 15 pCi/L, where the single event and integrated current measurement modes overlap. Calibration of the detector in the integrated current mode can be tied to readings obtained from the single event mode while operating in this overlap region. Because LRADs collect the air ionization from alpha decays directly without relying on gas amplification of the total charge, a major source of long-term drifts in the calibration of the detector in integrated current mode is removed.
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PVC REPLACEMENT AT SRS
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ABSTRACT
Shoe covers and waste bags made of polyvinylchloride (PVC) are used at the Savannah River Site (SRS). Plans are to incinerate shoe covers and disposal bags in the Consolidated Incineration Facility (CIF) after its scheduled start-up in the first quarter of 1996. The exhaust gas from the incineration of PVC must be processed through a hydroxide scrubber to remove the chlorides. This liquid scrubber waste stream is referred to as the CIF blowdown. The blowdown volume from processing the chlorides will be disposed of in a cement matrix in either the SRS E-Area Vaults (EAV) or in a proposed Hazardous Waste/Mixed Waste Disposal Facility (HW/MW DF) vault at a significant cost.
We have completed an investigation of alternative materials for PVC shoe covers and waste disposal bags to minimize disposal costs. Savings of about $2.89 millions a year in vault disposal costs are expected from replacing PVC shoe covers and standard-sized bags with non-PVC products, through a reduction in the volume of CIF blow-down generated. Recommended acceptable substitutes will cost about $450,000 a year more than the current PVC products. SRS plans are to implement PVC replacement starting June 1995, commensurate with the current CIF start-up schedule, to take advantage of the $2.45 million per year in net cost savings.
SHOE COVERS (1)
Manufactures' Information
Information from twelve (12) different shoe cover manufacturers were collected on seventeen (17) shoe cover samples and recorded on a spreadsheet for comparison. Specific information include the following: style, material(s), cost, size availability, tensile strength, puncture resistance, tear resistance, thickness, length, height, weight, carton weight, wear resistance, elongation percentage and miscellaneous information that appeared relevant. Some of the information was not available from a few of the Manufacturers.
The list of rnanufacturers is as follows:
- FRHAM Safety Products, Inc.
- Island Poly
- G/O Supply Company
- JSM Protective, Inc.
- Mar-Mac Manufacturing Co., Inc.
- Designs by Norvell, Inc.
- Associated Bag Company
- Euclid Garment Manufacturing
- Lancs Industries, Inc.
- RAD-SAFE
- FAB-Ohio
- Rich Industries

Acceptance Criteria
To be accepted, a replacement alternative had to meet certain requirements. It had to be waterproof, disposable versus re-usable, incinerable (but non-PVC), durable, cost effective and highly ranked within a "figure of merit" scale. A number of simple tests were conducted to quantify the acceptability of replacement products, as described below.
Puncture Resistance Tests
Using a force meter and two different puncture bits, each shoe cover sample was pressed separately on both bits (Fig. 1). The force meter measured the strength required to puncture the shoe covers. One bit was pointed and the other was wedge-shaped (Fig. 2). The diameter of the pointed bit was 250 mil, with an angle measure of 60. The diameter of the wedge-shaped bit was also 250 mil, with an angle measure of 60. The ridge on top was .013 mil wide.
To rank the samples, the two puncture bit readings were averaged. The highest average was found to be 56.72. Each shoe cover result was then divided by 56.72 and multiplied by 100 to calculate the percentage. The puncture resistance rankings, rounded to the nearest whole number, ranged from 1 to 100.
Coefficient of Friction Tats
Using a force meter, an 18.5 pound stainless steel brick, and kraft paper, the coefficient of friction between the shoe cover and the kraft paper was measured. Kraft paper is used extensively as a floor covering material to prepare for waste

removal, D&D and other activities in radiologically contaminated areas. Each shoe cover sample was fitted over the brick. The force meter was used to push the brick and shoe cover sample across the kraft paper. The force required to push the brick was measured (Fig. 3).

To tabulate the coefficient of friction, the formula of F/N was used. F is the force measured by the force meter and N is the normal pressure which is the weight of the brick. The highest coefficient of friction reading was 0.76. For ranking purposes, each shoe cover's coefficient of friction reading was divided by 0.76 and then multiplied by 100 to get the percentage. The coefficient of friction rankings, rounded to the nearest whole number, ranged from 28 to 100.

Durability Walking Tests

Two colleagues, each weighing around 200 pounds, walked in leather shoes wearing the shoe cover samples. They walked 1.5 miles over a trail constructed of "fines" gravel, which is smaller than "pea" gravel. The trail is shown in Fig. 4 and a close-up of the "fines" gravel is shown in Fig. 5.

After each walk, the description of each shoe cover was recorded in a journal. The shoe covers were rated on a scale from 1 to 17, with 1 being the least durable and 17 being the most durable.

Test Results

Of the 17 shoe covers evaluated, one brand of polyethylene shoe covers was found to be most satisfactory and was recommended to replace PVC shoe covers currently in use at SRS. These polyethylene shoe covers are both disposable and incinerable. They have a relatively high coefficient of friction and puncture resistance.

WASTE DISPOSAL BAGS (2)

Evaluation of waste disposal bags was conducted in two phases. In the first phase, bags made of poly-ethylene, PVC and polyurethane were evaluated. In phase two, only polyurethane bags of different thicknesses were evaluated, along with the PVC bags currently in use.

Manufacturers' Information

Information from four (4) different waste disposal bag manufacturers were collected on nine (9) waste disposal bag samples and recorded on a spreadsheet for comparison. Specific information include the following: description (material and dimensions), cost, tensile strength, puncture resistance, cutting resistance, and wear resistance.

The list of manufacturers and waste bag samples is as follows:

- JSM Protective, Inc. (6-mil, 8-mil polyethylene)
- Associated Bag Company (3-mil, 4-mil, 6-mil polyethylene)
- Rich Industries (12-mil PVC (2),12-mil polyurethane)
- Nuclear Power Outfitters (4-mil polyethylene)

Acceptance Criteria

To be accepted, a replacement alternative had to be non-PVC, yellow-tinted, capable of being incinerated without producing regulated emissions, cost effective and comparable in strength to the PVC bags being replaced. A number of simple tests were conducted to quantify the acceptability of replacement products. The tests and results are described below.

Puncture Resistance Tests

Using a force meter and two different puncture bits, as described above for testing shoe covers, each bag sample was pressed separately on both tips.

Polyurethane tested more resistant to puncture from the point bit (14.4 lb) and the wedge bit (25 lb) than any other material tested, including the PVC currently in use. The overall puncture resistance measurements ranged from 1.0 to 14.4 lb for the point bit and 1.2 to 25 lb for the wedge bit tests.

Cutting Tests

Each bag was cut with scissors connected to a force meter. The force meter determined the force required to cut each bag. The cutting force is thus also a function of the sharpness of the scissors.

The highest force required to cut the bags was 6.7 lb, which was observed for the polyurethane bags and the PVC bags currently in use. The cutting measurements ranged from 4.9 to 6.7 lb.

Wear Resistance Tests

The polyurethane and PVC bags were tested for wear resistance. The six polyethylene bag samples were deemed unacceptable, based on the puncture and cutting resistance test results, and were thus not tested for wear resistance. To conduct the wear

resistance test, the polyurethane and PVC bags were filled with atomic wipes and dragged across pavement in 50 ft intervals. After each interval the bag was inspected for any noticeable holes. When found, the hole was marked and the drag distance recorded. Distances were avenged for a number of tests. The average for polyurethane was found to be 1550 ft and for PVC bags, 1750 ft. This wear resistance test produced wide scatter of data. Phase two evaluation tests used a modified wear resistance test.

Phase Two Tests

Polyurethane bags of 12 mil, 10 mil, 8 mil and 6 mil thicknesses were requested from Rich Industries for additional testing. Thicker samples of polyethylene samples were considered, but they were not ordered for further evaluation because the thicker material would have made handling by area operators in protective gloves too difficult. Again, three tests were conducted: puncture resistance, cutting and (modified) wear resistance.

Modified Wear Resistance Test

The modified wear resistance test involved testing two different samples at the same time with a glass bead sprayer. Two 3" x 3" samples of the materials to be tested were clamped between two plates of carbon steel. Each samples was exposed through a " hole in the front plate. The holes had been filed smooth to prevent any burs in the metal from tearing the samples.

The test apparatus was placed in a bead tester (a glovebox used to clean or smooth machined parts). The blaster contains a pneumatically operated spray gun inside a glove box chamber. The gun sprays glass beads at high speeds against a sample set in the chamber that is to be cleaned or smoothed. The test apparatus was leaned against the back of the chamber at a 45 angle. The samples were blasted for varying lengths of time at a 45 angle with respect to the plane of the sample. Pictures were taken of the resulting worn patch using a scanning electron microscope. Relative information on the degree of resistance to the wearing effects of the pellets was recorded.

The phase two test results are summarized in Tables la and lb.

Summary of Test Results

A 10-mil polyurethane waste bag is recommended as a replacement for 12-mil PVC bags currently in use. 12-mil polyurethane waste disposal bags tested stronger than the 12-mil PVC in puncture tests and as strong as PVC in cutting tests. Polyurethane bags, however, cost more than PVC bags of the same thickness. 12-mil, 10-mil, 8-mil and 6-mil polyurethane bags were then tested. Results indicated that 10-rail polyurethane is as strong as the 12-mil PVC currently in use. 8-mil polyurethane is not as strong as 12-mil PVC. Substituting 10-mil polyurethane is expected to result in a cost savings over 12-mil polyurethane while still retaining the strength of the current PVC

COST-BENEFIT ANALYSIS(3)

SRS PVC Usage

Table II shows the breakdown of PVC operating materials use at SRS into shoe covers, two types of standard-sized bags (large and small), non-standard-sized bags and miscellaneous items, as surveyed in 1993(1,2). Shoe covers represent the majority of PVC used at SRS, about 60% by weight. The two standard-sized bag together make up 33% of the PVC used at SRS (28.5% large 42" x 50" bags and 4.5% small 22" x 36" bags). Together, shoe covers and standard-sized large and small waste bags comprise 93% of the PVC expected to be incinerated in the CIF.

Cost of PVC Replacements

Ten-mil polyurethane bags cost about twice as much per bag as 12-mil PVC bags currently in use (Table III). Based on the cost data from the recommended manufacturer, it is estimated that to replace both sizes of standard large and small sized PVC bags with 10-mil polyurethane bags will cost an additional $227,000 per year over current procurement costs.

Based on cost data from the recommended manu-factur, polyethylene shoe covers are expected to cost about twice as much as PVC shoe covers ($0.31/pr vs $0.15/pr). It is thus estimated that to replace PVC shoe covers with polyethylene shoecovers will cost an additional $222,000 per year over current procurement costs.

Disposal Fault Cost

SRS vault space costs have been estimated at $49/ft3 for the EAV and at $374/ft3 for the proposed HW/MW DF vaults.(4) These costs include both total project cost and operating cost

In a recent CIF Blowdown study(5), it is estimated that the yearly amount of waste generated from CIF blow-down is 185,000 gal/yr. With replacement of both shoe covers and standard-sized bags, the CIF blowdown waste is estimated to be 50,000 gal/yr, which represents a 73% reduction in volume. Thus PVC replacement will result in a 93% reduction in PVC usage and a 73% reduction in CIF blowdown volume. SRS stabilization studies have shown that 28 gallons of blowdown solution may be cement-stabilized in a 55-gallon drum. The cement-stabilized blowdown is planned to be disposed of in either the EAV or the HW/MW DF Vaults, depending on whether the stabilized waste qualifies as low level waste or as mixed waste.

Cost Savings

For the purpose of this cost-benefit analysis, it is assumed that 90% of the stabilized blowdown volume will be disposed of in the EAV and 10% disposed of in the HW/MW DF vaults. The reduction of 265,200 gal/yr in stabilized CIF blowdown volume thus equates to vault space savings of about 31,900 ft3/yr of EAV and 3,550 ft3/yr of HW/MW DF vaults.(4)

PVC repalcement of both shoe covers and standard sized bags thus will result in $1.563 million of savings in EAV space and in $1.328 million in HW/MW DF vault space. The total savings in vault space is $2.89 millions. Recommended acceptable substitutes will cost about $450,000 a year more than the current PVC products. SRS plans are to implement PVC replacement starting June 1995, commensurate with the current CIF start-up schedule, to take advantage of the $2.45 million per year in net cost savings.

Notes:

1. Reduced volumetric processing of CIF blowdown is also expected to result in reduced CIF operational costs associated with stabilization material and operating labor, increased life of the offgas processing equipment and easier compliance with proposed CIF air emission standards.

These savings have not been quantified and credit was not taken in this analysis.

2. It is also recognized that both SRS usage of PVC operating materials and costs of bags and shoe covers have changed since the completion of these studies. The actual savings may have changed, but the general conclusion is expected to remain the same.

Through this paper we hope to convey our strong belief in the need to quantify the "quality" of replacement products to avoid the griefs of "you get what you pay for". To that end, this paper has offered some simple tests that may be easily utilized by many organizations.

ACKNOWLEDGMENT

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IN SITU CONSTRUCTION OF HORIZONTAL SOIL CONTAINMENT BARRIER AT FERNALD*

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ABSTRACT
An innovative method of placing soil barriers to contain vertical flow is being prepared for demonstration by the Fernald Environmental Restoration Management Corporation (FERMCO), working in conjunction with the Department of Energy Office of Technology Development (DOE/OTD) and two principle subcontractors. The method employs proven directional drilling techniques, jet grouting technology and unique placement tooling to form horizontal soil barriers in situ. This is done without disturbance to existing land disposed wastes.
This paper is a summary report on the current state of that demonstration, including: a discussion of the construction methods, the results of the initial tool tests, an overview of the Fernald site conditions and, the resulting path of tooling development for the second phase of tool testing.
INTRODUCTION
Temporary or long term containment of mobile contaminants from existing land disposal waste site requires effective surrounding barriers. Current technologies typically require removal and repackaging/reprocessing of waste from its existing location or rely on existing impermeable soil layers beneath the waste. In situ constructed vertical barriers, or cut-off walls for the restriction of horizontal flow, have been used extensively and are relatively well understood. However, a full containment is difficult to achieve unless these walls can be tied into existing impermeable layers (clays). Those layers may be very deep or missing entirely. In either case, the containment then depends on the integrity of the existing geology. This is often difficult to assure.
The use of in situ constructed horizontal containment to restrict vertical flow is not highly developed. When dealing with existing sites, it is a problem to place containment without disturbing the waste. The excavation to or through the waste can create unacceptable health risks and/or regulatory problems.
At several DOE sites natural barriers to vertical migration do not exist at usable depths beneath existing sources of contamination. Examples are both Hanford and Savannah River sites (although their basic geology varies greatly). Like situations exist at government and commercial industrial, chemical and petroleum storage sites. Attempts to produce barriers to vertical flow have generally relied on permeation grouting, using large amounts of material, to form a floor beneath the source. The uncertain nature of these processes and the difficulty of conclusive inspection at depth raises regulatory concerns.
This paper describes a demonstration/innovative adaptation of jet grouting techniques to form a soil barrier in situ, the results of initial tooling tests and, the plan to bring the technology demonstration at Fernald to successful completion.
TECHNOLOGY/PROCESS DESCRIPTION
This innovative process for placing in situ barriers is a self verifying technique that actually moves a solid tool completely beneath existing sources of contamination, leaving a continuous soil and grout (soil-cement) mix behind, in the tool's wake, to form a containment. Multiple passes of the tool can be used to form a barrier of significant size.
The typical jet grouting process uses a cement grout pumped at very high pressure (up to 10k PSI) to a set of nozzles arranged in a single line on a pipe tool. When the grout exits the nozzles of the tool, the pressure is converted to velocity. If the tool is in contact with a soil formation, the high velocity grout transfers its kinetic energy to the soil, erodes the formation and mixes with the spoil. This forms a soil-cement. Both the soil and the grout and their respective ratios determine the properties of the resulting mix. Portland cement in a 1:1 ratio with water is a common grout mix. Bentonite, pozzolans and other additives are often used to modify properties and control costs.
A rod shaped jet-grouting tool (with a single line of jets down its length) may be placed vertically into the soil (by drilling or driving) and used to form a soil-cement cylinder, column or pier by rotating the tool and slowly withdrawing it leaving soil-cement behind. Conceptually, if the same vertically placed tool is

moved perpendicular to its axis and along the line of flow of the jets, it will cut the soil and leave a soil-cement "wall" in its wake. This action is the key to placing continuous horizontal barriers.

The equipment set necessary to perform successful jet grouting includes dry grout bulk storage bins or tanks, efficient mixing systems, effective grout cleaning equipment (screens), high pressure pumps (10k PSI) and power systems to support the production equipment. All parts of this supporting suit must deliver sufficient reliable capacity to ensure a steady flow of clean grout to the tooling. In addition, the horizontal barrier formation process requires powerful pulling equipment and directional drilling equipment.

The process of forming a horizontal grout barrier begins by placing two generally parallel directionally drilled holes from the near side (clean) surface, passing completely beneath the source of contamination and emerging again at the (clean) surface on the far side. The drilling stems remain in the holes and are attached to a draw bar and a pulling device at one end. The stems' other ends are attached to the front of a jet grouting tool. A grout feed line and a trailing drill stem are attached to the rear of the jet grouting bar, forming a capital "H"-shaped arrangement with the jet grouting tool as the crossbar of the "H" when viewed from above (see Fig. 1). A tractor, winch or pipe puller then pulls on the draw bar attached to the stems forming the top of the "H" and moves the jet grouting tool through the soil along a path between the directionally drilled holes.

Fig. 1. Solid bar tool ready for testing.

The jet grouting tool is an injector-mixer which leaves a soil-cement (grout and native soil) slab in its wake as it is pulled through the soil and beneath the source of contamination. It emplaces the soil-cement slab by employing grout delivered at high pressure, up to 10k PSI, to develop high velocity through use of jet nozzles. The high velocity grout streams emerging from the tool's jets cut soil formations by transferring their kinetic energy (through impact of the grout) to the formation around the tool. The high amount of energy involved and the abrasive nature of the grout currently in use gives a strong cutting action (which can be augmented by mechanical cutting). The remainder of the grout's energy is dissipated in a turbulent mixing action that forms the soil-cement. The formed barrier slab is continuous back along its path to the surface and remains fluid. Therefore, there is no significant build up of grouting pressure. A flow of excess grout and some spoils to the surface keeps the formation open and is expected to avoid pressure build up and issues of significant slab jacking or extensive fracturing of the soil formation.

Joined slabs, necessary to form an extensive bottom, are to be formed continuously and sequentially by using the previously mentioned trailing drill stem from the last slab and an additional, generally parallel, directionally drilled-in stem to make each additional slab. (see Fig. 2) The use of the trailing pipe ensures that each slab overlaps the previous one and that a continuous bottom is formed. Continuous forming and the grout formulation ensures wet joints with commingling along seams to enhance the integrity of the final barrier. This process is repeated until a bottom of the required width is formed.

This process produces grout overflow, drilling spoil, excess soil-cement and washout water, as secondary waste. Because the path of the directional drills and the barrier does not have to contact the waste, these secondary wastes may not be problematic.

Conceptually, the technique is simple to use. It relies on two proven technologies: grout high pressure pumping techniques and DOE/OTD demonstrated directional drilling methods. The difficulties of operating in rocky soil, cobbles, till and other challenging conditions remain to be determined and are challenging areas for additional work.

The horizontal barrier is potentially applicable to containment of a wide range of existing land disposal sites, underground storage tanks, spills and ruptures. The types of grout that may be placed are extensive, the jet grouting tool may be expected to place materials that can be fed to it as a liquid at the required volumes and pressures (grout tolerance to shear and impact may be issues). It should also be noted that this "horizontal barrier" can vary from the horizontal plane through the vertical for special applications, simply by varying the arrangement of the directionally drilled holes.

THE FERNALD DEMONSTRATION PROJECT

The objective of this work is to place a joined, four pass, demonstration horizontal barrier on a radiologically "clean" site at the Fernald Environmental Management Project (FEMP) site in southwest Ohio. The demonstration will build various tooling combinations and test them at a separate location prior to mobilizing to FEMP with at least the two tools thought best suited to the local conditions. Placement techniques, surface support systems operations and longitudinal joining integrity (seaming) are of primary interest in the work. A simple portland cement-based grout will be employed to control costs.

The FEMP site is a 600' x 200', area clear of surface and below grade obstructions. The site is accessible from plant roadways for trucks and heavy equipment. The completed demonstration barrier will cover approximately 100' x 40' of the test area and will begin near the surface and drop to a nominal placement depth of 12', rising to the surface at the far end. Soils at the site were classified as Pleistocene (6,000 - 10,000 year) glacial till to a depth of 30'. Well developed, historically cultivated A and B loam horizons are located to a depth of approximately 18". As a homogeneous, semi consolidated unit, the unweathered tills generally consist of approximately 75% mixed brown and gray clays, with 25% silt and fine sand in approximately equal proportion. Organic concentration below the cultivation level is less than 5% and moisture content varies from 3 - 10%. Lenses of moist to saturated sand, 1' to 3' thick, exist at random through the body of the till. Exploratory excavations revealed that the predominantly brown clays are underlain by a discontinuous rocky layer containing both limestone slabs and glacially transported granite schist, at a 12' depth below grade. The gray clays below that depth were distinctly less sandy and silty, and more plastic.

The existence of the rocky layer and strata change at 12' depth on the FEMP site is a concern for both tooling and testing and will impact the final operating depth for the demonstration.

The FEMP demonstration barrier will be placed, allowed to cure and then excavated for testing by an independent academic agency. The testing will include: surveying the site, measuring displacements and tilts during placement, physically sectioning and characterizing the barrier and conducting applicable strength and permeability tests to evaluate the uniformity and integrity of the barrier across seams. Key to the test effort is the issue of determining if adjacent barrier panels, formed on sequential passes of the tooling, can be joined in a reliable and reproducible manner to form a uniform and competent barrier.

PROGRESS/STATUS

The work involved in this project covers more than two years of effort directly aimed at establishing a viable capability to place in situ soil barriers to vertical flow. Two separate series of tool tests have been conducted and the knowledge gained from those tests was incorporated into the design of a third generation of tooling for up-coming field testing. For all testing completed and planned, the directional drilling was/will be simulated by trenching in the drill pipe, backfilling and compacting the overburden. Directional drilling is considered a DOE proven "enabling technology" at this point. The cost savings to place the drill pipe by trenching, rather than by drilling, were significant to the project.

The initial field work, begun in 1992, involved simple proof of concept work to answer the issues of "can this be done at all." Two basic types of tool were successfully tested in sandy soil and soft sandstone formations in Oklahoma.

The first tool was a solid bar with a line of forward-pointing jets for distributing grout forward, in the direction of travel, in streams like the teeth on a pocket comb. Also developed was a prototype rotary tool shaped like an automotive camshaft with angled, cross cutting jets distributing grout from the circumference of the lobes. The rotary tool was driven directly by a down hole hydraulic motor with fluid lines to the surface. Each tool was approximately 10' wide.

The solid bar tool produced the strongest cutting action, as its 61 jets were all aimed forward, concentrating the available kinetic energy on the formation directly ahead, shattering the soil and liquefying the path of travel. The solid bar was also relatively efficient in the use of grout since the cutting action all took place forward of the path of travel. Panels could be formed at the rate of 20 to 100 square feet/minute. Several shortcomings were apparent: the barrier formed was thin, barely exceeding the width of the bar; the barrier was irregular, due to the non-homogeneity of the formation and the fixed direction of the jets and; the solid bar was very susceptible to failures from the blockage of one or more jets. These

drawbacks to the solid bar led to increased soil resistance and bent or broken tools. While the simplicity of the tool was a plus, improvements were deemed necessary.
The prototype rotary tool produced a thicker and more uniform barrier slab, one that was 10' wide, 12" thick and literally looked like an underground highway when a section of it was excavated. Because the same available energy (as the solid bar) was spread over a greater surface area, the rotary tool formed a panel more slowly; about 10 to 30 square feet/ min. The rotary tool also experienced jet blockage, but it could compensate to a degree as the rotation of the tool brought other jets into play at the cutting face. Nevertheless, the tool exhibited what was considered to be excessive grout use and back flow through the slab, as only the jets at the face aided forward progress. In addition, the size of the down hole motor caused excessive resistance. The use of a down hole hydraulic motor had a second negative: in the event of a mechanical failure, the hydraulic fluid (a RCRA material) could escape into the soil. A combination of drag from the motor case and blocked jets ultimately resulted in the structural failure of the prototype rotary tool during testing. It had to be dug out with a back hoe and was destroyed in the process.
The first round of testing showed that barriers could be formed in situ using the horizontal grout barrier technique. The project goals were redefined to gain increased effectiveness in terms of barrier thickness, surface uniformity, cutting effectiveness and grout use.
The second round of tooling design, fabrication and evaluation concluded in mid 1994 with testing at a more challenging area of the Oklahoma test site. Three second generation tools were produced with the specific objective of optimizing against the new goals. The new tooling items were also about 10' wide but were widely different in design from the first generation and from each other:
The solid bar tool #1 was retained to gain a performance baseline for other tooling.
A second solid bar tool, #2, used a 54-jet array placed in a crosscutting arrangement in two horizontal rows.
A mechanical head tool (shuttle tool), the most complex new tool, consisted of a drive mechanism with a reciprocating mobile cutting head consisting of a steel box with two sets of 12 jets arranged in two rows of six (total 24 jets) at each end of the box. This device was approximately the size and shape of a baby grand piano with a shoe box on the keyboard. As the main tool body was moved forward (with the keyboard leading) the shoe box-shaped cutting head shuttled across the keys. The intent was to reduce grout usage through reducing the number of jets (24 vs 61) and to maintain the tool's cutting effectiveness by concentrating the remaining jets close together on the shuttle. The mechanical complexity of this tool, the cable linkage to the shuttle head, and its related surface support equipment with the long flexible drive shaft necessary to reciprocate the shuttle head was recognized as a negative at the inception. The potential for reduction of grout waste and backflow of spoils through the barrier was considered significant enough to warrant testing the concept.
A catenary (flex) tool was the most innovative of the second generation tools. This tool consisted of two separate cutting heads (subs) attached to a curved structural steel member. This assembly was, in turn, connected to the drill stems by two flexible spring steel straps. The design was to allow the tool to reciprocate across the face of the cut like a cable saw cutting wood, propelled by alternately pulling and playing out alternating drill stems in a coordinated back and forth stroking action. Each sub was equipped with 10 jets in two rows of five. The resulting total of 20 jets was expected to reduce grout waste and retain cutting effectiveness.
The supporting equipment for the second test was a custom flatbed mixing plant with mechanical drive equipment for the shuttle and flex tools, a 1,500 gal. grout surge tank and a tracked power unit to advance (pull) the unit and tool through the length of the barrier. The mixer capacity was 300 - 400 gal./min. with direct transfer to the grout surge tank. High pressure pumping was provided through a 1000 horsepower twin positive displacement pumping unit.
The grout formulation was cement kiln dust and water mixed to an original density of 12.6 lbs./gal. Grout density was later lowered to 11.4 lbs./gal. to increase cutting power. Lignosuphonate was added to act as a retarder. This mix was chosen for reasons of low cost and low hydration shrinkage.
The second round of testing did not produce the same successes as the first. A combination of more challenging soils, mistaken design assumptions and, support

equipment shortcomings resulted in multiple failures to form a satisfactory barrier. The original solid bar tool #1 with 61 jets was tested at both 3,300 PSI and 5,000 PSI. The bar appeared to improve in cutting performance at the higher pressure. In both cases the bar moved forward about 25' and stopped due to exhausting the grout supply in the 1,500 gal. surge tank and exceeding the 300 - 400 gal./min. grout make up capacity. The forward progress of the bar could not be resumed in either test after halting to replenish the grout supply because of plugged jets at the center of the bar.

Solid bar tool #2 with 54 jets was tested at 5,700 PSI on a single twin pump (1,000 horsepower). The tool traveled 15' - 20' into the soil and the tracks on the advancing mechanism began to slip. The grout flow began to drop and the test was halted. The bar was excavated and 25% - 30% of the jets were plugged with kiln dust from the grout mix and the bar was bowed up from horizontal. After cleaning and straightening the bar, the test was repeated with essentially the same result: 15' of travel, 10% of jets plugged and the bar bowed.

The shuttle tool functioned as designed during surface check out, and then was positioned at the face of the cut. When the tool was moved forward into the cut, the shuttle appeared to bind and the cable drive mechanism stretched. This reduced the travel of the shuttle to less than the full width of the tool body. The shuttle, therefore, could not clear a path for the tool, and progress halted at the face of the cut. The stretched cable could not be expediently corrected and the test was ended.

The flex tool was tested at 5,000 PSI and it was unable to construct a barrier. The system could stroke or reciprocate smoothly, pulling the tool through a 180 degree arc on the surface. When the tool was moved into the face of the cut however, it was able to complete only half a stroke. Upon attempting to recover from that stroke, the tool stuck in the face of the cut and the 40,000 lb. pull of the support system hydraulics could not move it. The main body of the tool was damaged when it was extracted from the cut.

LESSONS LEARNED

It is now clear that barrier formation is more difficult than the initial tests indicated. The second generation tooling did not create any satisfactory panels in the more challenging soils at the Oklahoma test site. Operational problems with the tools included jet plugging on solid bar tools #1 and #2, stretching of the drive cable and possible binding of the shuttle on the shuttle tool, and inability of the stroking mechanism to reciprocate the flex tool across the face of the cut.

A new understanding has been developed of the criticality of the surface support equipment. The grout required for successful employment of jetting methods must be much cleaner than grouts used for other soil amendment. It is necessary to use a high quality screening system to avoid jet plugging. Grout mixing must be more complete, and equipment must be maintained thoroughly clean and free from trash that can plug the grouting jets. The grout mixing, holding and delivery capacity must be sized to ensure no interruption in supply or pressure in order to maintain progress through a complete pass of the barrier. A halt invites failure due to jet plugging. The skin friction of the emplaced drill stems and the pulling angles for the flex tool may dictate considerably more hydraulic power requirements and/or consideration of scale factors when this type tooling is applied. The belief is that the geometry of a much larger catenary arc with a greater number of jet subs and much more bearing area in the cut would be a better design.

The excess grout flow that is produced with the use of a rotary tool may be a cost of enjoying less impact from the plugging of a few jets. In addition, the backflow of excess grout through to the surface may clear spoils and mix them better to help in barrier formation.

The use of solid bars with fixed jet patterns can cut formations effectively, but, the barriers produced are thin and the tools are highly susceptible to failure through plugging of a small percentage of jets.

NEXT STEPS

The success of the prototype rotary tool and the realization that the backflow of grout through the barrier slab to the surface may be acceptable factors have prompted the development of two third generation rotary tools. The first, rotary #1, is driven mechanically through a drive shaft from a surface support power unit. The cutting and mixing action come from the kinetic energy of the grout jetting through the tool. The second, rotary #2, will employ the same mechanical drive system to

drive mechanical cutting teeth mounted on the circumference of the lobes of the tool. Grout jets will be used to assist the teeth and to generate the turbulent mixing required to form a uniform slab.
The initial failure of the flex tool pointed up the importance of scale in the design of this tool. A wider tool has been developed and is expected to cut less sharply into the face of a soil formation on each stroke, and ,therefore operate at more "gentle" angles, reducing the tendency to bind in the cut.
The third field testing session will evaluate these tools in progressively more difficult situations beginning with open air testing with water and progressing through shallow cuts through loose material with water, deeper cuts, cuts into unexcavated strata with water and finally cuts into unexcavated strata using grout at full operating pressures. The two most successful tools will then be taken to Fernald for a full scale demonstration on the prepared site.
The Fernald demonstration is to take place in the summer of 1995 with the full report being completed in winter of 1996.

ACKNOWLEDGEMENTS

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DESIGN OF A FULLY MOBILE SUPERCOMPACTION SYSTEM

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ABSTRACT

A design review was undertaken on the feasibility of a solid low level radioactive waste supercompaction system, requiring only a concrete hard standing. The prime aim of such a system was to be totally independent of any reactor site services, except for the disposal of any liquids arising from the supercompaction process.

THE SUPERCOMPACTOR

Although a number of proven high force compactors were investigated in this study, the unit shown in Fig. 1 most closely met the reviewed specification, i.e.:

- Adequate compaction force of 1600 kN,
- Light weight, suitable for transportable use,
- Compact dimensions, particularly the overall height,
- Relatively easily enclosed with primary containment,
- Automated drum feed and puck delivery systems,
- Programmable electrical and hydraulic control systems.

Basically, the main body of the compactor (1) is a circular forged structure, the top section of which forms the hydraulic, high pressure cylinder, while the bottom section is partially removed to permit the drum being compacted to pass through the structure. The hoop stresses produced by the compaction process are absorbed by the bolster (5) and the longitudinal forces are accepted by the remaining material of the structure. The bolster is designed to be 'parked' inside the cylinder when the high preecontamination.
Bottom and top platens (10 &11) are made convex to produce a concave form of pellet, which relaxes to a virtually flat form on cessation of the compaction force. Wear has been reviewed by the original designers of this high force compactor and as a result, it is possible to change the bolster liner, the top and bottom platens and the hydraulic seals in the cylinder.
The overall sizes of this high force compactor are 1.53 meters width x 1.53 meters length and 3.3 meters height. The weight is 22,000 Kg.

THE TRANSPORTABLE SUPERCOMPACTOR

In considering supercompactor concept was, therefore, assumed to meet the following:

- that nothing short of total contamination control would be acceptable - the

compactor operation, together with the drum feeding and the puck discharge would be completedly enclosed within a primary containment envelope.
  the containment envelope would be provided, if possible, with a ventilation system designed to capture radioactive particulate released from the compacted drums.
  a means to be found to ease the difficulty of transferring contaminated pucks from the primary containment, without spreading airborne particulate to the exterior environment.
  suitable drainage system to be provided in the likely event that some drums will contain radioactive liquors.
The Primary Containment is arranged in the classical manner with the minimum of equipment installed inside the envelope and the maximum on the outside to reduce the problems of maintenance. It should be fabricated from stainless steel, polished on the interior surfaces for ease of decontamination.
The entry of drums into the Containment is via the Air Lock system and the exit of the compacted pucks is achieved using a double lid and overpack drum technique or the more traditional, cost effective bagging transfer. There is space around the Transfer Port to park 6 pucks, the height of each puck having been recorded by the Control System, to aid optimum filling of overpack drum or bag.
Roller conveyors, with their drive motors installed outside the Primary Containment, have been employed to provide the horizontal movement of the drums. Prior to entering the Compactor, the drum is pierced in 4 places, close to the bottom rim. The number and position of these holes has been carefully chosen to get optimum draining of any liquors in the drum and to prevent the drum bursting under the compaction load. The liquors draining from the drums are collected in a large capacity Sump, which is an integral part of the Containment and is emptied by a system of pump and centrifuge, to separate any oil from water.
Although the controller of the Supercompactor can see a limited degree of the operations through a small window, all of the processes within the Primary Containment can be viewed by means of a series of CCTV cameras relaying pictures to the Control Room monitor. The control of the High Force Compactor is an automatic function of the programmable system, all other operations being the responsibility of the controller. The control supervisor can override the automatic programme in an emergency, if required.
The various process steps in the Transportable Supercompactor system, and the extent of the Primary Containment envelope are indicated in Fig. 2.
The Hydraulic Power Pack for the compaction process and the filters and fans of the Active Ventilation system are all mounted at the rear of the Trailer in sound proofed enclosures. Suitable openings in the Trailer exterior weather covering enable the circular coarse and HEPA filters to be changed employing the traditional bagging technique. There is also a HEPA filter fitted to the inlet to the Ventilation system to prevent any back diffusion of radioactive particulate escaping from the Containment when the HFC is closed down overnight. All filters, being of circular form, can be disposed of in standard 200 liter drums, suitable for compaction in this Transportable Supercompactor.
The estimated overall weight of trailer and equipment was 38,000 Kg and using a standard 3 axle trailer configuration, pulled by a 3 axle tractor unit, it has been calculated that the trailer axle loadings would all be within European traffic regulations, as were the overall dimensions of the finished trailer.
An illustration of the Transportable Supercompactor arrangement is provided in Fig. 3

FULLY MOBILE SUPERCOMPACTION SYSTEM

The Transportable Supercompactor briefly described above was always going to need some support services from the sites being visited. The prime aim of investigating a Fully Mobile System was to review the possibility of transporting all of the services and equipment with the TSC and, if so, the complications of a totally self supporting unit.
Before commencing the investigation, it was necessary to establish a number of important factors, as shown below:

Number of reactor sites needing compaction services	:	10
Number of sites to be visited (Note 1)	:	5
Average number of 200 liter drums per site visited	:	573
Assumed throughput of compactor per hour	:	10
Average number of pucks per bag	:	4

Time required to pack 4 pucks per bag in minutes : 24
Bag production rate per hour : 2.5
Average number of pucks per 1/2 ISO container (Note 2) : 148
Average number of 1/2 ISO containers required per site visited : 4
Average time to fill 1/2 ISO container in hours : 14.8
Note 1 - It was assumed that it would be more cost effective to set up the Supercompaction service on a reactor site where there was a second site in close proximity, the second site transporting drummed waste to the set-up site. . The subsequent financing calculations took into account the cost of transporting the drums between sites.
Note 2 - The Low Level radioactive waste disposal regulations in the UK stipulate the use of 1/2 height ISO freight containers for the emplacement of compacted material in the Drigg repository. These containers are specially designed to be stacked when full of waste and cement grout, the grouting being undertaken at Drigg through dedicated holes provided in the container lid. The number of pucks assumed to be placed in the container is derived from having 4 rows of bags (4 pucks per bag) with 9 bags per row plus some 4 single bagged pucks filling the odd spaces, making 148 bagged pucks (drums) per 1/2 height ISO containers.
The first requirement of a self supporting system was a Building in which to carry out the compaction service. There are available on the market a number of portable buildings which are easily erected in a short time scale, with a minimum of manpower or auxiliary equipment requirements. Generally, these consist of a series of quickly erected portal frames covered with a reinforced pvc covering.
It was decided to use two different buildings, based upon manageable erection size, one to act as a Drum Buffer Store and the other as the 1/2 height ISO container Loading Building. The Buffer Store had a floor area of 47m2 with a ridge height of 4.4 m. The Loading Building had a floor area of 81m2 and a ridge height of 5.25 m. Both Buildings would be used side by side and in direct contact with each other through a curtain door.
It has been demonstrated that portable buildings of this size could be erected on a prepared hard standing by an experienced team of 4 people in under than two days. It is proposed that one side of the reinforced pvc covering of these Buildings are provided with openings which coincide with the service openings in the Transportable Supercompactor trailer. On the arrival of the TSC, the openings in the Building covering will be attached to the trailer by means of Velcro fastenings, making the complete assembly water and wind proof.
Another important requirement for a self supporting system is a radiological Change Room, particularly as the chosen location for the TSC might not be convenient to a site change room. For the same reason, a Messing Cabin was also provided. The Change Room provided would be in the form of a towable cabin, equipped with monitoring instruments, toilets, washbasins and shower. The toilets should be of the pumped/macerator type, the products of which would be automatically transferred into a dedicated, portable Bowser.
The potentially contaminated water arising from operator washing or showering would also be automatically pumped into a second dedicated, radioactive Bowser, which would also receive active water from the Sump below the High Force Compactor.
As the Drum Buffer Store and the container Loading Building would be classed as Secondary Containment, a classification which would be particularly necessary if, for example, a bag of pucks was accidently dropped and punctured, thus releasing some contamination into that Building, a small emergency type ventilation system is connected to the two portable Buildings and the Change Room. This portable, wheeled Air Mover is fitted with a fan and circular HEPA filter which can be changed using the bagging technique.
To permit the heavy 1/2 height ISO freight containers to enter the Loading Building, rails are installed inside and outside the Building. A portable, dedicated Crane is also assembled and placed on its own rails, straddling the container rails. As the utility being visited has a little used side loading truck, specifically designed to handle ISO freight containers, there was no call for this type of equipment to be included. The dedicated Crane could, if required, be modified to undertake the off-loading of 1/2 height ISO containers.
Once the container has been positioned on the external rails, it can be rolled into the Loading Building through the curtain door provided until it is located under the Crane, which, combined with a Tilting Lifting Frame, can be used to remove the

container lid and park it in the Lid Stillage provided. With the Roller Conveyor placed under the TSC puck Transfer Port and an empty bag sealed to the Port, the only item requiring connecting and starting is the diesel Generator to provide electrical energy to power the complete system.
The estimated convoy of vehicles likely to be required to transport the Fully Mobile Supercompaction System from site to site, would be:

Unit 1 - the Transportable Supercompactor
Unit 2 - a flat bed Truck towing the Change Room and carrying:
- 2 Portable Building Structures and pvc covering, accessories,etc.
- 1 Power Generator, cables and fuel
- 2 Bagging Stillages and Bags
- 1 Liquids Bowser
- 2 Roller Systems for 1/2 height ISO container

Unit 3 - a flat bed Truck towing the Messing Cabin and carrying:
- 1 Liquids Bowser
- 1 Electric Fork with Drum Grab
- 1 Crane rail, A frame, hoist unit and tracks
- 1 Container cover Lift and Tilt Frame
- 1 Working Platform (for erecting buildings)
- 1 Portable Air Mover

METHOD OF OPERATION
The operating team of four people would erect the complete system, as shown in Fig. 4, over a two day period, using the portable Crane, Fork Lift and Working Platform. The Transportable Supercompactor would be planned to arrive at the end of the second day and would be connected and tested before the compaction process commenced on day three. The same four personnel would also be responsible for operating the plant to compact drums and load bagged pucks into the ISO container. At the end of the campaign, the team would also dismantle the complete system.
To spread the radiation dose uptake, it is recommended that the operators regularly change stations to share the work load. The 4 work stations are:
1. Unload drums of solid LLW from the site transporter into the Buffer Store Building and the loading of drums on to the Supercompactor feed conveyor, all by means of the small Fork Lift provided.
2. Control room supervision, logging drum information into data bank and generally controlling the process via window and tv monitors, mimic diagram, etc
3. Bagging out of pucks in contamination free manner to a point on the output conveyor where they are ready for lifting into container.
4. Loading of bagged pucks, each of which can weigh up to 1 tonne, from the end of the conveyor to the ultimate position in the 1/2 height ISO container. The operator at this station, together with operator No 3 will also be responsible for changing the filled ISO container for an empty unit.
It has been estimated that the average dose uptake for each of the operators would amount to some 3.4 mSv per year, based upon compacting 3000 drums of typical waste in that year. This calculation indicates that a higher utilisation factor for the TSC of, say 8000 drums per year, would still allow the same team of 4 to operate the unit without exceeding the recommended target radiation dose uptake Fig. recommended by ICRP.
The safety review undertaken during this design study investigated, among a wide variety of subjects, the problems of decontaminating the System for transport on public roads. The Primary Containment meets the transport regulatory authorities requirements of preventing any radioactive material being released into the transport vehicle. It was, however, recommended that other precautions were observed to ensure radiological safety, i.e., that the Sump was completely evacuated of any liquids, by swabbing if necessary, until the surfaces were dry; any loose waste in the puck posting area was removed and transferred out in the final bag; a security hinged cover be fitted externally over the puck Transfer Port; and all suspect surfaces to be monitored and swabbed clean, if necessary, rather than washed down. After all those precautions had been observed and the site health physicist satisfied that the TSC was clean enough to issue the clearance certification, the unit should be free to leave the reactor site.
The complete layout of the Fully Mobile Supercompaction System is show in Fig. 4
COST
A careful analysis was used to establish the likely capital equipment costs involved

in building the Fully Mobile Supercompaction System.
The authors have then calculated the actual compaction cost per drum, based on depreciating the complete Mobile Supercompaction System over a 10 year period, compacting only 3000 drums per year and that 5 sites were visited and including in that calculation all operating costs of UK staff, consumable, transport from site to site, etc. With all of those factors taken into account, the actual cost for compacting Low Level Waste is in the order of 100 ($150) per drum. With better utilization of the System, a lower cost per drum would result. For example, if the throughput was raised to 5000 drums per year, the cost for the compaction operation reduces to around 77 ($116) per drum, even with the increased operating cost.

CONCLUSIONS

It is possible to design a Fully Mobile Supercompaction System which can be transported from site to site on two flat bed Trucks and the High Force Compactor Trailer. It also proved possible to assemble the System and have it operating in 3 days. The disassembly and radiological checking required to release the vehicle from a licensed site should also consume no more than three days.
It also proved feasible to be totally independent of any site services apart from a concrete hardstanding of 630 m2 with a capacity to accept axle loads of up to 10,000 Kg.

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REBAR CORROSION DUE TO CARBONATION IN STRUCTURAL REINFORCED CONCRETES FOR NEAR-SURFACE LLW REPOSITORIES - A CRITICAL FAILURE MECHANISM

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ABSTRACT

The concrete roof of a near-surface radioactive waste repository is the principle protection against water infiltration and intrusion. The following potential roof failure mechanism is examined: Carbon dioxide generated by the biodegradation of organic materials in the repository initiates corrosion of reinforcing steel embedded in the concrete roof. Because the bottom surface of the roof is mostly under tension, it is susceptible to cracking. The migration path for carbon dioxide is through cracks in the concrete between the bottom of the roof and the reinforcing bars. Carbonate corrosion of the reinforcing bars may result in concrete spalling, more extensive rebar corrosion and ultimately structural failure. Attention is brought to this failure mechanism, because it has generally been overlooked in repository performance assessment.
Literature relevant to the above failure is reviewed. Prerequisites for rebar corrosion are the presence of carbon dioxide and oxygen in the repository gas, high relative humidity and through cracks in the concrete. High carbon dioxide concentrations and relative humidity are expected in the repository. The oxygen concentration in the repository is expected to be very low, and that is expected to minimize rebar corrosion rates. Cracks are likely to form in locations with high tensile stresses. Healing of the cracks could be a mitigating factor, but based on our analysis, it can not be relied on. To minimize the potential of this failure mechanism for the Intrusion Resistant Underground Structure (IRUS), Canada's proposed near-surface repository, the carbon dioxide will be absorbed from the repository gas by reactive, porous concrete placed between the waste and the roof.

INTRODUCTION

Reinforced concrete is the principle structural material of construction for most radioactive waste repositories sited in the vadose zone. A concrete roof acts both as an intrusion barrier and as an infiltration barrier. The natural degradation of this barrier material will ultimately lead to a loss of integrity and the infiltration of water into the repository.

The assessment of degradation processes of reinforced concrete has been the subject of significant R&D, with the objective to design durable structures and estimate the rate of degradation. Radioactive wastes placed in near-surface repositories require 300 to 500 years of isolation from the environment. During that time period the radionuclides such as 137Cs, 90Sr, and 3H, with half lives 30a decay to insignificant concentration levels.

The carbon steel reinforcing bars can be one of the weak links in the long-term stability of the concrete structure. As long as the steel is encapsulated in the concrete, the steel is in its passive state and hence its corrosion rate is very low. However, the ingress of corrosive agents, by diffusion through the concrete or through cracks, will result in accelerated corrosion of the reinforcing bars and the ultimate failure of the structure. The most important chemicals responsible for accelerated corrosion are chlorides and carbon dioxide. Carbon dioxide is generated in the repository by the biodegradation of organic waste material. Carbon dioxide reacts with the alkaline (mostly calcium hydroxide) components of concrete shifting the pH of the concrete pore water from the alkaline towards neutral, where carbon steel is in the active corrosion range.

Requirements for the retrievability of waste from near-surface repositories vary with jurisdiction. Where retrievability is not a requirement, the space between the waste packages is generally backfilled with porous concrete. The concrete backfill is reactive with the carbon dioxide and thus the repository pore gas is not expected to contain carbon dioxide. Where retrievability is a requirement, usually only small quantities of materials are present that are capable of absorbing the carbon dioxide generated by the decomposition of organic material. In those repositories, a high concentration of carbon dioxide can be present in the repository gas, and hence structural failure initiated by carbonate-induced rebar corrosion is a possibility. In this paper we examine the potential mechanisms that can lead to an early failure of the repository structure, review the literature, and its relevance to the expected environment, and suggest an approach for the removal of the carbon dioxide generated in that repository to reduce the potential of rebar failure.

POSTULATED MECHANISM FOR REBAR FAILURE

The prerequisites for rebar corrosion are the migration of chloride and/or carbonate to the rebar, the migration of oxygen to the rebar and the presence of moisture on the rebar surface. Papadakis et. al. (1) examined the parameters that determine the rate of carbonation of concrete. They found that the rate of carbonation is highly dependent on the relative humidity (RH) of the environment the concrete is placed in. Carbon dioxide can migrate to the concrete through the gas phase, or as carbonate through the liquid phase. In dry environments, most of the pores in the concrete are filled with air. Since the rate of the gas phase diffusion is about four orders of magnitude higher than liquid phase diffusion, the carbonation reaction proceeds fastest in low RH environments. Experimental results (1) indicate that the carbonation rates are asymptotic to zero at 100% relative humidity. The RH of repositories placed in humid climates is expected to be very close to, or equal to 100%. In such an environment the migration rate of the carbonate front to the rebar is negligibly slow, and the repository concrete structure is not expected to absorb a significant portion of the carbon dioxide generated in the vault. In arid climates the carbonation of the concrete is of no concern for rebar corrosion, since the rebar is expected to be dry, and hence corrosion rates will be negligible.

Another, but more important route for carbon dioxide penetration to the rebar is by gas-phase diffusion through the fractures in the concrete, when the fractures are dry or only partially filled with water. Fracturing of the concrete takes place in regions where the structure is under tension. When the concrete structure is subjected to design loads, mostly micro cracks are formed. Through cracks, that lead from the surface, through the concrete cover, to the rebar, can also develop in the structure. They can be conduits for carbon dioxide and oxygen diffusion from the repository or the outside soil pore gas to the rebar. There are two factors that can mitigate rebar corrosion: the expected low oxygen concentration in repository pore gas, and the potential for the healing of the cracks in concrete. The examination of both of these factors follow.

The oxygen concentration in soil pore gas, outside the repository, is not much lower than in air. This gas can penetrate the repository mostly by atmospheric pumping, driven by atmospheric pressure changes. While most of this will take place through the bottom of the repository, such as the drain line leading to the sump, some may

occur at joints in the concrete. In the repository, oxygen is scavenged by steel corrosion and the aerobic degradation of organic matter. However, the presence of oxygen inside the repository in regions where air ingress takes place can not be ruled out.

The healing of fractured concrete has been widely observed, and has been the subject of several investigations. Guppy (2) noted that only a limited amount of experimental and archival work is available on crack healing. The mechanism of crack healing is not well understood, but the most likely process responsible is calcite formation under wet or high humidity conditions. Investigators found no evidence of amorphous material, such as CSH gel formation in the cracks. The following components are required for crack healing:

- flooded or high humidity;
- supply of carbonate or carbon dioxide;
- and a supply of portlandite.

Portlandite ($Ca(OH)_2$) in cement is produced by the hydration of Ordinary Portland Cement (OPC). In the newer, blended cements generally used for repository construction, a portion of the OPC is replaced by Blast Furnace Slag (BFS) and/or Pulverized Fly Ash (PFA), resulting in reduced portlandite production that is partly due to the reduced OPC content, and partly due to the pozzolanic reaction between portlandite and BFS/PFA. We were unable to locate literature on the crack-healing properties of these advanced cement compositions.

The carbon dioxide concentration in air is approximately 0.03 vol%. Most of the experience in crack healing is with air containing ambient carbon dioxide concentrations, or water in equilibrium with ambient air. The carbon dioxide concentration in the repository gas is expected to be at least several percent, which is two to three orders of magnitude higher than ambient.

For crack healing to take place, calcite must form within the crack. For this, the carbon dioxide (or carbonate) supply to the crack controls the reaction rate. Under those circumstances, there is ample supply of portlandite in the concrete pore water adjacent to the crack. When there is a large concentration of carbonate ion in the pore, and limited portlandite supply, calcite formation in the cement matrix, rather than in the crack is favored. The carbonation front advances into the concrete structure. This does not lead to crack healing. Brodersen et al. (3) assessed crack healing in flooded conditions, and advanced a similar argument to explain why cracks only heal for a short time after their formation. Thus the concrete used for repository construction and the repository environment are different in at least two aspects from those associated with in current field experience, and both differences disfavor crack healing. There is a need for research in crack healing, where both the repository-concrete and repository-environment are simulated.

LOCATION AND CONSEQUENCES OF REBAR FAILURE

The physical integrity of the repository concrete roof is essential for both infiltration and intrusion prevention. The function of the walls are less critical, their primary role is to provide a stable support to the roof. Only extensive deterioration of the walls will compromise the integrity of the roof. Tensile stresses are expected to be highest in the roof, since it has to carry its own weight, the weight of the water sheading layers and the soil cover. Thus premature roof failure has been the focus of our analysis for the IRUS (Intrusion Resistant Underground Structure) planned for the Chalk River Site. While we expect to have significant quantities of chloride ions present in the repository pore water, the chloride initiated corrosion of the rebars in the roof is not considered to be an issue. The repository roof will be supported by the walls, but it will be poured over repository contents covered by sand and porous concrete. With time the contents of the repository are expected to subside, creating an empty space between the waste and the roof, that will form a diffusion barrier for the salts leached from the waste. Our concern with carbonate induced rebar corrosion is due to the following factors:

The formation of cracks in the roof and the ingress of carbon dioxide into the cracks can start soon after the repository roof is poured.

Once corrosion is initiated, the deterioration of the structure is a progressive process. As corrosion proceeds, the expansion of the corrosion film can apply pressure on the concrete surrounding it, resulting in the widening of the fracture, and the eventual spalling of the concrete cover (4). This process than exposes more rebar surface area to the corrosive environment.

PREVENTATIVE MEASURES.

While the projected low oxygen concentrations in the repository provide some assurance of low corrosion rates of rebars exposed to a high carbonate environment, an additional preventative design feature was deemed desirable. A layer of porous concrete with a high surface area will be placed between the waste and the repository roof. The portlandite and other basic components of the porous concrete will react with and hence remove the carbon dioxide generated in the waste, and thus minimize the carbon dioxide concentration at the inner surface of the roof. The composition of the porous concrete has not been finalized yet, but it will have sufficient capacity to remove carbon dioxide generated during the first 500 years following repository closure.

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SAFETY EVALUATION FOR THE CS-137 ILW IN REPOSITORY OF ABADIA DE GOIS, BRAZIL

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ABSTRACT

A shallow ground intermediate level waste repository is being designed for the final disposal of the radioactive wastes which resulted from the Goinia accident. Its site has already been selected and approved by the regulatory authorities. It lies in Abadia de Gois in the outskirts of the city of Goinia. A brief description of the repository is given and a simple but otherwise conservative safety analysis is presented. This analysis consisted in the evaluation of the effective dose arising from the use of water from a hypothetic well located at various distances from the edge of the repository building along the direction of the groundwater flow. The source term has been evaluated on the basis of a sudden disruption of the building right after construction assuming that all the activity present at that time reaches without delay the water table. The dose limit of 0.25 mSv/y set by the authorities is reached if the well is located 5.5 m away from the building edge at a time equal 768 years after the repository disruption.

INTRODUCTION

In September 1987 a shielded strongly radioactive Cs-137 source was removed from its protective housing, in a teletherapy machine in an abandoned clinic in Goinia, in the Brazilian state of Gois. The source capsule was subsequently ruptured, and the remnants of the source assembly were sold as scrap to a junkyard owner, who noticed that it glowed blue in the dark. Over a period of days friends and relatives came to see the fascinating phenomenon. Small fragments of the source ended up in the homes of several families and some were dispersed to places throughout the city. By the time the accident was discovered and counter-measures were initiated, many people had incurred doses of radiation and many residences and public places were contaminated. Four persons ultimately died and 28 people suffered severe radiation burns. The decontamination and clean up activities resulted in a total of 3500 cubic meters of radioactive wastes which were accommodated in over 5000 packages of various types and activity concentrations. These packages have been temporarily stored in the open on concrete platforms built for this purpose at a site near the village of Abadia de Gois, about 23 Km from the center of Goinia (1).
Subsequently the Brazilian authorities, namely, Comisso Nacional de Energia Nuclear (CNEN), determined the site-selection procedures for the final disposal of the radioactive wastes from Goinia. In all 189 preliminary areas within the state of

Gois were identified and evaluated and from these, three candidate sites were chosen and presented to the state government for the final site selection. It was then jointly decided with CNEN to start the environmental impact assessment in the area nearest the temporary storage site. The selected site will house two different repositories. The first one will receive the packages whose activity concentration is lower than 87 Bq/g, which is the exemption limit, and the other one which will receive the 1078 packages with activity concentrations higher than that value and which will be dealt with in this paper.
A brief description of the repository is given and then a simple but otherwise conservative safety evaluation is presented.

DESIGN FEATURES

The repository design, for the final disposal of low- and intermediate-level radioactive waste makes use of a multiple barrier concept, in which the release of radioactive material to the environment is avoided by a succession of barriers, called engineered barriers. It consists of a reinforced concrete structure which is to be constructed in an excavation in the unsaturated soil and covered with seven different layers. The concrete building has external dimensions of 65.20m x 20.50m x 5.85m, and is situated in the center of a 300m x 300m terrain. The concrete structure is partitioned in three blocks each one with a different height. Within the concrete building 1078 packages will be deposited totaling a volume of 3200 m3 (2). The spaces inbetween packages and between the packages and the concrete building will be filled with a buffer material, reducing the voids to a minimum. The compacted buffer material (of low hydraulic conductivity) is composed of sand plus a minimum of 5% clay.
The external top layers are constituted by different materials with main function of impermeabilization (avoid and reduce a water infiltration), drainage and filtration, as indicated in Fig.1. The minimum distance between the top of the repository and the soil surface is 1.60 m. The inclination of the top layers will permit the runoff of the water in direction of the side region beyond the structure, conducting this water to the aquifer.
The engineered barriers like the steel/concrete barrels, the concrete containers, the buffer material and the repository concrete structure will retard the migration of activity for a long period of time when its level can be considered relatively high.
A fence around the repository, permanently inspected in order to guarantee its integrity by the local security service, has the purpose of avoiding intrusion.
The design and construction of the repository, including the transportation and the handling of the radioactive wastes, are conceived to be performed so that the Cs-137 release is of negligible level. The main planned measures to minimization of leakage possibility, during the transportation and handling of the packages are: maximum shortening of transportation route of the packages; utilization of safe equipments and vehicles; special conditions of protection during package handling, if necessary utilization of impermeable plastic cover; and operation supervision under safety and radiologic control, in strict accordance with a Radioprotection Plan.

SAFETY ANALYSIS

Preliminary safety analyses have indicated that the most important dose pathway to members of the public would be the use, for whatever purpose, of contaminated groundwater around the repository site. Thus the following scenario was selected for the safety evaluation: groundwater contamination, transport of radioactivity to a well along the direction of the groundwater flow and use of the well water for various purposes. The concentration of Cs-137 in repository, soil and biosphere is estimated by analytical methods, with help of parameters and input data resulting from laboratory measurements and field tests.

Source Term

It has been deterministically assumed that the groundwater below the repository building got contaminated in such a way that its activity concentration became equal to that, that would result from the dilution of the whole amount of caesium chloride (total activity 4.580E+13 Bq) present in the repository in a volume of water corresponding to the void volume of all the waste packages. This conservative assumption neglects the retention of caesium in the waste material, by the packages structure, by the repository itself and in the unsaturated soil layer above the water table.

Transport of Activity

The Darcy's Law for porous medium was employed, which allows determination of the average velocity of the groundwater. The representative parameters of groundwater movement (velocity, flow rate and direction) were determined based in properties of the local soil/aquifer and water table variations. The fundamental flow parameter (average pore water velocity or seepage velocity) was evaluated according to (3). The radionuclide migration velocity will be significantly lower than that of the groundwater, due to the sorption phenomena that takes place in the soil. It must be remarked that along the aquifer path, only a fraction of the radionuclide amount that leaked from repository will be present in the water, while the remaining part will be absorbed by the soil. This retardation mechanism prevents that a high concentration of activity can reach the water of an hypothetical well. The average velocity of a Cs-137 ion through a porous medium can be connected with the movement of ground water by the following equation:
See Eq. 1
where: Vr = average velocity of Cs-137 radionuclide; Va = average velocity (average pore water velocity) of the groundwater, whose value is 6.0 m/year for the terrain. The retardation factor, F, is a term that condensates the several retardation mechanisms of physical-chemical nature, expressed by:
$$Fr = (1 + r.e.Kd) \qquad (2)$$
where: e = number of porosity = (1-f)/f (4) and (5); r = density of the solid phase = 1.715 g/cm3; Kd = distribution coefficient. All the parameters contained in the retardation factor resulted from measurements and are characteristic of specific repository site. The distribution coefficient, Kd, is the most important parameter that constitutes the retardation term. The distribution coefficient represents the ratio of the concentration of the radionuclide sorbed on the soil and the concentration of the same radionuclide in the ground water (4,5). The employed value of the distribution coefficient was obtained from measurements performed by the "Instituto de Radioproteo e Dosimetria (IRD)" for a total of 32 samples of the soil of the local of the repository. The recomended value for the Kd is of 430 cm3/g (average value).
The radionuclide concentration along the aquifer path (distance X), neglecting the effects of convection and diffusion, can be determined by following equation (5):
See Eq. 3
where, C = Cs-137 concentration along the X-axis; C0 = 1.434E+08 Bq/l = Cs-137 concentration at t=X=0; l = 0.0231 year-1 = constant of radioactive decay of the Cs-137. The time necessary for the Cs-137 radioisotope to move through a distance X from a release point, called migration time, t,can be determinated by the following equation:
See Eq. 4
The calculation results are presented in Table I (columns 2 and 3), that contain the calculated activity concentration of Cs-137 in groundwater and the migration time, for various distances along the groundwater flow direction from the repository building edge.
Dose Evaluation
The analysis is based on the evaluation of the dose due to the use of water from a well hypothetically situated at the site of the repository. The radiation dose rate was estimated for a person that would belong to the critical group, considering the following dose pathways:
1) drinking water;
2) milk (water - cow - milk - man (ingestion));
3) meat (water - cattle - meat - man (ingestion));
4) milk (water - pasture - cow - milk - man (ingestion));
5) meat (water - pasture - cattle - meat - man (ingestion));
6) vegetables (water - soil - vegetable - man (ingestion)).
The ingestion dose rate, D, can be determined by the following equation:
See Eq. 5
where, Fc = 1.4E-08 Sv/Bq = Dose factor for internal exposure via ingestion of Cs-137; Ai = Activity of Cs-137 incorporated annually, Bq/y, and the index i, varying from 1 to 6, represents each one of the previously cited dose pathways. The activity incorporated annually, Ai, was estimated by using the equations and parameters recommended by (6).
Calculated values of dose rate (Sv/year) for several migration distances from the repository site, are presented in Fig. 2.

CONCLUSIONS

With help of the Table I and Fig.2, it is possible to estimate the annual dose due to the use of the water of a well in the repository terrain, as function of the distance repository-well. It must be pointed out here that the minimum distance from repository edge to fence is about 150 m.

In Table II one can find the main results in a condensed form. It can be observed that the Maximum Permissible Dose Rate established by the Brazilian regulatory authorities (0.25 mSv/y) (7,8), is reached when the Cs-137 concentration in the well water corresponds to 2.8E+03 Bq/m3. In this case, the migration time is 768 years and the well is situated at a distance of 5.5 meters from the repository structure edge.

It can be concluded that the resulting effective dose will be lower than the legal limit established by the Brazilian regulatory authorities, if the well is located more than 5.5m away from the repository edge in the direction of the ground water flow

This simplified, but conservative model used in this analysis gives a good indication of the minimum distance around the repository where the use of local ground water will not lead to the incurrence of doses above the established limit (7).

The resulting time span, as presented in Table II, is long for all practical purposes to be used as the period of institutional control. A more precise evaluation of the period of institutional control requires the use of a more realistic model, that considers the migration through several engineered barries, taking in account others processes (leaching, dissolution, diffusion, etc).

Work is been developed presently in Brazil for detailed modeling of the processes of Cs-137 migration in repository and adjacent soil, by considering saturated and unsaturated flow.

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NATIONAL EXPERIENCE IN COLLECTION AND STORAGE OF UNCONDITIONED RADIOACTIVE WASTE IN GHANA

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ABSTRACT

Radoiactive materials have been used in Ghana for more than three decades. The areas of application include agriculture, hydrology, medicine, research, industry and environmental studies. They also find applications in well logging, thickness gauging, radiography, radiotherapy and as well as calibration standards. From all these uses, radioactive wastes produced that must be managed safely and efficiently. Presently the waste management program in the country is handled

by the Radiation Protection Unit. Liquid, solid and mixed radioactive wastes are collected separately. Ghana has no treatment and conditioning facilities at the moment, however, there are plans to acquire the necessary facilities to immobilize the sealed sources in concrete matrix using 200l drums.

INTRODUCTION

The management of radioactive waste in Ghana is the responsibility of the Ghana Atomic Energy Commission. The Commission operates the National Nuclear Research Institute, NNRI, which through its Radiation Protection Unit, RPU, keeps records of all radioactive sources in the country. The Commission was established by an Act of Parliament in 1963 but the necessary legislative instrument which would enable the Commission perform its regulatory functions was not passed until January 5, 1993.

The law has not been published in the official gazette but it is known that it establishes a Radiation Protection Board under the Atomic Energy Commission. The Board is the licensing authority in Ghana for the issuance of licenses required for the purpose of radiation protection. A waste management committee, set up at NNRI, has now acquired the legal support to carry out its duties. Current waste management practices have been confined mainly to collection and storage of waste in a temporary facility controlled by the RPU.

Sources of Radioactive Waste in Ghana

The main producers of radioactive waste in Ghana are NNRI, the universities, the nuclear medicine unit and to a smaller extent the industry. At the NNRI, the main type of waste is that which arise from beta-scintillation counting; they are organic liquid waste. The waste from research activities in the universities is basically of the same type.

The activity concentrations range from few bequerels to hundreds of mega-bequerels or more. The composition of the wastes range from contaminated items such as paper, vials, filters, plastics, gloves and glassware to liquid effluents, carcasses and sealed sources. The radionuclides composition include C-14, Cs-137, Am-241, H-3, S-35, P-32, I-125, Tc-99m, etc. Another source of potential waste is the use of some gas chromatographs with electron capture detectors. One such chromatograph recently received in the Chemistry Department at NNRI employs 10mCi of Ni-63 as the source of electrons from the detector system; nickel-63 is a 100-year half-life beta emitter.

The NNRI, together with the University of Ghana, operates a Nuclear Medicine Unit at the teaching hospital. From this unit there arise two main types of wastes, i.e. solid and liquid wastes. The solid waste is the aggregate of expired technetium-99m generators. These are imported into the country at the rate of one a month and upon expiration are stored at the storage facility at NNRI. Eventually we plan to dismember the generators and remove the lead housing and thereby reduce the volume of waste significantly. The other type of waste generated from the Unit is that from radioimmunoassay, RIA, procedures. These are liquid wastes containing very low activities of iodine-125. RIA procedures are also conducted at the rate of about once a month.

There is a 30kW research reactor in the Department of Nuclear Engineering of Chinese design and a Co-60 source (7.5kCi, 1975) for food irradiation research. The operation of the research reactor will result in the generation of the following waste: spent ion exchange resins from water purification (cation and anion, volume of resins in columns is about 50l), compatible solid waste from mechanical filters, non-compatible solid waste such as ion-exchange resins and liquid waste aqueous and organic. It is anticipated that about two 55 gallons waste drums will be needed per year for the research reactor.

There is a reduced utilization of radioactive sources in industry compared to NNRI. Their sources are either sealed beta-sources used to measure thicknesses of materials, or those which have been imported into the country by oil companies for exploration. The latter are returned to the places of origin when the exploration work is over and the RPU keeps a record of all such sources. Another feature of our documentation process is the cooperation of RPU with the Customs Excise and Preventive Services, CEPS. In this regard the CEPS instructs all importers to obtain clearance from the Commission prior to actual importation. In this way the RPU knows

which sources are imported into the country and who is responsible for them. There is a radiotherapy unit at the School for Medical Sciences in Kumasi which uses Ra-226 needles in brachytherapy. The radiation safety aspect of its operation is under the Radiation Protection Unit of NNRI.

WASTE PROCESSING AND STORAGE TECHNIQUES

There is no treatment of radioactive waste in Ghana apart from storage for decay.

Collection and Storage of Liquid Wastes

Liquid wastes which are mainly produced from the hospitals and research centers are made up of mainly liquid effluents containing scintillation materials used to measure
C-14, H-3, I-131, Tc-99m and I-125 from nuclear medicine. For volumes of waste below exemption level, discharge into the sewage system is recommended after decay and dilution with sufficiently large volume of water and with the approval of the Radiation Protection Board. For volumes of liquid wastes which are not suitable for discharge into the sewage system, plastic or glass bottles are used. These are provided with appropriate labels bearing the name of the person or institute, type of radionuclide(s), activity, date of storage and source identity. They are then kept in a store room meant for radioactive wastes for subsequent disposal. Glass bottles are used for the organic wastes which contain mostly scintillation materials and plastic bottles for inorganic wastes. For relatively high activity wastes, the glass or plastic bottles are kept in lead or concrete shielding.

Establishments which find it difficult to manage their wastes are advised to send them to the NNRI after preparing the wastes in accordance with requirements. In this case, the person or establishment generating the waste notify the NNRI. The necessary transport arrangements to collect the wastes from the premises are made. The transportation is usually provided by the sender, but where necessary the institute finds its own means of transport.

Collection and Storage of Solid Waste

In order to facilitate subsequent handling and storage, the waste producers are advised to segregate the combustible wastes from the non-combustible wastes. The combustible wastes are made up of mainly contaminated vials, tissues, paper, filters,
plastics and carcasses from animals used for experiments. The non-combustible wastes compose of sealed sources, glassware and contaminated metallic scraps.

For wastes which can be handled by the individual, advice is sought from the NNRI. Wastes with half lives less than one year are advised to be stored for at least five half lives to reduce the activity below exemption. All sealed sources are however asked to be sent to NNRI. Wastes below the exemption level are advised to be disposed off onto the landfill. The combustible wastes below exemption can be incinerated in an incinerator meant for domestic wastes and the ashes treated as solid wastes which can be dumped onto the landfill. In the hospitals, the wastes are incinerated with other toxic chemicals after decay. For wastes which cannot be handled by the individual, it is recommended that they should be sent to the suppliers if there exist a contract agreement between the two parties (in the case of sealed sources). If there is no such an agreement the NNRI is notified, and the necessary arrangements made to collect the wastes. The combustible wastes are normally packaged in plastic bags and the sealed sources in lead containers.All types of sealed sources are exempted from ordinary disposal.

At the NNRI, treatment and conditioning facilities are not available at the moment apart from interim storage. there are however, plans to acquire the necessary facilities to immobilize the sealed sources in concrete matrix 200l drums.

STORAGE FACILITIES

The storage room currently in use consists of a concrete building with room size 5mX3m. This is located about 30 meters from the Chemistry department building. It is provided with a lock and a radiation warning sign.

A second storage facility was constructed in the early sixties for the storage of spent fuel and interim storage of other wastes from an anticipated 2Mw research reactor which was never built. The facility is presently not in use. This facility will require rehabilitation and safety and performance assessment. When completed, it can be used as a disposal site for spent sources and wastes from the 30Kw research reactor which has just been commissioned.

RADIATION SOURCES REGISTRY

The unit has developed a computerized database package to identify, monitor and

control of spent radiation source in Ghana. The database allows on-line queries by source name, characteristic, activity and location etc., and supplies status reports on any source in the country. The package creates, updates, sorts and organizes data and generates various reports.

CONCLUSION

In conclusion, one can say that the radioactive waste management is in its infancy in Ghana. With the promulgation of the Radiation Protection Law we are ready to carry out our duties to safeguard and protect people and the environment from the possible dangers associated with the use of radioactivity.

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TREATMENT OF IODINE-125 EFFLUENTS BY COLUMN METHOD USING MALAYSIAN LATERITE SOIL

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ABSTRACT

Soil column experiments were conducted to treat effluent wastes containing 125I by using four different samples of laterite clay materials commonly found in the local environment. XRD and XRF analyses showed that kaolinite, illites, aluminum hydroxides and iron oxide (geothite, a-FeOOH) were the major components of the soils. Iron oxide content was high and responsible for the yellowish-brown or reddish color of the laterite clays. Batch experiments using kaolin were conducted from pH 3.0 to 9 using spiked solution of 125I to determine the optimum pH condition for treatment. pH range from 4.5 to 5.0 were the determined values for removal of the 125I in the effluent where kaolin exhibited net positive charge and attracted the 125I anions. Soil column experiments using effluent wastes from hospitals, adjusted to pH 4.5 showed that about 87 % of 125I radioactivity was removed after treatment. At the moderately acid soil condition, pH between 4.5 to 6.5, positive charges predominated on the kaolinite, geothite and aluminum hydroxides. The negative 125I anions were electrostatically held by the positive charges on the mineral components of the laterite clay materials. Environmental parameters, C.O.D and pH values were within the local environmental quality criteria's values required. The treated effluents released to the environment will undergo further dilution, decay and will not harm the environment. This work shows treatment process by soil column provides another alternative for present and future producers of 125I effluent wastes to treat their wastes at their premises.

INTRODUCTION

Iodine-125 radioisotope is routinely used in clinics, hospitals and research institutes for blood cell screening of hospital patients and blood donors. Radioactive effluent wastes if not managed properly have the potential to contaminate the local environment. Even though the 125I radioactivity content is not high, it can pose negative effects on the environment. The radionuclide has toxic properties and a half-life of 59.7 days (1).

Choice of Soil Samples

Many soil do not exhibit significant ion-exchange absorption of iodine (2-4). Soil material samples in the presence of kaolinite, gibbsite and geothite with low net negative charge gives low positive charges between 2 - 5 cmol/kg depending on the soil pH (5). For soil with pH value between 4.0 and 6.5, the net overall charge for the soil is positive (5). The formation of the positive charges enables the negative anion 125I from effluent wastes to be attracted to the soil. The process lowered the aqueous effluent radioactivity. The charge characteristics of selected soil colloids are shown in Table I (5).

Laterite soil samples collected from four different locations in Malaysia were used in the experiments. Laterite is a residual ferruginous material in the soil and frequently exists in the form of lateritic pellets or nodules; locally it has developed into true laterite. The samples have different colors and composition contents of clay minerals and minerals. The minerals have a pH dependent charge characteristic. The reddish brown and yellowish color of laterite soil samples shows high composition content of iron oxide. In the literature, it was reported that the soils have texture and composition that vary. Normally a matrix of finely granular gibbsite containing different proportions of limonite, hematite, chamosite, and kaolinite that forms the actual ores (6). There are also soils which have low-grade iron-rich bauxite (or aluminum laterite) (6). Specific adsorption of anions and

cations by iron oxides at different pH equilibrium solution was reported in the literature (7). Literature reviews on the sorption of iodine radioisotopes by various minerals can be obtained from (8-9)
The choice of the soil material are due to its acidity where the soil samples were collected, presence of kaolinite clay minerals, aluminum hydroxides and high content of iron oxides. The minerals have variable (pH - dependent) negative charges and exhibits modest positive charges at low pH values and produced overall net positive charge. The positive charges can attract the anions of 125I from the effluent and eventually being released to the environment satisfying the local environment regulations. The laterite soil is easily available in Malaysia and its characteristic properties can lead to an effective effluent treatment by the column method.

Objective of Experiment

The objective of this study is to treat aqueous effluent wastes containing 125I by using soil column techniques using laterite soils. This can provide another alternative to treat effluent wastes containing 125I radioisotope. The wastes are currently generated from clinics, hospitals and research institutes in Malaysia. This was accomplished by : (1) Batch experiment using kaolin clay minerals to determine the optimum pH values suitable for optimum removal of the 125I radioisotope. This is to identify the required optimum pH values that are suitable for the conditioning of the laterite soil columns (2) Proper preparation and conditioning of the soil columns at the required experimental conditions (3) Four different soil column treatment experiments conducted simultaneously with the same experimental conditions and duration.

EXPERIMENTAL PROCEDURES

In the soil column experiments, aqueous effluent wastes containing 125I radioisotope was directly obtained from a local hospital. In the batch experiment, 125I in the form of NaI in sodium solution from Amersham International was used. High Resolution Hyper Pure Germanium Coaxial Detector was used to resolve gamma ray of 35.5 keV energy emitted from the 125I radioisotopes (1). High spiked 125I effluent activity was used in the batch experiments while in column experiments the effluent wastes activity was slightly higher than the wastes normally received from users. The aim is to produce 125I spectrum peak with the gamma spectroscopy method.
The batch experiment was conducted using kaolin clay minerals obtained from Source Clay Minerals Repository, Geological Science Department, University of Missouri-Columbia.(10). Ten stock solutions of 100 ml spiked with 125I were prepared between pH 3 to pH 9. Dilute NaOH or HCl was used to adjust the stock solution pH. The kaolin clay minerals were weighed precisely 1.00.001 gm. Twenty ml of each stock solution were added into a 50 ml polycarbonate centrifuge tube and equilibrated for 14 days at room temperature. The tubes were then centrifuged at 4000 R.P.M. for one hour and the final pH values of the solutions were measured. Two ml from each solution were collected with a 0.2 mm disposable filter, acidified by adding 10 ml of concentrated hydrochloric acid or nitric acid (6M or 12M) and then analyzed for 125I gamma. The solid materials were removed from each tube by filtering through a 0.2 mm filter, then allowed to dry. The solid materials were transferred to counting tubes for gamma analysis.
In the column experiment, laterite soils with different composition contents were used which were collected from 4 locations in Malaysia. Each soil type was send for XRD and XRF analyses to identify the types of clay and mineral present. The soils were air dried, filtered by filters (size 0.250-0.350 mm) and then mixed with sand (0.250 mm) with ratio of 3 to 1, respectively. The sand was used to enable the column to be porous in order to allow the aqueous effluent to pass through the soil column. Each mixture soil was packed every 2 cm thick into a polyethylene tube (5.0 cm diameter, 12.0 cm height) and then compacted uniformly. The tube was filled up to 90 % full. The four laterite soil columns were prepared with the same method and dimensions. Prior to the actual experiment, preequilibrated distilled water of pH 4.5 was run into the carefully saturated packed columns for 3 days. The conditioning of the soil column will enable the soil at the required pH 4.5 and maximum contact between solution and the soil. The experimental schematic diagram is shown in Fig. 1.

Fig. 1. Experimental schematic of diagram of soil column.

Twenty-five liters of untreated aqueous effluent wastes was stirred for 24 hours after being conditioned to pH 4.5. For each column experiment, three liters of the

wastes were used and poured into a holding container and allowed to pass through the saturated soil column at a constant rate for 4 days. The four experiments were conducted simultaneously but using different apparatus. The treated effluents were collected at few hours intervals by using a fractional collector at the bottom of the column into ten ml test tubes. The 10 ml treated samples were transferred to 20 ml counting vials, acidified by adding 10 ml of concentrated hydrochloric acid and then analyzed for 125I gamma. Handling of treated samples were carefully performed to prevent contamination in order to minimize error.

Counting Procedures and Data Processing

Prior to counting of samples, aliquots of a standard 125I solutions were counted. Samples' activities (corrected for background) were calculated from the 125I standard count rate and activity. In batch experiments, solid sample's activities were calculated from count rate and activity of standard solution mixed uniformly with the same solid material. All the standard 125I aliquots, solution and solid samples were collected at the bottom of the counting vials of the same geometry. In the column experiments, solution samples were only analyzed for 125I gamma before and after treatment in 20 ml counting vials. Counting was conducted the same day to minimize error due to decay of the 125I. Therefore, no correction for radioactive decay of the 125I radionuclide was included.

Decontamination factor value was calculated by:

$$\text{Decontamination Factor (DF)} = \frac{\text{Counts in Effluent before Treatment (cps)}}{\text{Counts in Effluent after Treatment (cps)}} \quad (1)$$

where, cps = counts per second

Decontamination factor was determined at each time collection of the treated samples. The average DF value for each column experiment was used as index comparison of the efficiency of treatment of each laterite soil used in the treatment.

The distribution coefficient Kd was calculated by :

$$Kd = \frac{\text{Activity of solid samples (KBq/gm)}}{\text{Activity of effluent samples (KBq/ml)}} \quad (2)$$

RESULTS AND DISCUSSION

Batch experiment

Table II summarizes the results of the batch experiments. Figure 2 shows a plotted comparison graphs effluent activity versus initial and final effluent pH before and after equilibrium, respectively. The initial effluent pH at low pH and high pH, increased and decreased, respectively, after the equilibration period. At low effluent pH values between 3.37 to 5.60, the effluent activity values were smaller than effluent activity values at high pH between 5.77 to 6.45. Figure 3 shows a plotted graph of distribution coefficient Kd versus final effluent pH. Between pH 4.5 to 5, the distribution coefficient Kd values are larger than at pH between 6.18 to 6.45. When the soils become more acidic, protonation-the attachment of H+ ions to the surface OH groups take's place. The reaction for kaolin clay mineral can be shown simply as:

Eq. (3)

At moderate acid soil conditions pH 4.5 to 5.0, the kaolin clay minerals exhibited positive charges. This caused the 125I anions to be attracted to the kaolin clay minerals, causing high Kd values.

Column Experiment

Figure 4 shows a plotted graph comparison effluent counts of 125 after treatment by the four laterite soils. From XRD and XRF analyses' results, the four laterite soils showed presence of kaolinite and illite clay minerals; and high percentage of iron oxide minerals such as geothite. From (5, 7, 11), kaolinite is one of the most widespread clay minerals in soils and is abundant in soils of warm moist climates in the equatorial belts such as conditions in Malaysia.

Table III shows the results obtained from the four column experiments conducted. Initial counts for untreated effluent were between 0.46 cps to 0.64 cps. The pH values of the untreated effluent wastes were 4.3. The average decontamination factor values were 90, 98, 33, 99 for samples A, B, C and D, respectively. Sample A, B and D showed to have the highest average DF values, where the treated effluent average counts have decreased to between 0.07 cps and 0.09 cps after 100 hour's treatment. Data showed that removal of 125I from effluent wastes by samples A, B, D was 88 %,

86 % and 86%, respectively. From Fig. 4, efficiency treatments were very effective before the 10 hour's treatment period, however as the treatment period increases, the counts of the treated samples became constant at about 0.13 cps. There were no large differences in the counts from treated effluent collected between 10.00 hours to 100 hours treatment period. The three liter effluent wastes were completely treated after about 100 hour's period and the treated effluent count were maintained at about 0.13 cps for the four columns. For treated effluent collected, the pH values have varied to 5.5, 6.2, 6.4, 6.3 for samples A, B, C, and D, respectively. Table IV shows the comparison of environmental quality criteria values for domestic effluent and the treated effluent environmental quality parameters after treatment by the four laterite columns. The pH treated effluent 6.2, 6.4, 6.3 by sample B, C and D were according to the requirements of the local environmental regulations. Determined C.O.D. values by column A, B, and D produce 36, 32 and 48, respectively, also attained local environmental regulations. However, the total solid values after treatment by the four columns were slightly above the regulation value of 1500 mg/l. The removal of the 125I was mainly due to the presence of the kaolinite, ferrous oxide and aluminum hydrous oxides that exhibited positive charges at pH 4.5. 125I anions were attracted to the positive particles when the effluent wastes passed through the column. The surface charge of ferrous oxide is created by an adsorption or desorption of H+ or a desorption or adsorption of OH-, respectively, in the potential-determining layer consisting of surface O, OH, and OH2 groups. The results were comparable to the model proposed by Parks and de Bruyn (7) shown as follows:
Eq (4)
The pH-dependent charges ferrous oxide (positive surface charge) was balanced by an equivalent amount of 125I anions that were electrostatically held in the outer diffuse electric double layer. The study showed high efficiency treatment about 87 % removal of 125I from effluent wastes were produced by A, B and D column. Environmental parameters C.O.D and pH values were within the requirements of the local environmental quality criteria. The low counts of 125I indicated that the laterite soils were suitable and able to treat effluent wastes containing 125I. Treated effluents release to the environment will undergo further dilution and decay and will not harm the environment.

CONCLUSION

From the study conducted, several conclusions can be drawn:
In the batch experiments using kaolin clay minerals, it was found that the optimum removal of 125I was at pH 4.5 to 5. In laterite soil, there was presence of high content of ferrous oxide, aluminum oxides and kaolinite clay minerals that can exhibit positive charges, formed at the optimum pH condition. This resulted in the 125I anions to be electrostatically held on the minerals. High average decontamination factor values were obtained using laterite A, B, and D where about 87 % removal of the 125I radioisotopes from the effluent wastes. Environmental parameters C.O.D and pH values were within the requirements of the local environmental quality criteria's values. The treated effluents released to the environment will undergo further dilution and decay and will not harm the environment. Treatment process by the soil column method was suitable, easy to handle and this would enable present and future producers of 125I effluent wastes to treat their wastes at their premises. Study is currently continuing to treat larger volume of effluent wastes consisting of other radioisotopes by using the laterite soils.

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INVESTIGATION OF ISOTOPE MIGRATION RATE IN CONCRETE USED AS BACKFILL MATERIAL IN SHALLOW LAND DISPOSAL FACILITY FOR LOW- AND INTERMEDIATE-LEVEL RADIOACTIVE WASTE

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ABSTRACT

Shallow land burial is being considered in Hungary as an option for the disposal of low- and intermediate-level radioactive waste from NPP. According to the conceptual plan the waste packages containing solid wastes and liquid concentrates immobilized by cementation will be placed in vaults constructed of reinforced concrete. Voids between the waste packages will then be filled with cement solution, thus forming a concrete monolith. This monolith structure improves the mechanical strength of the repository and represents a part of the engineered barrier system delaying the dispersion of radionuclides in case of water penetration.

The retention capability of the backfill concrete has been investigated by determining the migration rate of some radioisotopes which are typical in low- and intermediate-level NPP-wastes. Sorption coefficients (Kd) for 125I, 137Cs, 60Co and 85Sr isotopes (in their ionic form) were determined in static sorption experiments. Mass conductivity coefficients (D*) were determined in laboratory-scale migration experiments using cylindrical concrete probes of various moisture content and eliminating water movement and evaporation.

The results of the static sorption experiments showed that the sorption effect was the weakest for 125I and the strongest for 60Co, in order of 125I < 137Cs < 85Sr < 60Co. This order was also observable in the migration experiments. The mass conductivity coefficients were found in the range of 10-11 - 10-12 m2s-1, being higher at the beginning of the migration process and decreasing about one order of magnitude afterwards. Average linear migration rates were also calculated from the concentration distribution along the axis of the probes.

INTRODUCTION

In the course of selecting proper sites for the disposal of low- and intermediate-level radioactive waste from NPP the shallow land burial option has also been considered. A conceptual plan has been developed for establishing a system of engineered barriers to improve the isolation of the radioactive waste from the environment. This system consists of the waste packages containing solid wastes and liquid concentrates immobilized by cementation, the vaults constructed of reinforced concrete in which the packages are to be placed and the concrete filling in the voids between the packages. When assessing the performance of the system each barrier can be characterized by describing the rate of the process leading to the dispersion of radioisotopes from the waste packages. Such processes are the corrosion of the steel drums, the leaching of the radioisotopes from the solid

matrix and the migration of radioisotopes through the surrounding material, in this case through the backfill concrete.
Since concrete may be described as a porous solid with water-filled pores, a mass transfer by diffusion in the pores can be expected to be the dominating transport mechanism (assuming a very low or zero water flow rate). Diffusion due to a concentration gradient can in general be described by the Fick-law, which, in case of one-dimensional diffusion, is formulated as (1)
Eq. (1)
where
c = concentration in the liquid phase (mol m-3)
t = time (s)
D = diffusion coefficient (m2 s-1)
x = length (m)
In case of sorption on the solid due to reversible reactions (such as adsorption and ion exchange) apparent mass transfer rate will be decreased. Concrete may be expected to act as ion-exchanger and the sorption process can be regarded as linear as long as the concentration of the interacting ion is low (less than 5-10% of the total ion exchange capacity) (2). In this case the sorption process can be described by a linear isotherm equation
Eq. (2)
where
q = concentration in the solid phase (mol kg-1)
Kd = sorption coefficient (m3 kg-1)
Assuming that Kd is independent of concentration the apparent diffusion rate will be described by the following equation:
Eq. (3)
where
r = density of solid (kg m-3)
Inserting (2) into (3) gives
Eq. (4)
Equation 4 describes mass transfer rate when diffusion and sorption are the rate determining processes. The mass conductivity coefficient (D*) is defined as
Eq. (5)
Thus, D* can be calculated using experimental data for D (from diffusion measurements in pure aqueous phase) and Kd (from static sorption experiments), if the structure of the solid phase (porosity, apparent density) is accurately known. It is, however, expedient that mass conductivity coefficient (D*) data be obtained from direct migration experiments using representative samples of the given solid matrix (3). Such experiments have been carried out and reported here.
EXPERIMENTAL
The type of cement selected for preparing the backfill concrete was 350 PPC-10, the water/cement ratio was 0.49, the gravel/cement ratio was 4.76, gravel fractions of particle sizes less than 2 mm were used only. The concrete after one week was crushed and a size fraction of 0.045 - 0.063 mm was collected by sieving for the static sorption measurements. The mass ratio of the concrete sample and the solution in these experiments was 1:10. The solution of 125I, 137Cs, 60Co and 85Sr radioisotopes (each of 5 MBq cm-3) was prepared by dissolving the salts of the radioisotopes in natural ground water sample. The time period of 48 h needed for reaching the equilibrium distribution of the isotopes between the solid and liquid phase was determined in preliminary experiments.
The migration experiments were carried out by using cylindrical concrete probes of a diameter of 46 mm and length of 200 mm. The cement mixture for a 10 mm thick layer at half length of the cylinder was prepared with the solution containing the radioisotopes while for the rest of the probe pure water was used. Thus, a one dimensional isotope distribution along the longitudinal axis of the cylinder could be expected, which was symmetrical to the half length. The probes were sliced up after different time periods and the activity of the radioisotopes in each slice was measured as a function of distance from the half length of the probe.
RESULTS AND DISCUSSION
The results of the static sorption experiments were calculated by the following equation:
Eq. (6)
where

Ao = initial radioisotope concentration in the liquid phase
Ae = equilibrium radioisotope concentration in the liquid phase
V = volume of the liquid
m = mass of the solid
Kd = sorption coefficient

The sorption coefficients obtained from static sorption experiments are shown in TABLE I.

The Kd obtained for 125I was just comparable with the experimental error, the sorption of iodide was negligible compared to that of the other three isotopes.

When evaluating the results of the migration experiments the concentrations of the radioisotopes in each slice as the function of distance from the central layer were plotted first. As an example, the distribution of 60Co isotopes (in percentage of the maximum concentration) after different migration times is shown in Fig. 1.

Mass transfer coefficients (D*) were obtained by fitting the appropriate solutions of Eq. 4 to the experimental distribution curves4. When choosing the appropriate solution care was taken that in case of the curves measured after 112, 238, 378 and 643 days the migration was not limited by the length of the probe while in case of longer migration times increasing isotope concentration was observed in both outermost slices (see Fig. 1). Mass transfer coefficients obtained from experimental results are summarized in TABLE II.

It is seen from TABLE II. that the mass transfer coefficients in most cases decrease about one order of magnitude with increasing migration time. This effect can be caused by some moisture transport during the preparation of the probes and solidification of the concrete, or by diffusion on the surfaces.

The average migration rate of the radioisotopes could also be calculated from the shifting of the concentration profiles. These migration rates relating to 1% of the maximum concentration (calculated from the distance between the 1 % points of two concentration profiles and the respective time period between the two measurements) are shown in TABLE III.

Similarly to mass transfer coefficients, the average migration rates relating to shorter migration periods are also higher, decrease after longer migration and tend stabilizing around 10-3 cm day-1. In this case, beyond the above possible reasons, the decreasing concentration gradient is an additional explanation.

The influence of moisture content of the probes up to 30 % (by weight) was also investigated. Increasing moisture content resulted in increasing migration rates but this increase was less than one order of magnitude and could be observed for the short migration periods only. However, numerical evaluation of these results was not possible due to the relatively high experimental errors caused by deficiencies of the long term sealing of the probes.

CONCLUSION

The retention capability of the backfill concrete has been investigated by determining the migration rate of some radioisotopes which are typical in low- and intermediate-level NPP-wastes. Sorption coefficients (Kd) for 125I, 137Cs, 60Co and 85Sr isotopes (in their ionic form) were determined in static sorption experiments. Mass conductivity coefficients (D*) and average migration rates were determined in long-term laboratory-scale migration experiments. The obtained values can be used as input data when modelling the performance of this part of the engineered barrier system.

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FACILITY ACCIDENT ANALYSIS FOR LOW-LEVEL WASTE MANAGEMENT ALTERNATIVES IN THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAM*

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ABSTRACT
The Office of Environmental Management of the U.S. Department of Energy is currently preparing a Programmatic Environmental Impact Statement that evaluates the risks associated with managing five types of radiological and chemical wastes. The release source terms for potential facility accidents are evaluated for storage operations (current and projected waste storage and post-treatment storage) and for waste treatment facilities. This paper summarizes the facility accident analysis that was performed for low-level (radioactive) waste. A set of dominant accident scenarios and a set of dominant facilities and waste treatment processes were selected for analysis by means of a screening process. A subset of results (release source terms) from this analysis for storage and treatment facilities is presented, including storage of waste management and environmental restoration wastes and treatment of alpha and non-alpha contaminated low-level wastes.
INTRODUCTION
The risk to human health of potential radiological releases resulting from facility accidents constitutes an important consideration in the U.S. Department of Energy (DOE) waste management program. The DOE Office of Environmental Management (EM) is currently preparing a Programmatic Environmental Impact Statement (PEIS) that evaluates the risks associated with managing five types of radiological and chemical wastes in the DOE complex. Several alternatives for managing each of the five waste types are defined and compared in the EM PEIS. The alternatives cover a variety of options for storing, treating, and disposing of the wastes. Several treatment methods and operation locations are evaluated as part of the alternatives. The risk induced by potential facility accidents is evaluated for storage operations (current and projected waste storage and post-treatment storage) and for waste treatment facilities. For some of the five waste types considered, facility accidents cover both radiological and chemical releases. This paper summarizes the facility accident analysis that was performed for low-level (radioactive) waste (LLW). As defined in the EM PEIS, LLW includes all radioactive waste not classified as high-level, transuranic, or spent nuclear fuel. LLW that is also contaminated with chemically hazardous components is treated separately as low-level mixed waste (LLMW). LLW with radionuclide concentrations exceeding thresholds specified in Title 10, Part 61.55 of the Code of Federal Regulations is also treated separately as greater-than-class C waste.
LLW results from a variety of DOE operations at more than 30 different sites and ranges from low-activity waste that can be disposed of without further treatment to higher activity waste that requires treatment and disposal techniques involving greater confinement. LLW is also generated by environmental restoration (ER) activities of DOE sites. Operations LLW can include a wide variety of items such as contaminated equipment, dry solids, clothing, solidified sludge, spent ion exchange resins, and contaminated water and oils. ER waste can also contain contaminated metal and equipment and possibly soils.
Treatment technologies considered in the EM PEIS alternatives are based on the physical characteristics of the waste and the final form acceptable for disposal. Because of the nature of LLW, low radioactivity levels are frequently contained in large volumes. To minimize land disposal requirements, all DOE sites must minimize the quantities of LLW generated and emphasize volume reduction. Thus, in the EM PEIS treatment stage for LLW, processes are sometimes limited to volume reduction methods; these methods can include incineration of combustible waste. Other treatment processes considered are solidification and stabilization, which are intended for liquid wastes, incineration by-products, and higher activity waste.
The LLW accident analysis covers storage and treatment facilities for the various EM PEIS alternatives defined for LLW. The facility accident analysis, in accordance with DOE guidelines, addresses a spectrum of accidents ranging from internal events (initiated within operations carried out at a facility) to external events either man-made or initiated by natural phenomena. The following section briefly describes the accident analysis approach and is followed by the results for storage and treatment facilities. The facility accident analysis was also conducted for one EM PEIS alternative that includes management of ER waste. The results of including ER waste in the analysis are discussed.
OVERVIEW OF FACILITY ACCIDENT ANALYSIS

The purpose of the DOE EM PEIS is to evaluate a number of alternatives for dealing with the different types of waste in the DOE complex. The four major groups of alternatives can be summarized as 1) no action, in which existing sites will store and treat their own waste streams, 2) decentralization, 3) regionalization, and 4) centralization, depending on the degree of consolidation and the number of sites involved in waste treatment and disposal. The comparison of alternatives involves a variety of criteria, including economic aspects and the estimated environmental and health impact risks. To assess the environmental and health risks, routine and accidental release source terms from waste transportation and waste management facilities were developed at Argonne National Laboratory (ANL); health effects calculations were performed by Oak Ridge National Laboratory (ORNL).

Each EM PEIS alternative siting option involves existing, planned, and new facilities; the alternatives also evaluate a number of treatment technologies currently in different stages of development. Given the number of alternatives, sites, and treatment options, the variety of facilities existing and nonexisting involved in the EM PEIS, and the limited information about waste characterization and specific facility designs, a generic framework for generating facility accident source term estimates was needed. A generic accident analysis methodology and a uniform accident source term computational framework were developed for the facility accident analysis for all alternatives and waste types. The method consists of selecting a set of initiating accidents and developing a set of accident progression and radioactive release models sufficiently generic to permit adaptation to the peculiarities of the different waste and facility characteristics. A more extensive description of the generic methodology has been published previously (1).

LLW FACILITY ACCIDENT ANALYSIS

To facilitate analyzing and comparing numerous facilities and alternatives, risk-dominant scenarios were selected. These were selected on the basis of (1) risk-significant initiating events and accident progression scenarios and (2) risk-significant sites and facilities.

The facility accidents selected for analysis must provide an adequate basis for comparing alternatives and must be representative of the significant sources of risk expected at the types of facilities and operations involved. The initiating events selected range from high-frequency operational events to low-frequency, high-consequence events induced by natural phenomena. The representative accident progression scenarios expected to dominate the risk are those involving potential dispersion sources, such as fires or explosions, particularly when the material affected has a high specific activity, for example, after volume reduction processes.

Representative accidents were selected in accordance with the type of facility in use and the waste management operation being performed. This approach required a second screening that resulted in the selection of facilities and processes expected to be risk-dominant among those involved in the different LLW EM PEIS alternatives. The facility and treatment technology screening process was based on consideration of waste volumes, waste characterization, and waste physical and chemical form. Liquid and combustible wastes are more susceptible to accidents that would result in material dispersion. Similarly, dispersion of waste with higher specific activities is expected to dominate the consequences. On the other hand, processes limited to packaging, stabilizing, or compacting non-combustible solid LLW are not expected to be involved in accidents among the risk-dominant scenarios.

After determining the representative accidents and the risk-dominant facilities and processes, the LLW alternatives were more closely examined to reduce the number of cases to a manageable size. If the difference between two alternatives was limited to operations that were screened out as non-risk-dominant, only the risk-dominant alternative was explicitly included in the analysis. For each of the facilities and accidents selected, an accident sequence scenario was constructed that includes the conditional probabilities of different effects that could influence the material dispersion (fires, explosions), damage ratios (the fraction of the total amount of waste in the facility that is affected by the accident), and release fractions (the airborne fraction of the material involved in the accident). Because the storage facilities and the treatment facilities are significantly different, a slightly different analytical approach and set of assumptions were required for the accident sequences. The following sections describe the accident analysis methodology for the two types of facilities.

LLW Storage Facility Accident Analysis

In the EM PEIS, two types of storage are involved in LLW management: pre-treatment (often referred to as current) and post-treatment. The purposes of treatment are volume reduction and waste stabilization so that the waste can be converted into a form suitable for disposal. Because of the additional stability, treated LLW is in a physical form that is not readily dispersable; consequently, post-treatment storage is considered a much smaller risk than pre-treatment storage. For these reasons, pre-treatment (current and projected) storage was the focus of the accident analysis. The waste characterization, including waste form and radionuclide profile, for the different alternatives and sites, was obtained from the waste characterization database developed at ANL for the EM PEIS (2). The database indicates that, with very few exceptions, the LLW inventories in storage are mostly in solid form. Pre-treatment storage of solid LLW usually takes place in drums or containers located in storage pads or buildings with a low structural design. Uncertainties in the initial and maximum storage volume and uncertainties in the storage capacity fluctuations over the period covered by the EM PEIS led to the selection of a single facility capacity for all storage facilities analyzed, that is, those at the major LLW-generating sites. A generic facility design was selected in which storage takes place in containers that are unprotected from outside forces and are located on concrete pads. Ten major storage sites, as provided by the waste characterization database, were selected for accident analysis. The accident analysis for these 10 storage sites applies to all EM PEIS alternatives since the EM PEIS assumes that pre-treatment storage occurs at the generation site as opposed to the treatment site.

The representative accidents analyzed include operational accidents (representative of container drops, punctures, and overpressurization), facility fires, and external events, including seismic events to represent natural phenomena and aircraft impacts to represent man-made external events. The operational or handling events were assumed to involve a single container of waste, and the initiating event frequency was assumed to be dominated by the estimated frequency of incidents in container handling (container loading, stacking, etc). Thus, the operational event frequency is proportional to the number of handling operations to be performed at the facilities. For the facility fire and external events, a fraction of the entire contents of the facility, depending on the accident severity, was assumed to be involved in the event. The frequency of fire initiators was assumed to be site-independent, but the fraction of material involved in the accident progression is a function of the facility contents, in particular, of the fraction of combustibles in storage. Aircraft accident frequencies are site-dependent and were obtained from aviation statistics and the locations of DOE sites with respect to major airports and aviation routes. The frequency of damaging natural phenomena events was evaluated from DOE site natural hazard data (3,4) by selecting the expected annual frequency of the hazard whose magnitude exceeds a threshold estimated to cause damage to the stored containers (i.e., hazard magnitude at which containers can be thrown from their stacking position). The source terms for the radiological releases were estimated as functions of the waste amount and characterization involved in the accident.

Treatment Facility Accident Analysis

The waste volumes to be treated, the waste stream characterization, the characteristics of the treatment methodologies, and the physical form of the waste resulted in selecting incineration as the technology treatment with the potential for dominating the risk. Incineration facilities present opportunities for fires and pressurized releases that can heighten the dispersion of radioactive material. Furthermore, if the material affected consists of incineration by-products, there is a potential for more significant amounts of radioactive material being dispersed. Therefore, incineration was selected as the risk-dominant treatment for several alternatives. Because of the potential difference in the source term composition, a distinction was made between incineration of alpha-bearing LLW and incineration of LLW without alpha-activity.

The range of accidents analyzed was the same as that for storage facilities from operational events (i.e., an overpressurization or explosion in the incinerator chamber) to facility fires to external events (i.e., seismic events and airplane crashes). In all these accidents, the material at risk would consist of the ashes resulting from the incineration process since they contain the radioactivity

concentrated in a reduced volume and are readily dispersable. Following this consideration, the facility fire selected was a fire in the baghouse that affects accumulated ashes. The representative natural phenomenon analyzed was a seismic event because of its potential to affect the entire facility; aircraft impacts were also analyzed as potential man-made external events. The accident frequency for seismic events was estimated on the basis of the performance goal for an incineration facility, that is, a moderate hazard facility, as defined in DOE guidelines, and the aircraft impact event frequency was estimated as for storage facilities. The event frequency is site-dependent for aircraft accidents but not for seismic events. The accident sequences that were dominant in terms of releases were those involving large fires. The source terms are a function of the amount of waste at the facility (proportional to the annual throughput) and the waste characterization (radiological composition) (2).

RESULTS AND DISCUSSION

The results of the accident analysis were obtained in the form of a detailed source term and an associated estimated annual frequency. The source term includes the amount of each radionuclide released to the atmosphere during the accident and corrected for the respirable fraction (1). The accidents have been grouped into four categories on the basis of their estimated frequency: the categories range from anticipated (frequency higher than 10-2 per year) to extremely unlikely (frequency less than 10-6 per year) events.

Sample results are provided in Tables I and II for storage and incineration, respectively. The tables provide some important parameters, such as the waste volume in the facility, the damage fraction, and the total amount of activity released (without the breakdown by nuclide). Because of space restrictions, only the storage analysis results for four of the largest LLW-generating sites (Hanford, Idaho National Engineering Laboratory [INEL], ORNL, and Savannah River Site [SRS]) are included in Table I. Similarly, only one EM PEIS alternative, number 18, which involves treatment at the same 4 storage sites displayed in Table I, is shown for incineration in Table II.

The possibility of including Environmental Restoration (ER) waste in the LLW accident analysis was also addressed. Because it is anticipated that the bulk of ER waste will only undergo benign treatment, probably limited to size reduction and packaging, the radiological risk impact of ER waste in LLW treatment is expected to be low. Therefore, only ER waste storage was addressed. The analysis was performed under the same assumptions that were applicable to operations LLW, that is, a site-independent facility size (storage pad) and a site-dependent waste composition. The results of the ER waste storage accident analysis are also shown in Table I for the same four sites displayed for operations waste.

The results in the tables suggest that, in general, the risk of releases from LLW waste management facilities due to accidental causes would be low. Preliminary screening estimates for the maximally exposed individual confirmed that the risks to human health involved in LLW management would be relatively low. Generally, releases of large activities are associated with a very low estimated frequency, while frequent events potentially result in small released activities. The releases associated with LLW incineration are, in general, very small, in part a result of the small LLW throughput at the facility.

The accident source terms in Table I also indicate that the risk associated with generic storage of ER LLW in similar amounts and facilities as operations LLW Waste Management (WM) LLW can be slightly higher or lower than the risk of the same accidents for WM LLW. The increase or decrease in the estimated amount of radioactivity released depends on the physical characteristics of the ER waste and its radiological profile, both of which are site-dependent. Because of the physical form of ER waste (contaminated metal, debris, and soils with a lower fraction of combustible material), the dominant accident sequences are slightly changed; an earthquake followed by a spill is more risk-dominant (in terms of radioactivity released) than a seismic event followed by a fire. For the same reasons, the relative importance of a facility fire (normalized to the total activity in storage) is lower for ER waste. In general, the release estimates are the same order of magnitude as those for WM LLW, and, therefore, suggest a low risk impact.

The results of alpha-incineration for Alternative 18 (4-site treatment) are also shown in Table II for comparison of the relative releases. The amounts released in the scenarios analyzed are considerably lower than those of non-alpha LLW, a

function of the waste throughput and, more significantly, of its radiological profile. Although the dose conversion factors for alpha-emitters are considerably higher than for other nuclides, the doses are not expected to exceed those of the non-alpha LLW. These results are not unexpected since only waste with very low levels of alpha-activity can qualify as LLW. With the projected inventories and waste characteristics available, including alpha-LLW in LLW incineration operations does not appear to alter the releases due to facility accidents.

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EXPERIMENTAL CHEMICAL MILLING PROCESS TO DECONTAMINATE AND DISPOSITION REACTOR FUEL AT THE SAVANNAH RIVER SITE

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ABSTRACT

This paper presents the results of a scoping test to determine if a large scale chemical etching process would be effective at extraction of radionuclide and other hazardous contaminants from manufactured fuel tubes and assemblies. A review of the effectiveness of this process against other similar solid materials is considered. Because contaminants in contact with these types of materials tend to become electrostatically bonded to the substrate material, successful removal requires a chemical reaction to break these bonds. The process proposed is effective at bond deactivation and maintains an advantage over other decontamination alternatives due to its highly effective extraction of subsurface as well as surface contaminants. The large scale repeatability of the process makes it applicable to many areas of DOE decontamination and dismantlement (D&D) activities complex wide.

INTRODUCTION

The Savannah River Site (SRS) has operated since the early 1950s producing nuclear materials for the U.S. Government. The reactor materials production mission of the site has declined in recent years, and the facility is currently in a DOE-mandated decontamination and de-inventory mode (1). All building inventory of HEU is being melted and recast into solid cylindrical U-Al alloy ingots for shipment and storage at the Oak Ridge Y-12 Plant.

It has recently been proposed to return fuel assemblies from the SRS reactors to the Fuel Fabrication Facility to decontaminate, disassemble, melt, and recast this material as well. Because the fuel assemblies were contaminated by contact with reactor coolant, the Fuel Fabrication Facility's Tube Cleaning Process was proposed to decontaminate the assemblies sufficiently such that the disassembly, tube handling, and casting operations can be safely performed using the existing facilities. Since it is not certain that this process can effectively decontaminate the assemblies and because the process will introduce new, solid and liquid wastes containing radioisotopes which are not present in the facilities existing waste streams, a scoping test to demonstrate the effectiveness of the proposed activity was initiated.

BACKGROUND

Over time, contaminants have migrated into the substrate of the assemblies through the pores in the material. This migration has occurred naturally with the assemblies in contact with the contaminated moderator. The depth to which this migration has occurred is dependent upon many factors, including the porosity of the cladding, the mobility and solubility of the contaminants, and the existence of other drivers. To preclude personnel contamination during the casting operations, this surface and subsurface contamination must be eliminated.

The proposed tube cleaning technology is a sequential process, which prepares and then removes the outer layers of the selected material. The process was preferred over other decontamination alternatives because of its high likelihood of effective extraction of the subsurface as well as surface contaminants. This is important since total destruction and recasting of the material in a virgin metal casting environment would vaporize and release most of the radioactive materials present. Because the chemical treatment facility remains operational in the Fuel Fabrication Facility area, it would be more economical than other decontamination methods. The chemical formulations of this process are nonflammable and nonexplosive, and the process does not pose additional health risks for facility operators.

Since the assemblies would be shipped in aluminum cans, a special receiving area would not be required. Minor modifications to the Tube Cleaning Area would be needed to control contamination as the assemblies are removed from the transport cans and cleaned. A breathing air system and exhaust gas tritium monitors would likely be required during the initial handling steps.

EXPERIMENTAL PROCEDURE

To demonstrate a process for decontaminating the fuel assemblies, one contaminated inner target tube was returned from the SRS Reactors to the Metallurgical Laboratory. Prior to the transfer, surface contamination levels had been reduced by flushing the target tube. This was necessary since the Metallurgical Laboratory was not designed to process high contamination products, especially tritium. Post-rinse analysis indicated acceptable transferable contamination up to ~400 dpm g, ~200 dpm a; and ~90,000 dpm tritium. Although the surface radioactive contaminants were present in low quantities, the detection system was precise enough to measure the alpha emissions and allowed a decontamination factor (DF) to be calculated for the beta emitting tritium.

The tube was cut into 12" long sections and processed in ~8 liter solutions of caustic (NaOH), nitric acid (HNO3), and hot and cold water rinses to simulate the full-scale cleaning process. The samples were first submersed in a caustic solution to remove oxidation and etch the aluminum. The chemical representation of this action is given below.

$$Al2O3\ (s) + 6NaOH\ (aq) \rightarrow 2Na3AlO3\ (aq) + 3H3O\ (1) \qquad (1)$$

The action causes a sticky sludge to develop on the sample surface which is removed by submersing the specimen in a nitric acid rinse followed by hot and cold water rinses:

$$Contaminant\ (C) + HNO3\ (aq) \rightarrow Cx(NO3)y\ (aq) + NOz(aq) + H2O\ (1) \qquad (2)$$

The specimens were then allowed to air dry before making post-cleaning measurements. The complete laboratory experiment was carried out in a ventilated enclosure and monitors were in place to evaluate the air emissions.

The effectiveness of the cleaning process was determined by standard radiological smear tests before and after the chemical etching process. Each of the smears were evaluated by liquid scintillation spectroscopy. The processing time per cycle was maintained at 75 seconds, but multiple cycles were evaluated to determine the depth to which the contamination had migrated. The chemical bath temperature (150F), the caustic molarity (1.45M), and nitric normality (~5N) were all set to normal operating values expected in the full-scale operation, and were not varied. The hot water rinse was maintained at ~100F.

RESULTS

Test results show that the process proposed is very effective at removing radioactive surface contamination bound in the oxide layers of the sample material. Results after the first cycle showed an average increase in tritium levels of ~3 over original activity levels in 9 out of 15 samples analyzed as the subsurface was exposed. Individual samples ranged from 1.4 to 5.2 times higher. Six of the samples analyzed actually decreased in activity levels after the first cycle. Reduction in activity levels ranged from 1.1 to ~5 times lower. Three samples were run through

multiple cycles for a total process time of ~5 minutes. The retesting showed only a slight amount of residual beta activity from tritium contamination with no other activity present. Calculated decontamination factors of greater than 102 were demonstrated following repeat cycles.
A review of Fig. 1 demonstrates that tritium had permeated into the cladding to a maximum depth of ~3.5 mils. Unaltered surface contamination indicated a much lower value until the subsurface was exposed. The average depth of maximum activity within the substrate was around 1.4 mils. The tritium activity levels per cycle as measured are given in Table I. Based on the limited experience with this sectioned target tube, no airborne tritium contamination was detected external to the ventilated enclosure.

CONCLUSIONS

The test results show that the process proposed is very effective at removing radioactive surface and subsurface contamination from fuel tubes. The entire process can be repeated several times to achieve the desired levels of decontamination. Sampling is easily performed at the end of each cycle to determine remaining contaminant levels. Since chemical etching to a depth of 3.5 mils appears to completely decontaminate the surface of the tube which maintains a 30 mil thick cladding, no exposure of the fissile core material is expected during the full-scale operation.
Since the scoping test utilized a sectioned, flushed, target tube, there is uncertainty that the surfaces of concentrically nested tubes making up an assembly can be satisfactorily decontaminated using this process, especially at the contact locations between tubes. However, air sparging which is present in the full-scale operation may be able to effectively assist with the decontamination process in these reduced flow areas. The calculated decontamination factor was limited by the low contamination present on and within the tube sections evaluated. A much greater decontamination factor would be expected with non-flushed tubes.
Based on these preliminary results, it is planned to obtain additional contaminated subassemblies to further test and evaluate this process. A mechanical agitation apparatus will be used to simulate operational air sparging. Further research and tests are being performed to improve the process and broaden the applications into other areas of decontamination and dismantlement (D&D) activities complex wide.

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WASTE MANAGEMENT BUSINESS MODELING

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ABSTRACT

The U. S. Department of Energy, Nevada Operations Office, has been analyzing its hazardous, low-level, and mixed-waste management operations with an eye toward improving productivity and efficiency using business modeling, a graphical systems engineering analytical technique. The low-level waste management operations on the Nevada Test Site were analyzed, along with mixed-waste operations as presently configured. Results provided a basis for making process improvements and minimization of noncompliance with federal and state regulations.
Professional Analysis, Inc., is using Computer-Assisted Software Engineering (CASE) software to develop the business models. The software has the capability of extracting and exporting information from the model, which can then be imported into a variety of databases. Business models already completed (Hazardous Waste Management, LLW/MW Management) are being used to reengineer operations in order to minimize noncompliance and increase productivity and efficiency.

INTRODUCTION

The U. S. Department of Energy, Nevada Operations Office, (DOE/NV) has taken the first step in the application of systems engineering to improve its routine waste manage-ment activities. Managers have developed baseline graphical representations

of their waste management activities and have begun to engineer new and improved methods. Their goal is to ensure a world-class operation into the future -- a cost-effective, safe operation that is continually improving. The DOE/NV Waste Management Division has been analyzing its hazardous, low-level, and mixed-waste management operations with an eye toward improving productivity and efficiency. Business modeling was proposed for operations at the Nevada Test Site (NTS) where noncompliance with federal and state regulations could result in action by a regulatory agency. The business modeling used is an application of systems engineering methods for analyzing and graphically displaying activities performed to accomplish a business function or operation. The graphical output from the business modeling effort provides a tool that can be used as a basis of communication between interested parties. It serves as a means of documenting activities necessary for carrying out a business function.

BUSINESS MODELING DESCRIPTION

A business model is extremely useful in developing a view of how activities are related to one another across organizational and functional boundaries. It is not uncommon for managers responsible for a facility or single operation to attempt to optimize that function without clear understanding of the role their function has in the larger system. Focus on only one part of the system typically produces improvements only in that function that may be detrimental to the overall system operation. By using a business model focused on the total system, each operation and activity can be viewed as a contributing part, regardless of organization or functional boundary. Note that this type of model is intended to represent system structure only, not behavioral structure.

A business model is also means of identifying the actual activities performed, i.e., defining what steps the specialists are really doing, not just what the procedures say they should be doing. Unlike process flow models, along with determining actual activities, these models are used to identify the actual information and documentation used between activities. The models are particularly useful for processes that have been evolving over many years. They are equally useful in engineering new functions that have not yet been performed. An example is future activities associated with disposal of mixed waste under the to-be-approved RCRA Part B permit at the NTS.

Because it documents the way things are really done, a business model is a powerful tool for managers to develop better ways of performing and organizing tasks. Reviewing current practices allows a manager the opportunity to identify redundancies, question why certain activities are being performed, question the validity of the sequence of activities, evaluate whether activities may be outdated, and to prepare a new model of proposed improvements. Since the modeling uses graphical computer tools, it permits instant changes and many iterations to obtain the best mix of activities for achieving the desired end product. It is an effective management tool that can be applied to re-engineer an operation.

Business models can also serve as a means of verifying procedures by validating actual practice to the procedural requirements. This permits managers the opportunity to modify procedures to reflect actual practice or to modify actual practice to match procedural requirements, or to modify both to improve productivity.

Models can be designed to identify all known standards and requirements imposed on individual activities within a process. This can be very useful in identifying specific regulatory standards or requirements to individual work activities, rather than arbitrarily imposing these controls throughout an operation on all activities. Additionally, because standards and requirements can be identified to individual activities, the communication of applied standards to external regulatory reviewers is facilitated.

The diagram in Fig. 1 provides information on the activity (in the box), inputs and outputs to and from each activity, and controls and mechanisms needed to accomplish the task(s). Activities are arranged in the order they are performed. All activities are evaluated to determine whether they should be "decomposed" into more detailed steps in order to study the process more thoroughly. The major summary of activities is the "A0 (A Zero)" level. The number of activities displayed on an 8-1/2 x 11" page is generally restricted to six and no less than three. For every activity, inputs and outputs are defined, and procedural, regulatory, standard, and code controls are identified. Mechanisms can include the type of specialized personnel,

equipment, or databases needed to perform the activity.
Figure 2 shows how the decomposition works. Each decomposition is on its own page. The Tree Structure of the model provides an outline and guide to the flow of decompositions. An example of a portion of the Tree Structure for Low-Level and Mixed Waste (LLW/MW) is shown in Fig. 3.
Fig. 2 - Sample of How Decomposition Works
Fig. 3 - Sample Tree Structure
MODELING METHODOLOGY
Professional Analysis, Inc. is using Computer-Assisted Software Engineering (CASE) software to develop the business models. The software has the capability of extracting and exporting information from the model, which can then be imported into a variety of data bases. Business models already completed (Hazardous Waste Management, LLW/MW Management) are being used to reengineer operations in order to minimize noncompliance and increase productivity and efficiency.
To initiate the process, modelers interview personnel involved in the day-to-day operations and prepare a graphical representation of how all elements of the work process are actually being performed, ensuring that the operations personnel are involved in the iterative steps needed to understand the process.
During preparation of the baseline model (i.e., representation of how things are really done), the many suggestions for improvements from knowledgeable managers and staff are brought together with observations from the modelers and are combined into an initial proposed model. The output is a combination work process model and a graphical baseline record of how work is really done. Actual activities are compared with how the work is described in procedures as part of the proposed model; any differences between actual and procedural are defined and presented to the contractor for resolution.
A proposed model is then developed as a first step in re-engineering the process for improvement. It is essentially a place to start for managers and staff to consider how the overall system in the model might improve the process. For example, the proposed model may represent an "ideal" method of executing a process such as implementation of new or improved automated tools.
Both the baseline and proposed models can be used as a basis for understanding the total, obtainable direct cost savings by comparing the actual cost for each activity in the baseline model with the anticipated costs from the proposed model. The proposed model can thus be used for preparing process-change cost justifications.
Since all users of the model may not be familiar with reading and interpreting the graphical output of the model, a report describing both the baseline and proposed models is prepared. As part of this improvement analysis, the modelers observations and recommendations are recorded for possible action by senior management.
The model can represent the process from the perspective of DOE management or from the contractor management perspective. The LLW/MW and hazardous waste models were done from the DOE/NV management perspective.
A review of the standards and requirements controlling the activities associated with the LLW/MW model was also performed. A separate listing was developed supplementing the original model. This listing, recorded in simple, straightforward relational database tables, provides the details associated with each identified control. It also permits easy identification of activities affected by changes to regulations.
A static modeling paradigm was applied to represent the LLW/MW system structure, as opposed to using a dynamic modeling paradigm that represents behavior over time. The initial intent was to use the modeling to develop procedural and physical-barrier mechanisms to minimize noncompliance. However, not only was the business model valuable for modifying the operational processes to minimize noncompliance, it was also found to be valuable for improving the process overall. Examples are updating or correcting procedures, improving database capabilities, eliminating redundant steps, consolidating activities, clarifying responsibilities that cross organizational lines, reducing the amount of paperwork currently used, and providing a cross-procedural view as a communication tool.
APPLICATIONS
A hazardous waste management system model from the DOE/NV perspective was the first prepared. Immediate productivity increases were realized. Short- and long-term savings may more than offset the investment in the business model. While preparing the hazardous waste management model, the value of business modeling became apparent

as a tool for implementation of the Defense Nuclear Facility Safety Board Recommendation 90-2 (that DOE redefine all operating standards and requirements) because controls are identified on the model for each discrete activity. The DOE/NV 90-2 Implementation Plan requires that baseline process models be developed for each facility subject to 90-2 Recommendation requirements. In order to maximize benefit from the process models, they must be developed and analyzed from both the DOE management perspective and from facility operator viewpoints. The facility-specific standards and requirements identified during both processes can then be compared and evaluated to ensure that applicable requirements have been identified, are adequate for the work being performed, and are implemented by the contractor.
Because the regulations cited in facility operating procedures are applied to each specific activity in the business model, 90-2 Implementation functional area experts, peer reviewers, or facility experts who review the required regulations will use the business model as a "map" of functions related to specific regulations and standards for each discrete activity, as well as using models as a valuable communications tool.
Business modeling can also be used as a tool to engineer a business process prior to inception, resulting in a baseline process model. The process identifies resources necessary to perform each activity (mechanisms), so initial resource loading can be determined by skill or specialty required to perform the tasks. In addition, a work breakdown structure can be prepared based on the model because the process and products are already mapped out. A baseline model can be used to monitor and record a management system's evolution. For example, to facilitate development of a management system for implementing 90-2 Recommendation at DOE/NV, the process was modeled. This model facilitated understanding of the complex process and provided the mechanisms and sequence of steps needed to successfully complete the Generic and Facility-specific Standards/Requirements Identification Documents required by the 90-2 Recommendation.

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REACTOR FACILITIES ALTERNATIVE USE SCOPING STUDIES
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ABSTRACT
The purpose of the Reactor Facilities Alternative Use Scoping Studies (1) was to determine if excess Savannah River Site (SRS) reactor facilities were suitable for interim alternative uses and to determine if such uses were preferable to immediate Decontamination and Decommissioning (D&D).
Westinghouse Savannah River Company (WSRC) and Raytheon Engineers & Constructors considered over 50 potential alternative uses for the excess reactor facilities. Study priorities were determined with the help of a matrix of potential alternative uses that evaluated each alternative against 14 pre-selected criteria including immediate need, cost to implement, involvement of private industry, benefit to the Department of Energy (DOE) complex, etc.
Scoping studies for 20 of these alternative uses have been completed to date. Each study evaluated the viability, suitability, and economic value of the alternative, listed its advantages and disadvantages, and made recommendations based on its conclusions. Due to the nuclear nature of the facilities, most alternative uses involved radioactive material and, therefore, were generally under the cognizance of the DOE Office of Environmental Restoration and Waste Management. Some of the proposed uses (e.g., transuranic waste treatment and storage, and lead decontamination) can take advantage of the unique design, radiation control, and security features available at the reactor areas.
The results of these alternative use studies have led to the conclusion that DOE, as well as commercial and private operators, could benefit by using the excess SRS facilities. Outside industry may also benefit by employing their own surplus facilities for alternative uses. Utilizing an excess facility for other than its original function could eliminate the need to build costly new structures, and could be a valuable means for forming profitable business relationships in the local communities.
A separate comprehensive document entitled Alternative Uses for Reactor Facilities

(2), often referred to as the main report, incorporated the completed alternative uses studies as well as other site and facility information into a single-source document that could be used by other departments, DOE sites, or private industries. The main report describes the physical layout, support services, and status of the excess facilities, and includes a database of facility configuration information useful for determining the ability and practicality of using the facilities for other purposes. The main report was written to be understood by those unfamiliar with SRS facilities.

The completed main report is being reviewed by WSRC and the Department of Energy-Savannah River (DOE-SR). Direction to implement some of the more promising alternative uses is expected to be forthcoming.

WSRC has many assets available to other DOE sites and to private industry. These assets include:

Substantial experience in the beneficial reuse of excess facilities and resources.

Numerous excess facilities that could be made available for alternative government and private sector projects or other business endeavors. Available services and utilities are typically more than adequate to allow a wide range of operations to be performed.

Technical support provided by the Savannah River Technology Center (SRTC). SRTC can provide research, testing, development, and analytical services in the areas of robotics, metallurgy, thermodynamics, isotope development, radiology, and computer science.

WSRC is available to perform alternative use studies for facilities at other SRS locations, at other DOE sites, or at commercial nuclear operations.

BACKGROUND

In the 1950s, five nuclear reactor facilities were built on the SRS in South Carolina to produce Special Nuclear Materials for the nation's defense programs and special isotopes for non-defense applications. Four of the reactors are now considered "excess" by the DOE and are being evaluated as to their disposition. D&D, the ultimate disposition for all of the facilities by entombment or dismantlement, could be initiated immediately or delayed to allow short-lived, high-energy radioisotopes to decay.

If D&D is delayed, the facilities may be used for alternative functions or allowed to lie dormant. Both options require continued financial support for surveillance and maintenance to assure public health and safety. Unless an interim alternative use is identified that would be economically and/or environmentally preferable, these facilities will either be entombed (buried in place) or completely dismantled and removed. Using these excess facilities for other than their original purpose is potentially the most cost effective alternative to storage or immediate D&D.

ALTERNATIVE USE EVALUATION

The initial potential alternative uses for the reactor facilities were identified by senior reactor division personnel participating in brainstorming sessions. These personnel were familiar with the reactor facilities and the unique nuclear-related challenges and concerns of the DOE defense organization. A weighted value was assigned to each of 14 evaluation criteria:

1. Impact on worker safety
2. Impact on environmental safety
3. Impact on public safety
4. Benefit to waste reduction
5. SRS short-term need
6. Impact on public perception
7. SRS employment opportunities
8. Estimated capital cost to implement
9. SRS long-term need
10. Political impact
11. Benefit to other DOE sites
12. Net Operating & Management costs
13. Value as a national asset
14. Benefit to outside industry

These criteria were arranged from the most significant (impact on worker safety) to the least significant (benefit to outside industry). Each potential alternative use was then evaluated against each criterion and given a rating from +5 to -5 based on the relative benefit or liability, respectively, incurred by implementing the

alternative. Some alternative uses were determined to be impractical due to radiological, environmental, or public perception considerations. Other alternative use studies were deferred due to budget constraints or higher priorities. The remaining alternative uses were researched, analyzed, and documented for presentation in stand-alone reports. To date, the project team has evaluated the following alternative uses:

- Lead Decontamination
- Decontamination Technology Center
- Plutonium Vitrification and Storage
- Transuranic Waste Storage
- Transuranic Waste Treatment
- R-Reactor Transuranic Waste Storage
- Basin Aquaculture
- Engineered Trenches for Low-Level Radioactive Waste Disposal
- Engineered Trenches for Monitored Retrievable Storage
- Suspect Soil Treatment/Disposal
- Waste Volume Reduction
- UO3 Ultimate Disposition
- Stainless Steel Recycling
- Hydraulic Test Facility
- Museum/Visitor Information Center

Additional studies were requested and funded by departments other than WSRC Reactor Engineering Division based on their specific priorities. These studies were not necessarily subjected to this selection process.

Where applicable, environmental permitting considerations were discussed in the individual scoping studies. Most studies concluded that alternative use of SRS reactor facilities would not only be cost beneficial, but would also provide solutions to several SRS and DOE complex concerns.

FACILITIES DATABASE

As the first alternative use studies were being developed, it became apparent that current and accurate data on certain plant attributes were universally required to properly evaluate the facilities for the proposed alternatives. Among other things, this information included:

- Physical building attributes (floor loading, door sizes, room locations and volumes, etc.),
- Support equipment (location and condition of fans, cranes, pumps, etc.), and
- Utilities and services (location and capacity of electric, telephone, steam, sewage, and water systems).

A dedicated research and documentation effort was performed by a team of engineers whose purpose was to gather data for the alternative use teams. Numerous documents were examined including Reactor Safety Analysis Reports (SARs); site historical documents; and Cold Shutdown, Lay-up, and Transfer Packages.

Walkdowns were performed to determine the existing condition of each facility and to confirm and supplement information obtained from research documentation. Valuable information from the walkdowns and research was compiled and sorted in a computerized database. The database consisted of four forms (Area, Building, Elevation, and Room) which were provided as an appendix to the main report. A database search of needed facility requirements can be performed to determine what available facilities meet or exceed the specified criteria.

The General Facility Descriptions section of the main report includes a general description of the reactor areas, a tabular record of differences between the areas, and the status of services and structures in each. Specific details of the reactor areas are contained in the Specific Reactor Facility Information section, also in tabular form. These tables provide for a quick comparison of areas and are good references, but they were not Intended to present the level of detail available in the appendices. The reactor facility information sections discuss the following services and features:

- Communications
- Radiation Monitoring Equipment
- Contamination Control
- Security
- Water Systems
- Steam Supply

Electrical Service
Ventilation
Fire Detection/Protection
Water
Gas Systems
General Conditions
Licenses

SCOPING STUDIES

To date, there have been 20 scoping studies performed to evaluate the potential use of excess SRS reactor facilities for alternative uses. A brief description of the major Alternative Use Scoping Studies are given below.

Lead Decontamination

This study evaluated various lead decontamination methods, the feasibility of using excess reactor buildings for the interim storage of lead, and the feasibility of using these same facilities for housing a lead decontamination operation and its secondary waste treatment operations.

The best decontamination method for lead was found to be a two-step process consisting of abrasive or CO2 blasting to remove coatings and paint from the lead, followed by chemical immersion to remove remaining contaminants. It was concluded that using SRS reactor buildings for the interim storage of lead, and the further use of one of these buildings for a Lead Decontamination Facility (LDF) would be both feasible and desirable.

It was recommended that more analyses be performed concerning specific modifications and costs required for using reactor facilities for the interim storage of lead, performing abrasive blasting and chemical immersion decontamination operations, and performing appropriate secondary waste treatment for these methods. The additional analyses should also consider the possibility and feasibility of decontaminating lead from other SRS facilities as well as other DOE sites and private industry.

Decontamination Technology Center

A major objective of the DOE's decontamination plans is to reuse and recycle as much material from the D&D process as is practical, both to conserve valuable raw materials and to minimize the amount of waste requiring disposal. Existing technologies are inadequate to solve many problems, including how to cost-effectively decontaminate structures and equipment, what to do with materials and wastes that are generated, and how to adequately protect workers and the environment. New and improved technologies resulting from research and development at a Decontamination Technology Center could increase decontamination efficiency and cost-effectiveness.

The SRS must manage radioactively contaminated materials that are generated as a result of past operations and as an unavoidable consequence of current cleanup efforts. This study evaluated the possibility of using excess reactor buildings to house a decontamination research facility and its associated laboratories. The Decontamination Technology Center would also include operating decontamination processes for demonstrating decontamination technologies.

Based on this initial assessment, the establishment of a Decontamination Technology Center would be a feasible and desirable alternative use for the SRS reactor buildings. It was recommended that a detailed analysis be performed concerning the specific modifications and costs required for using the reactor facilities for decontamination research and development, and for demonstrating decontamination processes.

Plutonium Vitrification and Storage

This study examined the feasibility of converting an existing SRS facility for the vitrification and storage of plutonium. The Defense Waste Processing Facility, F-Area facilities, and reactor facilities were evaluated. This initial assessment led to the conclusion that plutoninm would best be vitrified at an F-Area facility and stored at a reactor facility. The F-Area facilities were designed to handle plutonium and have much of the required process equipment already in place. The reactor facilities have large amounts of storage space and would need only minor modifications to upgrade necessary equipment.

Consideration should be given to the possible public perception and political impact regarding the storage of vitrified plutonium from commercial facilities and/or other DOE facilities. If plutonium is to be shipped from other facilities, then consideration should also be given to the public concern and the political impact

related to transportation on public roads.

Transuranic Waste Storage

This study examined the possible use of one or more of the four excess reactor facilities for storage of transuranic (TRU) waste. The findings of this study support the use of R Reactor for non-mixed TRU waste storage and P Reactor for mixed TRU waste storage. Mixed and non-mixed TRU waste should be handled separately for optimum management of resources.

Several factors contribute to the need for additional storage space for TRU and other solid waste. Waste containers that have been stored for twenty years may be compromised or may already be leaking into the environment. Treatment requirements, the condition of storage containers, more stringent environmental protection requirements, and fund limitations are important concerns to examine when addressing optimum use of resources available for TRU waste storage.

Storage of non-mixed TRU waste in the R-Reactor facilities is of immediate interest since the present storage accommodations may be saturated before the end of fiscal year 1995. The use of the R-Reactor facility to store the present inventory of TRU waste would alleviate the shortage of space in less time than may be required to build new facilities.

Transuranic Waste Treatment

A large majority of the TRU waste generated through the 1980s contains hazardous materials and is therefore categorized as mixed-TRU waste. The current plan is to ship the TRU waste to Carlsbad, New Mexico for final disposal at the Waste Isolation Pilot Plant (WIPP), when it becomes operational. However, the facility will reject waste that has not been certified to be in compliance with the WIPP Waste Acceptance Criteria (WIPP/WAC).

Prior to shipment, the SRS-generated TRU waste will need to be assayed to verify that the waste does not contain:

- Hazardous materials that are not co-contaminants
- Excessive free liquids
- Explosives, pyrophorics, PCBs, or asbestos
- Unidentified waste materials

Uncertifiable TRU waste must be processed to remove, stabilize, and dispose of the unwanted hazardous constituents.

With modifications and upgrades, C-, L-, or P-Reactor facilities contain the necessary support systems and floor space to be used as a TRU waste treatment facility. When compared to the estimated $200 million dollar design and construction cost of a new treatment facility, the reactor reuse option may be economically preferable.

R-Reactor Transuranic Waste Storage Evaluation

This report evaluated the feasibility of converting vacant space in the R-Reactor building, the largest of the four excess SRS reactor buildings, for TRU waste storage.

There is an immediate need for additional interim storage space for TRU waste at the SRS. The waste is presently stored on outdoor concrete pads in E Area. The majority of these pads are permitted for storing "mixed" waste, which contains both hazardous and radioactive constituents. Some pads contain non-mixed TRU waste, consuming valuable space that could be used for storing more problematic mixed waste. Since a storage permit is not required for non-mixed waste, this waste can be relocated to other areas to make permitted space available for mixed waste.

R Reactor has been out-of-service since 1964 and has been extensively cannibalized, with equipment and components salvaged for use at other reactors. Structural modifications would be needed to allow storage of the large TRU waste containers that will not fit through existing entrances. The estimated cost for this work totaled $750,000 and included activities such as equipment removal, wall demolition, and floor decontamination.

Federal and State environmental permitting for interim storage of mixed-TRU waste in R-Reactor building would be difficult and time consuming due to the large number of radioactively contaminated areas and the potential for water intrusion into the lower elevations of the main building. However, the use of 0' elevation floor space for the storage of non-mixed TRU is achievable. It is estimated that the use of R Reactor for non-mixed TRU waste interim storage would clear two existing, permitted E-Area pads for continued mixed-TRU waste interim storage. This scenario, when compared to the cost of $1.0 to $1.5 million to design, construct, and permit

additional E-Area storage pads or building a new storage facility, is feasible and cost beneficial.

Basin Aquaculture

This study evaluated the feasibility of using the SRS C-, L-, P-, and R-Reactor 186 cooling water basins and 904 retention basins for aquacultural purposes. Proposed options included growing plant life, aquacultural research, and raising various aquatic life. The earthen 904 Basins were found unsuitable for fish production, and several of the proposed options were excluded due to time restrictions and/or a low probability of success. The study ultimately focused on using the 186 Basins for raising fish, crayfish, shrimp, or prawns for food.

Of the species that could grow in the basins, the hybrid striped bass was found to be superior in marketability, growth rate, disease resistance, temperature tolerance, and general compatibility with existing facility design. A hybrid striped bass operation could also have a large profit potential. The basins would not, however, support shrimp or prawn production, but could be used to profitably raise Australian crayfish.

The initial assessment shows that raising hybrid striped bass and Australian crayfish would be a feasible and potentially profitable alternative use for the SRS 186 Basins. It was recommended that a detailed analysis be performed and a pilot program run to test the feasibility of using the basins for crayfish or food-fish production.

Engineered Trenches for Low-Level Radioactive Waste Disposal

This alternative use study evaluated the economic feasibility of using spaces within the C-, L-, P-, and R-Reactor Areas for the permanent disposal of Low-Level Radioactive Waste (LLRW). Results show that the reactor cooling water basins (186 Basins), the disassembly basins, and the P- and R-Reactor pipe pit areas offer impressive disposal capacity which could be converted into LLRW disposal sites. Structural modifications would be required to ensure the 100+ year service life requirement for a LLRW disposal facility.

Calculations show that up to 300,000 cubic meters of LLRW could be disposed in the 186 Basins. Compared to the planned construction of new disposal vaults, the converted basins could replace up to six vaults. Assuming a construction cost of $23 million per vault, minus the estimated $40.8 million for facility improvements, the total SRS disposal cost savings could approach $97 million.

Engineered Trenches for Monitored Retrievable Storage

SRS solid wastes that require treatment are being stored onsite until the DOE and SRS long-term waste treatment and disposal plans are finalized. Once approved, these plans could take several years to implement. In the interim, one or more Monitored Retrievable Storage (MRS) facilities will be needed to store the future inventory of solid waste generated from environmental restoration and D&D projects.

This study examined the technical feasibility and cost associated with converting the C-, L-, P-, and R-Reactor 186 cooling water basins and the P- and R-Reactor pipe pit areas into usable MRS facilities.

Results of this study revealed that adequate storage capacity already exists through fiscal year 1998. After this date, existing TRU waste storage capacity will be exhausted. Similar indicators exist for mixed and hazardous waste in fiscal year 2000 and fiscal year 2001, respectively. Unless additional mixed waste storage space is permitted by the South Carolina Department of Health and Environmental Control (SCDHEC), or unless effective site waste reduction methods are implemented over the next one to two years, the site will exhaust its permitted storage capacity.

The conversion of the 186 Basins or pipe pit areas into suitable MRS facilities is feasible. Based on current SRS solid waste generation rates, the storage capacity gained from the MRS conversion of one 186 Basin would add approximately 25 years of non-LLRW storage capacity. The estimated MRS conversion cost is between $3.28 and $8.4 million per basin, which includes the cost of facility enhancements and the installation of a roof (choice of two types) for protection from the weather. The P- and R-Reactor pipe pit areas also offer considerable storage capacity but will require removal of the heat exchanger inlet and discharge piping.

Suspect Soil

This study investigated past and present methods for disposing of contaminated and potentially contaminated SRS soil, and evaluated soil treatment and disposal plans that could utilize reactor facilities. The reactor areas were considered for contaminated soil treatment applications. The 904 and 186 Basins, in particular,

were considered for interim soil storage and remediation operations.
A Soil Treatment Facility in a reactor area would provide many benefits. The volume of soil disposed of as contaminated waste could be reduced to as little as one percent of the current volume using remediation techniques that have been proven or investigated by the Savannah River Technology Center (SRTC). More actual waste could be disposed of in existing burial space, and new or expanded soil burial facilities may not be needed. Finally, hazardous and radioactively contaminated soil from most onsite locations would not have to be moved across public roads, thereby reducing regulatory requirements.
It was recommended that a Soil Treatment Facility be planned for one or more of the reactor areas. The new Soil Treatment Facility would process hazardous and radioactively contaminated soil. The existing petroleum-contaminated soil remediation facility in D Area and new bioremediation facilities to be built at T Area would process soil contaminated with non-hazardous and non-radioactive materials (e.g., kerosene and diesel fuel). These facilities would initially be used to meet site needs but could be expanded to process offsite soil contaminated with hazardous materials.
Waste Volume Reduction
This study examined the need for, and feasibility of, establishing a Waste Volume Reduction Facility at the SRS. Wastes to be handled by the facility would include transuranic waste, mixed waste, low-level radioactive waste, and hazardous waste, but would exclude high-level waste. The function of the Waste Volume Reduction Facility would be to assay, characterize, and physically reduce the size of accepted wastes. Emphasis was placed on using existing equipment and excess reactor facilities for this purpose.
It was concluded that although a Waste Volume Reduction Facility is not immediately needed, the expected increase in waste from planned D&D operations will likely exceed existing volume reduction capabilities. Since plans have been made for much of the site to undergo D&D at some point, it is recommended that a centralized Waste Volume Reduction Facility be established.
The existing reactor facilities appear suitable for a Waste Volume Reduction Facility. While some of the needed equipment can be obtained onsite, a fair amount must be purchased. The Waste Volume Reduction Facility would initially accommodate site needs but could be expanded to serve other DOE sites and commercial generators. Additional studies need to be performed to address the specific waste types and generation rates, desired level of volume reduction, specific equipment to be used, and modification requirements for site facilities that are proposed for volume reduction operations.
UO3 Ultimate Disposition
Uranium trioxide (UO3) is one of the by-products of past DOE nuclear production operations. Thousands of drums of UO3 are now stored in the reactor facilities and other areas at SRS. The purpose of this study was to identify beneficial uses for the UO3 that may be employed as an ultimate disposition and alternative to storage. This study investigated the possibility of mixing UO3 with cement to act as a shield in concrete used in waste disposal vaults or engineered trenches. Disposal of the UO3 as a LLRW may also be possible; however, final disposal would likely require a special license due to the high uranium content of UO3 (toxicity concerns). This option is considered undesirable due to its complexity and cost.
R was concluded that no obvious beneficial uses currently exist for UO3. If neither of the above options are exercised, continued storage may be the most desirable option until a viable alternative use is found.
It was recommended that the regulatory limits associated with mixing UO3 in concrete be further investigated and that tests be performed on concrete-UO3 mixes to determine structural adequacy and leachability characteristics. Proposals have recently been made to perform a concrete-UO3 demonstration project.
Stainless Steel Recycling: Final Report
This study evaluated the potential alternative use of excess SRS reactor facilities for recycling slightly contaminated stainless steel (SS). Using one of the reactor facilities for recycling contaminated SS would have two benefits: one, it would prevent burial of the SS and two, it would delay D&D of the facilities selected for the recycling tasks.
The proposed SS recycling program would convert contaminated SS scrap into two types of LLRW containers: 2.8-cubic-meter boxes and 208-liter drums. The cost benefit of

the SS recycling program was studied in four phases: decontamination, sizing, melting, and fabrication.
A Final Report integrated the cost results of the four phases and recommended locations, onsite or offsite, for performing the phase activities. lf onsite performance was recommended, a specific reactor facility was suggested. The lowest cost combination of phase activities and locations was estimated to result in costs of $202 per SS drum and $1192 per SS box compared to new drum and box prices of $370 and $1240, respectively. For the recycled containers, the least expensive container capacity is provided by boxes at $0.47 per liter compared to drums at $0.97 per liter.
A $51.3 million dollar savings could be realized by using recycled SS containers for waste disposal instead of purchasing new containers. This savings includes both the difference in container costs (cost to manufacture waste containers from recycled SS venus cost to purchase new waste containers--$37.1 million) and the savings from not having to dispose of the contaminated SS as a separate waste ($14.2 million).
Locations within C-, L-, and P-Reactor buildings appear feasible for all SS recycling activities except rolling sheet metal from cast slabs; the rolling mill requirements for space and floor loading cannot be accommodated.
Hydraulic Test Facility
This study evaluated the potential for using an excess SRS reactor facility as a Hydraulic Test Facility. Emphasis was placed on using the cooling water system pumps and/or the K-Reactor cooling tower recirculation pumps to provide the motive force for the hydraulic testing. Four hydraulic testing alternatives were investigated:
- Flow-testing large mechanical equipment
- Testing civil-hydraulic structures
- Dynamically testing boat hull designs or other water-borne devices
- Cooling tower research and testing using the K-Reactor cooling tower

Alternative uses found to have promise and warrant additional consideration are a permanent testing station for medium to large pumps and use of the cooling water basin(s) for short-term research projects.
Museum/Visitor Information Center
This alternative use scoping study evaluated the possibility of using reactor facilities for a museum and visitor information center. Part of the Museum/Visitor Information Center would include exhibits and displays that describe the history of the SRS, its mission in the DOE complex, and its many operations. Actual reactor system equipment could be viewed as individual pieces in a museum-like display and/or in a reactor building in their original setting.
The Museum/Visitor Information Center could also include exhibits, displays, and demonstrations associated with environmental restoration and management, energy production and use, nuclear weapons, nuclear medicine, and other related subjects. The capabilities and expertise available at the SRTC could also be highlighted. SRTC has produced many patents and is involved in high technology projects such as the NASA space program. The center could provide research, testing, development, and analytical services in the areas of robotics, metallurgy, thermodynamics, isotope development, radiology, and computer science. An extensive technical library at SRTC has recently opened to the public.
A basic Museum/Visitor Information Center that uses existing reactor facilities is expected to cost approximately $1.9 million. Costs would increase proportionately with the number and complexity of exhibits.
Based on the results of this initial evaluation, it was recommended that a Museum/Visitor Information Center be planned for C-, L-, or P-Reactor Area. A site Museum/Visitor Information Center could be an educational and public relations asset for the DOE, SRS, and surrounding communities.
CONCLUSION
The results of the Reactor Facilities Alternative Use Scoping Studies generally reinforced the idea that the excess SRS facilities are valuable assets and should be used for alternative purposes versus immediate D&D. Implementing recommended alternative uses generally provides the following benefits:
- Economically beneficial reuse of existing structures
- Cooperative and mutually advantageous relationships with private and commercial industries Preclusion or delay of D&D of valuable assets
- Stimulation of local economy
- Solution to several DOE and site facility needs

Reconciliation of many environmental concerns

Other alternative use studies have shown comparable advantages. It is likely, therefore, that further studies of other excess facilities (DOE and commercial) would see similar results. The completed Reactor Facilities Alternative Uses Study comprehensive report is now being reviewed by WSRC and DOE, and direction to implement promising alternative uses is expected to be forthcoming.

It is apparent that full advantage should be taken of surplus facilities whenever possible. Many facility operators (including DOE, commercial, and private owners) could very likely benefit from using the excess facilities available at SRS, or from using their own surplus facilities for alternative purposes. Reusing excess facilities for other than their original functions could eliminate the need for building costly new structures, and could be a valuable means for forming profitable business relationships in local communities--possibly through the use of Cooperative Research and Development Agreements (CRADAs).

WSRC has many assets available to other DOE sites and to private industry. These assets include:

Substantial experience in the investigation of beneficial reuse of excess facilities and resources.

Numerous excess facilities that could be made available for alternative government and private sector projects or other business endeavors. Available services and utilities are typically more than adequate to allow a wide range of operations to be performed.

Technical support provided by the Savannah River Technology Center (SRTC). SRTC has an impressive amount of expertise in the areas of robotics, metallurgy, thermodynamics, isotope development, radiology, and computer science. Research, testing, development, and analytical services are now being offered to other DOE sites and to outside industries. Work can be performed in-house or at the client's facilities.

The facilities now available for alternative uses are four excess reactor buildings and their support buildings, each consisting of several office-type structures, two 115kV substations, and a 31,000 cubic meter water basin supplied from the Savannah River. Each 291,000- to 303,000-square-meter reactor area is cleared and level, has road and railroad access, and is surrounded by a security fence.

Electric service is supplied by a 13.8kV electrical distribution system, with 240 and 480VAC power available throughout the facility. Low-pressure steam, potable water, telephone, sewage, and fire protection systems are also available in each area. The reactor buildings are constructed of reinforced concrete with total volumes of approximately 255,000 to 340,000 cubic meters and between 37,200 and 46,500 square meters of floor space. These structures are ideal for applications ranging from office space to heavy industrial use.

WSRC is available to perform alternative use studies for excess facilities at other SRS locations, other DOE sites, or commercial nuclear operations.

For more information concerning the resources, capabilities, and availability of SRS facilities, services, and technical support contact Mr. Brian Getson, Westinghouse Savannah River Company, Building 707-C, Rm. 349, Aiken, SC 29802, telephone number (803) 557-9765.

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APPROACH FOR UPGRADING PWR WASTE PROCESSING SYSTEMS IN KOREAN NUCLEAR POWER PLANTS

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ABSTRACT

Currently in Korea, there are 12 Pressurized Water Reactors (PWR) either operating or under construction. These units encompass several different designs for

radioactive waste processing systems. However, the different designs may be separated into three(3) basic groups based upon waste processing technology. This paper will describe the design concepts and operating experiences for the each waste processing group.
Based upon design and operating experience from these 12 units, KOPEC implemented to improve waste processing systems by simplification (i.e. elimination of unnecessary equipment) and employing current technology. This paper will also describe some of the features and benefits of upgrading the waste processing systems.
Table I
INTRODUCTION
Since the first nuclear power plant went commercial in 1978, Korea has fully developed 3 sites for Pressurized Water Reactors (PWR), with eight (8) operating PWRs. There are an additional four (4) more PWRs under construction at these sites, scheduled to begin operation through out 1995-1998. Thus there are 12 PWRs either operating or under construction. These units encompass several different designs for waste processing systems. However, the different designs may be separated into three (3) basic groups of waste processing technologies. This paper will describe the design concepts and operating experiences for each waste processing group. It will also describe some of the features and benefits of upgrading the waste processing systems based upon design and operating experience from these 12 units.
DESIGN CONCEPTS AND OPERATING EXPERIENCE OF THE EXISTING WASTE PROCESSING SYSTEMS
Group 1 Waste Processing System
Group 1, which is an older design, has no waste recycle capability. However, the concept of Unit Operations were considered in the system design for liquid, solid and gaseous wastes. The two (2) nuclear plants, Kori-1 and Kori-2 constitute Group 1.
Liquid Radwaste System
Liquid wastes generated in the plant are collected in three (3) tanks depending upon the processing required: the laundry & hot shower tank, the floor drain tank and the waste holdup tank. The laundry & hot shower tank wastes are filtered only and the floor drain tank wastes are filtered and demineralized. But, the waste holdup tank wastes are processed in an evaporator, and the evaporator distillate is normally demineralized. All processed liquid are collected in the monitor tanks for holdup prior to discharge.
The system capacity may be less than input flow during the abnormal operation since the capacity is based upon normal operation inputs. Since an evaporator, which has no redundancy, is incorporated into the system, an out-of-service evaporator may also cause an increase of radioactive material release to the environment. As a result of operation, however, the actual releases of radionuclides in the liquid effluent has been about 0.11 Ci/yr/unit.
Gaseous Radwaste System
The Gaseous Radwaste System (GRS) is a closed loop comprised of two waste gas compressors and several gas decay tanks to collect fission product gases. Gas decay tank capacity permits at least 45 days for decay before discharge. Prior to releasing waste gas to the environment, it is sampled and analyzed to determine the activity.
The pressurized-tank gaseous radwaste system has several limitations of operation, for example waste gas leaks due to high operating pressure and pressure transients in the waste gas input sources. Another unfavorable design of this system is that the required tank volumes are large and the system operation is more complicate than other operating plants.
The actual release of radionuclides in the gaseous effluent has been about 183 Ci/yr/unit.
Solid Radwaste System
Spent resin is collected from the plant demineralizers, stored in the spent resin storage tank to permit decay of short lived fission products, and then transferred to the drumming room for dewatering and packaging. Concentrates from the waste evaporator are also transferred to the drumming room. These concentrates are blended with cement and chemicals in a mixer, and injected into a drum. Spent filter cartridges are transferred to filter cartridge drums and stored for later disposition. Miscellaneous solid wastes are compressed into drums and also stored for later disposition.
Since system operation is performed manually, occupational radiation exposure(ORE)

to workers is relatively high in comparison with other operating plants.
Solid waste volumes shipped to offsite is about 600 drum/yr/unit, with 55 gallon drum.

Group 2 Waste Processing System

Group 2 has a similar configuration as Group 1, but Group 2 has a larger waste processing capability and increased redundancy to improve the waste processing system availability. Six (6) nuclear plants, Kori-3&4, Yonggwang-1&2 and Ulchin-1&2, constitute Group 2.

Liquid Radwaste System

Input into the Liquid Radwaste System(LRS) is divided into three (3) process trains based on the processing required : high total dissolves solids (TDS) waste, low TDS wastes, and chemical wastes. Decontamination wastes are directed to the chemical waste train which normally filters wastes prior to discharge. Low TDS wastes are routed to the low TDS train which usually filters and demineralizes the waste. High TDS wastes are collected and then processed in the evaporator, which has a capacity of 30 gpm, the evaporator distillate is normally demineralized. All processed liquid is collected in the LRS monitor tanks for holdup prior to discharge or recycling. Discharge from the monitor tanks is usually directed to the condensate storage tanks for reuse, but may be discharged offsite.
To improve system operability, the storage capacity has been increased based on anticipated operational occurrences, and system reliability has increased by providing two(2) evaporators. However some troubles have been at the group 2, such as frequent replacement of filters, degradation of ion exchangers and evaporator loss due to the waste oil and the suspended solid in the liquid wastes.
The actual release of radionuclides in the liquid effluent has been about 0.02 Ci/yr/unit.

Gaseous Radwaste System

The GRS has been designed to utilize the technology of ambient temperature (104 F) charcoal adsorption to reduce the radioactivity in a gas stream by delay and decay of the radionuclides. The system consists of a surge tank, a header drain tank, two (2) parallel 100% dehumidification trains, four (4) charcoal delay beds and a HEPA filter to process gaseous wastes generated in the plant during normal operation. The system provides at least 45 days holdup for Xenon gases during normal operation. Design flowrate of the system is 2 scfm.
System operation is relatively simple and the area for the equipment is small. In addition, off-site dose has been reduced and the actual releases of radionuclides in the gaseous effluent has been about 150 Ci/yr/unit.

Solid Radwaste System

The SRS consists of the Radwaste Solidification Subsystem(RSS), the Resin Transfer Subsystem(RTS), the Filter Handling Subsystem and the Dry Active Waste Subsystem. The RSS which is In-Line Mixing Process operates remotely on a batch process basis to solidify radioactive spent resins, liquid, evaporator bottoms, and to immobilize filters and other miscellaneous contaminated objects in the 55-gallon drums for shipment offsite. The system uses portland cement mixed with lime as the solidification agent. The RTS provides for removing and storing radioactive spent resins from the plant, and consists of a spent resin tank, for storage of resins to permit decay, a resin sluice pump, valves, and piping. Following decay, resin is removed from the spent resin tank and sent to the RSS for solidification. The Filter Handling Subsystem provides the capability to replace normally radioactive filters with a minimum of personnel exposure. The Dry Active Waste (DAW) Subsystem is used to reduce the volume of compressible, slightly radioactive dry wastes. Wastes are compacted into 55-gallon drums for eventual shipment offsite.
Since the system is operated remotely, ORE to workers has been reduced, compared to Group 1 systems. In-line mixing RSS has also, however, several troubles, such as cement setup in the components, poor drum capping operation, undesirable drum surface decontamination capability and unoperable level sensors. These are reasons to increase the ORE to worker in the system operation.
Solid waste volumes shipped offsite are about 500 drum/yr/unit.

Group 3 Waste Processing System

Group 3's main design improvements were the addition of waste tank sludge removal and oily waste removal processes for the Liquid Radwaste System. Adding these features improves operability by permitting the processing of additional types of waste without shutting down the system. However, with additional features, system

complexity is also added. The system now requires more work from the operating staff, and the possibility of component failures and maintenance is expected to increase. The four(4) nuclear units Yonggwang-3&4, Ulchin-3&4 constitute Group 3.

Liquid Radwaste System

The LRS has been designed to operate in the maximum reuse mode (Zero Release Concept) with releases as required to maintain the plant water balance and to control tritium concentrations in the primary and secondary systems.

Segregation of wastes are the same as Group 2's system. All liquid wastes are processed by filtration, oil removal, evaporation, and ion exchange methods by processing required for recycling or discharge. Prior to processing, settled sludge and waste oil in the waste holdup tank are separated from liquid wastes to improve processing efficiency. Another features of this Group 3 is that all processing paths consists of batch operation concepts, so that many feed tanks are installed in the system.

Gaseous Radwaste System

The GRS has the same configuration as Group 2, because it is one of the most improved technologies to process the gaseous wastes.

Solid Radwaste System

A subsystem, Sludge Waste Subsystem, is added to the configuration of the Group 2' SRS which are necessary to meet the interface requirements of the LRS, and to satisfy the waste conditioning requirements. The Sludge Waste Subsystem provides the means to receive and store sludge wastes from the LRS holdup tanks for radioactive decay prior to transferring to the RSS.

To eliminate the limitations of Group 2 RSS, In-Drum Mixing Solidification System is used in Group 3.

UPGRADING PLAN OF WASTE PROCESSING SYSTEM

General

Based upon design and operating experience from these 12 units, the following design goals were developed to be incorporated into future nuclear power plants.

a. Reduce the release of radioactive material to the environment
b. Reduce the radwaste volume shipped to disposal sites
c. Reduce the occupational radiation exposure (ORE) to plant personnel
d. Increase the waste processing system reliability using proven technology

KOPEC implemented aggressive feasibility studies for the improvement of waste processing systems at future nuclear power plants and conducted evaluations of available technologies related to waste processing from all over the world (1).

As a result of these studies, the conceptual design for upgrading PWR waste processing systems for the future plants were modified by using the following features to meet the basic design goals;

a. Reduction of Activity Release
- Use an effective liquid-solid separation system, such as a centrifuge system
- Use a selective ion exchange media for liquid waste processing
- Prolong Xe delay time (45 days) for gaseous waste processing

b. Reduction of Waste Volume Shipped to Disposal Site
- Eliminate evaporators to eliminate evaporator bottoms
- Volume reduction of Dry Active Waste
- Volume reduction of spent resin

c. Reduction of Occupational Radiation Exposure
- Eliminate troublesome equipment
- Incorporate ALARA goals into equipment layout
- Establish long term storage of high activity spent resin prior to processing.

d. Improved System Reliability
- Use duplex components and processing trains
- Use mobile processing equipment

Description of the Waste Processing System

Liquid Radwaste System

To remove radioactive materials from the radioactive and potentially radioactive liquid wastes generated in the plant, all liquid wastes are divided into three (3) process trains according to the processing required: high radioactive liquid waste train, low radioactive liquid waste train, and decontamination liquid waste train. High radioactive liquid wastes are processed by centrifuge and then ion exchangers. Low radioactive liquid wastes and decontamination liquid wastes are processed by centrifuge only. After the liquid wastes are processed through each train, it is

transferred to the monitor tank for sampling and analysis to determine if it should be released to the environment or recycled to the feed tanks. The monitor tank is the only tank in the LRS from which processed waste can be released to the environment.

Gaseous Radwaste System

The GRS collects and processes the radioactive waste gas. The gaseous wastes are delayed in a low pressure, ambient temperature, charcoal delay bed. The GRS is sized to provide the capability of delaying Xenon for at least 45 days. The GRS limits the release of gaseous activity so that personnel exposure and activity releases in the environment are meet ALARA goals.

Solid Radwaste System

The SRS is designed to collect, process, and package solid radwaste in a suitable manner for storage until the material is shipped offsite for disposal. The SRS consists of the Resin Handling Subsystem, the Dry Active Waste sorting and Segregation Subsystem, and the Filter Handling Subsystem

a. Resin Handling Subsystem

The spent resin beads used in the plant are transferred to the Spent Resin Dewatering Equipment or to the Spent Resin Long Term Storage Tank, depending upon the resin activity. Low activity resins are transferred to the spent resin dewatering equipment, where they are dewatered and capped in the container in preparation for disposal. High activity resins are stored for approximately 10 years prior to processing to permit radioactive decay in the long term storage tanks. After storage, the high activity resin will be processed in the same manner as the low activity spent resin.

b. Dry Active Waste sorting and Segregation Subsystem

For DAW minimization and volume reduction, the system allows for waste processing to be classified into three (3) separate operations, such as the sorting and shredding operation, the pre-compacting operation, and the supercompacting operation.

To perform above operations the system consists of the various equipment, such as a sorting table, a shredder, a bag monitor, three waste compactors, a screw compactor for polyethylene waste (mobile equipment), and a supercompactor (mobile equipment).

c. Filter Handling Subsystem

This subsystem provides for removal of expended filters from filter housings with a minimum of personnel exposure. Removed filters are transferred to the filter capping area and securely stored in a special drum.

Expected Release Activity and Waste Volume

The estimated release rates for radionuclides using the PWR-GALE computer program is 0.62 Ci/yr/unit for liquid waste and 2,300 Ci/yr/unit for gaseous wastes. These estimates are based upon the improved radwaste system design.

Solid radwaste volumes shipped to disposal site are expected to be approximately 250 drums/year/unit with 55 gallon drum.

SUMMARY

Sufficient operating and design experiences on PWR waste processing systems has enabled us to upgrade waste processing systems. Plant operational experience has proven to be an important source of information allowing KOPEC to achieve an improved waste processing system design.

Using the improved conceptual design, KOPEC is currently performing the design work for the next nuclear power plants, Yonggwang Unit 5&6, which is in the early design stage.

KOPEC will make an effort continuously to improve the waste processing systems incorporate with advanced processing technologies.

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CELEBRATING SUCCESS AT FERNALD

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ABSTRACT
Restoration of the Fernald Environmental Management Project is now moving from the environmental investigation stage to real, tangible remediation progress. Using a variety of programmatic innovations, DOE and FERMCO continue to strengthen an effective partnership that supports a mutually-developed mission of safe, least-cost, earliest final remediation of the Fernald Site while complying with all applicable DOE Orders, regulatory requirements and commitments and addressing the concerns of the many stakeholders who have an interest in how remediation at Fernald progresses. The progress that is occurring at Fernald is testimony to a productive DOE/FERMCO partnership that will continue to be an essential part of the difficult environmental restoration task at this site.
DOE/FERMCO PARTNERSHIP KEY TO SUCCESS
The U. S. Department of Energy (DOE) and the Fernald Environmental Restoration Management Corporation (FERMCO) have developed an effective partnership that has resulted in real, tangible progress in the environmental restoration of the DOE's Fernald Environmental Management Project near Cincinnati, Ohio.
FERMCO, which is managing the remediation under a performance-based contract with DOE, assumed responsibility for the Fernald Site in December 1992. FERMCO was the first DOE Environmental Restoration Management Contractor (ERMC) in the DOE complex. Together, DOE and FERMCO established the Fernald mission as the safe, least-cost, earliest final remediation of the site. The mission also commits Fernald to compliance with all applicable DOE Orders, meeting all regulatory requirements and commitments, and -- most importantly --addressing the concerns of the many stakeholders who have an interest in what happens at Fernald.
In July 1994, DOE and FERMCO successfully negotiated the first performance-based contract in support of DOE's new contract reform initiatives. The new contract includes more specific (and objective) criteria by which FERMCO's performance is assessed and fee is earned. The contract demonstrates the mutual trust and cooperation between DOE and FERMCO in planning and executing the environmental restoration effort at Fernald.
Since 1986, DOE has entered into several agreements with the U. S. Environmental Protection Agency (EPA) and the Ohio Environmental Protection Agency (OEPA) that guide the environmental restoration effort at the Fernald Site. These agreements require DOE to conduct the remediation in compliance with a variety of environmental regulations, including the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) and the Resource Conservation and Recovery Act (RCRA). CERCLA regulations, in particular, require exhaustive environmental studies and development of remediation alternatives before final field remediation can begin. Today, much of that work has been completed. The Fernald skyline is changing as buildings are dismantled and new facilities are constructed in preparation for the final remediation work.
Originally called the Feed Materials Production Center (FMPC), the Fernald Site began producing highly-purified uranium metal products used as feed materials in U. S. defense programs in 1953. All production was suspended in July 1989 to allow concentration of resources on environmental restoration activities.
This paper discusses how DOE and FERMCO have used innovative approaches to make safe, least-cost, earliest remediation at Fernald a reality.
ENVIRONMENTAL RESTORATION REQUIRES CAREFUL PLANNING
Much of the initial remediation effort at Fernald centered on CERCLA requirements for exhaustive environmental sampling and analysis of soil, water, and other media to determine the nature and extent of contamination at the site. Based on results of this Remedial Investigation (RI) work, a Feasibility Study (FS) carefully evaluated various alternatives for removing or containing the contamination. The RI/FS supported generation of Proposed Plans (PPs) for final remediation of the site. The CERCLA agreement established five Operable Units based on their location or the potential for similar response actions. This concept allows final remediation design to begin for each Operable Unit as soon as its individual RI/FS-PP phase is completed and a Record of Decision (ROD) is approved by the EPA.
During the course of the RI/FS process, DOE and FERMCO have built solid

relationships with regulatory agencies overseeing the Fernald remediation. Fernald management recognized the need to partner with regulators during the development of environmental restoration plans to promote early resolution of issues and common understanding of approaches for remediation. Fernald worked with regulators to effectively integrate RCRA and CERCLA requirements, thereby reducing costs and eliminating duplicate requirements. This was done through technical exchanges to update OEPA on the status on the closure of various Hazardous Waste Management Units. Similar technical exchanges with EPA have resulted in steady progress toward reaching final remediation decisions.

Fernald management and the regulators have jointly developed innovative approaches that will speed the remediation effort significantly. For example, EPA has approved an Interim ROD for Operable Unit 3 -- the first of its kind in the DOE complex -- that will allow accelerated decommissioning and dismantling of some 125 production buildings and other structures located in the former uranium processing area. Thus, while the final ROD addressing treatment and disposition of soil and wastes from Operable Unit 3 is not scheduled until early 1997, work has already begun on the dismantling of the former uranium production facilities. In fact, the Interim ROD for Operable Unit 3 is expected to allow dismantling of the former production areas some 3-4 years ahead of schedule and at a savings of about $300 million from the original cost estimates.

Perhaps the most visible evidence that full-scale remediation at Fernald has begun was the dismantling of Plant 7, the tallest building on the site and the first former production building to be removed. That dismantling effort included implosion of the building's steel superstructure in September 1994. The Plant 7 project was completed well ahead of schedule and at a cost of about $11 million, compared with the $33 million estimated in the FY 1993 budget baseline.

As old production structures at Fernald are being dismantled, new facilities are under construction to handle final remediation work. The recent EPA approval of a ROD for Operable Unit 4 calls for radium-bearing waste in two concrete silos and production waste from a third silo to be removed and vitrified (converted to a s glass form) prior to its shipment off site for disposal. A vitrification pilot plant is under construction to house that operation. Fernald also has constructed an advanced waste water treatment facility to help reduce the uranium content of wastewater discharged from the site as the remediation work accelerates. The became operational in late January.

FINAL REMEDIATION DECISIONS NEAR

Fernald has received conditional EPA approval for Proposed Plans for two other Operable Units. DOE is currently reviewing a draft ROD for Operable Unit 1 that calls for the excavation and treatment of material stored in six waste pits at Fernald prior to its shipment off site for final disposition. Nearing completion is a draft ROD for Operable Unit 2 that proposes disposal of fly ash, lime sludge, slightly contaminated soil, and other material in an engineered cell on site. Work also is well under way toward a final Record of Decision for Operable Unit 5, which includes soil, water, and all other environmental media on and around the Fernald Site.

Concurrent with the detailed investigation and planning associated with the RI/FS process, Fernald is making steady progress on several interim risk reduction projects called Removal Actions. Removal Actions are initiated when there is a need to accelerate removal or containment of hazardous substances posing a significant potential threat to the environment or to the human population. Removal Actions are coordinated with both Ohio and U. S. EPA to ensure that they are consistent with the long-term remedial actions expected as a result of Records of Decision.

One of the more significant ongoing Removal Actions is the extraction of uranium-contaminated groundwater from an area south of the Fernald Site known as the South Plume. The years of production operations at Fernald and stormwater runoff from the site have contaminated the aquifer which underlies the area. Fernald also has completed a number of projects aimed at preventing any further introduction of contamination to the aquifer, including installing stormwater runoff controls and upgrading and sealing concrete pads where drummed wastes are stored.

Another continuing Removal Action is the off-site disposition of low-level radioactive and mixed wastes, including both residues from past production and construction rubble from remedial activities. In FY 1994 alone, Fernald shipped more than 577,000 cubic feet of low-level waste and more than 3,500 cubic feet of mixed

waste off site for disposal.
STAKEHOLDERS KEY TO DECISION-MAKING PROCESS
Fernald's success in moving from the environmental restoration planning process to actual field remediation is due in large part to an innovative public involvement program that emphasizes person-to-person communication and early stakeholder involvement in the decision-making process.
A key public involvement activity at Fernald was the formation of the Fernald Citizens Task Force. The Task Force, the first Site Specific Advisory Board established as part of the DOE Environmental Management program, is an independent panel comprised of representatives from virtually all of the major stakeholder interests, including local government and the Fernald Residents for Environmental Safety and Health (FRESH). DOE commissioned the Task Force to develop a public consensus on specific aspects of the Fernald remediation, including potential future uses of the site, appropriate maximum risk levels, final disposition of waste, and development of environmental restoration priorities. The Task Force meetings are open to the public, thereby offering still another avenue for community involvement in the decision-making process at Fernald. The Task Force delivered its interim report on risk levels and general future land use decisions on November 30, 1994. The final report, scheduled for July 1995, will contain recommendations on waste disposition, environmental restoration priorities, and specific future land uses.
CERCLA regulations include requirements for holding meetings and providing information to help the public understand the environmental restoration process and to provide input into the selection of remedial action alternatives. While public comment periods are an integral part of the CERCLA community involvement requirements, complying with the letter of the law limits the public involvement aspect of the remediation planning to essentially one-way communication. Traditionally, public involvement has consisted of publishing newsletters and fact sheets and holding public hearings. These hearings too often focused on providing information and asking for public comment on what has already been done, rather than genuinely soliciting public input on what should be done.
Fernald recognized that a significant stakeholder audience remained unaware of or uninterested in the decision-making process. To reach those groups, Fernald developed an Envoy Program. Some 75 DOE and FERMCO employees have volunteered to serve as envoys to various governmental bodies, civic groups, service clubs, and other organizations. Envoys establish face-to-face contact with these groups by attending their meetings and listening to any ideas and opinions their members may have to offer. This input is relayed directly to Fernald management for consideration in the ongoing decision-making process.
Fernald also recognized that stakeholder involvement will be necessary well beyond the RI/FS stage of the remediation effort. To ensure that established relationships continue, Fernald is revising and updating its Public Involvement Program to maintain a high level of stakeholder participation throughout the Remedial Design/Remedial Action phase of the environmental restoration.
An important element of Fernald's stakeholder involvement is reaching out to stakeholders of those facilities planning to receive remediation wastes from the Fernald clean-upthe Nevada Test Site (NTS) and Envirocare's facility at Clive, Utah. For over two years, Fernald has built and maintained open relationships with DOE's Nevada Operations Office, the NTS Community Advisory Board since its inception, and state regulators in Nevada and Utah. These relationships are based on candor, full disclosure of information and proposed plans, and a sincere desire to understand the needs and concerns of stakeholders at these sites of crucial importance to Fernald. This non-traditional outreach effort has been very successful.
STRENGTHENING THE SAFETY CULTURE
While external stakeholder involvement is an extremely important element of the environmental restoration effort at Fernald, management has placed an equally high priority on the involvement of internal stakeholders -- the Fernald work force -- in strengthening the safety culture at the site and improving productivity. In another first for DOE sites, a broad-based Safety First Team formed in early 1994 has developed several initiatives to encourage Fernald workers to find safer and more effective work practices. A pilot work group concept successfully demonstrated that teams of employees can effectively identify potential safety hazards and develop work methods that both reduce those hazards and increase productivity. The success of this pilot program has led to the establishment of similar work groups throughout

the Fernald organizations, with both DOE and FERMCO management playing an active role in nurturing the program.
Because the environmental restoration effort at Fernald is a complex program requiring careful balancing of regulatory requirements with available funding and other resources, Fernald management has placed strong emphasis on the area of project controls. FERMCO was the first DOE contractor to produce a project baseline containing thousands of schedule activities and hundreds of thousands of cost elements that serves as a yardstick for planning and measuring remediation progress. In March 1994, a DOE-HQ Cost Quality Management Assessment commended Fernald for putting a fully-integrated cost/scheduling system in place.
Fernald has established a formal, structured cost savings/cost avoidance procedure in support of cost savings goals established by DOE Headquarters. The procedure establishes criteria for measuring cost savings that require initiatives to be innovative, go beyond simply good business practices, and reduce the Fernald Performance Measurement Baseline.
During the second half of FY 1994, FERMCO achieved $17,880,154 in cost savings. Major contributing components for those cost savings included: streamlining and accelerating certain Operable Unit 1 studies and expediting the remediation of Waste Pit No. 6 (a savings of $1.5 million); acceleration of Remedial Action decommissioning and dismantling design packages in Operable Unit 3 ($8.2 million savings); and changes in scope for a Removal Action for improved storage of soils and debris ($2.4 million savings).
The Cost Savings Program is closely allied with the DOE-wide productivity improvement initiative, and Fernald management regularly shares productivity improvement ideas with other sites. Fernald also is developing a team to provide the additional productivity improvement focus required by the current climate of tightened federal budgets.

TECHNOLOGY DEVELOPMENT PRODUCES EFFICIENCIES

DOE and FERMCO also place heavy emphasis on developing new technologies and adapting existing technologies for use at Fernald. The introduction of safer and faster technologies will produce efficiencies in both cost and schedule for the remediation work.
For example, Fernald has used a Solid Block Modeling technology -- developed by FERMCO's parent company Fluor Daniel for use in the mining industry -- to aid in the difficult process of site characterization. Solid Block Modeling uses three-dimensional computer images to show the location, types, and amounts of contamination in waste pits and other areas. The process has been particularly helpful in identifying "gaps" in existing characterization data at Fernald and in guiding subsequent sampling efforts.
Other technologies have also been adapted for use at Fernald. Micro-Purging, a groundwater sample collection method that eliminates the need for excessive well-water purging, both increases the accuracy of sampling by reducing water dilution effects and reduces the amount of contaminated water requiring disposal. Rotasonic (spoil-less) Drilling, a method which captures almost all of a soil sample within the core barrel, eliminates sample loss or cross contamination and reduces drill cuttings by 95-100 percent. Using this technology, a sample at 130 feet below the surface can be obtained in two days, as opposed to two-three weeks using other methods.
Fernald is hosting the DOE Office of Technology Development's Uranium in Soils Integrated Demonstrations (USID). The USID program continues to evaluate various technologies for the removal of uranium from soil using chemical and physical separation methods, as well as Real Time Characterization, Cone Penetrometer probes, and related technologies. Fernald also has applied innovative robotics in the management of wastes and the remediation operations.
Finally, Fernald pioneered Minimum Additive Waste Stabilization (MAWS), a process which converts optimum mixtures of waste to a much safer glass form using specially-designed furnaces. The MAWS developments led directly to the construction of a vitrification plant now under way at Fernald to remediate wastes from the K-65 silos.

REMEDIATION NOW A REALITY

The environmental restoration effort at Fernald has, by necessity, required an enormous investment in preliminary activities related to characterization, analysis, and planning. It also continues to require significant effort in the development of

stakeholder consensus on the appropriate level of restoration and the most desirable remediation alternatives. Fernald is demonstrating that it is among the leaders in the DOE environmental restoration effort and is achieving tangible success in the environmental restoration that all who have an interest in the site have been anticipating. Together, DOE and FERMCO have developed an effective partnership that is successfully melding the interests . . . and the ideas . . . of both internal and external stakeholders in the common goal of environmental restoration at Fernald. See Fig. 1

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STATUS REPORT: FERNALD SITE REMEDIATION

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ABSTRACT

The Fernald site is rapidly transitioning from an Remedial Investigation/ Feasibility Study (RI/FS) site to one where design and construction of the remedies dominates. Fernald is one of the first sites in the Department of Energy (DOE) complex to accomplish this task and real physical progress is being made in moving the five operable units through the CERCLA process. Two of the required Records of Decision (ROD) are in hand and all five operable units will have received their RODs (IROD for OU3) by the end of 1995. Pre-design investigations, design work or construction are now in progress on the operable units.

The lessons learned from the work done to date include implementing innovations in the RI and FS process as well as effective use of Removal Actions to begin the actual site remediation. Also, forging close working relationships with the Federal and State Regulators, citizens action groups and the Fernald Citizens Task Force has helped move the program forward. The Fernald successes have been achieved by close coordination and cooperation among all groups working on the projects and by application of innovative technologies within the decision making process.

INTRODUCTION

The Fernald site is located approximately 20 miles northwest of Cincinnati and has been managed by Fernald Environmental Restoration Management Corporation (FERMCO) since December, 1992. For more than 37 years, the facility manufactured uranium metal products for use at other DOE sites to satisfy Defense Program demands. Production operations were suspended on July 10, 1989 due to a sharp reduction in the demand for uranium metal products by user sites and continuing problems in achieving full regulatory compliance. Following necessary Congressional notifications, the facility was formally shut down on June 19, 1991 when its mission became environmental restoration.

The primary law regulating cleanup of the Fernald site is the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) as amended by the Superfund Amendments and Reauthorization Act (SARA) in 1986. Another major regulation, the Resource Conservation and Recovery Act of 1976 (RCRA), governs the generation, transportation, treatment, and disposal of the hazardous waste at the site. A Consent Decree was signed by the State of Ohio and the DOE in December, 1988, establishing milestones to bring the FEMP into full compliance with RCRA and other regulatory requirements. Amendments establishing additional milestones regarding the management of hazardous waste were later proposed to the Consent Decree, and in January 1993, the amendments were approved, resulting in the Consent Decree and its Stipulated Amendments (SACD).

In July 1990, the United States Environmental Protection Agency (USEPA) and the DOE entered into a Consent Agreement establishing milestone schedules for the completion of necessary studies to support the CERCLA clean-up process. The agreement established schedules for implementing near term clean-up actions (Removal Actions) while final clean-up solutions were being evaluated and selected. In September 1991, the DOE and USEPA jointly signed the Amended Consent Agreement establishing revised milestones for the completion of the required studies and identifying a series of

additional near-term actions for implementation by the DOE.
Here are some background notes of significance:
Production
Production initiated in 1952
Operations placed on hot standby in July, 1989
Formal plant closing - August, 1990
Total Uranium Production (1952 to 1989) - 462 million pounds U delivered product
Average Waste Generation Rate - 2.2 lbs waste per pound U product
Total Waste Product (37 year product history) - 1 billion pounds
Maximum Annual U Production - 10 metric tons (1960)
Environmental Issues
Plant designed to spray calcine raffinate, with pit usage exclusive to lower volume off specification material
1958 economic confrontation with Weldon Spring led to decision to abandon spray calcining process and build raffinate pits
All process residues sent to pits/silos from startup to August, 1984
August, 1984 began drumming all raffinates
80,000 drum waste inventory created by end of 1987
Pits ceased to be used in April, 1987
Plant estimates 400,000 lbs U discharged to atmosphere over 37 year production history
The Fernald site is broken down into five operable units. The following paragraphs discuss the scope and current status of each, along with a description of the remedy and discussion of some of the innovations employed on each and the lessons learned. Figure 1 provides an overall schedule for the five OUs.
OPERABLE UNIT 1 - WASTE PITS
Scope
Waste Pits 1 through 6
Clearwell
Burnpit
Berms, Liners and Soils Within the OU Boundary
Status
RI and FS/PP Reports submitted and approved
Final ROD submitted January 26, 1995
ROD signature expected February 1995
Pre-Design investigation in process
Description of Remedy
The Proposed Plan for the remediation of OU1 involves the excavation of an estimated 710,000 cubic yards of waste materials from six waste pits and two ancillary contaminated areas, the drying of this waste to meet waste acceptance criteria and its disposal in a permitted commercial disposal facility.
Highlights
In order to facilitate obtaining the Record of Decision for OU1, a successful strategy for informing the stakeholders of the Proposed Plan for Operable Unit 1 was employed. One of the greatest concerns to the public is the safety and integrity of transshipment by rail of waste materials along the spur connecting the FEMP with the main rail line to the west. In anticipation of the issues that the stakeholders would raise at the CERCLA public meeting scheduled for August 23, 1994 where formal comments from the public would be received on the Proposed Plan, OU1 held a series of public meetings prior to the official meeting to assure a thorough airing of the issues. A Rail Transportation Workshop was held on August 9, 1994 where a complete disclosure of DOE's plans for rail transport was laid out. A followup availability Session was held a week later on August 16, 1994 with CSX, the rail carrier along the spur line and the carrier for the waste freight from the FEMP to East St. Louis, Illinois. DOE's unit train concept and sealed gondola plans, Emergency Response activities, potential use of adjacent rail facilities were described and discussed with the objective of determining the public's overall reaction to the DOE Transportation plans. These reactions were then addressed and incorporated into the presentations which were made to the public at the formal public meeting held on August 23.
Ohio EPA's Office of Federal Facilities Oversight held an availability session with members of the public during the public comment period and prior to the public hearing for the OU1 Proposed Plan. Availability sessions are held by Ohio EPA during

the public comment period for each operable unit's Proposed Plan. The meetings provide an informal atmosphere for the citizens to ask questions and share concerns one on one with the agency. In addition, Ohio EPA is able to inform the public and address some issues that would otherwise have to arise through the formal public comment process.
As a result the public meeting was a very smooth, non-confrontational exchange of ideas about which USEPA's remedial project manager stated that "this was the best public meeting that he had attended."
It is estimated that $300-400 M could be saved if OU1 were allowed to ship waste in bulk to a commercial disposal facility such as the Envirocare facility currently in operation at Clive, Utah instead of sending the waste materials to the Nevada Test Site (NTS). These estimated savings derive both from the packaging requirements associated with NTS disposal and from the intermodal transfer required to dispose at NTS since NTS does not currently have any direct rail service to its disposal areas.
Working closely with DOE HQ, OU1 sought and successfully obtained an exemption from the portion of the DOE order 5820.2A which requires the DOE to dispose of low level radioactive waste at a DOE facility. The exemption, which is specific to OU1 at Fernald, allows OU1 to dispose of Low Level Waste materials at a permitted commercial disposal facility.
In order to optimize the excavation of the waste, a program entitled the Dewatering, Excavation Evaluation Program (DEEP) project was launched in 1994. DEEP consists of a 4 phase project; geotechnical testing, wet excavation, dewatering, and dry excavation for waste pits 1, 2, and 3. Waste pits 1, 2, and 3 were selected because these pits represent over 80% of the total waste OU1 plans to dry excavate during remediation.
Geotechnical testing will be utilized to evaluate the characteristics and geotechnical properties of the waste before, during, and after dewatering tests have been conducted. Wet excavations consisting of pre-dewatering back hoe excavations and waste reslurrying and pump tests will be performed after initial geotechnical testing. The feasibility of dewatering will be established by comparatively testing alternative well configuration and well types and through the testing of electro-osmosis and vacuum assist enhancements to determine the optimal method to employ during final remediation. Dry excavations, to include dry trench excavation and ramp excavation, will be performed to determine the efficiency of the dewatering techniques, amenability of the waste to excavation and handling, and the ability of the waste to support heavy equipment. To date, the initial geotechnical testing has been completed in the waste pits. Wet excavations are scheduled to begin in mid-January, followed by dewatering, and dry excavation.
OPERABLE UNIT 2 - OTHER WASTE UNITS
Scope
Fly Ash Piles
Lime Sludge Ponds
Solid Waste Landfill
Other South Field Disposal Areas
Berms, Liners and Soils Within the OU Boundary
Status
RI and FS/PP Reports submitted and approved
Draft ROD submittal scheduled February 1995
ROD signature expected June 1995
Pre-Design investigations in process
Description of Remedy
Excavation of materials and disposal in an on-site disposal cell. Shipment of any hot spots off-site.
Figure 2 shows the preliminary size and location of the on-site cell.
Highlights
Three Dimensional Modeling
Throughout the Operable Unit 2 remedial investigation process, soil and groundwater sampling results were fed into a modeling process that accounted for spatial variations in the data and created an estimate in 3D of the extent of contamination. The resulting graphics provided images that were easy to understand and easy to remember. Those images simplified interaction between the DOE and the EPAs and provided an important tool for informing the public about the operable unit.
Reduced Time Frame & Cost for Laboratory Services

The new FEMP program for preapproval of chemical analysis laboratories resulted in shorter procurement periods and reduced overall costs during recent Operable Unit 2 groundwater monitoring and the second phase of the disposal facility predesign investigation.

Cone Penetrometer Testing

Cone penetrometer tests (CPTs) are being used in the proposed disposal facility predesign investigation. The objective of this preliminary work is to better define the lithology in the potential facility location. The CPTs have proven to be five to ten times faster than routine drilling and to have associated cost savings. While CPTs do not provide samples for ex situ testing, they do provide full classification over the depth of the borehole.

The information learned from the CPTs also allows better placement of future wells and borings.

Meetings for Citizen Involvement on the Proposed Disposal Facility

DOE organized three opportunities for the public to discuss the proposed on-site disposal facility. Two public workshops were held, on June 28 and October 25, 1994, to discuss the Operable Unit 2 Feasibility Study/Proposed Plan and details of the preferred remedial alternative. A public meeting took place on November 8, 1994. During this meeting, DOE answered questions and accepted formal public comments on the Operable Unit 2 Proposed Plan and preferred remedial alternative. In addition to the DOE workshops and meeting, OEPA sponsored two availability sessions, in September and November, to discuss the Operable Unit 2 Proposed Plan and preferred remedial alternative. DOE also made presentations to smaller groups of local elected officials at their request. A priority has been placed on public outreach for this remedial action because of the sensitivity and strong public opinions associated with leaving waste on-site in an engineered disposal facility. Important perspective and useful input on this set of issues resulted from deliberations by the Fernald Citizens' Task Force. Indeed, Task Force input was instrumental in determination of the proposed remedy for OU2.

OPERABLE UNIT 3 - FORMER PRODUCTION AREA

Scope

All Man-Made Structures and Facilities at the Fernald Site, Above and Below Ground.

Status

Interim ROD issued July 1994, which allows D&D of all OU3 facilities

A D&D contract has been placed to take down the Plant 4 complex.

RI/FS/PP Report submittal scheduled to EPA September 11, 1995.

Draft Final ROD (disposition of waste material) submittal scheduled July 25, 1996.

Description of Remedy

Dismantlement of structures, excavation of materials and disposal in an on-site storage cell. Some items and material may be disposed offsite. Considerable emphasis on recycling.

Highlights

The Record of Decision for the Interim Remedial Action, signed by EPA on July 22, 1994, represents the first Operable Unit Record of Decision for the Fernald Site. This landmark decision allows remediation of the site buildings and structures to proceed, resulting in significant cost savings and schedule acceleration.

Plant 7 represents the first Fernald Facility to undergo the D&D process. As Fernald's tallest structure, the Plant 7 takedown became a symbolic gateway to future work at the site. (See Fig. 3) The task was difficult and frustrating at times, and the successful takedown was the result of diligent planning, hard work, and perseverance by all involved. The project, in spite of the building's resistance to our planned efforts, was completed significantly ahead of schedule and below cost.

The CRU3 RI/FS Field Investigations Program initiated the field sampling activity in late September 1993 and completed it on August 4, 1994. The program was completed 6 weeks ahead of schedule and under budget. During this time 1000 construction samples were collected. The samples were of excellent quality, resulting in a zero resample rate. Also, an impressive record of zero down time in the field and no lost time injuries are testimony to the good planning and enthusiasm of the program's personnel.

Other significant highlights

Plant 1 Ore Silos - completed demolition

Fire training facility - completed demolition

Plant 1 Pad - expanded/upgraded capability to stage and process remediation waste streams.
Awarded contract to recycle 700 tons of structural steel.
Completed demolition of Plant 7.
Established a prioritization and sequencing schedule and associated cost estimate for decommissioning of all Fernald facilities consistent with site wide remediation goals.

OPERABLE UNIT 4 - SILOS

Scope

Silos 1 and 2 (K-65 Silos)
Silo 3 - Metal Oxides
Silo 4 - Empty
Decant Sump System
Soils and any Buried Concrete
Berm Surrounding Silos 1 and 2
Perched Water (encountered During Remediation)

Status

RI and FS/PP Reports submitted and approved
ROD signed December 1994
Pilot Plant under construction, glass making to begin in July 1995, with radioactive glass scheduled for late 1995.

Description of Remedy

Remove silo contents by hydraulic excavation, vitrify the waste, ship to Nevada Test Site for final burial.

Highlights

Operable Unit 4 was the first operable unit to incorporate NEPA requirements into the CERCLA process at Fernald. This had a dramatic effect on the time and cost for responding to the requirements of each law, taking some extra time to begin with, but reducing the overall requirements by elimination of duplicate efforts. This was done through a Notice of Intent which integrated the process and documentation packages into a Feasibility Study/Proposed Plan-Environmental Impact Statement (FS/PP-EIS). The first final ROD for the Fernald site was also OU4's and was signed on December 7, 1994. This started the clock under the CERCLA process for remediation to commence within 15 months.

Several technical innovations are part of the OU4 project and revolve around the need to handle the silo material with care. The K65 silos are the largest radon concentration source in the United States (and maybe the world). Gas headspace in silo 2 is approximately 3 million pico curies per liter and the interstitial gas space in the solids may approach 40 million pico curies per liter. To deal with this concentration, the K65 material will be removed with a hydraulic mining pump which will contain the radon to a large extent as a soluble gas in the water and the vitrification plant itself will use activated carbon for removal of radon from the process gas streams. A small robotic device is planned for final cleanout of the heels in the tank so that the last traces of the spent ore can be removed before the silos are dismantled.

New analytical techniques are being tried and some are being custom designed to deal with the large range of radon concentrations that will be seen in the pilot plant. Concentrations are expected to range from environmental levels of around 1 to 4 PCi/L to over 60,000 PCi/L in the process itself. Advances in scintillation counting and in personnel monitors are being tracked as part of the project.

A new approach to high temperature vitrification furnace design is incorporated into the vitrifier for this facility because of the very high (approximately 1400 degrees C) temperatures expected to be needed for portions of the waste stabilization. The vitrification process will reduce the radon emanation rate by a factor of approximately 500,000 versus the silo material in its present form. The tighter glass structure of the vitrified product will retain the radon within the matrix so that it decays in place to a much larger extent that is presently the case in the powder in the silos.

The headspace in the K65 silos was very high in radon concentration (over 20 million picocuries per liter) and contributing to high levels of radon near the fenceline of the facility due to natural diurnal aspiration of the silos. A removal action was initiated to reduce the headspace concentration and thus reduce the overall release rate.

The approach selected was to cover the silo material with a cap of bentonite clay which acts as a barrier to diffusion of the radon from the underlying silo material. The radon was retarded in its ability to reach the headspace and the natural decay process of 3.8 days half life captured more of the radon progeny in the clay and silo material. The radon headspace concentration was reduced by approximately a factor of 10 to just over 2 million picocuries per liter so that the radon flux from the silos was reduced.

OPERABLE UNIT 5 - ENVIRONMENTAL MEDIA

Scope

Groundwater (Great Miami Aquifer)
Groundwater (Perched Water)
Surface Water
Soils (Not Associated with OU1 - OU4)
Sediment
Flora and Fauna

Status

RI and FS/PP Reports submitted
ROD submittal scheduled October 1995
ROD approval expected November 1995

Description of Remedy

Pump and treat ground water, place soils in cell where WAC is acceptable, ship hot spots off-site.

Highlights

The high degree of success of the OU5 RI/FS effort is directly attributable to the evolution of successful communication between DOE and the regulatory agencies. This has been developed through frequent meetings and discussions with the regulatory agencies during the preparation and implementation of work plans for field sampling programs, fate and transport model improvement process, and the development of procedures and protocols for risk assessment. Although initial efforts were tentative in the late 1980s, by 1993 a high level of communication had been achieved through technical information exchange meetings, informal data presentations and discussions between technical staffs which continue today. As a result, the reports and work plans submitted to the agencies by OU 5 have been accepted with a minimum of revision and comment by the agencies, because the work performed was consistent with agency expectations.

The second principal factor in the success of OU5 has been the use of a phased approach to conducting the remedial investigation and developing models for fate and transport in groundwater as well as the risk assessments. The 1988 RI/FS Work Plan set the goals of the investigation and the initial steps to be taken in the RI/FS. During the investigation, frequent review of the data as it was gathered resulted in work plan addenda that were focused on specific, narrowing, data needs. This approach ensured the best scientific and engineering practices were effectively applied to the investigation using the available resources. These reviews culminated in the 1993 field sampling programs in which remaining data needs were adequately addressed to complete the RI and conduct the FS.

These data reviews also led to the development of significant removal actions such as the South Groundwater Plume Removal Action. Implementation of the removal actions began the transition from RI/FS activities into actual cleanup activities well in advance of the ROD. Implementation of the removal actions has also provided for the phased development and scale up of treatment facilities based on operating experience with the constituents at the FEMP.

The size of the investigation area and the complex interplay of the pathways involving air deposition, surface water infiltration to perched groundwater, surface water transport and infiltration to the Great Miami Aquifer, and infiltration from perched groundwater to the Great Miami Aquifer resulted in the installation of over 1300 borings and 750 monitoring wells. Subsurface soil samples were collected from the borings and groundwater samples were collected from the wells. In addition, samples of surface soil, surface water, and sediment samples were collected. These samples were analyzed for one or more parameters from a list of radiological constituents, hazardous substances, and pesticides/PCBs. As a result a database of over 1,000,000 records of analytical data was organized, verified, and validated for the OU 5 RI and FS. The database includes RI data collected under work plans developed for each of the five operable units and incorporates pre-RI data and FEMP

routine environmental monitoring data with appropriate validation qualifiers. Environmental data collected during routine monitoring and RD/RA sampling for each operable unit at the FEMP will continue to be incorporated into this database. The database will be an invaluable resource to the development of RD/RAs for OU 5.
Fernald developed a successful ecological risk assessment strategy in conjunction with USEPA in early 1993, and incorporated the site-wide ecological risk assessment into the OU5 RI Report. Risks were evaluated in specified study areas of the FEMP to estimate potential risks to specific ecological receptors (as defined in the approved strategy). The risk assessment strategy was negotiated early with the regulators to facilitate prompt review and approval. These efforts were successful as reflected in the receipt of only minor comments from the USEPA, OEPA and support contractors.
Overall, a 17 volume RI Report, including an attached plate of over 350 maps, was submitted to the EPAs for approval. The document contained over 35,000 pages and represented the anthology of the more than 11 years of investigations of the environmental conditions at the FEMP and surrounding areas. Its approval represents a consensus on the part of all involved organizations that the nature and extent of environmental contamination at the FEMP site has been appropriately defined. Under a rigorous review by the regulatory agencies and their support subcontractors, the RI Report received only relatively minor review comments. The level of comments was reflective of the cooperative spirit that had been fostered between OU5 and the regulators through the use of this highly interactive document planning process.
The progressive data reviews associated with the field investigations program of the RI/FS led to the identification of near-term environmental issues. The identification of these issues facilitated the performance of a series of removal actions focused on attenuating the further migration of contaminants or minimizing the release of contaminants from the facility.
Issues relating to future use of the Fernald Site are mainly addressed in Operable Unit 5, and they incorporate input and recommendations from the Fernald Citizens' Task Force on future use.
Contaminated surface water controls
1986 - Storm water retention basin installed - Controlled the bulk of the contaminated runoff from the production area which was the primary source of contamination to the Great Miami Aquifer
1988 - Surface water control of the plant 1 pad - Rerouted contaminated runoff from the pad and periphery to the stormwater retention basin
1992 - Waste pit area runoff control (Removal Action No. 2) - Controlled contaminated runoff from the waste storage area (OU1) and the K-65 silo area (OU4)
1993 - Waste pit area containment improvement (Removal Action No. 22) - Involved minimizing the potential for wind and water erosion of contaminated materials by seeding exposed and stressed surfaces in the OU1 study area.
1993 - Collect uncontrolled production area runoff (Removal Action No. 16) -Controlled additional contaminated areas around the perimeter of the production area
Groundwater
1989 - Contaminated water below FEMP buildings (Removal Action No. 1) - Pumps highly contaminated perched water to reduce the potential for contamination to migrate to the Great Miami Aquifer - Water beneath plants 2/3, 6, 8, and 9 is being pumped and treated for VOCs and uranium
South Groundwater Contamination Plume (Removal Action No. 3) - implementation of its five parts is phasing into the OU5 remedy for groundwater contamination. See Fig. 4 for a schematic of the South Plume.
Part 1 - provide an alternate water source to an affected industry - completed
Part 2 - install a groundwater recovery well system to extract and pump groundwater back to the FEMP site for monitoring and discharge to the Great Miami River -Initiated Pumping August 1993. Also, installed a new HDPE outfall pipeline with fusion welded joints thus addressing the questionable integrity of the existing outfall and providing for future remediation waste water flow capacity.
Part 3 - construct an interim advanced wastewater treatment facility to remove additional uranium from site waste streams to reduce uranium discharges to the Great Miami River - Online since July 1992; treats 300 gpm from storm water retention basins (500 - 700 ppb to <.1 ppb total uranium) and up to 100 gpm other plant process/ storm water (1000 - 1500 ppb to approximately 10 ppb total uranium)

Part 4 - monitor groundwater more frequently and prevent use of contaminated groundwater - Ongoing, began in 1992
Part 5 - identify the location and extent of any remaining contamination attributable to the FEMP site (i.e., locate the leading edge of the plume exceeding the 20 ppb level) - Complete 1992
South Plume Interim Treatment project
  Dedicated treatment for 200 gpm South Plume groundwater
  Treats water containing 20-30 ppb uranium to <.1 ppb uranium
  Initiated continuous treatment in March 1994
Advanced Wastewater Treatment facility
  Scheduled to be online January 1995
  Phase I will treat 700 gpm of stormwater from SWRB or groundwater as capacity exists
  Phase II is 400 gpm treatment capacity for existing process waste water, South Plume groundwater, and future remediation waste water from other OUs
  Both Phases will focus on uranium removal
Provision of alternate potable water supplies to homeowners whose private wells were impacted by the South Plume - Supplies bottled drinking water to homeowners whose water supply has been affected by the FEMP (Uranium concentrations < 2.7 ppb).
Providing funding and technical support to the Hamilton County Department of Public Works to construct a reliable public water supply for residences around the FEMP.
SUMMARY
Fernald's environmental restoration program is well along the path from site investigation into genuine remediation. Success in this transition has been driven by innovations in regulatory relations; extensive stakeholder involvement; judicious application of technology; and innovative approaches to project management which have improved safety, accelerated schedules, and reduced costs.

8-3
DOE'S PERSPECTIVE: REACHING SUCCESS BY STANDING ON A THREE LEGGED STOOL
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ABSTRACT
Gridlock, inertia, conflict, outrage, bureaucracy, obstruction, media sensationalizing, courts, and politicians. These are the things that characterize any attempt to implement a public policy today.
It is worse today than it has ever been because the middle has dropped out of public opinion. We have today no consensus of public values. At Fernald, we have come to recognize that in order to achieve any success we must first build a public consensus about what success will look like.
We do this through a three-part approach we call the three legged stool. It includes public information, management involvement, and person-to-person communication. Each of these elements is essential.
PUBLIC INVOLVEMENT HELPS GET THINGS DONE
The reason we need to do public involvement is to make better decisions. Most of the Department of Energy's projects have failed or been stalled, not for technical reasons, but for political ones. In hindsight, we can say those projects were bad public policy decisions. This is not a phenomenon unique to the Department of Energy. All around the country, there are important projects that have failed because the appropriate political work has not accompanied the technical work. In Cincinnati there is a road like that. It was supposed to be a limited access inner belt across the north part of the city. It is called the cross-county highway. Unfortunately it doesn't cross the county. At one end of the county it begins, runs a couple of exits and then stops. At the other end of the county it begins, runs a couple or exits and then stops. The cross county highway never crosses the county. It would be more aptly named the highway at either side of the county. Good money was spent to begin this project and plan it. It would be a useful part of the highway net but the necessary political support for it was never generated.
Managers who believe that their only responsibility for their project is technical are likely to fail. In our society, ordinary citizens have the power to stop projects by demonstrating, lobbying their congressional delegation, going to court,

using the mass media, and even violence. In order to succeed, managers have to adopt good public policies. Good policies meet the needs of the majority without harming the minority. Such a policy is usually not clear. It is not clear to the public and it is not clear to us as managers. Although all too often we get an idea in our head that is all too clear and all too wrong. If we had better feedback we might not embark on cross-county highway disasters. Choosing a good public policy is a process of discovery and creation. At the same time it is also a process of promotion. The way to make the best decisions and get support for that decision is by public involvement.

People have a fundamental desire to participate in decisions that affect their lives. There are many people whose lives can be affected by our operations. Some of these are residents near our operations, business leaders, people concerned about the environment, and perhaps most important, our own employees. If these people feel they cannot participate in decisions that affect them they might not accept those decisions. If they don't accept them, they may try to get them overturned. At the very least they will passively resist them.

This one fact about human behavior would be enough to justify a vigorous public involvement program. But there is another reason. Time has shown that many minds, working together, can often come up with better solutions for problems. It may take extra time at the beginning to involve more people in decision making, but it can save time and money before the process is over.

WHAT IS PUBLIC INVOLVEMENT?

Public involvement is not a vote, or a popularity contest. At the outset, someone must accept responsibility for making policy decisions. Public policy decisions are often difficult because conflicting interests must be balanced. Public involvement can make these decisions a little easier and help predict the level of public acceptance for a particular decision. Public involvement is a process to give those officials the best information to make an informed decision. It also makes sure that all the people who have concerns are heard.

The three legs of public involvement

Think of public involvement as a three-legged stool. Leg one is public information. Leg two is management involvement. Leg three is person-to-person communication.

1. Leg one: Public information

When people think of public involvement, they often think of public meetings, brochures, video production, and displays. These are elements of a good public information program, but they are only one leg of the public involvement stool. In most operations, it is the strongest and most well-developed leg. We have a talented and experienced Public Affairs staff that do these things well. Do not imagine, however, that you have good public involvement by just holding good meetings and producing good brochures.

2. Leg two: Management Involvement

Public involvement starts to become a reality when managers accept it as their responsibility and make it a regular part of their decision-making process. That process has the following steps:

Identify the problem -- in clear terms.

Identify why it must be solved and whether or not you and your organization are theright people to solve it.

Identify the people and institutions like to be affected by the problem or your attempts to resolve it.

Speak to them and identify their concerns.

Identify a set of alternatives that show different ways of taking action. Every person affected should be able to see at least one alternative that includes measures to protect their interests.

Make sure the persons affected understand the alternatives and can see how at least one of the alternatives clearly addresses their concerns.

Evaluate and refine the alternatives. Show the effects on everyone's values of each alternative.

Give all affected parties an opportunity to say whether they think the alternatives have been described and evaluated accurately and fairly. Change the alternative if the new information warrants.

Make the decision.

Continue to evaluate progress using these same steps.

This may seem like a very long and complicated process. Sometimes it can be. For

large projects it may take many studies and several years. For other, simple projects, it may only mean a meeting and a few phone calls. The important thing is to make it a natural way of doing things. It will save time in the long run.
When management becomes involved in this way, the public involvement stool becomes much stronger. The first leg of the stool, public information, starts to become more effective and meaningful to the public. The second leg, management involvement, places responsibility for public involvement with decision makers.
3. Leg three: Person-to-person communication
The last leg of public involvement is person-to-person communication. This is communication through personal relationships between site employees and members of the public.
Our society is overwhelmed with information. Most people are bombarded day and night by information, most of it considered irrelevant and perceived as serving someone else's interest, or being just plain false.
People tend to trust someone they know over a newspaper article or a television show. Experience tells us that in most American communities, even big cities, there is a relatively small group of opinion leaders. The opinion leaders reflect the values of a group of people who share a common interest. If their concerns are addressed before a decision is made, the decision will rarely be overturned.
At Fernald, we have designated employees to act as representatives to opinion leaders. These employees are called envoys. It is their job to cultivate a relationship with one or more opinion leaders. In some cases, this is a natural development of existing relationships. The important thing is the relationship not the message. It is not their job to promote the site or sell a program. There most important job is to build and preserve a relationship of trust.
How the envoy concept works
For instance, the Fernald fire chief already has business contacts with fire chiefs and other emergency management people in the surrounding community. The envoy concept takes advantage of this relationship by making the fire chief the Fernald representative to the opinion group he or she is already contacting.
After all, the group naturally turns to the Fernald employee for information about site activities and related issues. Making that person the envoy merely recognizes that relationship and ensures that the envoy has complete, current and accurate information. Public Affairs gathers that information; envoys only need to tap into it.
Although the envoys can provide information,they should think of themselves as a pathway between the site and the public. Along this pathway travel messages, and concerns in both directions. Really important and complex issues may require dialogue between a site manager and some interested or concerned party. If the envoy has built trust in his relationships that trust will be, to some extent, extended to whoever the envoy introduces. Instant credibility. Because communication is two-way either through envoys or face-to-face management gets a much better understanding of the concerns of the public.
After each contact, the envoys report on what they have learned. Public Affairs then passes along that information to managers. The envoys then ask Public Affairs for any information they need to pass on to their groups. The envoys also prepare a debriefing report for Public Affairs.
The fire chief is just one example of how an envoy might be recruited. Thirty or so envoys would be a powerful communication tool. With envoys, public meetings become less important than one-on-ones, which then may only serve as group affirmation of understandings reached through person-to-person contacts. Brochures or fact sheets become calling cards to reinforce messages the envoys deliver.
The envoy system has proven useful in some unexpected ways.
It is an early warning system to alert project managers that their actions could become a cause of controversy or that bad feelings are being developed because of operations.
It has become a source of new ideas and solutions.
It is a powerful communication tool. Through the network of envoys information can be conveyed swiftly and effectively.
It reduces media attention to your operations. If something happens at our site, like an environmental spill, or an accident, we can get rapid, accurate information to community opinion leaders through a credible source. The outrage factor goes down, by the time reporters start getting quotable quotes, the quotes are less

likely to have anger toward us.
Other techniques to enhance public participation
The three-legged stool requires new communication techniques to support it. Traditional methods, most of which only promote one-way communication, are not sufficient to fully involve the public.
Although a certain level of public participation is required by various laws and regulations governing radioactive and toxic waste, most of these activities are retroactive; the public gets its say only after the initial recommendation has been made. As discussed earlier, the consequences of the public not supporting a particular decision can be time-consuming and costly.
At Fernald, we are changing the old way of doing things in order to maximize public involvement in an efficient and effective manner. While this requires more initial planning, the rewards promise to be great.
Some of the techniques we are introducing at Fernald modify traditional public information techniques to make them more interactive. Some of those include:
Publishing a community bulletin that summarizes the status of studies and activities supporting various phases of the Remedial Investigation and Feasibility Study, as well as other cleanup actions. For example, the bulletin would discuss all the disposal options being considered for the site, even before the actual studies are underway. The community bulletin is primarily an external communication tool.
Introducing a question-and-answer column in the monthly bulletin, much like the question-and-answer feature in the Fernald employee publication Focus. With a Q&A column, the public will have a way to get information about the site without having to wait for a public meeting. The benefit for technical managers is that they get specific feedback about site-related activities. For example, the Q&A can be a barometer of the community's acceptance and opinions on such issues as transportation options.
Inviting the public to comment on ideas as a partner in brainstorming sessions. Think of these public sessions as the CERCLA equivalent of NEPA scoping meetings. For example, if DOE decides to develop studies on various treatment options, there would be a public work session to solicit ideas and feedback from the community.
The creation of a citizens advisory board commissioned to solve specific problems. Public involvement supports the work of this board, the Fernald Citizens Task Force. When information is provided to the task force, it is given to the public. The intention is to thwart any potential tendency to see the task force as the only group of stakeholders worth listening to. While the task force's membership is a diverse and balanced group that typifies the major stakeholders for the Fernald site, it cannot possibly represent every interest without becoming too large and unwieldy. Sharing information with other constituencies, particularly those not directly represented on the task force, guarantees that public participation at Fernald is more than seeking consensus from the task force.
Disseminating a "management memo" that provides information about policy and site issues for supervisory personnel. The management memo is primarily an internal communication tool.
Creating a "newsroom" that provides accurate and up-to-date information about site activities and issues. This innovation, adapted from the operations of daily newspapers, is especially important for supporting the envoys. In the Fernald "newsroom," public affairs staff gather information from other divisions or departments, such as strategic planning, regulatory compliance, risk assessment, and construction. Envoys then draw upon that information as needed to fulfill the information needs of their constituencies.
The newsroom works like this:
Public Affairs staff -- primarily the counselors -- will gather information on all aspects of site operations and cleanup activities. Of course, anyone can be a "beat" reporter in the Public Affairs newsroom; managers may report information they received at a meeting or through other means. All managers or employees need to do is call or e-mail Public Affairs -- much the way members of the public call news organizations with news tips.
That information is compiled by the Public Affairs equivalent of a "city editor." The editor keeps the information in a easily accessible, centralized database. This database functions much like a "wire service," and serves as a source of information available electronically to managers, employees, and envoys as talking points for person-to-person contacts. The "wire service" information can be used for

internal or external publications and presentations.
In addition to compiling and maintaining the information database, the editor serves as a gatekeeper, making sure the information is complete, verified and timely. The information in the database is used for all public information tools, which ensures the consistency of the quality of information being disseminated to internal and external audiences.
At Fernald, these communication techniques are being developed with the goal of enhancing public involvement and providing accurate and timely feedback to managers. Although public involvement is the responsibility of program managers, public affairs specialists play an important role. At Fernald each project manager is assigned a public affairs counselor. These counselors draw upon their resources to help managers. They coach, train, and support. They help managers develop communication plans. They use their special communication skills in writing, graphics, video, and meeting facilitation to develop good communication tools for the manager. The counselors pay a special role in communicating through the mass media. Most importantly they integrate and coordinate the actions of the many different programs into a single site-wide public involvement program.
Public involvement is not a tidy process. It is not straightforward. There are no guarantees. However, huge public treasure has been lost because public policy was not accepted by the public. It is worth making the effort to use public involvement to discover and build support for a policy that will be accepted. The Fernald experience suggests that it is well worth the effort.
Planning for public involvement begins before a project starts
The DOE site manager and the Public Affairs staff use the information they get from the project plans to develop public involvement activities to support those efforts. They coordinate the many different public involvement demands of the various projects. That way we can avoid duplication or overkill and assure uniform quality. There are some site-wide activities that project managers can take advantage of as they develop their individual plans. The public involvement process will be simpler if project managers can take advantage of these site-wide activities for their own specific communication needs.
As soon as a project or change of direction starts to loom on the horizon, managers should consult with their Public Affairs counselors. When an activity requires public involvement, the Public Affairs counselors will prepare for their technical managers a communication plan. The communication plans average about 2 to 3 pages and specify the following information:
- Objective
- Audience
- Strategy & Messages
- Methods to deliver and receive messages
- Sensitivities

The Public Affairs counselors also create and maintain project-specific attachments that summarize community concerns. Because public involvement is a dynamic process, these attachments must be frequently updated.

8-4
PUBLIC PARTICIPATION AT FERNALD: THE TASK FORCE PROCESS
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ABSTRACT
This paper focuses on the unique aspects of the Fernald Citizens Task Force process that have contributed to a largely successful public participation effort at Fernald. The Fernald Citizens Task Force passed quickly by many procedural issues. Instead, the Task Force concentrated on (a) educating itself about the site, its problems, and possible solutions, and (b) choosing a directed way to approach its mandate: To make recommendations on several "big picture" issues, including future use of the site, cleanup levels, waste disposition, and cleanup priorities. This paper presents the approach used at Fernald for establishing and running a focused site-specific advisory board, the key issues that have been faced, and how these issues were resolved. The success of Fernald in establishing a strong and

functioning site-specific advisory board serves as a useful model for other DOE facilities, although the Fernald model is just one of many approaches that can be taken. However, the approach presented here has worked extremely well for Fernald.

INTRODUCTION

Site-specific advisory boards (SSABs), of which the Fernald Citizens Task Force is one, have been established by the U.S. Department of Energy (DOE) at several of its major remediation sites across the country. In the last five years, there has been a growing demand for the creation of citizen advisory groups at DOE's major complexes. In 1991, the Office of Technology Assessment published Complex Cleanup, which argued that there was a "need for a decision-making process -- acceptable to all interested parties -- through which public concerns can be addressed and resolved" as a way to secure public acceptance of cleanup decisions and future courses of actions at DOE sites. Advisory boards were suggested as one option for creating avenues of meaningful participation for interested and affected community members. In 1993, the Federal Facilities Environmental Restoration Dialogue Committee published its Interim Report, "Recommendations for Improving the Federal Facilities Environmental Restoration Decision-Making and Priority-Setting Processes" (often referred to as the Keystone Report), which also recommended creating site-specific advisory boards to assist in the decision-making process.

The Department of Energy moved quickly to implement these recommendations, and SSABs are up and running at several major DOE sites that require remediation, including all of the most severely contaminated ones. DOE at the Fernald Environmental Management Project (formerly the Feed Materials Production Center), in cooperation with the U.S. Environmental Protection Agency (EPA) and the Ohio Environmental Protection Agency (OEPA), asked key stakeholders if they wanted a site-specific advisory board. DOE, the regulators, and local stakeholders decided that a site-specific advisory board would be useful as an additional component of Fernald's existing and extensive public participation program. The Fernald Citizens Task Force was established and, a little more than a year later, it has delivered its Interim Report containing recommendations on cleanup levels and future use of the site once remediation is complete. The Task Force is on track to deliver a Final Report in July 1995.

Nevertheless, the precise role of SSABs in public involvement within the DOE complex is still being defined. What is clear from Fernald's experience, however, is that SSABs have great value for incorporating citizen involvement in public participation at DOE sites. So far, the Task Force process has enabled DOE to establish the personal relationships that are essential to public acceptance of its work, and the Task Force recommendations have avoided the pattern of extreme demands and out-of-hand rejection that all too often characterizes the governmental relationship with the public in environmental issues.

BACKGROUND

The Fernald Environmental Management Project is located approximately 17 miles northwest of downtown Cincinnati. Surrounding properties consist primarily of agricultural and residential development. The facility was established in 1951 as the Feed Materials Production Center to produce uranium metals for weapons production. The facility operated in this capacity until 1989 and the end of the Cold War. In 1991, its mission was changed to environmental restoration. Production and disposal activities, wind, and runoff during its 38 years of operation have resulted in widespread contamination from uranium and other hazardous and radioactive chemicals both on and off the 1,050-acre property. Of greatest concern is uranium contamination of soils on site and above background levels up to five miles from the facility. Also, uranium contamination of a sole-source drinking water aquifer beneath the site affected off-site water wells in the area. As a result, strong grassroots citizen activity -- notably the Fernald Residents for Environmental Safety and Health (FRESH) -- was established in the mid-1980s to press for clean-up of the Fernald site and to prod DOE and its contractors to open the decision-making process to stakeholders. In 1989, the Fernald site was added to the National Priorities List.

Fernald, in sum, presents several important characteristics that suggest the value of an alternative to the traditional present-and-defend approach to public participation. First, there is an active, vocal, and skeptical public. Public involvement in cleanup activities at Fernald is sophisticated and long-standing. The Fernald Citizens Task Force has had to overcome a great deal of mistrust of DOE in

order to work effectively on the problems that beset the site.
Second, there are large volumes of waste. Returning Fernald to background conditions would require the disposal of tens of millions of cubic yards of radioactive soils and materials. Task Force recommendations must take into account all of the complex issues of on-site versus off-site disposal, acceptable land uses, acceptable risks and cleanup levels, and the efficacy of available treatment and disposal technologies.
Third, the site has resulted in immediate impacts on stakeholders. The accessibility of the Fernald property makes existing health threats very real to the community and eliminates options for locating disposal facilities outside of the public's view and reach. For example, storage silos containing the most hazardous wastes at the Fernald facility are only 1,000 feet from the site boundary and literally right across the street and visible from surrounding homes and farms.
Fourth, there has been demonstrated a greater need for stakeholder access. Like all of DOE, Fernald has gone through a metamorphosis with regard to public access and input to its activities over a relatively short period of time. It was not long ago that many of the activities at Fernald were secret. Public access was prohibited and little was known by neighbors about site activities; indeed, some neighbors thought "Feed Materials" referred to animal feed. As the Cold War ended and the need for public participation became apparent, information was released in enormous volumes as public comment was solicited on cleanup decisions in accordance with federal law. Stakeholders at Fernald still felt like victims in a "decide-announce-defend" system, only now they had to deal with information overload as well. Today, Fernald managers recognize the need to develop a decision-making process that provides for the bona fide involvement of all stakeholders to make credible and implementable decisions that will stand up over time.
Fifth, budget constraints pose limits on the extent of cleanup. Budget considerations, prompted by federal efforts at deficit reduction and cost controls, will have significant impacts on the level of cleanup that will be achievable as well as the length of time needed to achieve results.
The Goals of Establishing an SSAB
Against this backdrop, then, it was decided to establish an SSAB at Fernald. In designing a process for establishing and operating a citizens advisory board, it is essential to keep constantly in mind the desired characteristics of the advice that DOE is seeking.
First, the advice must come from a group that is both reasonably representative of the wide variety of stakeholder views at a complex remediation site, and also credible to those who are not directly represented on the SSAB. This requires both a good process for choosing members and good judgment in making the selection. To be useful -- and hence actually used -- the advice also must meet several other criteria. It needs to be relevant to the actual remediation problems that DOE faces. This usually means that specific issues must be identified upon which public input would be helpful. Citizens' recommendations must also be informed. Further, they must be timely, that is, given at a point when the decision is not final or effectively final. Last, the advice must be realistic, recognizing the political, technological, and financial strictures within which all environmental decisions are inevitably made.
To achieve recommendations that have these characteristics, both DOE and its stakeholders have responsibilities. These are summarized in the following table:
CHARACTERISTICS OF SSAB RECOMMENDATIONS

Characteristics of Recommendations	DOE's Responsibilities	SSAB's Responsibilities
Representative and credible	Seek the full range of stakeholder views; assure independence	Reach out to the broader community
Relevant and specific	Participate fully with citizens	Address agreed upon issues
Informed	Provide full information	Inform selves of needed information
Timely	Involve the public early	Remain focused on workplan
Realistic	Give genuine consideraton to recommendation	Be flexible and seek consensus

With these characteristics in mind, we now turn first to the convening process, in

which DOE establishes a board that is likely to produce recommendations with the above characteristics, and then to the activities of the SSAB itself, which should be designed with the same goal.

Convening the Advisory Board

The first question for a site must be whether an SSAB is desirable at all. This involves at least three subquestions: 1) Are there specific, locally resolvable issues that need to be addressed? Nuclear nonproliferation cannot be resolved at a local level; future site use cannot successfully be dictated from above. DOE, working with its regulators, must be sure that there are appropriate issues for the SSAB to consider. Without a clear sense of what is needed, the process is likely to be frustrating for all parties concerned. 2) Are these issues on which DOE and its regulators are willing to accept advice different from its current plans? The worst possible outcome of an SSAB process is for the public and the board to find that the decision for which advice was sought was a "done deal," that the decision had in fact already been made and that public input was mere window-dressing. 3) Is there enough time remaining before the decision must be made? Even when they are working smoothly, citizen advisory groups take time to get organized, to inform themselves of the key issues, and to come to consensus. Moreover, a group that meets on a monthly basis cannot make decisions on lots of relatively small, individual activities. Therefore, an SSAB should not be convened unless an issue can be identified that gives the group plenty of lead time.

One of the key decisions made in establishing the Task Force was to charge it with addressing specific issues: future use, waste disposition, clean-up levels, and priorities. They are broad and involve nearly every aspect of clean-up decisionmaking, but the issues give direction and they signal where public input will be most useful to DOE. At Fernald, the clean-up decisionmaking was well underway when the Task Force was convened, but several important issues remained. The most important and controversial issues -- groundwater protection and waste disposition -- were at least a year from being decided. These are issues on which the law permits different answers, and they were to be definitively answered at the site level, with sufficient lead time.

Representativeness and credibility were established by the convening process. At Fernald, this began with an independent, credible convener who was to select a broad-based membership and identify a strong, unbiased chair. This process was selected in consultation with the stakeholder community to best balance the interests that need to be represented on the board with the need to get a board up and running in a reasonable time. Focus was placed on selecting a convener who would be able to identify balanced and representative members and avoid biasing the board toward any existing stakeholder groups. A single individual from the University of Cincinnati with no prior relationship to DOE or any stakeholder group was retained by DOE to serve as the independent convener. This worked well at Fernald because the individual was widely respected within the community and did not have any interaction with the board or site after the convening process was complete; thus, the convener was able to avoid any potential conflicts of interest in performing her duties.

Other convening approaches -- including empaneling a steering committee to select members, having DOE or a state agency select members, or holding some form of election or lottery --were considered and rejected, either because of the potential for conflicts of interest or because of the length of time that would be required to establish a board. The way in which Fernald's site-specific advisory board was convened set the tone for the overall effectiveness of the group. The convening process worked within an established set of values that defined the goals of the board. The primary objective of the convening process in creating a focused site-specific advisory board at Fernald was to result with a board that represents broad stakeholder interests while at the same time has the potential to work effectively together to make the decisions required. The goal was not to achieve complete stakeholder consensus on every individual selected for the site-specific advisory board, but to get broad stakeholder acceptance of the process used to identify specific members. The timing of decision-making activities at the site also required that a board be established as soon as possible in order to have meaningful input to decisions.

A size limit of 15 members was established at Fernald before the convening process began. This obviously resulted in some very difficult choices; not everyone can be

afforded a spot on the site-specific advisory board. At Fernald, the result could have been some nay-sayers and skeptics rejecting the board; however, because the convening process was developed in consultation with stakeholders and was open to the public, the composition of the board went largely unchallenged. Focus also was placed on getting a balanced representation without requiring any strict quotas for membership. The Fernald Citizens Task Force reflects the demographics and socioeconomic conditions surrounding the facility and all stakeholder groups can broadly identify with one or more members of the board. Local representation was given top priority in convening the Task Force. Both DOE and local stakeholders agreed that the issues at stake were local, and national representation should only be included if directly requested by a local group or if a clear stake were identified that was not encompassed by local interests.

The Fernald convening process, however, does not eliminate the need to make hard choices, and to accept responsibility for them. Not everyone can or should be selected, and indeed not every single interested group can be. Rather, the tough job is to select the key interests and then encourage those selected to keep in touch with their constituents. For example, a site may have several unions or several townships: choosing all would skew the content of a small board. It is incumbent on the convener to choose the best representative of the unions or townships based on the above criteria; and it is incumbent on the representative to keep in touch not only with her own union/township but those who are not on the board. For example, area residents -- through the Task Force process at Fernald -- discuss the consequences of proposed courses of action and now share in the responsibility for recommendations on cleanup levels and other issues. Ultimately, however, DOE remains responsible for decisions. It cannot completely stand aside, because it is being asked to pay particular attention to this group. Thus, DOE will be held responsible by the general public anyway, and it is in DOE's long-term interest to ensure up front, to the extent possible, that the group will be accepted by the general public.

Finally, several elements of the Task Force process were implicit in the convening. In the future, these should probably be made express in the charter or other appropriate document. DOE must assure administrative support for the group, adequate information, and independence. All of these are essential, and all cost money. Mailing, organizing meetings, and generally coordinating activities requires one or two full-time staffers. At Fernald, contractor and DOE employees were detailed to the Task Force; at other sites, a separate staff was hired. Information, however, is the most important support that DOE provides. It must be forthcoming, and DOE must be willing to obtain new data (within reason) and to reorganize information it already possesses. While the vast majority of the information used by the citizens will come from DOE and its contractor FERMCO, the board must have the ability to assess it independently. The Task Force hired an independent technical advisor, who also serves as facilitator, to develop information, assess it, and present it to the group. This has seemed to bridge the gap well between independence and duplicating work already performed.

Getting Organized: The Future Use Approach

Once convened, the advisory board has its own set of responsibilities. First and foremost, it must get itself organized and working on the designated issues. As mentioned above, at Fernald they were: future use, clean-up levels, waste disposition, and clean-up priorities. After convening in September 1993, the Task Force held three meetings to orient itself to the issues and basic information about the site. In November, we decided to address future use first because a recommendation on future use could be the foundation for decisions on other strategic issues. Future use not only gave further direction to our efforts, but also it provided an overarching structure for considering specific issues. As we found, it also allows us to deal with the many ongoing activities at a complex site like Fernald -- we know which are central and which peripheral to the main task at hand.

The future use focus also allowed us to target our search for an outside consultant to someone with the appropriate expertise, whose job was to gather, analyze, and sift relevant information for the Task Force to use in coming to its conclusions. This was essential for the process of developing detailed information about the designated issues. The Task Force had to educate itself about central concepts, like risk; understand the models and assumptions that go into the data presented; and

integrate the information about various parts of the site. Developing expertise is mostly about obtaining information, but ultimately this is trust-building. By providing reliable data -- which can be used to ask hard questions -- DOE exemplifies openness and builds confidence in itself.

The Task Force also adopted a detailed work plan that would allow it to address the issues in time to make an impact. Ideally, of course, the SSAB would be constituted at the very beginning of the process, so that its recommendations could be submitted at or before the time of the general public comment. The goal is for a roughly parallel process, in which the citizen group's knowledge of the site follows closely after the government's, so that both are developing positions together, instead of responsively. At Fernald, this was impossible given the preexisting schedules, but, helped by the future use orientation, we were able to focus on later, cumulative decisions.

Each Task Force meeting focuses on a specific set of issues as laid out in the board's work plan. This approach allows members to fully explore the relevant issues in an incremental fashion, thus avoiding becoming overloaded with information. Task Force members discuss the information, and technical representatives are on hand to answer questions. An effort has been made to keep meetings informal and accessible, while maintaining focus on specific issues.

Because an SSAB is unelected and must necessarily be comprised of less than all of the people or points of view at a site, its recommendations are entitled to relatively little weight if they are supported by only small majorities of board. Conversely, consensus views are entitled to great weight, as they represent a position that a broad cross-section of the community agrees on after a thorough investigation of the issues. Consensus does not necessarily mean unanimity, and in practice few positions on truly controversial issues are likely to win unanimous acceptance. It is therefore advisable to back up a consensus-based process with voting. The board has an obligation to attempt to reach consensus on everything, and if that fails, on as much as possible. Moreover, it should report all views fairly and completely. Hard choices remain for DOE and the individual members of the Task Force -- for example, the use of an on-site disposal cell. It is not realistic to think that this process makes all decisions easy; DOE will still, in all likelihood, have to conclude that one choice is better than another. A good SSAB process, however, can lend strong support to a decision that follows the board's recommendation.

Another consequence of the selective nature of the SSAB is that it must constantly reach out to the constituencies that it directly represents, as well as to the general public. To begin with, all Task Force meetings are well publicized and open to the public; an opportunity for public comment is always provided. To supplement this, the Task Force holds regular special meetings with the public to discuss specific issues and to share some of the learning tools (for example, the FutureSite exercise) it uses. The Task Force uses these meetings to receive as broad an input as possible, to update the public about its own activities, and to inform the public about the issues under consideration.

Results So Far

The ultimate goal of the SSAB process is a recommendation that is realistic given the legal, technological, and financial constraints on environmental remediation. To achieve this, the Task Force followed an orderly program of understanding the conditions at the Fernald site, identifying options for future use, analyzing available technology for waste management and disposal, and developing values to guide all future use recommendations. The consensus values agreed to by the Task Force encompass environmental values, economic values, long term management values, social and human values, and general use values. After identifying these values, the Task Force developed specific criteria to weigh against potential future land use options. Some of those criteria include minimizing impacts on the environment during remediation and maximizing restoration of the environment after remediation; ensuring that any waste that may be left on site be controlled to prevent further contamination of the Great Miami Aquifer, air and soils on- and off-site; developing future uses that provide some level of continuing employment for area residents, but not necessarily in categories that have traditionally been present at the site, and creating a long-term mechanism to ensure citizen involvement in the control, management, and future decisions at the site.

The Fernald Citizens Task Force has reached several preliminary recommendations on

future use and cleanup levels for the Fernald site, and I think that they meet the criterion of realism. These recommendations are contained in an Interim Report, which was submitted to DOE, EPA, and OEPA on November 30, 1994. They cover groundwater remediation and protection, removal of contaminated soil, and exclusion of certain future uses.

Because protection of the aquifer was one of the Task Force's consensus values, it took an in-depth look at the options dealing with groundwater contamination. The Task Force rejected the option of not remediating the aquifer, because the contamination would impact as much as 4,000 surface acres and 32 billion gallons of water if left unchecked. Instead, it has recommended 20 parts per billion as the cleanup level for the aquifer, which is the MCL level, after considering the marginal risk of going below 20 ppb, the difficulty of detecting the background level, and the length of time that treatment to the background level would require.

For soil, the Task Force evaluated risk throughout the range of risks considered acceptable by EPA for Superfund cleanups across a broad spectrum of land uses in evaluating the overall level of cleanup for the Fernald site. Evaluating the impacts of applying different risks across different land uses allowed the Task Force to compare numerous factors, including total soil volumes requiring excavation; off-site disposal requirements; on-site disposal requirements and disposal cell size; total cost; environmental impacts, and technical, legal, economic, and social implementability. The most striking concern in making this decision was the volume of soil that would require excavation beyond the Fernald site property boundary if a 1x10-6 risk level for a residential scenario were chosen. At this risk level, a total of 5.2 million cubic yards of soil would be removed from off-property locations alone. Evaluating the construction risks of excavating this volume of dirt, the size of disposal facility or transportation risk required, and the short-term disruption of stripping off the most productive layer of soil in this agricultural and residential area, the Task Force adopted an off-site risk level of 10-4, albeit using a very conservative exposure model. On-site, the amount of soil to be removed was driven less by direct exposure to the soil than by the need to protect the aquifer. Noncarcinogenic hazards were also taken into account, and a middle-range risk level was chosen.

The Task Force has not yet determined the specific use or uses for which the site is best suited. It must first fully evaluate the viability of on-site waste disposal and develop its own recommendations with regard to waste disposition before coming to detailed conclusions regarding land uses. The proximity to a long-term disposal facility and the Task Force's desire for a margin of safety make it unlikely that the Task Force would recommend uses that allow for intensive activities at a high level of exposure.

CONCLUSION

In these recommendations, and in the recommendations to come on waste disposition and priorities, the Task Force has tried hard to consider the values it developed, as well as all of the consequences of remediation decisions. It has informed itself, answered the questions put to it, and done so in a timely fashion. The members' willingness to put preconceived ideas aside and to remain flexible resulted in unanimous recommendations that are well within DOE's power to implement at a local level.

More broadly, the Task Force's deliberations have formed the basis of a relationship of trust among DOE, the regulators, and their stakeholders. A working relationship has developed not only between top management and the public, but also with the individuals who are doing the work to provide information for these decisions. The result is greater confidence in the remediation decisions that are being made and the people making them. By establishing a forum for informed, open, and easy communication among key stakeholders, DOE, and regulators, the Task Force allows decisions to be reached as a developing consensus, rather than as adversarial presentation and response. Moreover, the Task Force appears to be an excellent way to have public involvement during decision making post-Records of Decision at CERCLA sites; CERCLA and its implementing guidance provide only minimal mandated public involvement activities. The habits of cooperation and openness formed as part of the Task Force process are already in place and can continue easily. In the long run, this may be the most valuable product of the SSAB process.

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PUBLIC PARTICIPATION AT FERNALD: FERMCO'S EVOLVING ROLE

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ABSTRACT

The Fernald Environmental Management Project (FEMP) is a DOE facility near Fernald, Ohio, which operated from 1952 to 1989 to support United States defense programs. The FEMP occupies about 1050 acres, and is located about 18 miles northwest of downtown Cincinnati. The primary mission of the FEMP was to process, refine, and machine high-grade natural uranium ores into high-purity uranium metal. The storage and disposal of large quantities of wastes at the site and the potential for impacting human health led to the site being placed on the National Priorities (Superfund) List. Over 3 million cubic yards of contaminated material and a significant plume of uranium-contaminated groundwater must be managed at Fernald. In an effort to improve public involvement in the site restoration decision making process, the DOE has established site specific advisory boards, of which the Fernald Citizens Task Force is one. The Fernald Task Force is focused on making recommendations in four areas: 1) What should be the future use of the site? 2) Determinations of cleanup levels (how clean is clean?) 3) Where should the wastes be disposed of? 4) What should be the cleanup priorities? Because these questions are being asked very early in the decision-making process, the answers are necessarily qualified, and are based on a combination of preliminary data, assumptions, and professional judgement. The requirement to make progress in the absence of accurate data has necessitated FERMCO and the Task Force to employ an approach similar to sensitivity analysis, in which a range of possible data values are evaluated and the relative importance of the various factors is assessed.

Because of its charter to provide recommendations of future site use, the Task Force has developed a sitewide perspective, compared to the more common operable unit specific focus of public participation under CERCLA. The sitewide focus and accelerated schedule for public involvement have resulted in the exploration of technical project management issues far in advance of their traditional project sequence. Due to the early consideration of key issues, coupled with the sitewide perspective and sensitivity analysis, there has been a rapid progression from the esoteric realm of the CERCLA process to the early identification of the important decisions to be made and their specific consequences.

The relationship between FERMCO and the Task Force is evolving toward one of partnership with DOE in managing the obstacles and hidden opportunities for success. The Task Force likely will continue to participate in the Fernald project long after its initial recommendations have been made. DOE already has made the commitment that the process of public participation will extend into the Remedial Design phase. There is substantial reason for optimism that continuing the Task Force process through the design phase will assist in developing the appropriate balance of cost and engineered protectiveness.

INTRODUCTION

This paper presents observations and preliminary conclusions on the unique and still evolving relationship between the Fernald Citizens Task Force and FERMCO, the U.S. Department of Energy's (DOE's) environmental restoration contractor at Fernald,

Ohio. After a brief introduction to the site, specific examples are discussed of how the expanded public participation process at Fernald works and influences decision making.

THE PROJECT SETTING AND BACKGROUND

The Fernald Environmental Management Project (FEMP) is a DOE facility near Fernald, Ohio, which operated from 1952 to 1989 providing uranium metal products to support United States defense programs. In 1989 the mission of the facility was changed to one of environmental restoration. Also in 1989, the facility was placed on the National Priorities List ("Superfund List"). The FEMP is the subject of an Amended Consent Agreement between DOE and the U.S. Environmental Protection Agency (EPA), under authority of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), and a Resource Conservation and Recovery Act (RCRA) Consent Decree from the Ohio Environmental Protection Agency (OEPA).

The FEMP occupies about 1050 acres, and is located about 18 miles northwest of downtown Cincinnati. The primary mission of the FEMP was to process, refine, and machine high-grade natural uranium ores into high-purity uranium metal. The metal was shipped to other DOE or U.S. Department of Defense facilities for use as "feed materials" in the nuclear weapons program. These uranium production activities generated large quantities of waste materials. The storage and disposal of wastes at the site and their potential for impacting human health led to the site being placed on the National Priorities List.

The CERCLA Remedial Investigation/Feasibility Study (RI/FS) process has found that at least 3 million cubic yards of contaminated material must be managed at the site, and that groundwater has been contaminated. The Fernald site is located above the Great Miami Aquifer, a sole source drinking water aquifer. Off-property drinking water wells in the area have been contaminated with uranium.

The CERCLA process has defined five operable units at the Fernald site, each with its own schedule for completion of the RI, FS, and Record of Decision (ROD), per the Amended Consent Agreement. Operable Unit 1 is comprised of the Waste Pit Area, which includes about 700,000 cubic yards of contaminated material, including process wastes. Operable Unit 2 is comprised of a landfill, two surface impoundments, a flyash pile, and two uncontrolled disposal areas. Operable Unit 2 includes about 300,000 cubic yards of contaminated material. Operable Unit 3 is comprised of the above-grade structures at the site, which are about 200 in number and range from large, former ore processing and manufacturing facilities to small outbuildings. Operable Unit 4 is comprised of four waste silos, which contain about 15,000 cubic yards of waste process sludges. Operable Unit 5 is comprised of the site media, which include contaminated groundwater, sediment, and approximately 2 million cubic yards of contaminated soil.

CERCLA AND THE FERNALD CITIZENS TASK FORCE

Under the CERCLA Amended Consent Agreement, the decisions on the cleanup of the five operable units are made individually and sequentially, with each operable unit's ROD following the previous one by a few months. This has necessitated that much of the RI/FS work for all operable units be done concurrently. In order for the individual operable unit studies and decisions to be consistent and protective of human health and the environment, a common set of assumptions regarding the future use of the site was needed. For example, will the cleanup need to be protective of future resident farming families or will people be excluded from the property? In addition to the obvious effects of alternate future site uses on contaminant cleanup levels, there are also impacts on the evaluation of remediation technologies. In-situ containment and certain engineered disposal facilities, to cite two examples, may be incompatible with some desired future land uses.

The CERCLA process provides for public involvement in remedial decision making. Under CERCLA, there are specific time periods set aside for public comment on the remediation plans that have been developed previously. Although all comments are considered, the fact that the comment period comes late in the decision making process limits their range and impact. In an effort to improve public involvement in the decision making process, the DOE has established site specific advisory boards, of which the Fernald Citizens Task Force is one. The Task Force is focused on making recommendations in four areas that are important to remedial decision-making for each operable unit:

1) What should be the future use of the site?
2) Determinations of cleanup levels (how clean is clean?)

3) Where should radioactive and hazardous wastes which are generated as a result of site restoration activities be disposed of?

4) What should be the cleanup priorities?

To be truly effective in assisting the decision-making process, the Task Force must make these recommendations relatively early in the CERCLA process. Therefore, the Fernald Citizens Task Force becomes involved much earlier than in the traditional public comment period under CERCLA, and hence needs to obtain information from DOE and more specifically DOE's contractor, FERMCO, far in advance of the customary public comment period.

FERMCO AND THE FERNALD CITIZENS TASK FORCE

A traditional relationship between the public and a government contractor usually is constrained by relatively formal processes. However, the Department of Energy has directed FERMCO to cooperate fully with the Task Force and that Task Force needs should be given immediate priority. This, combined with the Task Force's broad charter and accelerated schedule, has resulted in a relationship between the public and contractor that is unusually direct and comprehensive in scope.

One of the initial functional departures from the traditional relationship was the need for FERMCO to provide the Task Force with preliminary data, estimates, and conclusions, whereas in a traditional relationship only final, carefully reviewed information would be released. For example, FERMCO was requested to provide its best guesses at final site cleanup levels and waste material volumes far in advance of our having the data to support definitive calculations. By responding with best, good faith efforts and honestly revealing the inaccuracies and uncertainties associated with the information being supplied, FERMCO has helped the Task Force focus on key site management issues, and equally importantly, gained a significant measure of mutual goodwill and trust.

The next phase of interactions went beyond the exchange of information into an exploration of indirect consequences of seemingly straightforward management decisions. One prominent example is the proposed on-property waste disposal facility. As discussed above, the cleanup of Fernald will result in the need to dispose of large quantities of contaminated materials. One alternative being considered for the less-contaminated fraction of material is construction of an on-site disposal facility. In order to carefully consider this alternative, the Task Force needed to know some basic facts. Why is on-site disposal needed? How large would the facility be? Where would it be located and what would it look like? How long would it be necessary and how long would it last?

Because these questions were being asked very early in the decision-making process, the answers were necessarily qualified, and were based on a combination of preliminary data, assumptions, and professional judgment. A good example is the question of the size of the potential on-property disposal facility. The size of the facility is a function of the quantity of contaminated material that meets the waste acceptance criteria for the disposal facility and the contaminant cleanup levels for the site. The necessary data normally are obtained through the CERCLA process: data on the nature and extent of contamination are developed through the Remedial Investigation and the cleanup levels are developed through the Feasibility Study and are finalized in the Record of Decision. However, at the time the Task Force needed the information, the relevant Remedial Investigation Reports, although on schedule, were not near completion. Feasibility Study and Record of Decision information was months to years away.

In order to make progress in the absence of accurate data, the Task Force, DOE, and FERMCO adopted an approach similar to sensitivity analysis. In the case of disposal facility size, the possible minimum and maximum sizes (material volumes) as well as the most probable size range were estimated from available data and the key assumptions for each estimate were documented. Naturally, the relatively high uncertainty led to a wide range of volume estimates. This very preliminary range was from 1.5 - 8.5 million cubic yards with a most probable volume of about 2.5 million cubic yards. The critical variable was found to be the cleanup levels (and hence volume) for the Operable Unit 5 soils in potentially contaminated, off-property locations.

The early estimation of a range of contaminated material volumes for potential on-property disposal enabled an evaluation of the required site area that would be dedicated for the facility within the context of the site's future use. The relative site space requirements or "footprints" for disposal facilities of 2.5 million and

8.5 million cubic yards are shown in Fig. 1.
The Task Force's consideration of possible disposal facility locations at Fernald was interesting and indicative of the technical planning issues on complex remediation projects. One of the insights gained from the earliest Task Force activities, especially through the use of the FutureSite exercise (Applegate 1995), was that the public, and in particular those persons living near the site, were especially concerned about project activities and future site use commitments that would occur near the site borders, compared to the interior of the site.
Concern about the relatively undisturbed border areas of the site led to the initial and informal public perspective that the preferred location for a potential disposal facility would be in the central area of the site. Given that a substantial portion of the property's interior already had been developed as the former process area, this location was suggested to FERMCO as an area that should be carefully evaluated for the disposal site location.
Clearly, a suitable hydrogeologic setting is essential for locating a disposal facility, and Ohio EPA has set appropriate technical requirements. However, if a large enough area of suitable hydrogeology were available, then DOE and FERMCO would have some flexibility in selecting the specific footprint to accommodate the preferences of the public. Regardless of hydrogeological conditions, other important factors associated with different potential locations could be evaluated.
Two of the potential locations for a typical 2.5 million cubic yard disposal facility are shown in Fig. 2. In the figure, Location A covers mostly unused land and is within 300 feet of the eastern property border and Location B is within the former process area, over 1,500 feet from any property border. At the integrated project level (as opposed to the operable unit level) the requirements for the sequencing of activities are very different for the two locations and have very different cost implications. For example, disposal facility construction is not physically constrained at Location A, whereas at Location B disposal facility construction cannot be implemented until the existing buildings are safely shutdown, decontaminated and demolished, and the contaminated foundations and soil beneath and around them are excavated. This would require the interim storage of any building debris to be disposed of on-site, as well as the stockpiling of the contaminated soil from the former process area and other site areas that would be remediated prior to the availability of the disposal facility.
The cost implications of the differences in project sequencing are significant. Additional costs are required for Location B due to double handling of the building debris; double handling of the process area soils and other site soils; construction of appropriate interim storage facilities for the soil and debris; monitoring and environmental control of the stored, contaminated material; and demolishing and disposing of the interim storage facilities. Given that over one million cubic yards of soil and debris would be stored and double-handled, the additional cost of locating the disposal facility in the central portion of the site would be on the order of $50-100 million. The Task Force will use this information in reaching its recommendations on future land use and waste disposition, which are due in the summer of 1995.

INITIAL CONCLUSIONS

Although the ultimate assessment of the utility of site specific advisory boards is years away, it is possible to draw initial observations and conclusions based on the experience at Fernald. The Fernald Citizens Task Force is providing input from the public to the decision making process far in advance of the traditional public participation process. Because of its charter to provide recommendations of future site use, the Task Force has developed a sitewide perspective, instead of the more common operable unit specific focus of public participation under CERCLA. The sitewide focus and accelerated schedule for public involvement have resulted in the Task Force's exploration of technical project management issues far in advance of the availability of accurate data.
The requirement to make progress in the absence of accurate data has necessitated an approach similar to sensitivity analysis, in which a range of possible data values are evaluated and the relative importance of the various factors is assessed. For example, in predicting the size of a potential on-site disposal facility it was determined that the key factor would be the final cleanup levels for the soils in Operable Unit 5, especially for the off-property areas that are under study. Another way of looking at this particular case is that the sitewide long-term

impacts of different cleanup levels for Operable Unit 5 were made tangible to the public at an early stage, and that FERMCO and DOE project managers obtained highly useful information months ahead of the time it would normally be developed. Similarly, in the case of evaluating different locations for an on-property disposal facility, the public and project planners were able to assess sitewide interactions and interferences far in advance of their traditional sequencing. In both cases, there was rapid progression from the esoteric realm of the CERCLA process to the early identification of the important decisions to be made and their specific consequences.

The Fernald Citizens Task Force is considering many more issues than the few that are discussed here, and to make informed recommendations the Task Force needs site data and cost and schedule information pertinent to each issue. Particularly helpful in this regard has been the use of state-of-the-art Geographical Information System (GIS) and three-dimensional, color imaging software to realistically depict possible future site conditions such as changes in site topography due to excavation of contaminants, or to simulate the appearance of a disposal facility from a public road. To quickly evaluate the cost and schedule impacts of several alternatives, an integrated site master schedule and cost model are essential.

The Task Force, DOE, and FERMCO must continue to be willing and able to develop, qualify, communicate, and accept the limitations of an information base that consists of preliminary data, assumptions, and professional judgment. One problem that has emerged as the result of doing so much, so fast, based on assumptions is that teams working on different specific issues may make different assumptions about the same site condition or assume different technical approaches for similar remediation tasks. These inconsistencies are best identified and corrected through regular integrated project planning meetings that result in a single set of documented management guidance for project personnel.

The relationship between FERMCO and the Task Force is evolving toward one of partnership with DOE in managing the obstacles and hidden opportunities for success. The enhanced public participation through the Fernald Citizens Task Force has led to an accelerated focus on sitewide technical and management issues and, through a project-level sensitivity analysis, has identified major opportunities to be managed. The Task Force likely will continue to participate in the Fernald project long after its initial recommendations have been made. DOE already has made the commitment that the process of public participation will extend into the Remedial Design phase at Fernald. There is substantial reason for optimism that continuing the Task Force process through the design phase will assist in developing the appropriate balance of cost and engineered protectiveness.

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THE FAST TRACK REMEDIATION DESIGN/BID PACKAGE

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ABSTRACT

The Fernald Environmental Management Project is a Department Of Energy (DOE) facility near Cincinnati, Ohio which provided high purity uranium metal products to support United States defense programs. Production operations were halted in 1989 to focus available resources on environmental restoration activities at the facility. Operable Unit 3 (OU3) is the designation given to the production area and production-associated facilities and equipment, including, but not limited to , all above and below ground structures, equipment, and utilities.

In late spring of 1994, two decisions were made that established the long range strategy on how remedial designs (RD) and the bidding of remedial actions (RA) are

approached for the decontamination and decommissioning (D&D) of the Fernald facilities. The first was to fast track the remediation design and bidding process for the first three D&D packages; and the second was to use standard performance specifications to streamline the decontamination and decommissioning process. This paper describes these strategies, identifies the key elements involved, and discusses the lessons learned that were associated with the approaches. A brief synopsis of these elements is as follows:

A facility complex which groups facilities into blocks of work was used for the bid packages.

A task force approach involving the required functional organizations was used to fast track the design and bidding process.

Standard Performance Specifications have been developed for each task to establish minimum acceptable criteria and yet provide the subcontractor flexibility to be creative within these boundaries. This approach also dramatically decreases costs for future D&D packages.

A key feature of value engineering has been the use of construction methods and techniques to drive the D&D cost significantly lower.

The bid package was structured around a modified IFB approach. This provides the successful bidder the opportunity to incorporate innovative ideas within the performance specification guidelines. In addition, several innovative concepts such as prequalifications of bidders, preparation of a task based bid package, and use of video and still shots were used to create a comprehensive bid package.

The D&D approach was structured to decrease safety risks by emphasizing mechanical methods over manual methods.

This paper will present the lessons learned in these strategies and illustrate how the various concepts described above were translated into a four month fast track design bid process, how the overall project costs were significantly reduced, and how safety risks were significantly mitigated.

INTRODUCTION

In Spring 1994, an opportunity was seized to establish the long range strategy on how remedial designs (RD) and the bidding of remedial actions (RA) are approached for the decontamination and decommissioning (D&D) of the Fernald facilities. Several key concepts contributed directly to this opportunity:

First, the CERCLA process provides for an Interim Record of Decision (IROD) in situations where RA can be performed that mitigates potential threat to public health or the environment, yet doesn't prevent regulatory and public participation in the ultimate remedy selection. Such is the case with building demolition at Fernald. Second, it was observed that the typical bidding process was too long, and that significant schedule gains could be made by fast-tracking the design and bid process. Lastly, it was concluded that we could successfully employ standard performance specifications to streamline the overall engineering design process, instead of specific procedures for each facility.

This paper describes these approaches, identifies the key elements, and discusses the lessons learned that were associated with these efforts.

FACILITY COMPLEX APPROACH

Initially various methodologies were created to prioritize the sequence of D&D of the 232 structures/facilities at Fernald. These included ranking and weighing of each facility with factors such as contamination levels, use, location, D&D difficulty, etc. However, these approaches ultimately resulted in using a simple approach based on three criteria: 1) Group the facilities into blocks of adjacent facilities which have obvious geographic boundaries. 2) Schedule those blocks that are available first. 3) Schedule the blocks to achieve the early start of a sub-surface cleanup taking into account site drainage and underground utility feeds. This resulted in the grouping of above-ground facilities into approximately 24 (3 underground) project packages called complexes.

COMPLEX DESCRIPTIONS

The initial complexes included in this approach are the Pilot Plant (12 Structures), Plant 4 (1 Structure) and Plant 1 (11 structures). The Pilot Plant has two large concrete framed structures and 10 steel framed structures. Plant 4 is a very large steel framed structure with asbestos transite panels. Plant 1 is composed of a large steel framed structure with asbestos transite panels, 1 masonry block structure and 9 smaller structures. Each of the complexes requires remediation activities for dismantlement of pipe, ductwork, electrical removal (1400 components); interior and

exterior transite siding and exterior features (doors/windows): structural steel/concrete framing removal including interior/exterior concrete slabs and walls and masonry walls and built up roofing systems.

TASK FORCE APPROACH

To accomplish a fast track design and bid process, a remediation task force was established to ensure that engineering, design, construction, safety, quality, regulatory, waste management, environmental, procurement, DOE input, schedule and budget aspects were considered in each remediation design package. The team was commissioned to participate in all aspects of the project, including but not limited to the inception of the design, Certification for Construction, contract award, execution of the remedial action, completion of the contract and the final verification report.

DEC (Design, Engineering, Construction) Team

Members of the CERCLA RCRA Unit 3 staff participated using representatives from functional groups to formulate an aggressive design package which included scope definition of the engineering, construction, and waste disposition aspects of a specific project. A work plan of construction tasks was prepared and brainstorm sessions including lessons learned from past and ongoing projects were incorporated into the approach to the work. Value engineering and constructability studies were performed and shared with the design engineer. Particular emphasis was placed on performing the remediation work in a safe, quickest and least cost method.

Fast Track Schedule

The work plan formed a basis for design specifications; a design/construction/waste management preliminary schedule; a task based safety matrix; waste stream characterization and documentation; quality plan and labor determination. A fast track schedule for the development of the Certified for Construction Package (CFC) was determined to be approximately ten (10) weeks. This period is combined with an additional two weeks for final preparation of the bid package; a four week bid period, a three week bid evaluation period and a two week DOE bid evaluation period. These are the components of the fast track schedule to award a contract for the remediation of the complexes in five to six months.

Integration of Environmental and Permitting Requirements

As the work plan described above was prepared, representatives from environmental, regulatory and safety and health divisions provided comments in parallel to the development of the design specifications. This ensured that regulatory requirements for construction plans were incorporated during the design process.

USE OF STANDARD PERFORMANCE SPECIFICATIONS

Generic performance, procurement specifications and project scope drawings were created to form a Certified for Construction design package to be issued for bids. The task force developed work planning packages based on the D&D work tasks. The design engineer then used these work planning packages to create specifications related to specific Decontamination and Dismantling construction tasks. Major standard specifications included dismantling actions such as asbestos, asbestos transite, piping, electrical components, ductwork, process equipment, concrete, and structural steel. Other specifications included methods and special requirements for rigging, mobilization, decontamination, site preparation, demobilization and waste handling.

A key decision to the fast track remediation design process was to develop a set of standard performance specifications. The foundation for standard performance specifications is built around two basic concepts:

1) Establish the minimum acceptable boundaries (performance standards) in which a task is to be accomplished.

2) Allow the performing subcontractor the flexibility to exercise creativity within these boundaries.

The objective is accomplished by breaking the project into discrete, homogeneous tasks. For each task category, a set of criteria is set forth in the form of standard performance specification that establishes the baseline requirements for accomplishing the respective category of work. The specification sets the requirements to accomplish the task in a manner that protects the individual and the environment, and also meets the site operating criteria. An example of the former would be dust control methods for removing concrete, and of the latter would be size reduction requirements for steel and associated appenditures to meet packaging criteria for shipping boxes.

The use of a standard specification eliminates the requirements to redesign the specification for each facility. For example, once a specification is developed for removing process piping, transite siding, and structural steel, it applies wherever that task is performed. A specification for a facility demolition then only requires the assembly of those specs that apply and the preparation of specifications for which a standard specification does not exist.

VALUE ENGINEERING AND D&D CONSTRUCTION METHODS

A key aspect of value engineering for D&D remediation projects is in the use of cost saving construction methods. There is a fundamental difference in the approach to design of a new facility versus the design for D&D of the same facility. New design requires specific calculation of loads, flows, stresses, etc. to establish size and configuration of equipment and facilities. This process results in a detailed set of drawings and specifications to describe these requirements. There is almost always no description of the construction methods that will be used to build the facility. However, in de-construction, the process is almost totally based on the construction methods that will be used to dismantle the equipment and the facility. The emphasis both from a safety and from a cost standpoint needs to focus on the construction methods and equipment to be used. Construction experts need to be involved in the early phase of the remedial design process to bring in the construction expertise that is required for a safe, successful, and low cost project.

BID APPROACH

Prequalification

In parallel to the design specification activities, prequalification of bidders was conducted. This consisted of preparing a Notice of Prequalification published in the Commerce Business Daily which included the key criteria to evaluate bidders in light of the type of work included in the three complexes to be bid. These criteria included performing government work (experience with government contracts); demolition experience (asbestos and regular); rigging experience; radiation experience (worker safety issues); safety experience (EMR rating in the industry) and verifiable proof of self performing 45% of the past qualified work. Prequalifications were evaluated and bidders were notified accordingly.

Modified Invitation for Bid (IFB)

A modified IFB consisted of the preparation of a bid package using performance specifications. The IFB package consists of commercial bid and award terms, general terms and conditions, special terms and conditions, the project agreement, the statement of work (including the work scope, performance criteria), drawings, specifications, photographs, videos, safety program (including the safety matrix), and the quality plan criteria. The successful bidder is required to incorporate innovative ideas within the guidelines of the performance specification to form a bid. While the award is made to the lowest, responsible, responsive bidder, the bidder must meet all the criteria specified in the performance specifications. After the award, the successful subcontractor submits detailed work plans for approval which incorporate the contractor's unique approach to meeting the performance guidelines.

Task Based

The entire construction process is built around discrete tasks of like work (i.e., ventilation, duct dismantling, transite siding removal, process equipment removal). The bid package as well as the management systems used to support the work are all structured to support the task concept. As was discussed previously, a standard performance specification is developed for each discrete task category. The estimate and schedule are developed to uniquely depict each of these tasks. This does not preclude greater detail than at the task level; however, the detail must uniquely roll up into the specific task. For example, the task may be transite siding removal, and the contractor may prepare a more detailed schedule that shows siding removal on all four sides of the building. The contractor's individual pay items are also established at the task level. In addition, safety and quality matrixes are developed at the task level to specify the safety and quality requirements that are required for the performance of each task.

Video and Picture Book

The use of videos and picture books greatly facilitate fast track remediation bidding. Typically, bid packages have reference drawings that the bidders use for a bid take off. The cost involvement in up dating these reference drawings to present as-built conditions is costly and time consuming. A detailed videotape which shows

inside and outside of the structures, miscellaneous outside structures, utilities, side photography, and construction boundaries is sent out with each bid package. A photo book showing a panoramic overview of each floor of the structure and outside of the structure is also sent out with each bid package. The use of videos and a picture book does the following: 1) reduces cost of engineering and drafting; 2) reduces bidder clarification questions; 3) reduces number of tour visits by the bidders; and 4) is used as a tool on claims and changes.

Bottoms Up Estimating

During the work plan development in the design phase, the estimate is based on D&D quantities and the associated construction methods, techniques and materials. This type of D&D work is approximately 70-80% craft labor; therefore, the duration to perform the tasks is based on the methods and manpower required to perform the tasks. Materials, construction equipment, indirect costs, overhead and profit were estimated based on a task based preliminary schedule. This schedule formed the basis of the contractor pay items.

By focusing all the management systems on the specific tasks, the bid package establishes the framework for success during the actual de-construction phase. The probability of success on a project is significantly enhanced if it is well planned and the management systems are "pulling in the same direction" as the work to be performed.

Finally, prior to the performance of a specific task, the subcontractor is required to submit a detailed work plan which embodies how the work is to be performed and how safety and quality issues are to be addressed.

EMPHASIS ON SAFETY APPROACH

Mechanical vs. Manual

D&D construction activities have one of the highest recordable incident rates in any industry. In order to cope with this reality, owners, construction managers, and engineering must find ways to reduce this liability and still get the job done. In reviewing the type of accident that occurs during D&D activities, most are caused by material and equipment removal and handling. Some accidents which occur during the activities include: slips, falls, pinch points, scrapes, bruises, burns, etc. These are all inherent with manual D&D tasks. A good safety program along with a good work plan reduces these dangers.

By emphasizing mechanical material handling methods, further reductions are achievable. Some mechanical methods that can reduce the manual labor in D&D tasks are:

1. Mechanically tripping of the structure can be accomplished by making strategic cuts on the structure, connecting cables to it and pulling the structure down using heavy equipment.
2. Shaped charges can also be utilized to use gravity to trip the structure. Shaped charges strategically placed on structural columns, beams, X-bracing, etc. can remove selective structural members so that the building will fall to the ground in a safe, controlled manner.
3. Large mechanical shears mounted on large track hoes can cut structural steel in almost any size and configuration vs. manual torch cutting.
4. Inside the structure smaller mechanical shears can remove piping, ducting, electrical conduit, etc.
5. Small "bob-cats" outfitted with grapples can eliminate manual handling and loading of debris.

By utilizing mechanical construction methods in removal and handling activities, the project cost and safety requirements are greatly enhanced.

Multiple Work Area - Fast Track Work

Radiological D&D work must be accomplished in a safe, controlled manner. The outer shell (siding and roofing) is used for primary containment. The sequencing of work tasks is somewhat straight forward: safe shutdown of the structure (utility disconnect, removal of holdup/hazardous waste material, etc.), asbestos abatement, construction debris removal (piping, equipment, conduit, ducting, etc.), interior wall panels, final decontamination of the outer shell (transite paneling, brick, block, steel sheeting), and finally structural steel.

Some of these activities can be accomplished concurrently to expedite the schedule. On multi-story structures, asbestos abatement can proceed on one floor while construction debris removal is ongoing on another floor. On a one story structure, this asbestos abatement work area can be sealed off and other D&D work performed on

the same floor as long as two different entrances and exits are established. The same goes with debris removal in one area while removing interior transite wall panels in another. It is important that work plans are comprehensive, integrated into the construction D&D schedule, reviewed by appropriate personnel, and that the work plan is followed. This allows simultaneous performance of multiple work tasks in a safe, productive manner.

LESSONS LEARNED

Keep it Simple

D&D work is not highly technical work. Use of basic demolition techniques with appropriate environmental controls will enhance wherever possible. Our experience shows that elaborate approaches have been unsuccessful, are costly, and create increased safety risk for the worker.

Involve Construction from the Beginning

Early involvement of Construction input will result in an integrated approach to a final product which meets the DOE, FERMCO (contractor) and stakeholders' requirements.

Involve Outside Expertise

When developing the scope of work for a D&D package, all construction methods must be evaluated, and the best one chosen. Even though you have a qualified construction staff, it is sometimes necessary to bring in outside expertise on a specific field (e.g., use of shape charges). This expert can help evaluate specific construction methods, review work plans, review specifications, evaluate safety concerns, and specify dust control measures. By bringing in outside expertise early on in the planning stage, you can evaluate each D&D method and select the appropriate technical approaches for the bid documents.

Key is Construction Techniques

The key to de-construction lies in the construction techniques. While a new facility requires great emphasis on analysis of loads, process flows, process control, etc., the emphasis for de-construction is on construction equipment and methods. Therefore, it is important to have construction expertise play a key role from the beginning of the remedial design process.

Bid Multiple Facilities

Grouping individual buildings into complexes by geographic area is an efficient and low cost way of performing D&D work. This is based on the following factors:

1) From a logistics standpoint, peripheral buildings and facilities must be removed prior to the removal of the main building.

2) Many main facilities are structurally connected, which requires a specific dismantling sequence.

3) The design and bidding costs for small, medium, and large contaminated structures is almost the same. By grouping facilities, the bid and design costs are significantly reduced.

4) Management and support costs are significantly lower for a complex of facilities than for the same set of facilities sequenced for D&D on an individual basis.

SUMMARY

Accelerating decommissioning projects can mean different things to different people. In the context of our demonstrated successful work at Fernald, accelerating decommissioning work means delivering projects to our clients safer, faster and for lower cost.

Our approach has yielded the following key results:

Improved safety performance. Employing this process on our recently completed Plant 7 dismantling project resulted in zero lost-time accidents, as compared to a nationwide incident rate for the construction industry of 6.5 lost time accidents per 200,000 hours worked;

The task force approach has yielded a structured design, engineering, and construction (DEC) team concept that has been formalized at Fernald by a standard operating procedure. The involvement of on-site internal stakeholder participation in the planning process greatly facilitates a smooth project flow;

Fast-track schedules can be achieved with this approach within the CERCLA and DOE arenas. Shorter project durations result in lower overall project costs. Our recently completed Plant 7 project was completed approximately $5M and 1 year ahead of its baseline estimate;

The specific task, performance specification approach yields improved change control. Subcontract change orders for the Plant 7 project (pending resolution) are

near 5% of the subcontract value, a goal of the project at the outset. Accelerating decommissioning activities can be achieved. To date, our experience with this approach leads to the conclusion that applying these identified concepts can result in a win-win situation for all.

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FERNALD WASTE MANAGEMENT AND DISPOSITION

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ABSTRACT

Historically waste management within the Department of Energy complex has evolved around the operating principle of packaging waste generated and storing until a later date. In many cases wastes were delivered to onsite waste management organizations with little or no traceability to origin of generation. Sites then stored their waste for later disposition offsite or onsite burial. While the wastes were stored, sites incurred additional labor costs for maintaining, inspecting and repackaging containers and capital costs for storage warehouses. Increased costs, combined with the inherent safety hazards associated with storage of hazardous material make these practices less attractive.

This paper will describe the methods used at the Department of Energy's Fernald site by the Waste Programs Management Division to integrate with other site divisions to plan in situ waste characterization prior to removal. This information was utilized to evaluate and select disposal options and then to package and ship removed wastes without storage.

INTRODUCTION/FERNALD HISTORY

The Fernald site is located approximately 20 miles northwest of Cincinnati and has been managed by Fernald Environmental Restoration Management Corporation (FERMCO) since December, 1992. For more than 37 years, the facility manufactured uranium metal products for use at other DOE sites to satisfy Defense Program demands. Production operations were suspended on July 10, 1989 due to a sharp reduction in the demand for uranium metal products by user sites and continuing problems in achieving full regulatory compliance. Following necessary Congressional notifications, the facility was formally shut down on June 19, 1991 when its mission became environmental restoration.

The primary law regulating cleanup of the Fernald site is the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) as amended by the Superfund Amendments and Reauthorization Act (SARA) in 1986. Another major regulation, the Resource Conservation and Recovery Act of 1976 (RCRA), governs the generation, transportation, treatment, and disposal of the hazardous waste at the site. A Consent Decree was signed by the State of Ohio and the DOE in December, 1988, establishing milestones to bring the FEMP into full compliance with RCRA and other regulatory requirements. Amendments establishing additional milestones regarding the management of hazardous waste were later proposed to the Consent Decree, and in January 1993, the amendments were approved, resulting in the Consent Decree and its Stipulated Amendments (SACD).

In July 1990, the United States Environmental Protection Agency (USEPA) and the DOE entered into a Consent Agreement establishing milestone schedules for the completion of necessary studies to support the CERCLA clean-up process. The agreement established schedules for implementing near term clean-up actions (Removal Actions) while final clean-up solutions were being evaluated and selected. In September 1991, the DOE and USEPA jointly signed the Amended Consent Agreement establishing revised milestones for the completion of the required studies and identifying a series of additional near-term actions for implementation by the DOE.

The overall mission of the FEMP is to provide the safe, least-cost, earliest, final cleanup of the Fernald site, within applicable DOE orders, regulations and commitments in a manner which addresses stakeholder concerns. To support this objective, the FEMP is currently conducting Remedial Investigation/Feasibility Studies, remedial design activities, removal actions, waste management operations, and other compliance and clean-up initiatives.

This paper discusses the reorganization of the Waste Programs Management Division of FERMCO and its integration with Removal Action #19 - Plant 7 Dismantling.

WASTE PROGRAMS MANAGEMENT REORGANIZATION

Historically, waste management within the Department of Energy complex has evolved around the operating principle of packaging waste and storing until a later date. In many cases wastes were delivered to onsite waste management organizations with little or no traceability to origin of generation. Sites then stored the wastes for later disposition offsite or for onsite burial. While the wastes were stored, sites incurred additional labor costs for maintaining, inspecting and repackaging containers and capital costs for storage warehouses. Increased costs, combined with the inherent safety hazards associated with storage of hazardous material make these practices less attractive.

In FY94, FERMCO organized its waste management factions, at the time spread across several organizations, into one Division that encompassed both professional and labor forces. The reorganization included the formation of 10 Departments. The Waste Programs Management organizational chart (with functional areas) is included as Fig. 1. The reorganization was a result of the need to bring together key personnel to improve the methods in which wastes were managed. The FEMP was realizing the problems stated above, maybe not to the degree of other sites, but the site definitely had a problem of deteriorating containerized "legacy wastes" monopolizing valuable storage space. Wastes were being introduced into the waste management system with little or no warning and identified or characterized only a portion of the time. This taxed resources (personnel and equipment) to the point where containers eventually had to be repackaged due to the waste not being characterized and deteriorating conditions of the containers.

A strategy was implemented as to how information flows within the WPM organization. Figure 2 displays the flow of information and the four distinct areas of activities: integration, planning, operations, and disposition. A central group within the WPM Division is responsible for integrating with other Divisions to assist in planning what WPM services will be needed for projects when wastes are generated. This ensures that information is passed on to the appropriate support organizations within the Division so work can be planned and budgeted effectively. This setup has allowed for fewer surprises (i.e., containers of uncharacterized waste) being introduced into the WPM system, resulting in a more efficiently run organization. The strategy not only saves time and money from a WPM perspective, but also for the project.

FY94 ACCOMPLISHMENTS

Major accomplishments were realized as a result of the reorganization of WPM into a Division. These accomplishments were tied to award fee milestones that required they be implemented by dates agreed upon by FERMCO and DOE. If the milestones were met FERMCO realized a certain percentage of fee, or award. Missing a milestone meant that FERMCO would not only lose the opportunity to gain fee, but could also potentially lose money if performance was deemed below acceptable. Therefore, the accomplishments were highly scrutinized for compliance by DOE and FERMCO. Some of the major accomplishments include:

Life Cycle Cost Analysis Program

Waste Minimization and Pollution Prevention (WMPP) principles were introduced to the site as a Life Cycle Cost Analysis Program instituted by WPM. All site project engineers and managers received training developed by WPM. The main theme of the program was to incorporate waste minimization principles and practices into the planning phases of a project. This has proven successful in that several waste minimization opportunities were incorporated into Removal Action #19 (D&D) as well as other ongoing and new projects.

Clean Trash Program

An effort that was initiated in late FY93 as a pilot program was the Clean Trash Program. This program involved segregating trash from selected radiologically controlled office areas. One hundred percent of the trash was frisked by hand and released to a local landfill if no contamination was discovered. In FY94, this effort was expanded to include all office areas, restrooms, and breakrooms from the radiologically controlled areas (excluding contamination zones). The trash is collected and 10% is hand frisked for contamination. If no contamination is detected, the entire 100% is released for disposal at a local landfill. The volume of the radiologically controlled area trash being diverted from the contaminated

trash waste stream was increased from 0% to 49%, realizing a cost savings of $140,000 to date. This has proven to be a successful process and other sites are emulating the program.

Hazardous And Mixed Waste

The Hazardous and Mixed Waste Program was able to reduce the mixed waste inventory by 31,234 ft3. This was realized by shipping to Envirocare 3,552 ft3 of mixed debris waste; consolidating mixed debris waste for a reduction of 932.4 ft3; and properly recharacterizing 26,751 ft3 as low-level radioactive waste. These efforts significantly reduced the amount of mixed waste to be managed, thus reducing management costs such as inspections, repackaging, and facility (warehouse) costs.

Waste Tracking

Development of an integrated waste tracking database system that incorporated several ancillary databases that were being maintained onsite. The system provides a singular location for information related to tracking of waste from the point of generation to disposal (cradle to grave). The implementation of the system will allow for a reduction in data entry redundancies and ensure more complete and accurate information is available to assist in the day to day management of site wastes.

Waste Storage

Waste operations were improved from a perspective of sheltered container storage. The main storage pad at the site (Plant 1 Pad) was upgraded in order to consolidate a majority of the waste containers being stored. Plant 1 Pad operations were able to segregate, inspect, overpack and relocate 28,000 drums of waste in support of the upgrade. As a result of the upgrade, the amount of containers in unsheltered areas was reduced by 43%. By consolidating a majority of the site's waste containers to one pad a reduction in pad management costs is realized.

Waste Shipping

Low Level Waste shipping operations were able to ship 582,595 ft3 of low-level radioactive waste to Nevada Test Site during FY94. This represents 78% of LLW disposed of at NTS for the year. The weekly average amount of waste shipped equated to 11,000 ft3. The FEMP also received approval in FY94 to ship four additional waste streams.

INTEGRATION WITH REMOVAL ACTION #19 - PLANT 7 DISMANTLING

Background

Plant 7 was constructed in May, 1953 to house the processes involved in the reduction of uranium hexafluoride (UF 6) to uranium tetrafluoride (UF 4). The seven story "Hexafluoride Reduction Plant" was designed to an established production rate converting 12 tons of UF 6 to UF 4 per 24-hour period. Production operations were initiated in June, 1954 but the plant has been idle since 1956, when its production processes were halted.

In 1967 the UF 6 reduction process was declared obsolete, and the equipment was dismantled and removed along with the majority of the process piping. The ammonia separation process, two 75,000 cubic feet per minute blower units and associated ducting, and several motor control centers were abandoned in place. Plant 7 was then used to store drums of intermediate product (UF 4) on the first and second floors, and empty drums on the third, fourth, and sixth floors. The use of this building to store low-level radioactive materials, as well as the previous process operations, resulted in the corresponding presence of radiological contamination. The presence of asbestos containing material and bird droppings, which presented a biological hazard, were additional factors which were considered in recommending the removal of this building.

The dismantling of Plant 7 included: relocation of the drums and debris to an alternate storage area; decontamination (surface cleaning to remove radiological contamination); removal of asbestos containing material (ACM); dismantling the structure; segregation of materials; size reduction of materials; packaging and certification of wastes; placing waste packages into interim storage (staged for shipment); and off-site disposal of low-level radioactive waste (LLW).

Planning

To coordinate these activities, a Design, Engineering and Construction (DEC) Team was formed that consisted of several Divisions within FERMCO. Participation on this team by Waste Programs Management (WPM) was essential for all waste management related activities to be effectively coordinated. A formal group was formed within WPM to integrate and coordinate all WPM activities and participate on the DEC Teams.

This group is/was involved in the project from start to finish to maintain a consistent interface. The purpose of the DEC Team was to include each organization that would be involved in the project to ensure everyone knew what to expect as the project progressed. The functions of Waste Programs Management were: identify, characterize and provide volume estimates of the waste streams; provide and coordinate the delivery of waste containers; remove full waste containers from the project queuing area; place waste into interim storage (stage for shipment); store any hazardous waste; and disposition the LLW to NTS.
The first function of WPM was to identify and characterize all waste streams generated from the project prior to the actual dismantling of the building. Prior to D&D activities, random sampling points were chosen in Plant 7. The areas with the greatest potential of having RCRA hazardous waste were identified based on historical use and process knowledge. Samples were collected and submitted for Toxicity Characteristic Leaching Procedure (TCLP) metals analysis. The results of these tests were used to verify the lack or presence of RCRA hazardous waste. The following sample media were sampled for either hazardous waste determination or to determine the disposition options: decontamination waste/residues; steel and metals; miscellaneous equipment; concrete/debris; and transite. The only hazardous waste identified was the lead bolt caps used to fasten the transite to the structure and the lead flashing around the windows.Once the waste streams were characterized, volume estimates of each waste stream were calculated based on data from "as-built" drawings. This information is included in a site-wide waste forecasting database that is utilized to provide volume estimates of media from all structures onsite. This information, waste stream characterization and volume estimates, is compiled onto a Project Waste Indentification and Disposition (PWID) Form. This form is utilized by the project engineer to assist in the planning and budgeting process. For instance, by estimating the bulking factors of the different material types, the number of containers needed can be determined. Also, based on the characterization, the disposition options are indicated on the PWID form. Therefore, the PWID provides the project with a complete synopsis of the types and amounts of wastes to be incurred and of the disposition options.
Another deliverable to the project was the Material Segregation and Containerization Criteria (MSCC). The MSCC details each step of the D&D effort and what waste streams will be anticipated as a result of that activity. The MSCC utilized segregation categories that allow the sub-contractor to combine certain waste materials that are compatible depending on the disposition options. The MSCC was developed specifically for dismantlement activities and was provided to the D&D sub-contractor in order to assist them in segregating the categories of waste. The MSCC also directed the sub-contractor what type of container was to be utilized for each waste stream, and what waste streams could be consolidated into one container. The MSCC proved very useful because all waste streams were known prior to the commencement of dismantlement and the number and type of containers needed was known in advance.
There were several waste minimization opportunities realized throughout the life of Removal Action #19. One major opportunity is the recycling of approximately 710 tons of structural steel. This effort involved the packaging of the structural steel framework of the building into reusable roll-off containers. The steel will be shipped to a recycling vendor where the decontamination of the surfaces will occur and the steel recycled into the free market. Other opportunities include restricting the sub-contractor to bringing only the necessary equipment and materials for the project to minimize the potential for contamination. Also, used wooden skids were utilized to prepare containers for shipment to NTS in lieu of purchasing new ones.
Operations
Waste Programs Management also provided field support (Waste Technician) during the construction phase of the Plant 7 D&D project to coordinate work and assist in the completion of paper work related to containers. A container staging area was established adjacent to the project where empty containers were delivered and full containers were removed. The staging area was sized to provide a three day capacity of both empty and full containers in the event that transportation of the containers was inhibited. The Waste Technician was responsible for coordinating the delivery of empty containers to the project and removal of full containers for shipment or interim storage. The Waste Technician's responsibilities also included completing the paperwork necessary to verify the contents and track the type and amount of waste in each container. Another function of the Waste Technician was to ensure that

filling of the containers is done as efficiently as possible (another waste minimization opportunity).
Other operations support included scheduling the containers of LLW into shipments for offsite disposition to NTS and movement of containers into onsite interim storage. This involved scheduling the personnel and equipment necessary to move the containers on-site and prepare the containers for offsite shipment (i.e., labeling, inspection). The scheduling of these activities was very often done on a short turnaround in order to meet the sub-contractor's needs which were weather and schedule dependent. The sub-contractor often worked four 10 hour days, whereas FERMCO was on a different schedule. However, for the most part, the on-site operations services were able to keep up with the sometime accelerated pace of the sub-contractor.
Disposition
Three distinct disposition alternatives existed for the Plant 7 Dismantling project: LLW shipments to NTS; the recycling of 710 tons of structural steel; and the restricted reuse of lead flashing.
All of the LLW from the project is scheduled to be shipped by February 15, 1995 in support of a milestone associated with the project. The total volume of containerized LLW to be shipped to NTS equates to 70,392 ft3. Part of the support in the area of waste shipments included the final preparation and labeling of the containers, weighing the containers, and loading onto shipments. The subsequent paperwork was certified to the NTS Waste Acceptance Criteria by FERMCO Quality Assurance in support of WPM shipping activities.
The structural steel remains onsite for shipment to the vendor facility for decontamination and free release. The structural steel will be shipped to the recycling facility by the vendor. The current shipment schedule to the recycling facility is March 28, 1995 through May 19, 1995. As a result of the decontamination process, hazardous waste in the form of lead based paint and grit blast will be shipped back to Fernald and stabilized for disposition to Envirocare of Utah. Approximately 163 ft3 of lead flashing remains onsite and is awaiting disposition through a potential recycling vendor. The lead was the only hazardous waste generated from the Plant 7 Dismantling project.
CONCLUSIONS
The reorganization of WPM and its integration with Removal Action #19 has proven that up-front involvement on a project to properly plan and coordinate activities is successful. Characterizing the waste prior to generation allowed for proper measures to be taken in planning for the packaging and disposition of the waste. The number of waste containers needed was determined utilizing "as-built" drawings and applying bulking factors, depending on the container used. The containers were then ordered in advance and ample supply was available for the project when needed. With one central group responsible for integration and planning efforts, the required information was passed along to the respective Departments within WPM to ensure that all activities were covered. This involved field support at the project site, movement of waste containers to and from the project, storage of containers, and shipment of LLW to NTS. Other integrated efforts include the recycling of 710 tons of structural steel.
The major key to success during this project and others currently ongoing is constant communication. The formation of the DEC Team allowed for the channels of communication to remain open and was the avenue for all project information to be distributed throughout the organizations involved.

8-8
TECHNOLOGY DEVELOPMENT AND APPLICATIONS AT FERNALD
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ABSTRACT
At the Fernald Environmental Management Project (FEMP) northwest of Cincinnati, Ohio, the U.S. Department of Energy and contractor Fernald Environmental Restoration Management Corporation (FERMCO) are aggressively pursuing both the development and the application of improved, innovative technology to the environmental restoration

task. Application of emerging technologies is particularly challenging in a regulatory environment that places pressure on operational managers to develop and meet tight schedules. The regulatory and operational needs make close communication essential between technology developers and technology users (CERCLA/RCRA Unit managers). At Fernald this cooperation and communication has led, not only to the development and demonstration of new technologies with applications at other sites, but also to application of new technologies directly to the Fernald clean up. New technologies have been applied to improve environmental safety and health, improve the effectiveness of restoration efforts, and to cut restoration costs.

The paper will describe successful efforts to develop and apply new technologies at the FEMP and will emphasize those technologies that have been applied and are planned for use in the clean up of this former uranium production facility.

Assistant Secretary of Energy Thomas P. Grumbly has emphasized the application of new technology to environmental restoration as an important key to cleaning up DOE facilities more quickly and cost effectively. Saving money through improved technology meets a strong need during these times of limited budgets.

The paper will summarize a range of technologies being developed or applied at Fernald but which may be applicable to many other DOE sites and commercial locations. These include a robotic trimmer/excavator to reduce risks in treating high level tank wastes, soil washing and supporting technology, real-time instrumentation that reduces characterization and sampling costs, first-of-a-kind development of horizonal grout barriers, controlled detonation for cost-effective and safe demolition of contaminated buildings, minimum additive waste stabilization for more efficient and cost-effective stabilization of multiple streams of radioactively contaminated wastes, post closure monitoring technologies, solid block modeling, and advanced decontamination technologies.

The authors will bring together in one paper the perspectives of both technology developers and technology users, and will outline how Fernald has successfully overcome obstacles to achieve an aggressive, productive Technology Program. Managers responsible for remedial actions have a paramount need for technologies with ascertainable performance, cost, and schedule, while managers responsible for identifying and developing innovative technologies must maintain a balance between the role of independent advocate and the role of "partner" with operations managers. In addition to introducing Waste Management '95 attendees to the new technologies emerging at Fernald, this paper will conclude with lessons learned about specific technologies and management/development methods that can be applied by other projects facing cleanup under stringent operations needs and compliance deadlines.

INTRODUCTION

At the Fernald Environmental Management Project (FEMP) northwest of Cincinnati, Ohio, the U.S. Department of Energy and contractor Fernald Environmental Restoration Management Corporation (FERMCO) are aggressively pursuing both the development and the application of improved, innovative technology to facilitate the mission of the site environmental restoration. The FEMP is a DOE-owned, former uranium production facility which is in the early stages of restoration operations.

Implementation of new technologies to restoration operations is particularly challenging in the current regulatory environment that places pressure on operations managers who are legally bound to meet tight remediation schedules. Managers responsible for remediation (technology users) have a paramount need to use methods with ascertainable performance, cost, and schedule. Although new technology can offer the potential to substantially reduce cost and schedules, remediation managers frequently are reluctant to entertain risks associated with new approaches. Because of the potential for improvement, a comprehensive restoration effort, such as the one at Fernald, employs technology developers whose role is to act as independent advocates and promote the use of new, improved technological approaches. The challenge is to maintain a balance between technology user and technology developer that achieves the most advantaged restoration effort.

At Fernald technology users and developers have achieved a special communication and cooperation that has led to the development, demonstration, and application of new technologies directly to the Fernald environmental restoration effort. Innovative technologies have been applied to improve environmental safety and health, improve the effectiveness of restoration efforts, and to cut restoration costs.

This paper describes successful efforts to develop and apply new technologies at the FEMP and emphasizes those technologies that have been applied and are planned for

use in the restoration of Fernald. The application of new technology to environmental restoration is an important key to cleaning up DOE facilities more quickly and cost effectively. Saving money through improved technology meets a strong need during these times of limited budgets.

This paper presents the perspectives of both technology developers and technology users, and outlines lessons learned on how Fernald has successfully overcome obstacles to achieve an aggressive, productive, user-oriented technology program.

MINIMUM ADDITIVE WASTE STABILIZATION

The Minimum Additive Waste Stabilization (MAWS) demonstration, which was conducted at Fernald during calendar year 1994, is illustrative of a number of key issues in the coordination of technology development and environmental remediation. Vitrification has been widely identified as a preferred method for stabilizing high-level radioactive waste. At the time the MAWS demonstration was initiated no vitrification plant was operating in the United States, however, vitrification was considered a potential technology for application at Fernald. Demonstration of feasibility was necessary to gain the commitment of operations management, regulators, and engineers to proceed with this technology.

The MAWS demonstration plant was built at Fernald to treat as many waste streams as possible from the contaminated areas of the site, while minimizing the addition of non-contaminated materials to the vitrification process. The primary waste stream processed was comprised of sludge from Fernald waste pits. Pits 3 and 5 contain approximately 350,000 cubic meters of residues from uranium ore processing. To provide the properties required for satisfactory classification of this sludge, a source of silica was needed. Fernald's uranium-contaminated soil provided a suitable source for this ingredient. To maximize recovery of non-contaminated materials, soil washing technology was employed. Contaminated soil was treated to concentrate the uranium-bearing fraction with the silica-bearing fraction, and to separate these constituents from the bulk of the soil. The contaminated silica fraction was then blended with sludge from the waste pits and heated to produce glass "gems." These flattened marbles of black glass constituted a more stable waste form which was easily packaged and transported, more stable, and was significantly less volume than the original waste. In addition to the soil washing and vitrification furnace technologies, the MAWS plant incorporated an ion exchange waste water treatment system to recycle water used in the soil washing process.

Before termination of the demonstration, the MAWS facility was processing at a rate of 300 kilograms of glass output a day. The work demonstrated that waste pit volumes could be reduced by up to 70 percent, a substantial potential for cost savings. In addition, soil was cleaned to a condition allowing return of the clean portion as backfill for pits or other use. The sludge, which is classified as "mixed waste" for regulatory purposes, was converted to "low level waste." Mixed waste disposal costs run in the vicinity of $360/cubic foot compared with a current cost of under $20/cubic foot for low level waste disposal.

The MAWS demonstration was conducted largely by the technology suppliers. GTS Duratek Corporation provided the vitrification furnace and waste water treatment system. Lockheed Environmental Management Systems Corporation provided the soil washing technology. Catholic University of American performed the glass composition development and testing, while Argonne National Laboratory conducted long-term durability and performance testing of the glass. FERMCO provided facility support, as well as technical and project support.

In the CERCLA regulatory environment, Operable Unit (remediation) managers are driven primarily by schedules established under Consent Agreements. Although new or emerging technology may have the potential to improve schedules, reduce costs, or provide a better quality outcome, the remediation manager often faces a dilemma in accepting new technology. The remediation manager cannot commit to new technology until performance and cost aspects have been clearly demonstrated. For new technology to be applied, technology demonstration timing must accommodate the remediation manager's timing for preparing and negotiating the remedial action plan. The remedial action benefits from flexibility to allow the introduction of the new technology late in the process. CERCLA provides sufficient flexibility to allow the substitution of new technologies after approval and publication of the Record of Decision. Moreover, in the case of Fernald, the Ohio Environmental Protection Agency encourages the application of new technologies. The pragmatic implication of these circumstances is that technology development managers and remediation managers must

operate on coordinated, parallel paths to pursue a common goal: improvement.

VITRIFICATION PILOT PLANT

Successful demonstration of vitrification through the MAWS project provided the validation to support selection of vitrification as the preferred alternative for stabilization of waste from Fernald's K-65 silos. Located within Fernald Operable Unit 4 (OU4), Silos 1 and 2 contain radium-bearing sludge which emits radon gas. A vitrification pilot plant to handle the OU4 silo wastes is currently under construction and is scheduled to begin processing material from the silos in late 1995. The plant will incorporate a new furnace concept to reach higher temperatures, "stretching" state-of-the-art electrode technology. The plant will also require integration of a system for handling radon off gases. Currently, carbon absorption appears to be the most promising technology for addressing radon, but alternatives need to be evaluated, as carbon systems have intrinsic deficiencies.

SOLID BLOCK MODELING

Solid Block Modeling technology has been used on the Fernald project to produce three dimensional images of subsurface contamination and geologic conditions. Using an Oracle relational database along with Intergraph CAD/CAM hardware and software, the FEMP has been able to map the contaminated groundwater plume, calculate volumes of contaminated material, and map layers of clay and sand to identify potential disposal facility sites. Application of Solid Block Modeling resolved the question of the primary source of aquifer contamination. The major contributor of contamination was identified as the Southfield Landfill. It also was established that contribution to the regional aquifer contamination from the solid waste landfill was minimal.

The improved understanding of complex data interrelationships gained through application of Solid Block Modeling resulted in significant schedule and cost savings. Savings were amplified in 1994 when data acquisition was enhanced through use of a cone penetrometer. The Site Characterization and Analysis Penetrometer System (SCAPS) was brought to Fernald to provide characterization data for both OU1 and OU2 under auspices of the Uranium in Soils Integrated Demonstration and the Characterization Integrated Program, both sponsored by the Department of Energy Office of Technology Development. SCAPS is a modified cone penetrometer mounted on a truck which was specifically developed for use at environmental restoration sites. The unit pushes a two-inch diameter probe into the ground to gather geological information deep below the surface.

The SCAPS assessed characteristics of waste in three OU1 pits and collected data needed for the Dewatering Excavation and Evaluation Program. For OU2, the unit gathered geological data on the eastern area of the site to evaluate geological conditions for potential siting of a disposal facility. The unit also conducted tests with a prototype radiation probe in the Fernald Southfield area, providing information to support further development of the technology. The cone penetrometer project at Fernald demonstrated an effective partnership of field application in parallel with technology development. Both parties, user and developer, profited. The OU1 and OU2 managers each obtained valuable data, while the developers were able to demonstrate the feasibility of the penetrometer under radioactive field conditions. Companion data were obtained by standard geophysical methods for comparison with the cone penetrometer results. By replacing slower laboratory analysis of samples with real-time field analysis, the cone penetrometer achieved significant cost and schedule savings on the actual remediation project. Cost savings were calculated at more than $620,000 compared to 1994 baseline cost, with additional savings expected in 1995. In addition, data from the penetrometer measurements were fed into the Solid Block Modeling system to more accurately model subsurface conditions.

OTHER APPLICATIONS FOR COST AND SCHEDULE SAVINGS

Application of Micro-Purge technology to groundwater sampling has also begun producing cost savings at the Fernald site. Micro-purge technology reduces the volume of water that must be purged from a well each time a sample is taken. This reduces the time required for sampling, increases sampling accuracy, and reduces the volumes of contaminated waste water. During 1994, savings of $100,000 at Fernald were demonstrated. Savings from this technology will continue to be realized for the life of the project. Pioneered at Fernald, the micro-purge technology will have applications throughout the DOE environmental restoration network.

In the summer of 1994, Fernald's tallest building, Plant 7, was removed using

implosion technology. Implosion uses shaped charges to remove structural columns and splice plates on various floors so that the building will fall in on itself. The shaped charges act as a cutting tool by delivering approximately three million pounds of cutting force on the structural member.

Implosion was selected at Fernald because it offered a technical approach that eliminated worker exposure to heights (the height of the building was 110 feet), it reduced exposure to burning of lead paint from 640 manhours to 25 manhours, reduced worker time in a radiation area by four weeks, and reduced torch cutting of steel coated with lead paint from 3,420 feet to 90 feet. Early concerns with the public perception of use of explosives on a contaminated structure were overcome when the DOE and FERMCO worked closely with stakeholders and explained the safety advantages and reviewed the mechanics of implosion in several public forums. Public confidence with the FERMCO-DOE approach supported implosion as an innovative approach in environmental restoration.

The superstructure take-down required two implosions, but was the fastest and least-cost method that reduced the risks to craft personnel. Schedule improvement approximated six months, compared to baseline, and actual cost savings were calculated at approximately $5,000,000 from the innovative approach.

ROBOTICS APPLICATIONS FOR SAFETY ENHANCEMENT

The FEMP has also aggressively pursued the use of robotics to reduce human exposure to radioactive or hazardous materials and risk situations. In the recent past, Fernald has had two incidents of mower capsize. While no one was injured, these incidents represent a significant hazard that must be reduced. In addition to time expended donning and removing protective clothing, during summer months workers can be at risk of heat related disorders and discomfort during mowing operations. During 1994, a remote control lawn mower, as illustrated in Fig. 1, was tested at Fernald to perform vegetation control in areas of the site which are contaminated or have dangerous slopes. Following successful field trials, the equipment was accepted by FEMP site operations management for use by workers entering contaminated areas where protective clothing is required. While remote-control mowing may not seem revolutionary at first consideration, use of the advanced mower reduces costs, and, more importantly, reduces risks to workers.

A robotic waste retrieval device, dubbed "Houdini" by the developers, is currently being planned for use in the cleanup of three silos at Fernald (See Fig. 2). Silos 1 and 2 contain K-65 waste, a putty-like radium-bearing material. The robotic device is being developed to excavate "heel" and other hard-to-reach wastes, and thereby supplement the main hydraulic excavator, a sluicing pump, for the primary removal of wastes from the silos. The Houdini mechanism also will be designed to retrieve small objects which block the intake of the main excavator. Houdini consists of a folding, tracked base vehicle with a heavy-duty six-axis arm. The unit will be equipped with several tools, including a gripper, shears, hose reel assembly, and a scoop. The unit can be collapsed to enter openings as small as 24 inches in diameter. It will be environmentally and radiologically hardened, and designed to be easily decontaminated.

In Fernald Silo 3, which contains dry, powdery metal oxides, the robot will serve as the primary waste removal system, manipulating a vacuum nozzle to remove material form the silo. In all three silos, the innovative robot will be operating in environments where human entry is prohibited.

Robotics development represents a good example of where technology users are relying on developers to provide a product for application of an improved remediation method. The advance was made possible by development of a good working relationship between the Fernald developer and user, and by the generous support and guidance of the DOE Office of Technology Development Robotics Program.

HORIZONTAL GROUT BARRIER

A wide range of technologies are being demonstrated at Fernald. Among those with potential broad application in both the DOE and private sectors is the Horizontal Grout Barrier. This promising new technology may be the solution to one of the most perplexing problems of environmental restoration. Leaching of contaminants from surface or near surface sources is one of the most serious threats to aquifers. Placing an impermeable barrier beneath the contamination source to block movement of contaminants into the aquifer has long been a goal of engineers. Fernald is pioneering efforts to develop enabling technology for installing horizontal grout barriers intended to provide a solution to total in situ containment of wastes. The

Horizontal Grout Barrier combines directional drilling technology with advanced grouting technology developed in the oil service industry. Using directional drilling rigs, holes are drilled under the contaminated source. The holes are parallel and extend completely under the contamination source. The steel drill rods are then used to pull the grout injection system beneath the waste. Grout, injected at high pressure, cuts through the soil and mixes with it to form a continuous slab beneath the contamination source.

URANIUM IN SOILS INTEGRATED DEMONSTRATION

The most comprehensive technology demonstration at Fernald has been the Uranium in Soils Integrated Demonstration (USID). The uranium contamination in the soils at Fernald provides the appropriate conditions for demonstrating technologies for all stages of the restoration of uranium-contaminated soils. Since 1990, Fernald has supported demonstrations of technologies for real-time characterization of uranium contamination in soils, excavation and separation of the contaminated fraction from the soil, disposal of the contaminated fraction, and evaluation of treatment effectiveness. Soil washing technologies demonstrated at the site are under consideration for application to some of the cleanup challenges at Fernald. Although onsite storage is the current leading remedial alternative for restoration of Fernald soils, many technologies developed by the USID are being implemented in direct support of the Fernald restoration, including technologies for characterization and risk models. Many of the technologies will also undergo further development in DOE's "Landfill Focus Area." The new Focus Area approach is designed to bring the technology developers into closer dialog with users to assure that end-user requirements can be met by developments, an approach similar to the method Fernald has used to successfully implement new technologies.

CONCLUSIONS

The demonstration and application of new technologies to the challenges of environmental restoration are essential if the costs of restoration are to be brought down. The technology program at Fernald offers an example of how development work can be conducted to yield a measurable, positive return on investment (savings). New technologies also promise to reduce risks to workers, provide quicker solutions to cleanup challenges, and improve the overall quality of environmental restoration. The regulatory drivers for environmental restoration, however, force remediation managers in the field to make compliance with schedules a primary goal. Pragmatically, this means that environmental restoration managers will have little time to focus on potential application of unproven technologies. The authors believe that the following approach will facilitate development and application of new technologies to environmental restoration problems:

Field restoration and demonstration of new technologies must be pursued on separate, parallel tracks, with close schedule integration.

The overall technology development program must operate as a service organization that treats the technology user as a true "client."

Technology users must work with developers to articulate definitive performance requirements and companion needs to be integrated into development efforts.

Technology developers must place the end-users' needs foremost, including the need to be confident that new technologies will meet schedules.

There must be a "champion" for technology in the user organization who will aggressively advocate new technologies.

Environmental restoration managers should have technology application included in their performance standards.

Technology development efforts must accommodate stakeholder considerations from the outset.

The DOE-FERMCO Technology Program has provided innovative and improved technologies to directly support FEMP restoration, which has resulted in demonstrated cost reductions, schedule savings, and improved operational safety. The measured savings and improvements at Fernald indicate the potential for significant cumulative savings and increased efficiencies through further applications at other sites.

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9-1
IAEA ACTIVITIES IN DEVELOPING AND IMPLEMENTING SAFETY PRINCIPLES AND CRITERIA FOR

GEOLOGICAL DISPOSAL OF RADIOACTIVE WASTE

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ABSTRACT

For decades, the IAEA has played an important role in fostering the cooperation and the coordination of activities that can be best handled at the international level to establish, reach consensus on and publish Safety Standards, Guides and Practices for the safe management and disposal of radioactive waste. Over the past several years, a special series of safety documents is being prepared within the framework of the RADWASS programme, which covers all aspects of radioactive waste management. A new Safety Standard on geological disposal, based on the safety principles developed within the RADWASS programme, is being prepared. The draft Safety Standard specifies the objectives and defines general requirements to be considered in geological disposal of radioactive waste, including legal, regulatory and administrative questions, organization and responsibilities, radiological and environmental protection, characteristics of acceptable waste and sites, design, construction and operation of disposal facilities, their closure and post-closure issues, safety and environmental impact assessments, quality assurance. It provides the basis for developing Safety Guides and Safety Practices that contain detailed guidance on its implementation.

While the RADWASS programme is planned to document the existing international consensus on approaches to radioactive waste management and disposal, the IAEA has also established a forum at which any unresolved issues may be discussed. This is known as the INWAC Subgroup on Principles and Criteria for Radioactive Waste Disposal. Recently the Subgroup has prepared and published a report "Safety Indicators in Different Time Frames for the Safety Assessment of Underground Radioactive Waste Repositories". The report concludes that the long-term safety case can be made most effectively by the combined use of several safety indicators, such as risk, dose, environmental concentration, biosphere flux, flux through barriers, and time, recognizing, however, that risk and dose remain the most fundamental of the indicators of safety. It also draws conclusions on the nature of the safety assessments and the types of safety indicators to be used in the different timeframes of interest in relation to radioactive waste disposal.

INTRODUCTION

In the past the IAEA has published a number of Safety Series documents on geological disposal, (1, 2, 3). Lately, as a contribution to the process of demonstrating that radioactive waste can be managed safely, the IAEA has engaged upon the Radioactive Waste Safety Standards (RADWASS) programme in which it intends to document the internationally agreed approaches to safe radioactive waste management, including disposal, and provide Member States with a comprehensive series of documents to assist in the derivation of and to complement national criteria, standards and practices (4). The RADWASS programme has been organized as a complete body of 55 planned publications on safety of radioactive waste management with hierarchical structure following the general framework of the IAEA Safety Series documents (5). The importance of the programme has been reaffirmed by the Agency's General Conference, which, at its thirty-eighth regular session held in September 1994, invited the Board of Governors and the Director General "to maintain the emphasis given to radioactive waste management, especially with regard to RADWASS, to commence with planning activities for a convention on the safety of waste management and begin the process of collecting relevant background information (including appropriate RADWASS documents) that would be useful in drafting the convention". While the RADWASS programme is intended to document the existing consensus in radioactive waste management, it is recognized that there are some issues of philosophical nature which require further discussion before a consensus can be reached. For this reason the IAEA has established an international forum where these issues can be discussed. The forum is with the INWAC Subgroup on Principles and Criteria for Radioactive Waste Disposal (PCRWD). To date, the Subgroup has prepared a report on safety indicators and timeframes (6) and has developed position and discussion papers on several other issues.

In this paper we describe the progress on the parts of the RADWASS programme relevant to geological disposal and also the work of the INWAC Subgroup on Principles and Criteria for Radioactive Disposal.

RADWASS SAFETY PRINCIPLES

The radioactive waste management safety principles, as defined in the draft version of the RADWASS Safety Fundamentals document "The Principles of Radioactive Waste Management", recommended for publication by the extended INWAC meeting in January 1995, are:

Principle 1: Protection of human health. Radioactive waste shall be managed in such a way as to secure an acceptable level of protection for human health.

Principle 2: Protection of the environment. Radioactive waste shall be managed in such a way as to provide an acceptable level of protection of the environment.

Principle 3: Protection beyond national borders. Radioactive waste shall be managed in such a way as to assure that possible effects on human health and the environment beyond national borders will be taken into account.

Principle 4: Protection of future generations. Radioactive waste shall be managed in a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today.

Principle 5: Burdens on future generations. Radioactive waste shall be managed in a way that will not impose undue burdens on future generations.

Principle 6: National legal framework. Radioactive waste shall be managed within an appropriate national legal framework including clear allocation of responsibilities and provision for independent regulatory functions.

Principle 7: Control of radioactive waste generation. Generation of radioactive waste shall be kept to the minimum practicable.

Principle 8: Radioactive waste generation and management interdependencies. Interdependencies among all steps in radioactive waste generation and management shall be appropriately taken into account.

Principle 9: Safety of facilities. Safety of facilities for radioactive waste management shall be appropriately assured during their lifetime.

In the implementation of these principles, the relevant recommendations of international bodies, such as IAEA and ICRP are typically taken into account. In particular, the recommendations related to radiological protection incorporated in the "International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources" (BSS) (7) are relevant.

In order to assist countries in implementing the above safety principles the IAEA is working to formulate as Safety Standards the definition of technical and safety requirements for each of the six RADWASS subject areas 1) Planning, 2) Predisposal, 3) Near surface disposal, 4) Geological disposal, 5) Uranium/thorium mining and milling waste, 6) Decommissioning/environmental restoration).

RADWASS SAFETY STANDARD ON GEOLOGICAL DISPOSAL

The objective of the Safety Standard on geological disposal is to set out the internationally agreed basic requirements for safe disposal of radioactive waste in suitable geological media at an appropriate depth (hundreds of meters) below the earth's surface. It is intended primarily to be applied to the disposal of solid high level and other long lived radioactive waste, in particular spent nuclear fuel (if disposed as a waste), the highly radioactive waste from reprocessing spent nuclear fuel and alpha bearing waste, in mined facilities and in deep boreholes. It is evident, however, that geological disposal systems, acceptable on a safety basis for disposing of high level and long lived waste, can be also used for disposing of solid low and intermediate level waste, provided the latter is properly conditioned and a country chooses this option.

Structure

The structure and the content of the document are still being developed. For the purpose of consistency, and reflecting the fact that a number of requirements for both near surface and geological disposal of radioactive waste are similar, the main structure of the already prepared Safety Standard on near surface disposal was preserved in the draft Safety Standard on geological disposal. Proper attention has also been devoted to harmonization of this Safety Standard with other RADWASS publications. Accordingly, the draft Safety Standard on geological disposal is structured in thirteen Sections, including an overview of the basic waste management principles, as they are stated in the Safety Fundamentals (Section 2), national framework for radioactive waste management (Section 3), an outline of the organization and responsibilities for each of the parties involved (Section 4), safety and environmental protection (Section 5). Sections 3, 4 and a part of the Section 5 represent incorporation into this Safety Standard of the basic radioactive

waste management requirements as they are formulated in the Safety Standard on "Establishing a National System for Radioactive Waste Management". They are followed by a description of specific technical and safety requirements for disposal components and activities, in particular, acceptable waste (Section 6), acceptable site (Section 7), design of disposal facilities (Section 8), construction (Section 9), operation (Section 10), closure (Section 11), post-closure (Section 12) and requirements for quality assurance during all disposal activities, from the planning through to the post-closure phase (Section 13).

Safety and Environmental Protection

For consideration of safety and environmental protection issues the lifetime of a geological disposal facility is subdivided into three phases: pre-operational, operational and post-closure. The pre-operational phase includes siting, design and construction. The operational phase includes commissioning, operations and closure. The post-closure phase may include institutional and post-institutional control periods. These phases may occur separately, or may overlap, in part. During all of these phases, safety of the facility relies on the appropriate application of several types of protective measures including multiple barriers, quality assurance procedures and administrative controls to prevent releases of radioactive materials to the environment and to mitigate their consequences should this occur.

The draft Safety Standard emphasizes that during the operational phase of a disposal facility humans and the environment must be protected against unacceptable levels of ionizing radiation in compliance with BSS requirements. For application to the post-closure phase of geological disposal additional requirements to those provided in the BSS are needed for dealing with exposures which are not certain to occur [2, 6, 8, 9]. The exact nature of the guidance in the Safety Standard on radiological and other safety criteria for disposal is still under discussion but it may be noted that in a previous IAEA Standard (2) a reference value of about 10-5 of annual risk, which should not be exceeded by an individual of the critical group from all radiation sources, was specified. A fraction of this value was said to be applicable to waste disposal.

Comprehensive safety and environmental impact assessments are considered in the draft Safety Standard to be key elements in ensuring safety and environmental protection. They are required to be performed and updated, as necessary, in support of an application to the regulatory body for approval to construct, to operate and to close a geological disposal facility and if significant changes in approved conditions appear. It is also required that these assessments consider the radiological and non-radiological safety and environmental impact within and beyond national boarders with particular attention being given to regulatory requirements and that waste is disposed in a manner that adequately ensures protection of humans and the environment without imposing undue burdens on future generations.

Other issues

Other issues of importance for geological disposal considered in the draft Safety Standard include retrievability, security arrangements and safeguards, closure of the repository and post-closure institutional arrangements and keeping records.

Although geological disposal means waste emplacement in an approved specified facility without the intention of retrieval, some countries may prefer to keep the possibility during a shorter or longer period of time for, a maximum, hundreds of years, to retrieve waste already emplaced in a repository. It is emphasized, however, that, if the ability to retrieve waste is a design requirement, provisions should be made for retrievability at the start of the project to exclude any threat to long term repository performance.

All reasonable security arrangements should be made to prevent unauthorized actions that jeopardize safety of the repository. In particular, provisions should be made to detect, delay and prevent any unauthorized entry into the security sensitive areas. Waste that does not meet the criteria for the termination of safeguards, must be subject to safeguards control during the operational phase of a disposal facility. Safeguards verification activities should obviously be matched with the disposal operations and should not have any negative impact on the operational and long term safety.

Permanent closure of a disposal facility comprises a set of systematic actions, after the waste emplacement operations are completed, with the objective of finalizing the isolation system. It may include sealing of engineered openings such as boreholes, tunnels, shafts and drifts and decommissioning of surface facilities.

Partial backfilling and sealing of the emplacement areas may already have been done during the operational phase. After the completion of the closure actions, a geological disposal facility is brought to its post-closure status. No further engineering measures are then expected to be necessary to ensure proper future performance of the disposal facility.

In accordance with the principle of limiting the burden on future generations, geological disposal of radioactive waste should, to the extent possible, not rely on any institutional arrangements or actions as a necessary safety feature. However, monitoring of the repository in the post-closure phase is not precluded for a limited period of time if it is justified and does not impair the safety of disposal. Safety can be enhanced by institutional control measures, for example, by the prevention of intrusion. The country hosting a geological disposal facility should consider the need for post-closure institutional arrangements and actions and the allocation of responsibilities and resources for such activities. Some Member States may be compelled to introduce long term institutional measures with respect to repositories containing spent nuclear fuel in response to international safeguard obligations.

Keeping records of the repository and site markers above and/or below ground is considered as a cost-effective way to maintain awareness of the potential long term hazard and is recommended as a measure to reduce the probability of inadvertent intrusion into the repository. Records are to be maintained of the layout of the repository, facility design, as built structures, waste inventory including location and physical form details, package identification, backfill materials, site characterization data, safety assessment results, models and computer codes used, environmental monitoring results, and data on the closure of the disposal facility.

Companion Documents

The Safety Standard on geological disposal will be accompanied by three Safety Guides and four Safety Practices documents. One of them, Safety Guide "Siting of Geological Disposal Facilities" has been published (10), the other two Safety Guides "Design, Construction, Operation and Closure of Geological Repositories" and "Safety Assessment for Geological Disposal" as all four Safety Practices documents "Validation and Verification of Models for Long Term Safety Assessment of Radioactive Waste Disposal Facilities", "Procedures for Closure of Radioactive Waste Disposal Facilities", "Waste Acceptance Requirements for Geological Disposal of Radioactive Waste" and "Selection of Scenarios for Safety Assessment of Geological Disposal Facilities" will be prepared and published later.

ACTIVITIES OF THE INWAC SUBGROUP ON PRINCIPLES AND CRITERIA FOR RADIOACTIVE WASTE DISPOSAL

As explained earlier, the INWAC PCRWD Subgroup plays an important role in examining unresolved issues related to radioactive waste disposal. Details of its terms of reference and the activities during the initial period are given in (11). Since then, two Subgroup meetings (October 1993 and November 1994) have been held at the IAEA Headquarters in Vienna.

Safety indicators and timescales

Recently the first report of the Subgroup was published (6). It examines the question of what type of safety assessment is needed to cover time periods far into the future and also what sort of safety criteria (or safety indicators) are appropriate at such times. The motivation for this discussion is the need to substantiate the principle of protection of future generations given the difficulties in showing compliance with safety criteria over long time-scales because of the increase with time of the uncertainty associated with the results of predictive models. On the other hand, the radiotoxicity of the wastes decreases with time due to radioactive decay. These contrary trends suggest that the meaningfulness of and the need for detailed quantitative assessments become less and less the further into the future the assessment is carried. It is generally considered appropriate, however, to continue assessments sufficiently far into the future to ensure that any peak in potential impact of the disposal facility has been taken into account. The only adjustment to this principle might arise if qualitative studies were to show the peak impact to be so low as to be considered trivial.

The results of the discussion provided in the report are summarized as follows:

1. The assessed long term consequences of disposal systems in terms of risk and dose can only be considered as indicators of safety.
2. The long term safety case can be made most effectively by the combined use of

several safety indicators, such as risk, dose, environmental concentration, biospheric flux, flux through barriers and time recognizing, however, that risk and dose remain the most fundamental of the indicators of safety.
3. Indicators become particularly valuable when they are supported by observations from natural analogues.
4. In the time period up to around 104 years after repository closure, the safety case should be based on quantitative safety assessments using dose/risk calculations supported by calculations involving other safety indicators.
5. In the period from around 104 years to about 106 years after repository closure, the safety case should be based either on quantitative safety assessments or on qualitative assessments using a combination of safety indicators. The emphasis may be expected to shift increasingly towards qualitative assessments as 106 years is approached.
6. Beyond about 106 years little credibility can be attached to integrated safety assessments.

Several position and discussion papers have been prepared and discussed at the Subgroup meetings. They include papers on: "Interface Issues between Nuclear Safeguards and Radioactive Waste Management", "Position Paper on Post-Closure Issues", "Optimization of Radiation Protection in Application to Radioactive Waste Disposal" and "Regulatory Decision Making in the Presence of Uncertainty". The publication of several of these was approved at the last meeting of the PCRWD Subgroup.

CONCLUSION

We have described some of the IAEA activities related to the solution of the problem of disposal of high level and long lived radioactive waste. Their results demonstrate that Agency has contributed to and can further assist in the solution of this problem by: 1) putting radioactive waste management and disposal into a global context; 2) documenting the existing international consensus on the approaches and methodologies for safe radioactive waste management and disposal; 3) providing a forum for discussion of principles and criteria for safe management and disposal of radioactive waste and creating a mechanism to establish consensus where it does not yet exist; and 4) providing Member States with a comprehensive series of internationally agreed documents to complement or set up national standards and criteria.

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9-2

INTERNATIONAL PERSPECTIVE ON U.S. HIGH-LEVEL WASTE LICENSING REQUIREMENTS

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ABSTRACT

In support of the National Academy of Sciences (NAS) study required by Section 801 of the Energy Policy Act of 1992, the U.S Department of Energy (DOE) developed, in April of 1994, specific recommendations pertaining to a standard for a potential repository at Yucca Mountain. These recommendations are summarized as follows:

The standard should be health-based.

The standard should focus on protecting those people who will be living in the vicinity of the potential repository.

The standard should require a quantitative demonstration of compliance for no more than 10,000 years.

The standard should provide qualitative requirements related to reducing the potential for inadvertent future human intrusion to cause radionuclide releases.

The quantitative aspects of the standard should be stated in deterministic terms, with probabilistic analyses used to demonstrate compliance.

A standard expressed in terms of individual dose would be appropriate, provided that the following aspects are clearly specified in a regulation or associated guidance:

- The individual dose limit should be commensurate with current dose limits for individual members of the public as specified in 10 CFR Part 20, and with recommendations of national and international advisory groups.
- The dose limit should be applied to an average individual in the population living in the vicinity of Yucca Mountain.
- Assumptions related to the future biosphere should be specified. These should include a water use scenario for the average individual which is based on the current practices of people living in the Amargosa Valley.

These recommendations are consistent with existing national and international standards and guidance. Most of the national and international community expresses the primary radiation protection criterion in terms of annual individual dose, with a range from 10 to 100 mrem/yr (0.1 to 1 mSv/yr), and over a time frame of 10,000 years.

BACKGROUND

In the Nuclear Waste Policy Act (NWPA) of 1982 (1), the Congress of the United States directed the Department of Energy (DOE) to characterize potential repository sites and to design, construct, license, and operate a geologic repository for spent nuclear fuel and high-level radioactive waste. The same act directed the Environmental Protection Agency (EPA) to promulgate generally applicable standards for protection of the general environment from offsite releases of radioactive material from repositories. Furthermore, responsibility for implementing this standard through the development and application of licensing requirements was assigned to the Nuclear Regulatory Commission (NRC). In 1985, the EPA issued repository standards in the form of the final rule 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste" (2). In 1987, the United States Court of Appeals for the First Circuit vacated and remanded 40 CFR Part 191; while the management and storage aspects were reinstated on appeal, the disposal standards were left in remand. Also in 1987, the Nuclear Waste Policy Amendments Act (3) redirected the DOE program toward the characterization of only one potential site for a geologic repository - the Yucca Mountain site in Nevada. The Energy Policy Act of 1992 (4) directed the EPA to contract with the National Academy of Sciences (NAS) to conduct a study to provide findings and recommendations on reasonable standards for protection of the public health and safety. The 1992 law also directed the EPA, based on the recommendations of the NAS, to promulgate public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site. Furthermore, the NRC was directed to modify its requirements to be consistent with the EPA standards. The NAS formed the Committee on Technical Bases for Yucca Mountain Standards to prepare its recommendations to the EPA. The committee held public meetings in May,

August, November, and December of 1993, and in April of 1994. A committee report, containing recommendations to the EPA, is expected some time in 1995.
In May of 1993, the DOE proposed to the NAS that a standard for the potential geologic repository at Yucca Mountain should be implementable, to allow the DOE to carry out its obligations to the citizens of the United States, and understandable, to foster public confidence in the safety of geologic disposal. The DOE noted that a standard should require a scientifically supportable demonstration of compliance, and should be consistent with other radiation standards and regulations. In April of 1994, the DOE provided more specific recommendations to the NAS on a health and safety standard, which will be discussed later in this paper. (5)

GENERAL CONSIDERATIONS IN DEVELOPING A STANDARD

A standard could be expressed in a variety of forms, including the number of expected health effects on future populations, doses to individuals or populations, and the amount of radionuclides released to the accessible environment. Whatever the form, demonstrating compliance with such a standard will necessarily involve making quantitative projections of repository performance far into the future. There is considerable uncertainty inherent in these long-term performance projections for any repository. Dealing with that uncertainty must be considered in the development of the standard itself. In some cases it will be appropriate to specify, as a part of the standard or associated regulatory guidance, how the uncertainty is to be treated. In particular, the characteristics of future human society are a major uncertainty that should be dealt with up front, in the NAS recommendations to the EPA, rather than during a licensing proceeding.

DEPARTMENT OF ENERGY RECOMMENDATIONS

With the general considerations discussed above in mind, in April of 1994, the DOE developed specific recommendations pertaining to a standard for the potential repository at Yucca Mountain. These recommendations are discussed below:

The standard should be health-based.

A health-based standard is considered preferable largely because it is easier to understand the link to public health and safety, and to compare risks between a repository and other human endeavors. A health-based standard should assist in fostering public confidence in the safety of a geologic repository. In addition, it should be noted that a health-based standard is consistent with both the express language and legislative history of the Energy Policy Act of 1992. While the EPA is given final discretion to promulgate the appropriate standard, it is nevertheless required to promulgate a health-based standard, absent a finding by the NAS that such a standard is unreasonable. (6)

The standard should focus on protecting those people who will be living in the vicinity of the potential repository.

A standard might focus its protection in a variety of ways. These include: an individual at the boundary of the accessible environment, people living in the vicinity of a repository, the world population, groundwater, or plants and animals. Due to the nature of geologic disposal in general and the Yucca Mountain site in particular, the people who will be living in the vicinity of a repository are those most at risk, and they are the proper focus of protection. Protection of a hypothetical, maximally-exposed individual living at the boundary of the accessible environment is an unrealistic, overly conservative worst case for the arid, isolated Yucca Mountain site. Protection for people in the vicinity of the repository must include consideration of exposure to radiation via a groundwater pathway. Therefore, no additional groundwater protection provisions are needed. Also, world population protection through release limits or population dose limits is superfluous so long as the people in the vicinity of the site are adequately protected. Population protection, if provided, should allow the truncation of very small doses to large numbers of people.

Consistent with the discussion above, a standard incorporating a single performance measure focused on protecting those people in the vicinity of a repository would make demonstration of compliance more straightforward in a licensing proceeding. Conversely, a standard specifying multiple long-term performance measures (such as the subsystem performance objectives in NRC's 10 CFR Part 60) (7) would complicate the licensing process without providing significant benefit in terms of public health and safety. Hence, the licensing regulations should not include quantitative subsystem requirements.

The standard should require a quantitative demonstration of compliance for no more

than 10,000 years.
The time period of regulatory concern is ultimately a policy decision. Other EPA regulations include time frames of 1,000 years for the regulation of uranium mill tailings (8), and 10,000 years for the regulation of underground injection of hazardous waste (9). The 10,000 year time frame of EPA's 40 CFR Part 191 was upheld in Federal court as adequately justified. Furthermore, it has been endorsed by the NRC as a practical time period for determining compliance (10). The DOE recognizes that health effects from a repository may be projected to occur at times greater than 10,000 years after permanent closure. However, the uncertainty in calculated quantitative results increases with the time of the projection, to the point that extremely long-term performance calculations may have no value beyond rough comparisons between alternative sites or designs. If time frames longer than 10,000 years are addressed in a standard, then the associated requirements should be qualitative, as opposed to quantitative total system performance assessment limits.

The standard should provide qualitative requirements related to reducing the potential for inadvertent future human intrusion to cause radionuclide releases. The DOE believes that the uncertainties associated with predictions of human intrusion are so great that including such calculations in the demonstration of compliance would significantly increase regulatory costs without any corresponding contribution to reasonable assurance of the long-term safety of disposal. For example, predicting the probability and effects of inadvertent human intrusion could involve forecasting the characteristics, technologies, and habits of human societies far into the future. Intrusion should be addressed with qualitative design requirements and active and passive institutional controls. Detailed specification of these requirements (e.g., numbers and types of permanent markers, land-use restrictions, records) can be deferred until near the time of permanent closure of the repository. At that time the appropriate specifications can best be determined, based on the information and technology available.

The quantitative aspects of the standard should be stated in deterministic terms, with probabilistic analyses used to demonstrate compliance.
It is recommended that the primary radiation protection criterion (e.g., dose, release, etc.) contained in a quantitative standard be stated in deterministic terms. In other words, the primary criterion would be stated in terms of a single value, not limits that are a function of the probability. This approach is easier to understand and would be easier to defend in a licensing proceeding. Even with a deterministic standard, the DOE proposes to perform a probabilistic calculation of future repository performance. Results would be presented as a range of values and associated probabilities (i.e., a probability distribution function), rather than a single number. This will allow consideration of low probability, high consequence events as a part of the evaluation. The DOE would compare the calculated results, a range of values with associated probabilities, against the deterministic value specified in the standard. This calculation would include both disturbed and undisturbed performance, but not human intrusion scenarios (see previous recommendation). It is recognized that there may be some probability of the calculated performance exceeding the specified limit. The demonstration of compliance would be required to show that the probability of exceeding the limit is acceptably low. The quantitative safety case for the repository would be supplemented by other arguments such as comparison to natural analogs and defense-in-depth. In this manner, based on the weight of all the evidence available, the DOE would demonstrate with reasonable assurance that the health and safety of the public are adequately protected.

A standard expressed in terms of individual dose would be appropriate, provided that certain aspects are clearly specified in a regulation or associated guidance. There are a variety of ways in which a health-based standard could be expressed. These include: acceptable risk, individual dose, radionuclide releases, subsystem performance, and specific engineered or natural system requirements. All of these are potentially acceptable, provided that they are expressed in a manner that provides for a demonstration of compliance that can be defended during the licensing process. The major difference between the options is the amount of derivation from the ultimate basis of the standard, which should be one of public health.
For a dose standard, three key aspects that need to be specified are:
- The individual dose limit should be commensurate with current dose limits for individual members of the public as specified in 10 CFR Part 20, and with

recommendations of national and international advisory groups (i.e., on the order of 100 millirem/year or less). (11,12,13)
- The dose limit should be applied to an average individual in the population living in the vicinity of Yucca Mountain.
- Assumptions related to the future biosphere (i.e., practices of humans in the future) should be specified.

COMPARISON TO OTHER NATIONAL AND INTERNATIONAL STANDARDS

As indicated above, the DOE recommended to the NAS that the new standard for the Yucca Mountain site should be consistent with other standards and regulations. This section compares the DOE recommendations with other national and international standards and guidance. This comparison is focused on three key characteristics: 1) the form of expression of the standard, 2) the limit for the primary radiation criterion, and 3) the disposal time frame regulated. Table I summarizes these characteristics for other national standards and guidance for nuclear facilities, while Table II presents this information for international standards and guidance.

The Form of Expression of the Standard

With respect to the form of expression of the standard, Table I shows that most of the national standards express the primary criterion in terms of annual individual dose. While EPA's 40 CFR Part 191 has, as its primary criterion, a radionuclide release standard, it does contain an individual dose limit for undisturbed conditions. While Canada and the United Kingdom use an individual risk objective as their primary criterion, Table II shows that the primary criterion used internationally is predominantly an individual dose limit.

Limit for the Primary Radiation Criterion

Since for the most part the primary radiation criterion being used is an individual dose limit, based on the discussion above, this comparison will be focused on these limits. For national standards, Table I shows that the range for such limits is 15 to 100 mrem/yr (0.15 to 1 mSv/yr). These numbers focus on normal operating conditions and undisturbed postclosure performance. National standards covering disturbed postclosure performance in terms of an individual dose limit have not as yet been established in this country. Within the international community, Table II indicates that the range of limits on individual dose is from 10 to 100 mrem/yr (0.1 to 1 mSv/yr). Again, these limits focus on undisturbed performance. For handling disturbed performance, several of the countries use the individual risk criterion of 10-6/yr.

Disposal Time Frame Regulated

With respect to the time frame regulated for disposal, 10,000 years appears to be the predominant time period. For the national standards in Table I, EPA's 40 CFR Part 191 is the only standard that applies to postclosure performance, and the time frame regulated is 10,000 years. In Table II, for those countries that explicitly specify a postclosure time frame, 10,000 years is the limit, but assessments for longer periods are required on a qualitative basis.

CONCLUSIONS

The process of promulgating a standard for high-level waste disposal is a complex one, and touches on many technical as well a policy considerations. This paper summarized the DOE recommendations regarding certain characteristics of a standard that, if incorporated, would result in a reasonable and implementable regulatory framework for disposal of high-level nuclear waste in this country, a framework that would ensure the protection of the health and safety of the public. When the DOE recommendations are compared to other national and international standards and guidance, this comparison demonstrates that the recommendations are within the range of existing national and international standards and guidance, and are consistent with the direction in which the global technical and policy community is headed.

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WHICH WAY TO YUCCA MOUNTAIN (STANDARDS)

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ABSTRACT

The U.S. Environmental Protection Agency (EPA) has recently begun the process of establishing environmental radiation protection standards for the potential repository at Yucca Mountain, Nevada. The standards are being developed under the author of the Energy Policy Act of 1992. It had been anticipated that the report from the National Academy of Sciences which was required by Federal law to provide advice to EPA would have been received in time to be the basis of this paper. Since that report has not been received, this paper will examine the history leading up to this new standards-setting process, EPA's initial positions on various issues, the status of the standards-setting project, and the future steps.

HISTORY

In 1985, the Environmental Protection Agency (EPA or "the Agency") issued 40 CFR Part 191, Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste (1). These standards were issued under the authority of the Atomic Energy Act of 1954, as amended, (2) and Reorganization Plan No. 3 of 1970 (3). The standards are generic and apply to all facilities for the management, storage, and disposal of the named wastes. The Nuclear Regulatory Commission (NRC) must adopt these standards into their licensing regulations for commercial facilities and the Department of Energy (DOE) must follow these standards for their disposal facilities. The disposal portion of Part 191 was vacated and remanded by a Federal court in 1987. The remanded portions included release limits, assurance requirements, and individual and groundwater protection requirements. The Agency pursued the reestablishment of the disposal standards until October, 1992.

At that time, two Federal laws were enacted which significantly affect the approach the Agency would take in writing environmental standards for spent nuclear fuel (SNF) and high-level radioactive waste (HLW). First was the Waste Isolation Pilot Plant Land Withdrawal Act of 1992 (WIPP LWA) (4). Section 8 is the part of the WIPP LWA which pertains to EPA's standards for SNF and HLW. It reinstated most of the disposal standards in Part 191. However, it also stated that the Part 191 standards

"shall not be applicable to the characterization, licensing, construction, operation, or closure of any site required to be characterized under section 113(a) of Public Law 97-425." Public Law 97-425 is the Nuclear Waste Policy Act of 1982; the only site being characterized under its provision is Yucca Mountain, Nevada.

THE ENERGY POLICY ACT OF 1992

The second law is the Energy Policy Act of 1992 (5). Section 801(a)(1) of the act directs the Agency to establish standards for Yucca Mountain:

"... the Administrator [of EPA] shall, based upon and consistent with the findings and recommendations of the National Academy of Sciences [NAS] promulgate, by rule, public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site."

There were two requirements given for the contents of the Yucca Mountain standards. One, as stated in the preceding quote, was that the standards be "based upon and consistent with the findings and recommendations of the National Academy of Sciences." The second was that the standards "shall prescribe the maximum annual effective dose equivalent to individual members of the public ...."

These requirements were further delineated in Section 801(a)(2). It requires the Administrator to "contract with the National Academy of Sciences to conduct a study to provide ... findings and recommendations on reasonable standard for protection of the public health and safety ...." Section 801(a)(2) further mandates answers to three specific questions i.e., the NAS is to provide "findings and recommendations on reasonable standards including -

A) whether a health-based standard based upon doses to individual members of the public from releases to the accessible environment ... will provide a reasonable standards for protection of the health and safety of the general public;

B) whether it is reasonable to assume that a system for post-closure oversight of the repository can be developed, based upon active institutional controls, that will prevent an unreasonable risk of breaching the repository's engineered or geologic barriers or increasing the exposure of individual members of the public to radiation beyond allowable limits; and

C) whether it is possible to make scientifically supportable predictions of the probability that the repository's engineered or geologic barriers will be breached as a result of human intrusion over a period of 10,000 years."

The contract between EPA and NAS was signed in February 1993. Shortly thereafter, the NAS established the Committee on Technical Bases for Yucca Mountain Standards ("the Committee"). The Committee is administered by the Board on Radioactive Waste Management of the NAS. This committee has fifteen members including the chairman Dr. Robert W. Fri of Resources for the Future. The members are a mixture of personnel from academia, contractors, consultants, and one member from a foreign radioactive waste management program.

FURTHER SUGGESTIONS FROM EPA TO THE NAS

Before the formation of the Committee, the Director of the Office of Radiation and Indoor Air, the EPA office in charge of radiation standards, sent a letter to the Staff Director of the Board on Radioactive Waste Management to provide "a clear sense of the questions EPA would like answered so that your study can be of the most use ..." (6). The letter was organized based upon the three questions (quoted above) posed by Congress in the Energy Policy Act. These questions will be referred to as Energy Policy Act Questions A, B, and C.

Regarding Energy Policy Act Question A (whether a health-based individual dose limit would provide for the health and safety of the general public), the Agency requested that the Committee examine the comparative effectiveness of various types of standards, i.e., individual, collective, or other types. Also, the Committee was requested to consider the implications and anticipated results, including collective dose, resulting from application of different levels of individual dose standards at Yucca Mountain. The range of dose limits should include 0.01 to 0.25 millisieverts [1.0 to 25 millirem] (mrem) per year (mSv/yr). The type of individual standard should also be indicated, e.g., probabilistic (risk), deterministic, expected value, maximum individual or average of a critical population group. Finally, the Committee might want to consider:

the uncertainty in the relationship between individual and collective dose;

the impact of different periods of assessment and the relationship of those

periods to individual and collective dose or other types of limits;
  whether a "static biosphere" is desirable and, if so, what it should include; and,
  the protectiveness of an individual or population dose standard or other types of limits in light of the possibility of natural or human-initiated disruptive events.
Energy Policy Act Question B asked if post-closure active institutional controls could be developed which would prevent an unreasonable risk of breaching the repository's barriers or increasing individual doses beyond allowable limits. The Agency suggested that the Committee considerations include different types of active control measures and their cost. Also, that the Committee might want to determine the types of intrusive events possible at Yucca Mountain and evaluate the effectiveness of different active controls to mitigate resulting releases. In addition, possible issues to consider were:
  types of actions considered to be "active institutional controls" and what credit for prevention or remediation these actions should be given;
  the historical record of institutional controls to prevent problems at other disposal sites; and
  durability of active institutional controls over time and how society could assure continuous control during any particular time.
The final Energy Policy Act Question, C, asks if "scientifically supportable predictions of the probability" of human intrusion breaching the repository's barriers over a 10,000-year period are possible to make. The Agency hopes to get a clear definition of "scientifically supportable predictions of the probability." In examining this issue, other issues to consider include:
  comparison of such predictions between human intrusion and naturally occurring disruptive events;
  is the determination of "scientifically supportable predictions of the probability" dependent on the probability value; and
  can "scientifically supportable predictions" result from simplifying assumptions about future human behavior which could be made to reduce uncertainty.
EPA'S CURRENT VIEWS OF THE ISSUES
It had been anticipated that the report from the National Academy of Sciences would have been received in time to be the basis for this paper. Since that report has not been received, in this section of the paper, rather than initial reactions to the findings and recommendations in the report, we will discuss the Agency's pre-report view of the issues. It should be understood that these positions could change based upon future input from the NAS as well as other parties.
The Agency is responsible for setting the environmental standards for Yucca Mountain and is required to use its own discretion and expertise during the formulation of those standards. The NAS input will, of course, be an integral part of the EPA deliberations. However, for the NAS input to be most useful, it must be accompanied with adequate background information and reasoning. In addition, there must be clear rationale because the EPA must base its decisions on the clearest, most technically based information, and, therefore, legally defensible reasons, available for the bases of provisions in the final standards.
The questions in the Energy Policy Act could be answered generically even though Section 801 of the Act is focused upon Yucca Mountain. Those familiar with studies such as the NAS study know that such groups rarely feel constrained by written guidance. It is clear that the committee's activities fit into this paradigm. According to the opening statement of the committee chair, Dr. Fri, " ... we are free to question all assumptions and evaluate all options with regard to the technical bases for EPA's standards. Everything is on the table. We have no preconceptions about the subject of our study" (7). This is understood and encouraged by EPA but it is important that the NAS makes it clear as to which type of guidance is being provided.
Finally, it is important that the NAS deal with the precedent of 40 CFR part 191. Any differences between the existing generic standards and the site-specific standards for Yucca Mountain must be clearly addressed. The most useful manner to deal with this would be for the NAS to clearly explain any advice that could lead to the establishment of standards different from the provisions in 40 CFR part 191. This does not mean that the NAS technical advice or the Yucca Mountain standards cannot differ from Part 191 but the differences must be recognized, clearly explained, and justifiable.
Energy Policy Act Question A

The Agency believes that an individual dose standard alone is not sufficiently protective and cannot replace the need for other requirements. First, an individual dose limit would not necessarily protect groundwater. The First Circuit Court of Appeals remanded 40 CFR Part 191 partly on the basis of inconsistency with the Safe Drinking Water Act. Rationale for solely an individual dose limit must conclusively demonstrate why, at Yucca Mountain, groundwater protection is not required or how it will be provided.

Second, use of an individual dose limit could encourage dilution since that could be the easiest method to comply with an individual dose limit. However, this could also lead to many people receiving a small dose. This could lead to a large number of health effects in the larger population. The EPA believes that the goal of these standards is to protect life and the environment as well as to assure that no one person is excessively exposed. A standard which keeps individual risk levels low but results in a large number of health effects is not adequately protective. Advice to develop an individual dose standard must consider the resulting risk to the population and how it would allow no more than an acceptable number of health effects.

Finally, an individual dose standard cannot easily take human intrusion scenarios into account. The existing 40 CFR Part 191 individual protection requirements do not include human intrusion mainly because the intent of the containment requirements is to keep the waste confined to the original emplacement location. This means that an intruder would likely receive very large doses, certainly larger than any reasonable dose limit. Such a high limit would not be protective for nonintrusion scenarios. Further, a reasonable limit would not allow for any intrusion events. A circumstance which the Agency currently believes is unrealistic.

If the NAS answers the question with a "yes," there are a number of issues the Agency would like addressed:

define the scientific basis for determining a "reasonable standard of protection" and explain why an individual dose standard provides it;

explain in detail the approach to setting the individual dose standard that is being suggested;

explain why protection of groundwater is either unimportant or irrelevant;

explain if Yucca Mountain is unique or if the NAS is generally endorsing dilution as an acceptable method of waste disposal; and

how human intrusion should be considered in relation to an individual dose stands.

Energy Policy Act Question B

The Agency believes that active institutional controls cannot be assumed to exist, not to mention provide significant protection, for long periods into the future. The concept of active controls assumes the direct and continuous involvement of governments or societies. No human institution has lasted through all of recorded history, approximately 5,000 years, which is only half the time of the 40 CFR Part 191 standards. Some monuments have survived for long periods but the governments that existed at the time they were built have not. History has many examples of powerful governments which, at their time, appeared invincible. For example, the Roman Empire, the Chinese dynasties, and, more recently, the Soviet Union.

In the relative short term, even over tens of years, economies and priorities, and therefore, budgets and government programs change. Many large projects in the United States have not been finished such as the breeder reactor and the supercollider. If these ambitious projects have not been finished, how can it be assumed that a government or society, even if they could exist that long, would support a program which should show no results or activity for hundreds or thousands of years?

Energy Policy Act Question C

The EPA believes that te answer to this question is yes. Many different approaches could be taken to form a reasonable regulatory basis. For example, one could assume that future drilling rates will continue as they have in the past. This would be similar to an approach which has been widely discussed with regard to future human beings and society for purposes of calculating releases of radionuclides and resulting doses, i.e., that they will remain the same in the future as they are now. To answer the question, NAS should define the phrase "scientifically supportable predictions of probability" and address the scientific bases of the regulatory assumptions necessary to implement the answer. If the NAS's answer to the question is "yes," there needs to be an explanation of how such estimates could be done. If the NAS's answer is "no," there should be an explanation of the scientific basis for

regulatory treatment of human intrusion and a recommended process for estimating its expected rate.

STATUS

The Agency is not as far along toward the establishment of the Yucca Mountain standards as had been planned. The report from the NAS has not been received. Until the time that it is received, it would not be responsible of the Agency to proceed vigorously. The report is expected to be received in April. The major activity thus far has been general discussions of procedures and preliminary steps necessary within the Agency to begin the rulemaking.

FUTURE STEPS

The EPA is planning to study and evaluate the NAS report upon its receipt in order to assess the need for additional work and related analytical support. There will also be an inter-EPA-office work group. The work group is the staff-level method used by the Agency to provide input from EPA program offices. It is expected to include personnel from the Offices of Water; Solid Waste and Emergency Response; General Counsel; Regional Operations and State/Local Relations; Research and Development; Policy, Planning and Evaluation; and Enforcement and Compliance Assurance. In addition, the Agency will involve interested parties from outside of EPA early in the process. The exact mechanism for this has not been decided but could include public meetings, early drafts of potential standards, or one-on-one or small group meetings.

Expressions of interest are solicited anytime by writing to the address given at the top of this paper.

Prior to issuing final standards, the Agency will publish proposed standards, in the Federal Register, for public comment. This will be followed by public hearings in the vicinity of Yucca Mountain. After considering all comments received, the final standards will be promulgated. Until the NAS report has been received, the Agency will not know what its schedule will be for issuing the standards. However, it will do so in the most expeditious manner possible while taking into account all points of view, the amount of resources available, and the technical and legal bases. The Energy Policy Act set a year from the receipt of the NAS report to accomplish promulgation. This will be our goal.

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9-4

PROTECTION OF MAN AND NATURE

Some Criteria for Acceptance of Nuclear Waste Repositories.

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ABSTRACT

The Swedish Radiation Protection Institute, SSI (Statens Strlskyddsinstitut), presents tentative criteria which will lead to promulgation of regulation for the disposal of high level waste.

In defining these criteria a general goal is that man and the environment must be ensured adequate protection today and in the future, including a distant future to account for the necessary protection of long-lived nuclides. SSI employs the general principles of the International Commission of Radiological Protection, ICRP, i. e. Justification, Optimization and Dose Limitation. The Institute considers that

optimization must refer to the whole chain of waste handling, which implies that the quantity which must be minimized include both doses to personnel and to the general public in a distant future.
Sub-criteria including various dose limits in existing Swedish legislation must be met. The dose limit for personnel is 20 mSv/a averaged over 5 years (50 mSv in a single year), the dose to an individual from the public must be below 0.1 mSv/a, and additional global collective dose criteria must be met (1 mSv = 100 mrem).
Protection of the environment must focus on protection of biodiversity. Protection of biodiversity is not restricted to well known endangered species, but also to the intricate web of interactions, nutrition and energy flow etc of which the ecosystem is built. Biodiversity can be assumed to be ensured in conditions where the organisms in the ecosystem are protected. To achieve this level of protection, it is usually assumed adequate to protect populations rather than individual organisms. To ensure protection of populations, however, criteria may have to be defined that will protect individuals. This is also relevant in cases of marginal populations with limited gene flow from the population's main habitat, endemic species, certain domestic or crop species, etc.
The long-term performance for a nuclear waste repository described in the licensing application should be judged by the treatment in 3 phases, a) the first 1000 years, b) the first 10 000 years, and c) the period after 10 000 years. The first two periods up to 10 000 years must be studied in detail, and special emphasis must be placed on the efforts to protect the environment in the first 1000 years during which time the activity decreases by 99%, largely due to the decay of caesium-137 and strontium-90.

BACKGROUND

The Swedish Radiation Protection Institute, SSI (Statens Strlskyddsinstitut), is responsible for defining criteria for radiation protection in society where several authorities and bodies coordinate their work, such as for nuclear waste handling and disposal in Sweden. The Institute is in the process of defining criteria for nuclear power reactor waste repositories and would like to stimulate the discussion of these issues by presenting some important components of the regulatory system.
SSI is responsible for defining criteria for radiation protection in society where several authorities and bodies coordinate their work, such as for nuclear waste handling and disposal in Sweden.
Pursuant to the Swedish Radiation Protection Act, SSI has already promulgated regulation for radiation protection from nuclear installations
- to personnel in nuclear installations,
- to individuals in the general population,
- for releases to the environment from nuclear power plants, and
- for releases from the Swedish repository for low and medium level waste.

The SSI employs the general principles of the International Commission of Radiological Protection, ICRP, i. e. Justification, Optimization and Dose Limitation. In principle, the decision to accept the practice which produces waste includes an acceptance of the justification of waste handling processes associated with this practice.
The justification principle is therefore not addressed here. The other two principles, Optimization and Dose Limitation, are applicable in the regulation of waste disposal. In addition to this, the latest formulation of the Radiation Protection Act includes provisions for protection of both man and nature.

A GLOBAL VIEW OF OPTIMIZATION

SSI requires that all societal practices using ionizing radiation are carried out so that man and nature are adequately protected. Doses (effective doses) should be held not only lower than stipulated limits but also as low as reasonably achievable. In this optimization, SSI considers doses from the whole fuel chain, today and in the future, and doses both in Sweden and abroad.
It is not enough to focus on one component in the chain, such as waste transport, encapsulation or the long term performance of the repository. Similarly, doses to personnel and to the public must be viewed as an integral whole in the optimization. It is therefore not acceptable - for example - that a change in the construction of spent fuel canisters should imply large doses to the personnel involved in the construction, in order to achieve a hypothetical dose limitation in the far future if a future dose reduction cannot be credibly substantiated.

DOSE LIMITS

Doses to Personnel

Nuclear waste installations and their transport systems must be designed and constructed with the same personnel effective dose limits as for nuclear power plants, that is

- 50 mSv per year,
- 100 mSv for a 5 year period, and
- 700 mSv as a lifetime dose.

This must also include a collective dose limit to personnel of 2 person-sievert per year and gigawatt installed electrical effect. It is now stated that the waste handling and disposal must be included in the same collective dose budget.

Doses to individuals in the general public

Nuclear installations and their transport systems must be designed and constructed so that it is unlikely that the dose exceeds the limit for the dose to the Critical Group. This implies that the protection is source-related and that the dose is less than 0.1 mSv/year. The Critical Group consists of individuals who may be expected to receive the highest dose resulting from a release from a nuclear installation.

In addition to this, the expected global collective dose must not exceed 5 personSv per operational year and per gigawatt installed effect. The collective dose must be integrated over 500 years. This collective dose requirement is added to ensure that the mean global individual dose is not likely to exceed 1 mSv after 500 years production of nuclear power. The long term perspective for releases from a repository is discussed below.

PROTECTION OF NATURE

SSI acknowledges that protection of man indirectly protects species in the important pathways for dose to the Critical Group. SSI feels, however, that this is too narrow a goal for practices which fall under the Radiation Protection Act which specifically requires protection of both man and nature.Since there is little consensus in this area, this section is held as a discussion putting forward examples of potentially useful and acceptable approaches.

General Considerations: Focus on Biodiversity

Expressed in general terms, measures taken by Swedish authorities to protect the natural environment focus on protection of human health, biodiversity, the production potential of biological resources, and the (agri)cultural landscape (1). We propose that emphasis should be put on protection of biodiversity in assessments of impact of geological nuclear waste repositories on the natural environment. Human health is already covered by the criteria discussed above. The term biodiversity (biological diversity) has been defined by the Rio conference on economy and development as "the variability among living organisms from all sources including, inter alia, terrestrial, marine and other aquatic ecosystems and the ecological complexes of which they are part; this includes diversity within species, between species and of ecosystems" (2). The fact that the ecosystem level is mentioned is interesting, since it implicates that not only particular organisms but also the transfer of material, energy, etc that constitutes the ecosystem back-bone are considered important. This highlights the importance of all organisms, not only certain conspicuous or in other respects publicly "interesting" species.

With "biological resources" is understood, again in the terminology of the Rio documents, the "genetic resources, organisms or parts thereof, populations, or any other biotic component of ecosystems with actual or potential use for humanity" (2). Biological resources are intimately coupled to biodiversity in a number of ways; biodiversity contributes to genetic resources of potential use, and the production potential of the natural environment as well as its capacity to withstand or recover from natural stress conditions is usually considered correlated to biodiversity. The (agri)cultural landscape is somewhat ambiguous to include in assessments of impact of geological repositories on the natural environment, also when considering the time-scale involved. However, in diverse uses of arable land, the concept of biodiversity is partially applicable, and the agricultural landscape is also considered in the exposure pathways to man. In conclusion, it appears that biological diversity has general relevance in environmental impact assessments in relation to nuclear waste repositories.

SSI is well aware, however, that biodiversity can not in all situations be considered an ultimate goal for efforts directed to environmental protection. It would seem possible that increased levels of ionizing radiation in the environment could increase biological diversity while at the same time being detrimental to

human health. Furthermore, there is no interest in protection of organisms that are in some way (e.g. for health reasons or on economic grounds) harmful to man. Finally, monitoring - or prediction - of biodiversity are non-practicable options when defining criteria for nuclear waste repositories. Measures directed to protection of the natural environment will rather have to ensure that the conditions for maintained biodiversity remain favorable.

Target Organizational Level

Biological life can be arranged in a hierarchy according to the level of organization, ranging from the molecular level to ecosystems. Although distinctions are not straight-forward in all cases, it appears justified to define a target organizational level for actions taken to protect the natural environment, so that higher organizational levels, by inference, should be automatically protected. The traditional view (3) is that for non-humans, populations should be protected. However, populations may be small, isolated, or have restricted gene exchange with other populations of the same species. Under such circumstances, protection should be directed towards individuals. We propose that rather than using different organizational levels for different organisms, the individual organism should be generally considered as target organizational level.

Criteria: is There Justification for Quantitative Criteria in Protection of the Natural Environment?

Quantitative criteria are helpful in assessments of environmental impacts of nuclear installations, and may also aid in defining the area around a waste repository that is likely to be significantly affected. Ideally, the area influenced by a repository should during the operational and post-closure phases not exceed the physical boundaries of the repository itself, i.e. the damage to natural life in the long run should not exceed that caused by the construction per se. Predicted harmful effects outside this area could, depending on magnitude of area and predicted effects, be interpreted as signs of warning or even failure of the chosen storage concept. Quantitative criteria are accepted elements when discussing protection of man, and are supported by a large body of scientific information that enable calculation of risk factors for e.g. cancer and genetic disease. In setting quantitative criteria for protection of nature, the goal(s) cannot normally be the same as for man. It is also evident that radiation sensitivity of the vast majority of the probably 30 to 40 million non-human species that exist on earth is by no means as well-documented as it is for man. However, it can be argued that the material is solid enough to indicate a range of biological sensitivities among organisms and developmental stages that would be of relevance for setting criteria. It is unlikely that these ranges will be substantially modified with more research into the subject, although increased insight in these problems would be interesting in its own right.

Targets for Criteria

The effect on the abiotic environment can be assessed using the environmental increment as a criterion. This defines the increment in radiation from individual nuclides in relation to the natural occurrence and variation, or inferred variation, of these nuclides in the environment (4). In cases where negligible environmental increments can be predicted, further assessments may be unnecessary. If required, further assessments will have to be based on predicted doses to generic target organisms occupying soil, water, air or vegetation, largely in accordance with an assessment concept developed by AECL (3). The choice of generic target organism has to be justified by the applicant and/or operator of the repository.

Setting Quantitative Criteria

A view on quantitative criteria could be developed along the following lines. Although variation between organisms and developmental stages is enormous, experimental data suggest that doses in the range 1 to 5 Gy per year are unlikely to cause observable detrimental effects in most organisms. It is proposed that a safety factor of between 100 and 1000 should be used to account for the fact that experimental studies usually do not consider the possible effects of long-term low-level radiation (consequently, risk estimates are scarce), and also to account for a reasonable compliance with the precautionary principle. This reasoning would lead to the conclusion that a quantitative criterion should be set within the range 1 to 50 mGy per year. A working hypothesis at the present is that 20 mGy per year could be possible to use as a quantitative criterion when assessing the feasibility of nuclear waste repositories.

Comparison can be made with "limits" set by the Swedish Radiation Protection

Institute for doses received by man caused by operation of nuclear facilities. As mentioned above, these are 0.1 mSv per year for individual members of the critical group and 20 mSv per year as a five-year average for occupational exposures.
It is interesting to note that the range thus defined corresponds to the maximum expected doses to organisms implied by a the current ICRP standard of a maximum dose of 1 mSv annually to members of the critical group (3). However, while protection of man will often protect other species, we feel that the applicant should go to reasonable length to prove that most species are adequately protected. Although further analysis is required, the standard set for man and discussed for organisms in the natural environment in this communication seem compatible.

THE LONG TERM PERSPECTIVE

Radiation protection must be similar in all areas in society. This implies that the same protection principles must apply to future and present generations.
For future releases from the repository, after closure, a discussion must be held addressing different time periods separately, as given below. For reasons relating to the decay of the different redionuclides these periods cannot be treated the same way in the analysis. The judgement of this analysis will take into account the uncertainty in the calculations which increase with distance to the discussed time period. For periods reaching more than 1000 years into the future, SSI considers that the protection philosophy must to the extent possible follow protection principles established for other genotoxic agents.

a) The First 1000 Years

This is the most important period considering the waste's potential danger. Special consideration must be given to assessment of the protection of man and the environment the first 1000 years after closure. The assessment must include the protection for people living near the repository site.

b) The Time Period from 1000 Years After Closure to the Next Expected Ice Age

The next epoch range from 1000 years up to the time for the next glaciation, that is of the order of 10 000 years (depending on where in Sweden the site is located). In this time perspective the nuclides caesium-137 and strontium-90 are short-lived; they have for all practical purposes decayed along with other nuclides with shorter half-lives. The activity from long-lived nuclides is large enough to warrant isolation from the environment during many thousands of years. With the distance to the time concerned, the uncertainty increases considering the barriers, society and nature. After a few 1000 years the Scandinavian peninsula may resemble a cold tundra and the beginning glaciation may be evident in the northern areas.

c) The Time Period from the Next Glaciation Onwards

The problems of assessing the repository's environmental impact are fundamental during this period. For this interval, SSI sees several difficulties in describing the repository quantitatively and in detail. Primarily, fundamental limitation of science as a predictive tool stand in the way of a credible detailed and quantitative description.
SSI will require quantitative analysis for the epochs a) and b) of dose to the critical group under different conditions. The calculations must include the environmental impact using biosphere modeling using the state of the biosphere prevailing at the time of closure. Assessments of doses to the critical group must be made using best, not over- or under-, estimates. The use of the period 10 000 years as the main time period for quantitative judgement is in line with views and regulations held by other environmental protection authorities, for instance in the USA, Canada and Germany. For epoch c), during and after the next glaciation of Scandinavia, a description must be made of the repository. The large uncertainties for any assessment during this period must also be described and SSI judgement of that description will be largely qualitative.

INTRUSION AND ACCESS

The method of waste isolation implies a risk to an intruder who may receive a dose in excess of existing dose limits. The possibility of a high dose cannot be used in the assessment of radiation protection from the repository, since it flows directly from the strategy of waste isolation.
SSI holds that there is no alternative to the principle that future societies must take responsibility for their own conscious actions. A conscious intrusion into the repository cannot be prevented. No actions should be taken to facilitate or prevent intrusion if such actions should impair the protection capability of the undisturbed repository.

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9-5
ANALYSIS OF SAFETY CRITERIA FOR HIGH LEVEL WASTE DISPOSAL

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ABSTRACT
The suitability of a deep geological formation for radioactive waste disposal mainly depends on its isolation potential. In Germany, radiological safety is expressed by a individual dose limit of 0.3 mSv/y for all reasonable radionuclide release scenarios. Parallel to the geological exploration program at the Gorleben site, BGR is currently evaluating a catalog of site selection criteria for alternative host rock formations. The purpose of the paper is to update the procedure for demonstrating the suitability of different types of host rock, and to explain specific steps of the site selection and the geological exploration program. Both geological host rock formations under consideration, that is a domal salt formation as a potential encapsulating system, and the granitic rock as a typically fractured formation can be explored in a way that an appropriate design for the repository can be established to satisfy the basic radiological safety criterion. The natural barrier of an intact salt formation provides a self-sealing system. The isolation potential of the crystalline formation depends on its low permeability. The geotechnical barriers as part of the multibarrier system become more important for crystalline rocks. The different role of the natural barrier in both host rocks requires an adequate geological exploration process.

INTRODUCTION

In accordance with the salient objectives for the disposal of high-radioactive wastes, namely (1):

to isolate radioactive wastes from the biosphere over long time scales without giving the responsibility to future generations to maintain the integrity of the disposal system, or imposing upon them significant constraints due to the existence of the repository, and

to ensure long-term radiological safety, that is to protect humans and the environment against inadmissible radiation in accordance with agreed radiation protection principles

world-wide national nuclear waste programs prefer a concept in which heat-generating high level wastes are considered to be disposed of in deep geological formations. The suitability of a particular site is achieved if the entire repository system fulfills the above mentioned protection aims. These objectives imply that no inadmissible release of radionuclides to the accessible environment may occur. The repository system therefore in particular must guarantee that its barrier function is able to prevent transport of radionuclides into the biosphere.

The necessity to keep risks associated with practices involving the radiation exposure of humans to acceptable levels forms the basis for a system of dose limitation developed by the International Commission on Radiation Protection (ICRP) (2). This system has been incorporated in the IAEA Basic Safety Standards (3) and has been accepted by many national authorities.

In Germany, the Reactor Safety Commission (RSK) has published recommendations on safety criteria for the disposal of radioactive wastes in a repository mine (4). With the understanding, that the safe design of the geological repository is determined by the site specific non-standardized geological situation the recommendations only establish the protection aims. These protection aims have to be converted into geological and geotechnical criteria for the site under investigation. With respect to the radiological safety as the ultimate protection

aim in Germany, all reasonable release scenarios must not lead to individual doses exceeding the limit of 0.3 mSv/y. This is the only quantitative criterion given in the recommendations. Willingly and intentionally, these recommendations allow for a margin of expert judgment. No formal procedure is specified. On the contrary, the realization of compliance with site specific criteria is demanded in the context of the application documents submitted by the applicant to the licensing authority. These application documents must therefore be appropriate to the current state-of-the-art and must be in agreement with the scientific knowledge at the time of submission of the documents.
This requirement implies that design supporting R&D-work has to be carried out parallel to the elaboration of the application documents. A research based design and in particular a research based demonstration of safety is requested by this demand.
In Germany, a comprehensive geological exploration program has been performed to assess the suitability of the Gorleben salt dome as a permanent repository for radioactive wastes. As a supplement, the Federal Institute for Geosciences and Natural Resources (BGR) is currently evaluating a catalog of site selection criteria for alternative host rock formations (crystalline rocks) for the disposal of high level wastes.
In accordance with the above mentioned principle, and to compare the different requirements for different host rocks, the purpose of the paper is:
to state our current knowledge about site selection,
to update the procedure for demonstration the suitability of a particular site, and
to explain how radiological safety criteria give the basis for geological and geotechnical criteria, and shall be converted to specific steps of the geological exploration program.

MAIN CRITERIA FOR SITE SELECTION IN GERMANY

Final disposal of heat-generating high level radioactive waste in Germany requires specific site selection criteria which are supposed to be applied for both rock salt and crystalline rock as potential host rocks. In principal the disposal drifts of the repository have to be excavated about 500 m deep in the host rock formation not exceeding the depth of 1200 m because of the expected higher rock temperature.
During the preselection process criteria that exclude a site from further consideration have to be examined (basic requirements in Fig. 1). Considering the situation in Germany, we can roughly distinguish between the following basic requirements:

Ecological factors:
- protected areas
- industrial areas and overcrowded regions
- areas with large water reservoirs
- areas with unsuitable hydrogeological prerequisites,

geologic factors:
- zones of tectonic and seismic activity
- high density of fractures
- zones of high vertical and lateral movements
- zones of recent or expected magmatism,

geotechnical factors:
- insufficient area (< 10 km)
- intense open pit mining and/or underground mining
- significant change of host rock properties.

All regions and areas which have not been excluded after application of the preselection factors are declared to be potentially suitable sites and have to undergo a subsequent process of site selection. This mostly host rock specific process requires two additional categories of criteria allowing the statement of a first hierarchic sequence of potentially suitable sites:
Criteria making the site favorable for further consideration (Cat. A) and
criteria permitting the site to be considered, but with reservations (Cat. B).
Whereas the category A - criteria are reflecting directly on the positive properties of a potential site, the B - type criteria require either an improvement of the isolation capacity by engineered barriers or have to be adjusted due to the results of further investigations.

BARRIER CONCEPT

The disposal of radioactive wastes into deep geological formations claims credit of the host rock to provide a sufficient barrier against the biosphere. However, the shafts as entries to the underground repository, perforate the natural barrier.

Therefore, one cannot rely on the only functioning of the natural barrier rather than on a multibarrier system (Fig. 2). The multibarrier system becomes even more important if the natural barrier given by the host rock and the geological formation has local imperfections or discontinuities with unacceptable permeability. However, it is the common understanding of competent experts that a multibarrier system in principle can be designed in a way that the release of radionuclides through the multibarrier system will be limited to the admissible dose rate even over long time frames. Obviously, the guarantee for the multibarrier function requires both a comprehensive knowledge of the properties of the host formation and qualified measures to build engineered barriers such as the waste package or various seals.
The overall disposal system consists of both the natural barrier and the engineered barrier. The properties of the natural barrier like mechanical integrity and permeability have to be explored. Since these properties can only be improved to a certain extent, local weakness of the natural barrier demands for special design requirements with respect to the engineered barriers. From this viewpoint it is worthwhile to distinguish between two different natural barrier systems, namely a salt type and a granite type.
Fig. 3. Natural barrier systems
The long-term tightness of intact rock salt is proven through the encapsulation of gases and fluids in salt formations over very long time. The natural barrier of a repository in an intact salt formation therefore may act as an encapsulating system, and may provide the ultimate barrier with the shaft sealing as the only important engineered barrier in this case.
The natural barrier of a repository in a granite formation, which is always fractured, will be incapable of protecting the waste from fluids over long time. However, a granite formation may possibly have large blocks of very low permeability which is sufficient to satisfy the protection aims in connection with engineered barriers.
Deep geological disposal has its advantages also in the case of a fractured natural barrier, namely with respect to the following aspects:
- The waste is disposed in areas sufficiently far away from major disruptive processes.
- Rates of natural processes affecting the waste are slow.
- Engineered barriers are protected.
- Radionuclide migration is retarded.

SAFETY ASSESSMENT
The methodology of the safety analysis with particular regard to the post-operational period of high level waste repositories, in our view, has already its impact on the site characterization procedure and therefore also on the judgment of suitability of a potential repository site. The assurance of compliance with safety objectives requires a dualism of safety demonstration (proof) (5), namely:
a) demonstration (proof) that the required quality (integrity) of the host rock is effective,
b) demonstration (proof) that risk conditions (scenarios) with respect to the barrier efficiency are non-evident or acceptable at a residual risk level.
Furthermore, the scenario analysis involves the identification of phenomena and processes which could initiate and/or influence the release and transport of radionuclides from the waste to man. Obviously, this is one of the major challenges of site specific geological exploration. Likewise, this is also valid for the demonstration of the efficiency of engineered barriers. With this in mind, one can question the differences with respect to the foundations of the scenario analysis either for a homogeneous formation (encapsulating systems, e.g. given by a diapiric salt formation), or a fractured formation (open systems containing blocks of low permeable rock, distinct water paths and extended engineered barriers).
For a repository in a salt diapir at respective depth there is evidence for the existence of a tight geological barrier all around the planned repository. It is therefore of great importance to prevent damage of this natural barrier during the exploration of the site. If this can be achieved, only the serviceability (stability) of the intact salt barrier over the lifetime of the repository as a prerequisite for its tightness has to be analyzed. This can be done by a stability computation.
Starting from a well characterized initial state of the natural barrier all further impact on the integrity of the salt barrier can be studied in a deterministic

analysis. The stability computations have to demonstrate that the important properties of the natural barrier remain unchanged. The analysis of system behavior can even be based on conservative assumptions, e.g. regarding events with random occurrence. A probabilistic analysis becomes not necessary. Probabilistic features, however, can be applied within the deterministic analysis to study the sensitivity of parameter uncertainty (6).
If we have to consider the barrier behavior of fractured rock the deterministic analysis cannot sufficiently be applied. Since it becomes already impossible to completely describe the exact initial spatial distribution of rock permeability, not to mention the mostly time dependent alteration due to termomechanical impact, probabilistic analysis including its inherent greater difficulties has to be applied.

GEOLOGICAL EXPLORATION

The ultimate objective of site-specific geological exploration is the determination of the suitability of a particular site for the purpose of nuclear waste disposal. In order to be able to fulfill this task the following sub-objectives are to be met (7):

- development of a geologic model for the host formation and the geologic environment based on a documentation and description of all relevant geological features,
- evaluation of all relevant data to perform safety assessment analysis, and identification of further geologic scenarios affecting the barrier function of the host rock.

As a consequence of the previous discussion, the methodology for site-specific geological exploration will differ with respect to the basic behavior of the host formation:

- rock salt as a typically encapsulating system,
- fractured rock mass as a typical permeable system.

Exploration of a salt diapir as an encapsulating system has to be carried out under the strict demand of a protection of the natural barrier. As an advantage, in principle only one set of data for one homogeneous host formation has to be provided. Certainly, the possible scatter of data and its origin has to be determined and taken into account for further investigations. As an example, for the Gorleben site in Germany, only a few slightly differing homogeneous domains have to be anticipated within the Zechstein 2 formation. Therefore, the exploration with respect to geotechnical data aims at confirming the expected data and identifying abnormal domains.
From a geological point of view, a main effort during the exploration at the Gorleben site will be taken to define the extension of the favorable Zechstein 2 formation against the extremely folded anhydrite structure and to ensure a safe distance to the brittle and potentially permeable anhydrite. Reliable measures to detect the anhydrite layer during exploration, like detailed mapping and geophysical testing are available.
Unlike the principle of the encapsulating system in rock salt, the function of the natural barrier in crystalline rocks is based on the assumption of a low permeable host rock and means direct consequences to the objectives of the exploration program. The main objectives of the site selection in crystalline rocks consist of searching, identifying and exploring already existing fixed natural conditions which satisfy the demands of the disposal concept (natural barrier). On the contrary to rock salt, the engineered barriers as a more important contribution to the multibarrier system have to be developed and improved according to the particular geological situation.
Although only 10% of all fractures are thought to be water-conducting, the disposal drift geometry has to be designed with respect to their density and spatial distribution. Exploration of a fractured rock mass is therefore characterized by the necessity to identify all major release paths (design controlling fractures), and to find domains of sufficiently low permeability. The basic aim of an investigation program is to create a conceptual model of the potential site (8). Geological criteria adapted to that include the following identifications:

- the size and depth of the rock formation,
- blocks of low permeability (solid host rock formation) bordered by fracture zones, major water-conducting faults,
- local heterogeneities and water-conducting features with their orientation and

their hydraulical and mechanical properties.
In addition, characterizing of the hydraulic parameters of the host formation dominates the evaluation of data. Since a fractured formation requires a probabilistic safety analysis, statistical data have to be elaborated so that a probabilistic model can be developed on that basis.

CONCLUDING REMARKS

In Germany, the basic requirements for the preselection of a site for disposal of heat generating high level waste in deep geological formations are established as valid for any type of host rock.
The subsequent site selection process, however, is dependent on the particular geological situation and therefore on the properties of the different types of host rock. Both geological host rock formations under consideration, that is a domal salt formation as a potential encapsulating system, and the granitic rock as a typically fractured formation can be explored in a way that an appropriate design for the repository can be established to satisfy the basic radiological safety criterion.
For domal salt in Germany, the geological/geotechnical criteria are well developed. Adequate criteria for a crystalline formation in Germany have still to be established in more details.
In principal, the isolation potential of rock salt and crystalline formations is very different. The natural barrier of an intact salt formation provides a self-sealing system, whereas the isolation potential of the crystalline formation depends on its low permeability. Therefore, the geotechnical barrier as a part of the multibarrier system becomes more important for crystalline rock.
The different role of the natural barrier in both host rocks requires an adequate geological exploration process. The main objectives of the exploration are to collect all necessary informations for the development of a geological model, and to evaluate all relevant data for the safety analysis. However, for a salt diapir the exploration aims must coincide with a careful protection of the salt barrier. The exploration of a crystalline rock, on the contrary, is concentrated on the identification of insufficiencies of the natural barrier and its spatial distribution.

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9-6

PROPOSED WIPP COMPLIANCE CRITERIA (40 CFR 194)

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ABSTRACT

The Environmental Protection Agency (EPA) is in the process of promulgating compliance criteria for the Waste Isolation Pilot Plant (WIPP). The criteria, which are officially titled "Criteria for the Certification and Determination of the Waste

Isolation Pilot Plant's Compliance with Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," will be codified at 40 CFR 194. The proposed compliance criteria was signed by the Administrator on January 11, 1995 and is expected to be published in the Federal Register in late January 1995. During the presentation of this paper, EPA will offer insight into the different points of view on the key compliance issues.

BACKGROUND

EPA regulates the release of radionuclides from the management, storage and disposal of radioactive waste in order to protect public health and the environment. The Waste Isolation Pilot Plant, under development by the U.S. Department of Energy (DOE), is a potential geologic disposal system for defense transuranic (TRU) radioactive waste. The WIPP is a mined salt repository 2100 feet below the surface in southeastern New Mexico, near the city of Carlsbad. It is designed to hold 6.2 million cubic feet of waste, which translates to about 800,000 55-gallon drums. Under the 1992 WIPP Land Withdrawal Act, EPA is required to perform several activities including, but not limited to: (1) finalizing safety standards for radioactive waste disposal, (2) issuing criteria for judging whether the WIPP complies with the radioactive waste disposal standards, (3) certifying whether the WIPP complies with the standards before waste disposal can begin, and, if EPA certifies that the WIPP complies with the disposal standards, (4) determining whether the WIPP continues to be in compliance every five years after initial receipt of waste and throughout the waste emplacement phase.

In December 1993, EPA accomplished the first of these activities by issuing the final radioactive waste disposal standards (40 CFR 191), which place limits on the releases of radionuclides from waste management, storage and disposal facilities. In January 1995, EPA issued proposed compliance criteria (40 CFR 194), which will implement the 40 CFR 191 disposal standards specifically at the WIPP. Since the WIPP is a first-of-a-kind facility, and the most likely to be considered for radioactive waste emplacement in the near future, EPA has taken extra steps to ensure that diverse viewpoints are presented, stakeholder input is considered, and relevant data are analyzed.

The promulgation of the compliance criteria represents the first time the Agency is implementing 40 CFR 191. EPA's approach at the WIPP may set precedent for future approaches taken at other radioactive disposal facilities. Many of the nuclear waste issues that are of considerable concern to the public will be addressed in this EPA rulemaking.

IMPLEMENTATION

EPA's proposed compliance criteria are specific to the WIPP and its compliance with the disposal standards found in subparts B and C of 40 CFR 191. The primary goal of the criteria is to make compliance at the WIPP as straightforward as possible. The criteria are also aimed at clarifying the requirements of the radioactive waste disposal standards and explaining the procedural aspects of EPA's certification or determination of compliance at the WIPP. The proposed criteria are intended to support a "reasonable expectation" of compliance and include four subparts: General Provisions, Compliance Certification and Determination Applications, Compliance Certification and Determination, and Public Participation.

Subpart A -- General Provisions

The General Provisions address the purpose, scope and applicability of the criteria, definitions of terms not already defined in 40 CFR 191, communications, and conditions of compliance certification and determination, which state that any certification or determination of compliance is subject to modification, suspension or revocation by EPA for cause. This provision requires DOE to inform EPA of changes in the disposal system and notify EPA of releases or expected releases.

Subpart B -- Compliance Certification and Determination Applications

This subpart discusses the general format and content of compliance applications. For example, DOE is required to provide a complete description of the disposal system, input parameters, waste acceptance criteria, climate conditions, background radiation, results of compliance assessments, topographic maps, and other information that demonstrates that the compliance application meets the requirements of the compliance criteria. This subpart also specifies the requirements for subsequent compliance determination applications, which must be submitted every five years if initial compliance certification is granted. In general, DOE must provide EPA with information regarding monitoring results, a description of waste emplaced

in the disposal system, and changes regarding the disposal system's performance.
Subpart C -- Compliance Certification and Determination
Subpart C can be considered the "core" of the compliance criteria. It outlines, in detail, the technical components EPA expects to see in a complete DOE compliance application. Like the disposal standards, Subpart C of the compliance criteria includes General Requirements, Containment Requirements, Assurance Requirements, and Individual and Ground-Water Protection Requirements. Each of these requirements are further divided into specific sections and discussed below.
General Requirements
The General Requirements include inspections, quality assurance (QA), models and codes, waste characterization, future state assumptions, expert judgment, and peer review. The inspections section provides for EPA access authority to the WIPP, provided EPA personnel follow applicable security guidelines. In performing inspections, EPA may gain access to written records and obtain samples from the facility. In the last few months, Agency staff have been participating in inspection and auditing training courses so they will be prepared to conduct DOE facility inspections when necessary.
A major focus of any auditing/inspection program is quality assurance. The compliance criteria require that all data that support compliance applications be collected in accordance with an approved QA program. The 40 CFR 194 proposal requires DOE to institute a QA program that is equivalent to the American Society of Mechanical Engineers' (ASME) NQA-1, 2 and 3 standards. EPA is further proposing that certain quality indicators be applied to the data, such as accuracy, precision, and comparability.
Models and computer codes will be employed in assessing whether the WIPP disposal system will comply with the 40 CFR 191 disposal standards, which require predictions of disposal system performance over 10,000 years. The compliance criteria state that DOE must provide a complete listing and description of all models and codes employed. The models and codes must be quality-assured so EPA can judge if the conceptual, mathematical, numerical and computer models used to support compliance applications are appropriate.
Waste characterization, another general requirement, has generated a great deal of discussion throughout this rulemaking process. In predicting the WIPP's performance over the 10,000-year time frame, it is necessary to have a comprehensive understanding of the characteristics of the waste proposed for disposal in the repository. The proposed compliance criteria require an analysis of the effects of certain waste characteristics on the containment of waste. A question still unanswered is the level of detail the Agency should require in DOE's waste characterization analysis and what specific characteristics should be evaluated for their significance to containment of waste over the 10,000-year time frame. Another issue before the Agency is the role of "process knowledge," which is the examination of waste generation documentation and associated records. In February 1995, these questions and others will be placed before experts attending an EPA-hosted technical workshop on the compliance criteria rulemaking.
The Agency recognizes that environmental and societal conditions are likely to change in significant and unforeseeable ways over 10,000 years. Because it is impossible to predict such changes, EPA is proposing that, unless otherwise specified in the criteria, future states will be assumed to remain what they are today. This approach will enable compliance assessments to focus on more predictable and significant features of disposal system performance. However, certain characteristics, such as geologic, hydrologic and climatologic conditions will not be held to the future state assumption. Climate change, for example, is a debatable topic slated for discussion during the February technical workshop.
The last two general requirements are expert judgment and peer review. EPA recognizes that in some cases actual data may not be available and, instead, expert judgment may have to be employed to obtain information. The Agency is proposing, however, that use of expert judgment be limited only to those situations where actual data are not reasonably attainable through data collection or experimentation. The compliance criteria describe the requirements of expert judgment elicitations, such as formal documentation procedures and independent panel selection. Peer review, sometimes confused with expert judgment, is a documented, critical review performed by peers who are independent of the work being reviewed. EPA believes that peer review can be helpful in validating information contained in

compliance applications and has specified in the proposed compliance criteria certain instances in which peer review is required.

Containment Requirements

In addressing the Containment Requirements of 40 CFR 191, the proposed compliance criteria articulate methodologies for considering inadvertent human-initiated processes and events and conducting probabilistic performance assessments. The containment requirements limit cumulative releases of radionuclides over 10,000 years and specify that there be less than one chance in ten of cumulative releases exceeding the limits specified in Appendix A of 40 CFR 191, and less than one chance in 1,000 of cumulative releases exceeding ten times those limits. The curie content of the waste inventory proposed to be disposed at the WIPP must then be determined. In the compliance criteria, EPA is proposing that the expected curie activity 100 years after disposal of the waste in the WIPP be used in calculating the release limits. The 100-year mark was chosen because it represents the point at which most of radioactive materials with short half-lives will have decayed to relatively low levels. The remaining waste will be the focus of attention because it is that waste that poses the biggest threat to public health and the environment.

The proposed criteria indicate that performance assessments consider all natural and human-initiated processes and events that may affect the disposal system; however, scenarios that have less than one chance in 10,000 of occurring over 10,000 years need not be considered.

Perhaps the most critical aspect of the compliance criteria is the consideration of human-initiated processes and events. The methodology by which to ascertain the likelihood of inadvertent human actions that may affect the WIPP's performance has been the subject of considerable research and discussion. The proposed approach considers the site-specific characteristics of the WIPP while incorporating many of the assumptions which underlie the containment requirements of 40 CFR 191. The Agency believes that the most productive consideration of inadvertent human-initiated processes and events concerns those realistic possibilities that may be usefully mitigated by disposal system design, site selection, and institutional controls. Therefore, the Agency is proposing that inadvertent drilling for resources (other than those resources provided by the waste in the disposal system or any engineered barriers designed to isolate the waste) be the most severe scenario considered.

Because drilling events do not necessarily have to penetrate the disposal system to have an impact on performance, the human-initiated processes and events considered should not be limited to those drilling operations that actually penetrate to the depth of the waste in the disposal system. For example, since drilling can create new pathways for ground-water flow and alter the geology of the site even if the repository horizon is not penetrated, non-penetrating drilling operations should also be considered.

The Agency is proposing to divide human-initiated processes and events into two distinct categories, "human intrusion" and "human activity," and is proposing a separate process to establish the drilling rate for each. "Human intrusion" includes those drilling events that reach the level of the waste in the disposal system or below, such as exploration for and development of oil and natural gas resources. The second category of human-initiated processes and events, "human activity," includes drilling events that may affect the disposal system, but do not reach the level of the waste in the disposal system. These drilling events may include exploration for potash, withdrawal of water and drilling for other resources. It is important to note that a given resource may exist at levels above and below the level of the waste in the disposal system and may therefore be included in establishing the rates for both human intrusion and human activity.

EPA is proposing that consideration be given to the record of human-initiated processes and events in the Delaware Basin (the largest contiguous area that shares similar geologic and hydrologic conditions with the WIPP site) over the past 50 years. Once the 50-year time period has been considered, EPA proposes that the rate of occurrence for each type of human-initiated process and event be held constant at the rate determined according to the method explained below for the duration of the regulatory time frame -- consistent with the "future state assumptions" discussed earlier.

In assessing the consequences of human-initiated processes and events, the Agency is proposing that the processes and events be assumed to occur at random intervals in

time and space throughout the regulatory time frame. The consequences of each process and event must be calculated in terms of the projected impact on the WIPP disposal system. If more than one process or event is predicted to occur, the consequences of any which occur subsequent to initial ones must be added to the consequences from previous disruptions. This cumulative consideration should be realized to take into account the fact that every drilling event introduces potential changes to the disposal system.

The Agency is proposing different criteria for establishing the frequency of "human intrusion" and the frequency of "human activity." While both are based on the historical record of resource exploration over the past 50 years in the Delaware Basin, an upper and lower limit is placed on the rate of human intrusion, while the rate of human activity is not limited to a set range. The rate of human intrusion is determined by identifying past occurrences in the Delaware Basin over the past 50 years. The sum of the individual rates of human intrusion for each resource then becomes the rate of human intrusion to be used in performance assessments, provided that the sum is not less than 25 and not greater than 62.5 boreholes per square kilometer per 10,000 years. In the event that the calculated total rate is less than 25, then the rate of human intrusion to be used in performance assessments should be adjusted upward proportionally to yield a total rate of 25. Likewise, if the calculated total rate exceeds 62.5, then the rate of each type of human intrusion should be adjusted downward proportionally to yield a maximum rate of 62.5.

By placing an upper and lower limit on the rate of human intrusion, the Agency is adhering to the assumptions made in formulating the containment requirements of the disposal standards as promulgated in 1985. As part of the development of the disposal standards, the Agency estimated the range of future human intrusion and human activity for the general case of a repository in bedded salt, the geologic setting of the WIPP. Assumptions were made about the presence near a repository of different types of resources -- including oil, gas, minerals and water -- though it was assumed that the most significant resources present would be oil and gas. Using drilling data from the contiguous 48 states as a rough guide, the Agency estimated that a region of bedded salt would experience 25 to 62.5 boreholes per square kilometer per 10,000 years.

Should DOE wish to forego the process of analyzing the historical rates of human intrusion events in the Delaware Basin, DOE can assume the maximum rate of 62.5 boreholes per square kilometer per 10,000 years. After the rate is calculated, it may be reduced in accordance with the criteria pertaining to institutional controls; however, in no instance will EPA assume that institutional controls will eliminate the likelihood of human intrusion.

For consideration of "human activity," the historical record of drilling must be examined, but without placing pre-set limits on the rates. The Agency recognizes that for some resources, such as water, use may depend upon the quality of the specific reservoir that is being exploited. A given reservoir of water, for example, may not be of potable quality but may still be usefully withdrawn for controlling dust. Therefore, it may be possible for DOE to demonstrate to EPA that a certain resource found within the controlled area differs in quality and drilling rate from the same resource found in rest of the Delaware Basin. Therefore, when examining the historical record of human activity associated with that resource, only that human activity that has been associated with resources of quality similar to that found within the controlled area need to be considered. As in the case of human intrusion, the rate of human activity may also be reduced in accordance with the criteria pertaining to institutional controls.

In assessing the results of performance assessments required under the containment requirements of 40 CFR 191, the Agency is proposing to require that the results be displayed as complementary cumulative distribution functions or "CCDFs." These CCDFs should display the releases of radionuclides and their probability over 10,000 years after disposal. In developing the radioactive waste disposal standards, EPA recognized that because of the long time period involved and the nature of the processes and events of interest, there will inevitably be substantial uncertainties in projecting disposal system performance. In conducting performance assessments, there will be many parameter values that can affect the results. Therefore, the Agency is proposing to require the development of probability distributions for parameter values which represent the probability of different values of the parameter occurring.

Assurance Requirements
The quantitative containment requirements set forth in the Agency's radioactive waste disposal standards are accompanied by a set of qualitative requirements to further assure that the desired level of protection will be achieved. EPA included these "assurance requirements" in the disposal standards to complement the numerical requirements because they can provide additional confidence in the performance of disposal systems. The first of these assurance requirements is active institutional controls. In accordance with 40 CFR 191, active institutional controls over disposal sites should be maintained for as long as possible after disposal; however, performance assessments cannot consider contributions from active institutional controls for more than 100 years after disposal. The proposed compliance criteria require that any application for certification of compliance contain detailed descriptions of proposed active institutional controls, their location and the period of time they are proposed to remain active. In addition to active institutional controls, 40 CFR 191 requires that "disposal systems shall be designated by the most permanent markers, records, and other passive institutional controls practicable to indicate the dangers of the wastes and their location." The standards define passive institutional controls as "1) permanent markers placed at a disposal site, 2) public records and archives, 3) government ownership and regulations regarding land or resource use, and 4) other methods of preserving knowledge about the location, design and contents of a disposal system." The Agency is proposing in the compliance criteria that any application for certification of compliance include detailed descriptions of the measures that will be employed to preserve knowledge about the location, design and contents of the disposal system.
A difficult question facing EPA is the extent to which the Agency should allow performance assessments to consider contributions from passive institutional controls in reducing the likelihood of human-initiated processes and events that may affect the disposal system. While the disposal standards address contributions from active institutional controls (up to 100 years), they do not specifically address contributions from passive institutional controls. The Agency may be willing to consider contributions if a persuasive case can be made that the passive institutional controls can be expected to endure and act as a deterrent to potential intruders for 10,000 years. Because of the uncertainty concerning the effectiveness of passive institutional controls in terms of influencing human activity, EPA must carefully scrutinize information about the controls. The Agency has considered the fact that markers exist in the world today that are thousands of years old. This would tend to support the view that passive institutional controls can survive for very long periods of time. Nevertheless, it is possible that markers have been created in the past and were destroyed or disintegrated. The actual percentage of surviving markers is thus unknown. Further uncertainty in the effectiveness of markers derives from the possibility that even if markers survive, they may not necessarily be understood by future generations.
Evaluating the effectiveness of institutional controls in deterring future human actions is one of four key compliance issues which will be addressed at EPA's planned Technical Workshop in February. In addition, EPA will continue to participate in key international discussions convened under the auspices of the Organization for Economic Co-operation and Development's Nuclear Energy Agency regarding consideration of future human actions at geologic disposal sites and the role of institutional controls in deterring such actions.
Since the predictions associated with long-term compliance with the disposal standards of 40 CFR 191 are inherently uncertain, a provision was included in the disposal standards which requires monitoring of disposal systems to confirm that they are performing as predicted. The proposed compliance criteria require that the WIPP be monitored after disposal (i.e., when all of the shafts are backfilled and sealed) to detect substantial and detrimental deviations from expected performance. The monitoring must be done with techniques that do not jeopardize the isolation of the wastes. EPA is proposing that monitoring programs be designed to detect the movement of radionuclides toward the accessible environment at the earliest practicable time. Any compliance certification application must include a detailed plan for monitoring the performance of the WIPP after disposal. At a minimum, this plan must identify parameters that will be monitored and how baseline states will be determined; indicate how each parameter will be used to evaluate the performance of the disposal system; and discuss the length of time over which each parameter will

be monitored to detect deviations from expected performance.
EPA is proposing that pre-closure monitoring of disposal system parameters be conducted to establish baseline performance. The Agency believes that baseline monitoring can provide important information about the disposal system and that such information can be used to verify the assumptions which form the basis of a compliance assessment. Since there may be additional disposal system parameters important to the containment of waste, EPA is proposing that DOE undertake a study to determine the effect of various disposal system parameters on the performance of the disposal system.
The assurance requirements of 40 CFR 191 require that disposal systems use different types of barriers to isolate the wastes from the accessible environment. 40 CFR 191 defines the term barrier as "any material or structure that prevents or substantially delays movement of water or radionuclides toward the accessible environment." For example, a barrier may be a geologic structure, canister, waste form, or material placed over and around waste. If designed properly, engineered barriers can significantly reduce the potential for waste migration away from the disposal system. While the disposal standards require use of engineered barriers, they do not specify how many or what kinds of engineered barriers must be used. The Agency is, therefore, proposing WIPP compliance criteria for selecting engineered barriers. Specifically, EPA is proposing that DOE complete a study of engineered barrier alternatives and their benefits and costs. The results of the study will be used to justify both the selection and rejection of engineered barriers at the WIPP.
EPA's disposal standards require that resource-rich places be avoided in selecting disposal sites. Resources to be considered include minerals, petroleum or natural gas, valuable geologic formations, and ground water that is either irreplaceable because there is no alternative source of drinking water available for substantial populations or that is vital to the preservation of unique and sensitive ecosystems. The criteria go on to state that sites potentially rich in resources may not be used for disposal of radioactive wastes unless the favorable characteristics compensate for the unfavorable characteristics, such as the greater likelihood of being disturbed in the future. EPA is therefore requiring DOE to demonstrate that the favorable characteristics of the WIPP compensate for the presence of resources and the likelihood of human-initiated process and events that may occur as a result of the presence of those resources.
The last assurance requirement included in the 40 CFR 191 disposal standards involves the removal of waste from the disposal system. Specifically, 40 CFR 191 mandates that disposal systems should be selected so that removal of most of the wastes is not precluded for a reasonable period of time after disposal. In order to address this requirement, EPA is proposing criteria that require a plan for removing waste from the disposal system using the best technology available at the time of application.
Individual and Ground-Water Protection Requirements
The individual protection requirements of 40 CFR 191 limit committed effective doses of radiation to members of the public to no more than 15 millirem per year. The ground-water protection requirements limit releases to ground water to no more than the limits set by the maximum contaminant level for radionuclides established in 40 CFR part 141 under section 1412 of the Safe Drinking Water Act (SDWA), 42 U.S.C. 300g-1. Both of these requirements are concerned with human exposure to radionuclides from disposal systems and, like the containment requirements, both limit exposure for 10,000 years.
EPA is proposing that any application for certification of compliance must include information which identifies the processes, events, or sequences of processes and events considered in compliance assessments. Once the processes and events have been identified, they must be incorporated into compliance assessments of the disposal system. The disposal standards require compliance assessments to include consideration of the uncertainties associated with the undisturbed performance of the disposal system. To do this, it is necessary to identify all disposal system parameters that can affect the performance of the WIPP, as well as to identify the uncertainty associated with each parameter.
When the disposal system parameters and their associated uncertainties have been identified, probability distributions must be developed for each parameter. In compiling compliance assessment results, computational techniques must be used which draw random samples from across the full range of probability distributions for

parameter values used in compliance assessments. This will help assure that all possible values of a parameter have been considered.
It is important to note that a "reasonable" expectation of compliance with the individual and ground-water protection requirements of 40 CFR 191 will not be based solely on a final statistical estimate of doses to individuals or radionuclide concentrations in ground water. A reasonable expectation of compliance will be determined by EPA based on the full record before the Agency and a thorough consideration of the methods and assumptions that produced compliance assessment results. For instance, in certifying and determining compliance, the Agency will consider certain factors, such as the reasonableness of the processes and events considered, the appropriateness of any expert judgment elicitation used to provide inputs to the assessments, the adequacy of peer review, the quality of the models, and the merit of data inputs to those models.
Subpart D -- Public Participation
The Agency intends to involve the public as much as possible in its regulatory oversight of the WIPP. Throughout the compliance criteria rulemaking process, EPA has interacted with stakeholders on all aspects of the criteria. EPA is committed to achieving logical, reasonable WIPP compliance criteria that are protective of the public and the environment. To this end, EPA has worked, and will continue to work, to gather input and promote understanding among key stakeholders. For example, the Agency: 1) circulated and obtained comments on an early draft of the criteria in January of 1994, 2) participated in open technical meetings with DOE on relevant topics, 3) will host a technical workshop on several key compliance-related issues, 4) will solicit comment on the proposed criteria and hold public hearings in the State of New Mexico, and 5) will convene a meeting of the National Advisory Council for Environmental Policy and Technology after the public comment period.
CONCLUSION
The WIPP must be designed to provide a reasonable expectation of compliance with 40 CFR 191. EPA will make a certification decision based upon performance assessments that demonstrate that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from processes and events that may affect the disposal system will not exceed specified quantities. DOE must also demonstrate that the WIPP will meet the assurance requirements of the disposal standards and be designed to provide a reasonable expectation that, for 10,000 years after disposal, undisturbed performance cannot cause any member of the public to receive a committed effective dose in excess of 15 millirems per year. Finally, the WIPP must be designed to provide a reasonable expectation that 10,000 years of undisturbed performance after disposal will not cause levels of radioactivity in any underground source of drinking water in the accessible environment to exceed maximum contaminant levels specified in the Safe Drinking Water Act.
EPA drafted the proposed WIPP compliance criteria for compliance with the final radioactive waste disposal standards. Once promulgated, the criteria will be codified at 40 CFR 194 and will specify the procedures by which DOE will submit its certification application and documentation of continued compliance under the Land Withdrawal Act. As long as DOE submits a technically and administratively complete application as specified in the compliance criteria, EPA expects to be able to evaluate the application and certify whether or not the WIPP will comply with the disposal standards.
EPA plans to finalize the compliance criteria approximately one year after the proposal, after comments from interested parties are considered. When the final criteria are promulgated, EPA will be responsible for assuring that the criteria are properly implemented.

9-7
EVALUATION OF THE PROPOSED COMPLIANCE CRITERIA (40 CFR 194) FOR THE WIPP
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INTRODUCTION
This paper is the Environmental Evaluation Group's initial evaluation of the U.S.

Environmental Protection Agency's draft Criteria for the Certification and Determination of the Waste Isolation Pilot Plant's Compliance with Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (40 CFR 194) (1). We discuss various provisions in the proposed criteria and remaining unresolved issues.

WIPP & TRU WASTE

The Waste Isolation Pilot Plant (WIPP) is a repository for disposal of defense-generated transuranic (TRU) waste, including chemically hazardous components. The WIPP is located in southeastern New Mexico, 40 km east of Carlsbad. The repository is in bedded salt of the Permian age, some 650 m below ground surface, in a 600-m thick salt formation. Work on WIPP began in 1974 and several miles of drifts and tunnels have been excavated. The U.S. Department of Energy (DOE) plans to ultimately dispose 178,000 m3 of contact-handled transuranic waste containing 3.3 x 1017 Bq along with 7100 m3 of remote-handled waste containing 1.85 x 1017 Bq.

Before waste disposal can begin at WIPP, the DOE must demonstrate that the WIPP will comply with the Environmental Protection Agency's (EPA) Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes (40 CFR 191) (2).

COMPLIANCE

The proposed criteria (40 CFR 194) are limited to consideration of the WIPP's compliance with the disposal regulations found in subparts B and C of 40 CFR 191, addressing containment requirements, assurance requirements, individual protection requirements, and groundwater protection requirements. These compliance criteria do not address compliance with the management and storage regulations found in subpart A of 40 CFR part 191.

The original standards (40 CFR 191) apply to both high-level and transuranic waste. For the spent nuclear fuel and defense high-level waste repository, the standards were implemented into licensing procedures by the U.S. Nuclear Regulatory Commission (3). However, the WIPP is exempt from regulation by the U.S. Nuclear Regulatory Commission. The WIPP Land Withdrawal Act (4) requires EPA to certify and determine whether or not the WIPP will comply with the Agency's radioactive waste disposal standards. Thus the certification/compliance criteria may be viewed as equivalent to 10 CFR 60, for the WIPP. The draft compliance criteria, published in the Federal Register on January 30, 1995, provide a public comment period until May 1, 1995. The EPA expects to issue the final compliance criteria early in 1996.

THE ENVIRONMENTAL EVALUATION GROUP

The Environmental Evaluation Group (EEG) was established in 1978 to perform independent analyses of health, safety and environmental issues related to the WIPP. From 1978 to 1992, the EEG was the only full-time technical oversight group for the WIPP.

THE DRAFT COMPLIANCE CRITERIA

The draft compliance criteria (1) provide elaboration on the standards. The criteria contain procedural requirements for the certification application, guidance on doing performance assessments to show compliance with the containment requirements, as well as guidance for meeting the assurance requirements. The following evaluation is divided into these three sections.

Procedural Provisions

The procedural part of the draft criteria deals with applications for certification and determination, EPA's access to WIPP, quality assurance, and public participation, among others. Procedures are given for the initial certification for compliance, and periodic renewal of compliance, known as determination. The draft covers contents of applications, and conditions which will be contained in certification. The draft also covers denial, suspension and revocation of certification. The public participation provisions follow that of the Administrative Procedure Act, in so much as certification amounts to a rule making

One of the conditions of certification and determination, 194.04(b)(6), requires that if the DOE determines that a release of waste from the disposal system to the accessible environment is in excess of what is permitted under the disposal regulations has occurred or is likely to occur, the Department shall notify EPA, who shall determine if emplacement of waste may continue and whether to modify, suspend, or revoke any previously issued certification or determination of compliance.

Because the disposal regulations, subparts B and C of 40 CFR 191, deal only with

cumulative releases to the accessible environment over 10,000 years, there is no reliable way to determine if any particular release will be in excess of what is permitted. For this requirement to be realistic, there has to be a different and additional measure of radioactivity discharge.

Compliance with the Containment Requirement

Most of the draft criteria is about the conduct of performance assessments to show compliance with the containment requirement, which can be expressed as follows. For all i nuclides, each with calculated release of Qi and EPA prescribed limit of RLi, the EPA containment requirement is

Eq. (1)

Eq. (2)

Performance assessment represents the risk of a geologic disposal system as

Eq. (3)

where Si are the scenarios, or what can happen,
pSi are the scenario probabilities, or how likely things will happen,
cSi are the consequences or outcomes of what can happen, and
nS is the number of scenarios.

Performance assessment involves the calculation of risk as

Eq. (4)

which says risk is a function of a vector of imprecisely known variables , such as rock permeability, human intrusion rates, and retardation coefficients of radionuclides.

We discuss below the specific requirements in the draft criteria.

Scenarios to be analyzed: The draft criteria allow some scenarios [Si in eq. (3) and (4)] not to be analyzed:

Performance assessments need not consider processes, events, or sequences of processes and events that have less than one chance in 10,000 of occurring over 10,000 years [194.32(b)].

This provision is consistent with the guidance for implementation of the former subpart B (now subpart C) of the disposal regulations, and we support it.

Radioactivity Baseline: The disposal regulations give release limits [RLi in eq (3) and (4)] in terms of radioactivity allowed per unit of radioactivity emplaced. However, the time to measure the amount of radioactivity emplaced was not given. Because radioactive waste decays, it is necessary to specify a standard time for determining the amount of radioactivity emplaced. The draft criteria specify that initial radioactivity is to be determined at 100 years.

The expected curie activity 100 years after disposal of the waste proposed for disposal in the disposal system shall be used in calculating applicable release limits under Appendix A of 40 CFR 191, Table I, Note 1(e) [194.31].

The EEG supports this sensible action.

Future States: Demonstrating compliance with the disposal regulations over 10,000 years involves calculating the transport of radionuclides from the disposal system to the accessible environment. Because of the long-term nature of these evaluations, uncertainty of values for many parameters important to the analysis may be very large. Environmental conditions and living habits of future populations and individuals may change in significant and unforeseeable ways over the lengthy timeframes that will be analyzed for compliance.

In light of the difficulty of assigning appropriate values with confidence, the EPA is proposing that, unless otherwise specified, applications for compliance shall assume that characteristics of the future remain what they are today, except for parameter values related to long-term geologic, hydrologic, or climatologic conditions of the repository and its vicinity.

Because the future state of civilization would fall under this future states assumption, one would assume that future technology, radioactivity exposure pathways, dietary habits, and population distribution would be identical to the present day. This also means that the applicant cannot project that radioactivity will no longer cause cancer years from now; but that oil and gas drilling rates near the repository will stay the same for the 10,000-year regulatory period.

The EEG reluctantly accepts this provision of the future states assumption, because we recognize the difficulty of alternative provisions. We urge this assumption be rigorously applied, with respect to the positive and negative consequences that might result.

Human Intrusion: For the WIPP, the chief disruptive event is human intrusion (5).

The 1985 guidance for the implementation of subpart B [40 CFR 191, Appendix B] suggested that human intrusion rate by drilling need not be
greater than 30 boreholes per square kilometer of repository area per 10,000 years for geologic repositories in proximity to sedimentary rock formations....
The draft criteria divide human-initiated processes and events into two distinct categories, "human intrusion" and "human activity," and propose a separate process to establish the drilling rate for each. "Human intrusion" includes those drilling events that reach the waste in the repository. Such events would include, but would not be limited to, exploration for and development of oil and natural gas resources. "Human activity" includes all drilling events that may affect the disposal system, without disturbing the waste in the disposal system. Such drilling events may include, but would not be limited to, exploration for potash, and withdrawal of water. EPA proposes examination of the record of human-initiated processes and events in the Delaware Basin over the past 50 years, and holding the rates constant throughout the future, consistent with the future states assumption above. The sum of the individual rates of human intrusion for each resource then becomes the rate of human intrusion to be used in performance assessments, provided that the sum is not less than 25 and not greater than 62.5 boreholes per square kilometer per 10,000 years. If the DOE does not analyze the historical rates of human intrusion events in the Delaware Basin, the Department shall assume the maximum rate of 62.5 boreholes per square kilometer per 10,000 years in performance assessment.
The drilling rate for any resource in the last fifty years in the Delaware Basin is certainly the highest for all time, and to extend that rate over the next 10,000 years betrays our understanding of minerals economics. The rational approach is to adopt a time-dependent drilling rate, that incorporates the history of drilling in this area, and the concept of mineral depletion (6). This suggestion was not accepted by the EPA because it is not consistent with the future states assumption. Another argument for not projecting future rates of drilling based only on the present experience is that we have no way of knowing what future societies may drill for, just as 200 years ago it would have been difficult to predict current drilling for oil, gas and potash.
In the draft criteria, EPA proposes to allow DOE to take credit for passive institutional control by reducing the rate of human intrusion!
(c) Any application for certification of compliance may include a proposed credit (which may vary over the regulatory time frame) for reducing the rate of human-initiated processes and events calculated using the procedures enumerated in 194.33. The Administrator shall allow such credit, or a smaller credit, to be taken if the Department demonstrates that such credit is justified because the passive institutional controls can be expected to endure, be understood, and act as a deterrent to potential intruders throughout the regulatory time frame. In no case, however, shall passive institutional controls be assumed to eliminate the likelihood of human-initiated processes and events entirely.
By using a time-dependent credit, the EPA can arrive at a drilling rate very much like we suggested (6), and step away from the simplistic assumption of a constant drilling rate for 10,000 years, projected on the basis of 50 years of data. However, we understand the need to uniformly apply the future states assumption, and we look to a rigorously applied future states assumption.
Models, Codes and Data: Computation for performance assessment is usually done by a series of computational models. Each model consists of a conceptual model, a computer code and related input data. The EPA requirements in the draft criteria deal with all three. The EPA proposes to require the applicant to show all conceptual models considered and how and why a particular model was chosen. This is a very important and wise requirement. The DOE has already started down this path with the Position Paper on technical issues in the systems prioritization method.
As for computer codes, the EPA requires full documentation as required in NQA-2a (7). Complete documentation is required for people who did not author the software to understand and pass judgment on the work. The EPA will also require the computer codes to be peer reviewed, a routine requirement in software quality assurance.
A model is only as good as the input data. For performance assessment, data required include waste characteristics, geologic site characteristics, and characteristics of rocks in the projected flow paths. The draft criteria address the topic of data obtained before quality assurance was adopted, and use of expert judgment.
The EEG participated in the collection of some early site data, before quality

assurance programs were in place. State-of-the-art methods were used. The EEG has no problem with a sensible approach to qualify data collected this way.
In EEG-57 (8) and elsewhere (9) we pointed out the fallacy of using so-called expert judgment in place of experimental data. The EPA agrees, saying
...expert judgement does not substitute for information that could reasonably be obtained through data collection or experimentation [194.26].
Upon the completion of performance assessments, the results have to be compared to the containment requirement. Under a myriad of assumptions, and using Monte Carlo simulation to investigate the effect of parameter uncertainty [in eq. (3) and (4), there will be many complementary cumulative distribution functions (CCDF) generated. Which of the CCDFs should be compared to the containment. The draft criteria specifies
Any application for certification of compliance shall provide information which demonstrates that there is at least a 95% level of statistical confidence that the mean of the population of CCDFs meets the requirements of section 13(a) of 40 CFR part 191 [194.34].
Our interpretation of this requirement is shown in Fig. 1. A family of compliance CCDFs is generated. Statistical moments can be generated for these curves. On the mean curve there is estimation error, shown by the bell-shaped curves. The mean curve can actually extend further to the upper right-hand corner, closer to the containment requirement. The draft criteria says use the mean curve, plus as much error as would be incurred for a 5% one-way Type-I error, to compare with the containment requirements, as shown in Fig. 1.
We find this requirement quite reasonable.
Compliance with the Assurance Requirements
The guidance in the draft criteria could have been much more specific in telling the applicant when they have satisfied these qualitative requirements. For example, the draft criteria could have specified that records for the WIPP will be placed in the National Archives, the New Mexico State Library, the Carlsbad Public Library, and all public libraries within 100 km of the WIPP. The draft criteria could have adopted an engineered barrier release rate requirement similar to the Nuclear Regulatory Commission's (3) to measure the contribution of engineered barriers at the WIPP. The draft criteria could have defined active institutional control as 3-meter high barb wires around the controlled zone. More specific guidance is needed and should be in the final criteria.
Uncharted Territory
The draft criteria also provides guidance on showing compliance with individual protection and ground-water protection provisions. Whereas DOE has published four iterations of performance assessments to show compliance with the containment requirement, we know of no published analysis showing calculation of dose to individuals. Thus requiring compliance analysis to consider all potential exposure pathways is next to meaningless. More specific guidance should be given in the final criteria.
CONCLUSION
The draft criteria for evaluation of applications for certification of compliance with the EPA standards is a major advance in terms of spelling out what EPA expects in an application. However, major uncertainties remain. Several provisions should be clarified and made more specific in the final criteria.
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Session 10 -- Site Specific Advisory Boards: Where They are and What They are Doing
Co-chairs: Sandy Perkins, USDOE

10-2

LICENSEE ADVISORY BOARDS FOR SITE DECOMMISSIONING PLANS

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ABSTRACT

Citizen advisory groups are in vogue in both the public and the private sectors. This paper discusses six issues that arise with advisory groups: 1) their specific functions and underlying purposes, 2) how they can be structured to achieve their objectives, 3) implementation details that should be thought out in advance, 4) whether guidance concerning the establishment of advisory groups should be prescriptive or performance-based, 5) special situations that can affect the need for a group and the participation of certain members, and 6) how and why the effectiveness of advisory groups should be evaluated. These issues are discussed within the context of a proposal by the U.S. Nuclear Regulatory Commission to require advisory boards during the development of restricted use decommissioning plans. The issues are broadly applicable, however, and should be confronted whenever an advisory group is contemplated.

INTRODUCTION

On August 22, 1994, the U.S. Nuclear Regulatory Commission (NRC) released its proposed rule for radiological criteria for decommissioning (1). This rule has potentially broad applicability: there are approximately 22,000 facilities in the United States that are licensed to use radioactive materials. In its rule, the NRC proposed the establishment of "site-specific advisory boards" (SSABs) in certain instances -- specifically, to provide advice to licensees that intend to develop "restricted use" decommissioning plans. Licensees that are likely to propose limitations on the future uses of their sites as part of their decommissioning plans comprise only a small subset of the total number of licensees. Nevertheless, they undoubtedly will include some large facilities -- nuclear power plants, for example, as well as other fuel-cycle facilities and some non-fuel cycle facilities.(2)

In proposing the establishment of SSABS for the development of restricted use decommissioning plans, the NRC drew upon a concept popularized by the Federal Facilities Environmental Restoration (FFER) Dialogue Committee, a national, 40-member committee initiated in 1992 by the U.S. Environmental Protection Agency and facilitated by the Keystone Center. The FFER Dialogue Committee articulated the concept of SSABs in its February 1993 interim report, "Recommendations for Improving the Federal Facilities Environmental Restoration Decision-Making and Priority-Setting Processes."(3) While having local advisory boards to federal sites was not an altogether new concept -- at some U.S. Department of Energy (DOE) and U.S. Department of Defense (DOD) sites, for example, it had been tried in various forms for several years -- the FFER Dialogue Committee report brought it to national attention, and it is now being explored by other agencies such as the NRC.

The NRC recognized, however, that the SSAB concept could not be transferred wholesale from the FFER Dialogue Committee's report, partly because the NRC was intending to put the concept to rather different purposes: one involving private companies, not federal facilities, with the advisory group to serve only on a temporary, ad hoc basis. Thus, the NRC sought ideas on how to flesh out the concept of SSABs for restricted use decommissioning plans. As part of this effort, we were requested to provide a background paper for a NRC workshop held in December 1994. In doing so, we drew on the wisdom of NRC staff; on comments made during earlier workshops on decommissioning (4); on the prior experience of somewhat similar advisory groups convened by DOE, DOD, and others (5); and on our own past work in this field (6).

Whenever an advisory board is contemplated, a multitude of questions need to be answered. Most critically, these include:

Why is the board being established? What topics will it address, and for what underlying reasons?

What authority will the board have? Is it simply advisory, or will it make decisions in some instances?

To whom will the board's advice be directed? What response to the advice will be required?

How will the board give its advice -- as a group, or as individuals?

Who should be on the board? How should its members be selected?

How can the board be operated to maximize its effectiveness? If the board is being set up at the behest of a public agency, how much control should that agency exert over how the board is established and run?

When will the board be terminated?

How can evaluations of the board's operation and output provide the greatest help both to the board and to other, similar efforts?

In the paper developed for the NRC workshop, we discussed six issues that together address these and other questions. While directed toward the possibility of SSABs for restricted use decommissioning plans (herein referred to as licensee advisory boards, or LABs), much of what is said below could apply to other advisory boards.

SIX KEY ISSUES

Issue 1. Specific Functions and Underlying Purposes

What specific functions and underlying purposes should a LAB serve?

According to the proposed rule, LABs could have such functions as recommending ways to reduce residual radioactivity, providing advice on proposed institutional controls, and advising on whether the licensee has provided sufficient financial assurance for necessary future site maintenance. But LABs might perform other functions as well: for example, advising licensees on special mechanisms for public input during a decommissioning plan's development and implementation, and airing environmental justice issues that otherwise might go unacknowledged. Some functions might be "off-limits," however: for example, prescribing who subsequent site owners should be, or otherwise infringing on various parties' authority and responsibility to carry out the decommissioning plan.

In performing its functions, a LAB could serve one or more underlying purposes. For example, it could 1) improve understanding of the decommissioning process and of its consequences through a sustained information exchange with representatives of key stakeholder interests; 2) improve the quality of the decommissioning plan by bringing to bear both the knowledge and the values of LAB members; 3) promote widespread acceptance and ownership of the decommissioning plan, by having it developed in a participatory fashion; and 4) promote the ideals of environmental justice, by empowering stakeholders that historically have lacked political and economic clout. But there may also be inappropriate underlying purposes for LABs: for example, co-optation of key stakeholders, or use of a LAB as mere "window-dressing" or as a substitute for other meaningful public input.

Issue 2. Achieving Objectives

What are the general objectives of a LAB, and how can they be achieved?

The primary goal of a LAB is to create a representative process that provides timely, useful input. In achieving this, a number of objectives arise. These concern the adequacy of stakeholder representation; the degree of LAB responsibility; the form, use, and timeliness of the advice; how the problem under consideration is characterized; and the LAB's ability to address the problem. But, as noted further below, it may not be possible to realize all objectives; instead, tradeoffs may be

necessary.
Adequacy of stakeholder representation. Achieving broad stakeholder representation of affected interests poses a dual problem: first, of deciding which interests merit representation, and second, of deciding who can speak for those interests. With a LAB, it may be necessary to have representation of such diverse interests as those of the licensee, subsequent site owners or managers, state or tribal officials, local officials, and people living near the site now or in the future, as well as non-human, ecological interests.
Given this array of interests, the simplest approach could be to appoint opinion leaders, including elected or administrative officials. A potential disadvantage, however, is that all of the affected interests may not be well-represented. For example, some interests (e.g., ecological interests) are associated with the welfare and well-being of the community at large, rather than being linked to specific individuals. To correct for this, opinion surveys could be used to identify such broad community interests, and may lead to the identification of particular individuals or groups able to represent those interests.
An additional problem arises if there are so many interests requiring representation that the size of the LAB would be cumbersome. This problem is addressed under Issue 3.
Degree of LAB responsibility. For a LAB to be effective, its degree of responsibility needs to be made clear. One option is for the LAB to be charged with deciding aspects of the licensee's proposed decommissioning plan. This would invest the LAB with authority, but it would burden the LAB with greater accountability, both to the licensee and, possibly, to other constituencies. A second option is for the LAB to simply offer recommendations. This reduces the LAB's responsibility, but it decreases its influence. As a third option, a hybrid is possible: the LAB could make general recommendations about the plan as a whole, but representatives of organizations with authority over parts of the plan could specify the acceptability of those parts. Form, use, and timeliness of LAB advice. The LAB could provide its advice as either a group or a set of individuals. If it is to provide advice as a group, then its "decision rules" must be made clear. For example, if a discrete judgment is sought, a voting procedure might be appropriate, whereas if the charge is to conduct open-ended discussions, then a consensus-type procedure might be more suitable. Both have pros and cons: reaching consensus often takes considerable time and can be especially difficult to achieve with a large group but produces fairly unified advice; voting procedures are usually more time-efficient and can be readily conducted with large groups but are likely to produce minority opinions.
At a minimum, the results of LAB deliberations on the proposed plan should be used by the licensee informally, to develop and refine the plan. Alternatively, as suggested in the proposed rule, the licensee could be required to include in the proposed decommissioning plan the disposition of the LAB recommendations. In reaching its decision on the proposed plan, the NRC might also be required to comment on LAB recommendations that it has decided not to follow. This has the advantage of demonstrating that all explicit, written LAB recommendations have been given consideration. However, if a recommendation is made by an individual LAB member, the licensee and the NRC may be unsure of the extent to which the recommendation is supported by the group (although having licensee and NRC representatives at LAB meetings would help to address this concern).
One reason for having a LAB is to provide a mechanism for early stakeholder involvement. To this purpose, the LAB could be created and its advice sought when the licensee has just begun to weigh possible decommissioning options. This might complicate LAB/licensee interactions, however, particularly if the licensee has not yet formulated a clear set of options. But the further along the decommissioning plan is, the more likely it is that the LAB's advice will not be readily entertained. As an intermediate approach, the LAB could be set up early on, with the understanding that it would hold off on options analysis and would assume as its first tasks developing its operating procedures and its background understanding of the site and related issues.
Problem characterization. To provide advice, a LAB will need to understand the dimensions of the problem before it. Doing so may require access to studies and analyses pertinent to the proposed decommissioning. These analyses may differ in many respects. Some may be more detailed than others. All are likely to reflect the goals and methodological preferences of their authors -- e.g., the licensee, state

or local agencies, or special interest groups. The LAB will need to assess the quality of these analyses, using such criteria as adequate explanation of technical language, specification of the full range of consequences for various scenarios posited, indication of the reliability of technical analyses, and identification of potential sources of bias.

Deliberative capability. Especially if the LAB is to provide group advice rather than simply individual opinions, its members need to deliberate together effectively. Deliberation involves a weighing of facts, values, and arguments in reaching conclusions. There are two general approaches. The first is informal: the group thinks through a topic or problem intuitively, using whatever means are readily available. The second is more formal: the group employs structured tools such as decision analysis and Delphi techniques. Informal deliberation has the advantage of requiring little or no outside expertise or technical support. However, it can result in the omission of important factors. Formal analysis helps ensure completeness and may make clearer distinctions between facts and values. However, it requires more external support and may be inappropriate for some problems, particularly those that do not lend themselves to quantification (e.g., the assessment of social impacts of decommissioning).

Apart from the analytic methods employed, the quality of the LAB's deliberations will be affected by the ability of individual members, first, to understand the proposed plan and its related analyses, and second, to tease out issues that might go unnoticed. A LAB composed mainly of experts on aspects of the decommissioning plan may succeed on the first score but fail on the second. At issue is how much technical expertise LAB members should have, individually and collectively, and whether technical expertise can be incorporated by means other than membership on the LAB.

Group interactions may be enhanced if external meetings between two or more members outside the group setting are discouraged. This may be especially important if group consensus is sought: by exchanging views and information only as a group, the LAB may be more likely to develop a shared viewpoint. However, this approach may be inefficient on highly technical issues, or it may lead to stalemates on volatile issues. In such cases, the LAB may be better able to deliberate in subgroups, possibly composed not only of LAB members but also of other experts or interest representatives.

Tradeoffs among objectives. While maximally achieving such objectives as broad stakeholder representation, timely establishment of the LAB, and optimum deliberative capability would be desirable, it may not be possible. For example, ensuring that the LAB is fully representative of stakeholder interests may make it so large as to impair its deliberative capability. To address the problem of tradeoffs among objectives, the licensee and/or NRC could consider ranking objectives according to the central concerns associated with the particular site involved. Alternatively, the NRC could decide a priori to emphasize one or more objectives, while giving the licensee discretion in the degree to which others are met.

Issue 3. Implementation Details

What implementation details need to be considered when establishing a LAB?

Once the need for a LAB is identified, someone must take responsibility for its start-up, operation, and termination. Success with implementation details will allow the LAB to focus on its main goal: providing sound advice to the licensee. Failure can lead to organizational problems that can jeopardize the entire advisory process. Inattention to implementation details may also call into question the licensee's and/or NRC's commitment to the LAB process. While it is assumed here that the licensee would be responsible for implementation, the NRC and others could share responsibility. Thus, allocation of responsibilities is one question discussed below.

Start-up. Start-up decisions include determining when to form the LAB, as discussed under Issue 2; how to announce its formation and invite stakeholders to participate; whether LAB members need to make any commitments, written or otherwise, to the LAB process and/or the licensee as a condition for membership (e.g., commitments not to disclose confidential business information); and whether to reimburse expenses (e.g., for travel and child care) and pay honoraria for participation. The question of honoraria merits especially careful consideration. On the one hand, honoraria may contribute to the impression that participants are being bought off; on the other

hand, honoraria may be needed if participation by some members would be precluded because they can not take time off from their paid jobs.

Perhaps the greatest challenge is to ensure that the LAB is large enough to permit representation of all relevant stakeholder interests yet small enough to permit meaningful interaction and deliberation. While there is no formula for determining optimum size, as a general rule a group of more than 10 to 15 members is likely to find informal interaction difficult. To address the problem of size, it may be necessary to consider, first, whether some interests are more crucial than others; second, whether some members of the LAB can represent more than one set of interests; and third, whether there are ways to informally involve peripherally affected parties without expanding the size of the LAB (e.g., through periodic open workshops, or by inviting representatives of certain interests to attend LAB meetings when topics particularly germane to them are to be addressed).

During the start-up phase, rules for participation may be needed: e.g., who has the final say on specific appointments to the LAB once stakeholder groups have been identified (the stakeholder groups, the licensee, the NRC, an independent third party?); whether alternates can serve in the absence of appointed members; and how appointed LAB members will be replaced if they terminate their LAB membership once the process has begun. Some of these "groundrules" may need to be specified by the licensee or by an independent third party; others can be developed by the LAB once it is operating.

Operation. Operational decisions include when and where to hold meetings, how agendas should be set, and whether meetings should be public or private (or a mix, depending on the topic under discussion). Decisions also need to made about how the LAB will be led (e.g., by a chair appointed by the licensee or by LAB members); whether the LAB should have an outside facilitator to help structure discussions and lead brainstorming sessions, and if so, how one should be selected; and, as discussed under Issue 2, how decisions on group advice should be made (e.g., by consensus, by majority vote, etc.) and advice communicated.

Administrative, staff, and technical support. The LAB may require administrative support, including meeting space, office supplies, copiers, phones, and fax machines. To disseminate information, prepare correspondence, compile reports, etc., it may require staff support. The LAB may also require technical support internally (e.g., someone to collect information and maintain an information data base) and externally (e.g., outside consultants to provide independent reviews of technical studies). Without some minimal level of support, the LAB's ability to deliberate effectively and provide useful advice to the licensee may be compromised. However, the LAB probably should not seek to (or be expected to) replicate credible scientific and technical studies conducted by the licensee and others. A good working relationship between the licensee and the LAB would help to ensure the timeliness and cost-effectiveness of administrative, staff, and technical support.

Cost drivers. The past experience of advisory groups suggests that there are five basic factors affecting advisory board costs: the advisory group's size, the frequency of its meetings, how long the group remains in existence, the extensiveness of support required, and the distances involved in convening members, support personnel, consultants, etc. The licensee's ability and willingness to bear LAB costs may be influenced by several factors -- in particular, the licensee's financial health and the perceived benefits of having a LAB. In some instances, it may be very important to the licensee to anticipate and control LAB costs. Means of controlling LAB costs could include, for example, using video or audio teleconferencing to reduce the number of meetings, and having a fixed annual budget (lump sum or itemized), with the LAB responsible for securing any additional funding desired through outside grants.

Conflict resolution. If differences arise between the licensee and the LAB (and, possibly, among LAB members), various approaches to conflict resolution could be considered. Optimally, the LAB, together with the licensee, should develop a strong working relationship with a shared sense of ownership in the outcome of the process and with the ability to resolve differences one-on-one, by negotiating with each other. Some issues may be fairly intractable, however, and an outside mediator may be needed to help reach compromise. While the prospect of using an independent third party to help resolve differences may erode the licensee's and group's sense that they are working together, mediation may help to ensure that the LAB process does not get stalled. Not all issues are suitable for mediation, however, and a degree of

cooperation from participants will be required. A first step is to have the prospective mediator assess whether the situation is appropriate for mediation.
LAB termination. Termination could become a divisive issue if the licensee wants to terminate the LAB but its members (individually or collectively) want to continue, or vice versa. To allow for this, forethought may be needed about who will make the final decision to terminate, and about whether consensus is required for continuation. Controversy over termination may be minimal, however, if all other issues have been handled properly. Then, it may be self-evident to all concerned that the LAB's mission has been completed. Notification of any future plans by the licensee to solicit advice from the public might still be needed, however. This could be done at the time of the LAB termination.
Allocating implementation responsibilities. The licensee, the NRC, or a neutral outside party or steering committee (possibly selected by the licensee in consultation with the NRC and the state) could take sole responsibility for initiating the LAB process, providing support services, offering guidelines on LAB operation, and managing the LAB termination process. Alternatively, these tasks could be allocated according to who would be most appropriate. For example, a steering committee could be established to select LAB members and offer guidelines on LAB operation; the LAB could expand upon those guidelines (and possibly its membership) in consultation with the licensee; the licensee could be responsible for providing a fixed LAB budget limit, with discretion for the expenditure of funds left to the LAB; the NRC, in consultation with the LAB and the licensee, could be responsible for determining when the LAB should terminate. Any number of combinations is possible. The main point is to decide who has the highest likelihood of implementing a part of the process in a manner that will meet the fundamental goal of the LAB concept.
Issue 4. Prescriptive v. Performance-based Rulemaking
In rulemaking concerning LABs, is a prescriptive or a performance-based approach preferable?
Prescriptive approaches to rule-making are means-oriented, whereas performance-based approaches are ends-oriented. Prescriptive approaches try to anticipate which design features and methods of implementation are most likely to realize certain objectives -- e.g., to ensure representativeness, a prescriptive approach might specify the size and composition of a LAB. The prescriptive approach ensures that a licensee will not overlook or ignore important considerations in the establishment and operation of an LAB. However, because variations among licensees and affected communities cannot be fully taken into account in rulemaking, highly prescriptive rules can be "Procrustean": they can force all licensees and LABs to fit a limited number of molds. In doing so, they may be burdensome and inefficient. They also may fail to achieve the objectives driving the rulemaking.
In contrast, performance-based approaches to rulemaking focus on the objectives themselves and permit a good deal of implementation discretion -- e.g., in a performance-based approach it might simply be stated that representation of all significant interests should be sought. The performance-based approach avoids the inflexibility inherent in a highly prescriptive approach. It allows the licensee and the affected community to tailor the LAB to their particular circumstances by granting a large measure of discretion in determining how the LAB should be set up and operated (and, perhaps, whether one is needed at all or whether another group can fulfill the LAB's roles). With a performance-based approach, however, explicit verification and evaluation measures become especially important. Without them, there is no objective means of determining whether the licensee has actually met the spirit of the regulatory objectives.
There is, in reality, no bright line between prescriptive and performance-based approaches. Even with prescriptive approaches, objectives are usually articulated and the prescriptions usually include a measure of flexibility. And even with performance-based approaches, certain design features and methods of implementation may be recommended. In addition, rules rarely spell out all of the details of implementation; they usually are supplemented with regulatory guidelines and other documents. To the extent that rules focus on objectives, with implementation details contained in non-mandatory guidelines, more flexibility is introduced. Thus, the distinction between prescriptive and performance-based approaches is usually one of degree, and in fact, hybrids of the two are the norm.
As suggested in the proposed rule, a two-step process for establishing a LAB could

be adopted -- a process which balances between the prescriptive and the performance-based. According to this process, when the licensee submits notification of intent to have a restricted use decommissioning, a plan for establishing and operating a LAB is also submitted. Following NRC approval, the licensee implements the LAB and develops the decommissioning plan. This two-step process allows the licensee to tailor the LAB approach but reduces the likelihood that the approach would be found deficient after a decommissioning plan had been developed and submitted. Pre-approval of a LAB plan could also tackle early on, before a LAB had been initiated, the problem of necessary tradeoffs between LAB objectives discussed under Issue 2.

Issue 5. Special Situations

Are there special conditions affecting the need for a LAB, or the ability of public officials to be members of a LAB?

The NRC has proposed that a LAB be established if a restricted use decommissioning is contemplated. However, some sites proposed for a restricted use decommissioning might not justify the effort and expense of establishing a LAB: for example, sites where the restrictions on future uses are to be temporary, short-lived measures. In addition, at some sites where the LAB requirement would apply, establishing a LAB could be unnecessary because existing organizations such as local environmental review boards could be adapted to provide advice on decommissioning issues. Substituting an existing organization has, however, both potential assets and potential drawbacks. Doing so could be time- and cost-efficient and could downplay questions that may arise concerning membership, but it could run the risk of distorting or trivializing the LAB's mission by tying it to existing agendas, and furthermore, the existing group might not have the appropriate composition or expertise.

In contrast with situations where exemptions from the LAB requirement might be considered, some sites not proposed for a restricted use decommissioning might still merit the establishment of a LAB. For example, a LAB might be needed on at least an interim basis if, at the time of the licensee's notification of intent to decommission, it cannot be determined whether the site can be cleaned up to permit unrestricted use. Another special situation where a LAB might be appropriate concerns licensees who plan to retain their licenses, but in an amended or modified form: e.g., licensees undertaking actions preparatory to decommissioning while not yet seeking license termination. In this situation, a LAB-like body might be needed to provide advice on the arrangements being made toward the eventual prospect of decommissioning, especially since these arrangements could immediately or prospectively affect human health, property values, and local jobs.

In addition, there may be impediments to the participation of members -- particularly state and local officials -- on LABs. Public agencies may have regulations that restrict employees from participating in activities that pose a potential conflict of interest with their work. In those cases, participation would have to be written into their jobs. Officials may also have concerns about tort liability -- i.e., liability for the consequences of decisions or non-action -- in instances where they are not clearly protected by sovereign immunity. One remedy is to have representatives of public agencies sit ex officio on the LAB. This would temper their liability concerns, but it also might lessen their input. Fundamentally at issue is whether public agency officials are to serve as representatives of their agencies or in their personal capacities.

Issue 6. Evaluating Effectiveness

How should the effectiveness of LABs be evaluated, and what are the regulatory implications of LAB evaluations?

In thinking about how the effectiveness of LABs might be evaluated, there are several considerations. First, evaluations could be done either of individual LABs, to make mid-course corrections or as "post-mortems", or of a suite of LABs, to suggest to the NRC whether changes in regulations and guidelines might be advisable. Second, an evaluation could have several purposes: it could be concerned with process (e.g., to identify difficulties in a LAB's initiation or operation), or it could be concerned with outcomes (e.g., to determine whether LAB advice has made a difference in subsequent licensee and NRC decisions). In either case, indicators of effectiveness need to be specified, and methods for their measurement established. Third, someone must be responsible for conducting the evaluation, which raises the question of whether an independent evaluator or a participant in the process is

preferable. Fourth, the results of an evaluation are often affected by its timing. Preliminary evaluations could be performed while the LAB is in process, but its overall impacts may be more readily assessed at the conclusion of its efforts. And fifth, there may be circumstances under which there is little point in conducting an evaluation: for example, if a LAB is convened only fleetingly. Also, if a particular decommissioning site is unusual, a LAB evaluation might be useful for "self-help" purposes, but its results probably would not be generalizable.

Evaluation results could be used in many ways. As noted above, cumulative evaluations of several LABs could contribute to an NRC decision to continue, abandon, or modify the LAB approach, and the NRC also might decide that the LAB approach should be extended to other circumstances, in either its current or a revised form. In addition, LAB evaluations could be used to provide informal guidance for licensees just embarking on their decommissioning planning, to inform them about LAB approaches that do and don't work. Similarly, when a LAB is convened its members might wish to review evaluations of other LABs. In any case, the ways in which evaluation results are reported should take into consideration who will use them. For example, licensees and LAB members may benefit most from case studies. In contrast, the NRC may be most interested in evaluations that systematically characterize and discuss findings from a number of LABs along several dimensions.

CONCLUSION

The small-group approach to stakeholder input is currently much in favor in the public and private sectors. It is not, however, an approach that should be undertaken blithely, without careful thought given to why and how it is to be carried out.

In our paper for the NRC workshop, and in the summary given above, we do not attempt to provide a template for how advisory boards should be set up and operated. To do so would ignore the many different circumstances that should be taken into account in establishing advisory groups. We do, though, seek to lay out the critical issues -- issues that should be resolved before an advisory group is established. With careful forethought, pitfalls may be avoided or minimized, and the advisory group is more likely to realize its goal of provide useful, timely advice.

In addition, if an advisory group approach is well-constructed and well-executed, it can over time help to overcome barriers that sometimes exist between individuals and groups with different perspectives. It can lead to a greater understanding of -- and possibly a greater appreciation for -- dissimilar views. But mutual understanding and respect is not an automatic outcome of advisory groups. They also can result in increased factionalization and distrust. There are no guarantees that advisory groups will not backfire, but they are less likely to do so if they are conceived and executed with care. In particular, the group's purpose must be made clear, members must be carefully selected, adequate information must be provided to the group, and those seeking advice must indicate how they have responded to the advice given.

Finally, it should be noted that advisory groups should not be regarded as substitutes for broader forms of citizen participation. This point has been recognized by the NRC and other federal agencies considering SSAB-like approaches, but it is worth reiterating. While the advice that can be obtained from the sustained, intensive efforts of a small group may be invaluable, it should be complemented with approaches that invite the opinions of the larger citizenry. It should augment, not replace, the concept of participatory democracy.

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Session 11 -- LLW Waste Processing Using Vitrification
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11-1
VITRIFICATION OF WET WASTE RESIN
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ABSTRACT
Ion exchange materials used for decontamination of nuclear power plant primary system water is often referred to as wet waste. Wet wastes consist mainly of organic resin loaded with radionuclides such as Co and hazardous components such as Cr. Ideally, the radionuclides and hazardous components should be concentrated into a small stable waste form. One such waste form due to its high density and the large volume reductions available upon vitrification of the wet waste is glass. Glass is also a very stable material, both mechanically and chemically, making it a desirable form for the storage of highly radioactive low-level wastes. Presented in this paper are the results of pilot-scale melter runs where greater than an order of magnitude volume reduction was achieved with the formation of a stable waste form as defined by the Toxic Characteristic Leaching Procedure (TCLP) and Product Consistency Test (PCT). In addition, off-gas emissions for these high organic content wastes can be controlled to less than 20 ppm CO, and less than 35 ppm NOx and SOx.
MELTER DESIGN
Joule-heated ceramic melters have been extensively developed for the vitrification of high-level radioactive wastes. In fact, they have become the internationally-preferred means to treat high-level radioactive liquid wastes (1) and the U.S. Environmental Protection Agency's Best Demonstrated Available Technology. Similar melters in our facility are based on high chromium-containing glass contact materials such as Inconel 690 and Monofrax K-3 refractory. These contact materials are backed by Zirmul AZS refractory and 95% alumina castable. The entire glass containment basin is housed in an interior liner constructed from Inconel. The interior liner permits the installation of a substantial refractory ceramic fiberboard insulation package around the melter to mitigate heat losses through the melter sidewalls. The upper plenum sidewalls are constructed of Zirmul Refractory, and backed by insulating fiberboard. Ceiling construction is from multiple layers of refractory ceramic fiberboard protected by an Inconel plate. The outer shell of the melter consists of 304L stainless steel.
Two routes of glass discharge exist: airlift and bottom drain. The airlift discharges glass through the sidewall to an Inconel 690 riser and pour trough for continuous melter runs. The bottom drain is used for batch mode operations.
Power is provided to the melter from four sources: electrodes, lid heaters, discharge heaters, and bottom drain heaters. The first set of pilot-scale experiments was performed in a melter with a nominal glass output of 10 kg/day, known as the DuraMelter 10.
The melter itself is only a small part of the DuraMelter system. Figure 1 is a schematic of the entire DuraMelter system, consisting the melter and off-gas train. The off-gas train consists of a quencher, scrubber, and filters to clean the off-gas

emissions from the melter. By adjusting the solution in the quencher and scrubber, various compounds, such as NOx, SOx, Cs and Co, can be filtered. The baghouse and HEPA filter capture particulates in the off-gas. Gas and particulate emissions are monitored after the main blower.

RESULTS AND DISCUSSION

Simulated feeds of anion, cation and mixed bed resins were prepared and fed to the DuraMelter 10. Anion resin consists mainly of amine groups while cation resins consist mainly of sulfonate groups. Mixed bed resin normally consists of 60% by volume of anion and 40% by volume of cation. However, since waste by nature is heterogeneous, the extremes were tested in addition to a mixed resin composition. The first tests with the DuraMelter 10 used pure anion and then pure cation to examine the extremes that could challenge the control of NOx and SOx emission, and the extreme in organic content per liter of resin, thus the extreme in heat formation. Table I presents the results of a pure anion and a pure cation run. Using an average feed rate of 1.4 liters of resin per hour for 40 hours, stable glass temperatures were maintained in addition to low CO and NOx emissions using pure anion resin. There is no sulfur in anion resin, therefore SOx was not monitored for the pure anion resin run. However, a typical cation resin contains 4.3 lbs of sulfur per cubic feet of resin. During the pure cation runs, the ability to control SOx emissions was assessed. Table I shows that by using the DuraMelter system, there were low CO, NOx, and SOx emissions, and heat generation was controlled as demonstrated by stable glass temperatures.

Having tested the high-end of sulfur- and nitrogen-containing wastes, mixed bed resin was then tested. A dirty resin was simulated by loading with the appropriate amounts of Na and Fe plus Co and Cs (using non-radioactive Co and Cs) at greater than eight orders of magnitude higher levels than reported by nuclear utilities. The high loading of Cs and Co was used to obtain accurate distribution data.

Figure 2 shows a schematic of wet waste vitrification. The melter is started with an appropriate high conductivity glass, and then chemicals are added to obtain a low-conductivity, high-viscosity glass. After adding 60 liters of wet waste, the conductivity of the melt will increase and the viscosity of the glass will decrease due to the sodium and iron from the wet waste resin. Studies are made to determine which compositions will remain processable throughout 60 liters of wet waste vitrification. After processing 60 liters of wet waste resin, zeolites, used in cleaning the scrubber and quencher, are added to the melter. This will produce a total of 10 kg of glass or 4 liters of waste glass (using a typical glass density of 2.5-2.7 g/ml); a volume reduction of 15.

The off-gas data for the mixed resin run is also presented in Table I. Using an average feed rate of 1.4 liters of mixed resin per hour, both the off-gas emissions and the glass temperature were under control during the entire 40-hour run. Off-gas emissions for CO were less than 20 ppm, while off-gas emissions for NOx and SOx were both below 35 ppm.

In addition to obtaining large volume reductions, vitrification has the advantage of producing a mechanically and chemically stable waste form. Glasses are known to have very high compression strengths, and their chemical durability often exceeds those of other stabilized waste forms. Glasses produced from the simulated wet waste were subjected to two different chemical durability tests: the EPA Toxic Characteristic Leaching Procedure (TCLP) and the High-Level Nuclear Waste Program's Product Consistency Test (PCT). These results are shown in Table II and Fig. 3a , (Fig. 3b) respectively. Wet waste glasses were spiked with 1000 ppm of Cr, Ni, Ba, Cd, Zn, Co, and Cs. This level of Cr, Ni, Ba, Cd, Zn, Co, and Cs is two to nine orders in magnitude higher than the contents expected in the actual waste corresponding to zero to seven orders of magnitude higher than for the actual waste glass. These high concentrations of all of the hazardous components and simulated radioactive components were used in order to clearly see the distribution of these components between waste glass and leachate. These glasses were then subjected to the EPA TCLP test. Analysis of the leachates after the TCLP test shows very small amounts of both hazardous and radioactive components. The leaching of hazardous components is certainly much less than the US EPA limits, as shown in Table II. Regarding the radioactive components, note that for Co, only one part in 10,000 leached out of the glass, and for Cs, only one part in 20,000 leached out of the glass. The same proportion of Co and Cs should leach out in the real waste glass, but the actual amounts will be orders of magnitude less since the actual Co and Cs concentrations

in the real waste glass is seven orders of magnitude and six orders of magnitude smaller, respectively.
The EPA TCLP test uses a surface area to volume ratio of 20 m-1. A more stringent leach test, the Product Consistency Test (PCT), uses a longer leach period (at least seven days and up to 2 years) and greater surface area to volume ratio (2000 m-1). For this test, the leach rate for the major components of glass (Si, B, and Na) are compared to the high-level nuclear waste glass standard, SRL-EA. Figure 3 presents the leach rate of eight of these wet waste glasses compared to the high-level nuclear waste glass standard, SRL-EA. These eight glasses represent compositional variations of wet waste glasses due to variation in wet waste compositions. All eight glasses produce a lower leach rate after seven days with respect to B, Na, and Si than the high-level nuclear waste glass standard. Similar data have been obtained for these glasses after 28 days of leaching. These glasses will continue to be monitored for at least two years.

SUMMARY

We have shown that by using the DuraMelter system, wastes with high organic contents can be vitrified to produce a stable waste form with at least an order of magnitude volume reduction. Vitrification is an ideal method for disposing of these low-level radioactive wastes due to the large volume reduction obtained when vitrifying wet wastes and due to the resulting waste form (glass), which is both mechanically and chemically stable. A major concern in the vitrification of these high organic content wastes is heat generation and CO, NOx, and SOx emissions. All of these problems have been addressed using the DuraMelter system.

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11-2

VITRIFICATION OF LOW-LEVEL RADIOACTIVE MIXED WASTE AT ARGONNE NATIONAL LABORATORY

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ABSTRACT

Argonne National Laboratory-East (ANL-E) is proceeding with plans to use vitrification to treat low-level radioactive mixed wastes (LLMW) generated on-site. The objective is to install a full-scale vitrification system at ANL-E capable of processing the entire annual generation of selected LLMW streams. Crucible glass studies with actual mixed waste streams have produced sodium borosilicate glasses under conditions achievable in commercially available melters. These same glass compositions, spiked with toxic metals above the expected levels in actual wastes, pass the Toxicity Characteristic Leaching Procedure (TCLP) test. Earlier evaluations of the likely off-gases that will result from vitrification indicated that the primary off-gases will include compounds of SOx, NOx, and CO2. These evaluations are being experimentally confirmed with a mass spectrometer analysis of the gases evolved from samples of the ANL-E wastes. The composition of the melter feed can be adjusted to minimize volatilization of some components, if necessary.
The full-scale melter will be designed to handle the annual generation of at least three LLMW waste streams: evaporator concentrator bottoms sludge (ECB), storage tank sludge (STS), and HEPA filter media. Each waste stream is mixed waste by virtue of its failure to pass the TCLP test with respect to toxic metal leaching. Additional LLMW streams under consideration for vitrification include historical mixed waste glass from past operations and spent abrasive from a planned decontamination facility.

INTRODUCTION

Argonne National Laboratory-East (ANL-E) is proceeding with plans to use vitrification to treat low-level radioactive mixed wastes (LLMW) generated on-site. The objective is to install a full-scale vitrification system at ANL-E capable of processing the entire annual generation of selected LLMW streams. We had previously established the feasibility of using vitrification as a treatment technology for the remediation of historical and current wastes generated at ANL-E (1). We reported that the ANL-E mixed waste streams studied were amenable to treatment by

commercially available vitrification technology.
The decision to proceed with this project was based on several factors, regulatory and technical. Section 105 of the Federal Facilities Compliance Act of 1992 (FFCA) requires all DOE facilities are required to prepare an inventory of existing and planned mixed waste and to develop plans for mixed waste treatment. The Department of Energy (DOE) is also required by Section 3021(b) of the Resource Conservation and Recovery Act (RCRA), as amended by the FFCA, to prepare plans describing the development of treatment capacities and methods for treating mixed wastes. The project described here is one of these treatment options.
The major technical factors that supported the decision to proceed with vitrification include the overall volume reduction achieved by the process (roughly a factor of eight) and the lower disposal costs (50% reduction) resulting from the stabilization of the waste from a mixed waste to a low-level radioactive waste. The total cost savings will be determined after the waste streams are treated and their total and annual volumes are finalized.
This project is in the process of producing a range of processible glass compositions from actual mixed wastes and low-cost additives. Emphasis is on maximizing the waste loading in the glass (currently up to 90 wt%) and reducing the overall waste volume, while producing a stabilized low-level radioactive waste glass. In crucible glass studies with actual mixed waste streams, sodium borosilicate glasses have been formed under conditions achievable in commercial melters. Some of these same early glass compositions, spiked with RCRA metals above the expected levels in actual wastes, pass the Toxicity Characteristic Leaching Procedure (TCLP) test. These results provide the technical evidence that the vitrification system and the glass waste form will be robust enough to accommodate expected variations in the ANL-E LLMW streams. Approximately 30 crucible melts are being made to establish a compositional envelope for vitrifying ANL-E mixed wastes.
EXPERIMENTAL
Representative samples of LLMW streams generated at ANL-E were used in this study. Three different waste streams were studied: evaporator concentrator bottoms sludge (ECB), storage tank sludge (STS), and HEPA filter media. The radioactive component of these waste streams included radionuclides such as 232U, 239/240Pu, 237Np, 210Po, 241Am, 137Cs, and 60Co, while the hazardous component included Ag, As, Ba, Cd, Cr, Hg, Se, and Pb at levels above the Environmental Protection Agency (EPA) limits for hazardous materials, as determined with the TCLP. The typical composition of each of these waste streams is given in Table I. The glass-forming compositions of HEPA and ECB are not expected to vary significantly, except for the amount of radioactivity or hazardous metals (hundreds of ppm). There will be considerable variation in the composition of STS. Scanning electron microscopy with energy dispersive x-ray spectroscopy (SEM/EDS) analyses of representative samples of STS found compositional variations much larger than what could be attributed to analytical error of the instrument and technique. The values reported in Table I were assumed to be typical of the material and were used in formulating glass compositions. The STS is a brown material, with the appearance of moist dirt clods that contains occasional root-like organic pieces, some up to 1 cm in diameter. This material has a moisture content of about 45 wt%. The ECB is a sticky yellow sludge-like material that is approximately 25% moisture and has the consistency of moist paste. The HEPA is dry, has a paper-like consistency, and has been shredded down to 1 cm2 squares. Some HEPA filters contained aluminum foil spacers, which make up about 5% of the total filter.
Two composite waste streams were blended from STS, ECB, and HEPA. Composite waste stream A was mixed in the ratio (by weight) of 4:1:1, STS:ECB:HEPA, and B was mixed in the ratio of 4:2.5:1, STS:ECB:HEPA. These formulas are based on estimated annual generation rates of waste streams STS, ECB, and HEPA at ANL-E. Feeds for crucible melts were prepared in the following manner. The waste stream, either A or B, was weighed out, according to the specifications of the formulation. The glass-forming materials were then added, and the mixture was blended to a degree representative of the conditions that may exist during full-scale vitrification. The material feed was neither dried nor processed any further. The mixture was immediately transferred to an alumina crucible, and melting was initiated.
All melting was performed in an electric resistance furnace at 1100C, with the melts being held at temperature for approximately two hours. This temperature was selected since it is readily achievable with commercial vitrification technology (1). Upon termination of the melt test, the glass was poured onto a graphite or stainless

steel surface, neither of which significantly interacted with the melt. The cooled glass was transferred to a clean container until analysis. Each of the glasses was examined with SEM/EDS approximate composition, and structure of the final glasses. The TCLP was performed according to standard procedures with high-density polyethylene leaching vessels. The leachates were semi-quantitatively analyzed for As, Ba, Cd, Cr, Hg, Se, and Pb with an inductively coupled plasma/mass spectrometer. Samples of the melter feed stream are being subjected to evolved gas analysis with a mass spectrometer. These analyses will allow us to experimentally determine which species (both radioactive and hazardous) are evolving from the melt and provide data that can be used in designing an off-gas control system for the full-scale vitrification system. Typically, milligram quantities of a glass formulation (melter feed and glass-forming additives) are gradually heated in a vacuum with a resistance furnace. The evolved gas is introduced into the mass spectrometer, where three values are continuously monitored and stored on a computer: temperature, the atomic mass associated with the analyzed signal, and the analyzed signal. A computer program controls the spectrometer so that a range of atomic masses is continuously scanned and recorded. The data are later processed to identify the evolved gases and their temperatures of evolution.

RESULTS

The glass formulations that have been made to date are summarized in Table II. The first ten melts were used to identify glass compositions that the three waste streams passing the TCLP criteria for a non-hazardous final waste form and meeting the criterion that the material be easily poured from its crucible immediately after removal from the 1100C furnace. Five spiked crucible melts were subsequently produced: melts 11, 12, 13, 14, and 15. These melts had compositions close to the processible formulations 2, 4, 5, and 9. The hazardous metals added to the formulations (Table III) consisted of reagent grade oxides of Ag, As, Ba, Cd, Cr, Pb, and Se or metallic Hg and Cd. These spiked melts all had acceptable viscosities at 1100C and passed the TCLP and are now low-level wastes.

Each of the melts formed a black to dark green glassy material. The volume reduction from the unmelted mixtures was approximately a factor of eight. Melts 3, 7, and 8 formed viscous glasses that could not be poured from their crucibles. Melts 7 and 8 did start to flow but not enough to remove glass from the crucible, probably the result of high melting temperatures and rapid cooling. Melt 3 melted but did not show any evidence of pouring. Examination of these glasses by SEM indicates that melt 8 is a phase-separated glass, while melts 3 and 7 are homogeneous. Since these glasses did meet the TCLP criteria, as discussed below, they may still be useful if a melter technology is identified that can handle mixtures with high viscosity and high melting temperatures. However, at this point they are excluded from the compositional envelope. Optical evaluations of the ten compositional envelope glasses (melts 1-10) and three of the spiked glasses (melts 13-15) indicated that a glassy material with no visible inhomogeneities had formed. Melts 11 and 12 contained inhomogeneities.

The SEM/EDS analyses provide a means of semi-quantitatively verifying the homogeneity and compositions of the glasses produced during crucible melting. In general, these analyses indicate that the compositions of the crucible glasses are approximately those calculated from the component materials. The variations could be attributed to analytical error in the SEM/EDS system, volatilization of elements during melting, or variations in the compositions of the as-received waste samples. The analyses of the crucible glasses indicate that, on a scale that can be probed with the SEM/EDS system, most of the glasses are homogeneous. Exceptions to these observations were melts 11 and 12. Analytical data from melt 11 are shown in Fig. 1. The qualitative composition of the entire area in the photograph is depicted by the SEM/EDS spectrum labeled "Bulk Glass" at the top of the figure. The middle figures shows light-colored phases in a dark matrix at a microscopic level. Typical SEM/EDS spectra for each of these components are presented in the bottom figures. These spectra indicate that the light-colored phase is composed primarily of calcium and phosphorus. Preliminary trace element analyses further indicate that the light-colored phase preferentially sequesters arsenic from the dark matrix. A similar light-colored phase was found in melt 12. The composition of these two phases, as measured by SEM/EDS, is presented in Table IV.

The results of TCLP testing of the crucible melts are summarized in Table V. The row labeled "Untreated HEPA" presents TCLP data for the HEPA sample blended in A. This

row indicates how the unvitrified (as-received) HEPA portion of A performed in a TCLP test when it failed for cadmium leachability. It also provides a measure of the effectiveness of vitrification in stabilizing the hazardous elements in this material, especially with the spiked melts 13, 14, and 15. Also presented in this table is the leachant concentration limit that is used by the EPA to define whether a material is hazardous. Except for melt 1, all glasses passed the TCLP for each of the listed RCRA metals. Melt 1 leached mercury at the TCLP limit, which would exclude this glass from the compositional envelope. These analyses and this glass formulation are being further examined to evaluate whether analytical errors for mercury of the TCLP leachate produced this result.

DISCUSSION

Figure 2 is a ternary diagram for the system SiO_2+Al_2O_3:B_2O_3:R_2O, where all values are in weight fraction, and R represents alkali metals. The stars depict the individual waste streams STS, ECB, and HEPA. Also labeled are waste streams A (STS:HEPA:ECB ratio of 4:1:1) and B (STS:HEPA:ECB ratio of 4:1:2.5). The light-shaded region is the known glass-forming region for alkali borosilicate glass (2). Each numbered point corresponds to a glass in Table II. The glasses represented by white numbers and black backgrounds were made with LLMW for compositional development, while the glasses represented by black numbers and white backgrounds are formulations with added amounts of RCRA metals (see Table III). Melts 1, 2, 4, 5, 6, and 9 define the current compositional envelope of glasses for ANL-E LLMW vitrification (the region filled in with the dark shading). These melts form a region of alkali borosilicate glasses that can contain large amounts of ANL-E wastes (72 to 90 wt%), have good viscosities at 1100C, and pass the TCLP criteria for being nonhazardous. Furthermore, melts with similar compositions and spiked with relatively large amounts of hazardous metals also have these same characteristics. Therefore, this region of the alkali borosilicate glass system will be included in the compositional envelope for full-scale vitrification of ANL-E wastes.

The TCLP results suggest that vitrification can successfully treat mixed waste streams with concentrations of RCRA metals much greater than those typically encountered at ANL-E. The results also suggest that the production of additional crucible melts will demonstrate that the entire compositional envelope will likely be able to accommodate mixed waste streams from ANL-E. This is not the final compositional envelope since fabrication and testing of five additional melts are planned. The remaining five melts will be used to expand the existing compositional envelope beyond the existing boundaries, which would provide additional flexibility to the eventual users of the full-scale vitrification system.

CONCLUSIONS

The results of this study have identified a range of alkali borosilicate glasses that meet the acceptance critiera: glasses that melt, are easily poured from their crucibles at 1100C, and pass the TCLP test. The glasses that make up the compositional envelope were produced with relatively small amounts (7 to 28 wt%) of glass-forming additives, and the glass-forming additives that were used (borax, boric acid, and sodium carbonate) are easily handled materials. The compositional envelope will be expanded after the next phase of crucible melting.

Characterizations of the final waste forms with SEM indicate that most of the glasses that comprise the compositional envelope are homogeneous. This finding suggests that hazardous metals are uniformly distributed into the glassy matrices of each glass. Several glasses spiked with RCRA metals are not homogeneous, but still are processible and pass the TCLP, characteristics that do not preclude them from inclusion in the compositional envelope.

The results achieved with the spiked melts indicate that the identified compositional envelope will be able to accommodate unusually hazardous waste streams. The spiked melts provide compelling evidence that alkali borosilicate glasses are a robust waste form. Furthermore, the identified compositional envelope is broad enough to allow ANL-E to treat a diversity of composite waste streams, not only composite waste streams similar to A or B.

Completion of our off-gas analyses will provide a technical basis for designing an off-gas control system for any identified radioactive or hazardous species generated during vitrification.

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FIGURE CAPTION
Figure 1. SEM/EDS Spectra of a Cross-Section of Melt 11. The area these spectra were collected from is shown in the photograph. The scale bar at the bottom the photograph is 20 microns.
Figure 2. Ternary Diagram for Alkali Borosilicate Glasses Indicating Glasses Made in this Study. See text for details.

11-3

VENDOR VITRIFICATION PROGRAMS FOR THE SAVANNAH RIVER SITE

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ABSTRACT
Two programs are underway, which will provide waste vitrification services to the Savannah River Site in South Carolina. The first of these programs, is a model privatized waste remediation effort, which will provide the on site vitrification of nearly 700,000 gallons of mixed wastes currently in storage at the M area portion of the site. The second program, evaluates two vitrification technologies: joule melting and plasma melting, for their possible use to destroy asbestos containing materials targeted for remediation thought the Savannah River Site. This paper provides an overview of both programs currently managed by GTS Duratek, Inc. Under contracts with the Westinghouse Savannah River Co.

SAVANNAH RIVER M AREA WASTE VITRIFICATION

Currently, mixed wastes (hazardous and radioactive) are stored in nine tanks of the Interim Treatment/Storage Facility (IT/SF) and in the Mixed Waste Storage Shed (MWSS) by the Reactor Materials Department (M Area) of the Savannah River Site. These wastes are aqueous-based sludges generated from a nickel plating line process and contain uranium, which respectively classify them as hazardous (listed F006) and radioactive. GTS Duratek will couple a single-stage treatment unit (vitrification glass melter) to these tanks and, with the addition of common glass-forming chemicals, convert these wastes into a smaller volume of chemically durable glass. The glass generated during production will be regularly tested to guarantee the converted waste form satisfies TCLP release criteria to a high statistical certainty (95%). It is anticipated, that the approximately 2,960,000 kg of wastes will be reduced to approximately 1,081,000 kg of solid glass in the time frame of one year. A total waste volume reduction of 82.4% is expected. At the conclusion of the vitrification process, the storage tanks will be decontaminated to RCRA clean closure standards. A more detailed overview of this treatment strategy using a subcontracted vendor has been provided by Pickett, et. al. (1) Some initial vitrification tests on these wastes have been performed by Jantzen. (2) This program will be a model privatized commercial-scale waste remediation effort when vitrification operations begin in 1996.

In the 1960s, vitrification first gained popularity in the U.S. as a means to stabilize high-level liquid nuclear waste. (3) By the mid 1980s, slurry/liquid feeding to a joule-heated electric melter had become the preferred international approach to immobilizing high-level radioactive liquid wastes. (4) The process to be utilized for the SRS M Area waste solidification is an evolution of the technology developed for high level waste vitrification. The cost of the process has been substantially reduced, and it is more tolerant of a wider range of waste compositions. The process described below is a larger-scale version of one GTS Duratek has placed at the Fernald Environmental Management Project to process mixed waste (5) and one constructed to process asbestos wastes located at the Vitreous State Laboratory of The Catholic University (Washington, D.C.). (6) The off-gas treatment system of this system has been modified to accommodate NOx released during

the calcination of nitrate salts contained in the M Area wastes.

Waste Feed Preparation

The wastes stored at the M Area site partially occupy three 500,000-gallon tanks, six 35,000-gallon tanks, and about 125, 55-gallon drums (Watts Plating Solution, and Mark 15 Filtercake). Prior to vitrification, these wastes will be blended into two larger batches of waste which will be nearly the same composition. Due to the limited available space in the existing tanks, blending the wastes into a single volume is not possible. It is planned to consolidate all the wastes into two large "master batches" in two of the 500,000 gallon tanks. When processing begins, glass forming chemical additives will be mixed into the waste in one week batches prior to delivery to the melter.

Vitrification Melter

The waste feed slurry mixture described above is pumped to a single-stage vitrification unit (DuraMelterTM 5000). The slurry enters the melter through water-cooled entrance ports and is deposited on the surface of a molten glass bath. Heat from the glass bath is transferred vertically through the accumulated feed pile and evaporates free water from the slurry, calcines inorganic salts to metal oxides, and fuses the resultant mixture into a uniform melt which mixes with the molten glass inventory. All organic species in the feed are oxidized to carbon dioxide and water.

The DuraMelterTM 5000 is an electric, joule-heated, refractory-lined melter, which will convert the M Area wastes into leach-resistant solid glass. The melt basin will hold 1,950 liters of molten glass. This glass pool is heated by parallel submerged electrodes. Nominally, the glass bath will be controlled to 1150oC and can be discharged through either of two side exiting pour spouts. The two pour spouts are coupled to machines to produce glass gems (glass pieces resembling flattened marbles, about 1.5 cm in diameter). Each pour spout can be activated by an airlift, for controlled glass discharge. The melter is also equipped with two bottom drains for complete removal of the glass inventory at the conclusion of the M Area waste processing. In the plenum space of the melter (area above the glass pool), are lid heaters to provide supplemental power during start-up and operation. The DuraMelterTM 5000 is also equipped with a patented air bubbling and stirring system, which accelerates melter production capacity and oxidizes any organic and metallic inclusions in the waste feed stream. This melter will produce a minimum of 5 tons glass/day, with a maximum capacity of 15 tons glass/day. The mass balance for normal plant operations will be based on the 5 ton/day minimum.

Vitrification Off-Gas System

The melter is coupled to a multistage off-gas treatment system, which maintains the melter at a constant slightly negative pressure (-5 in. w.c.) and entrains and treats emissions resulting during processing. The reactions that occur when the slurry enters the melter generate steam, a variety of gases (NOx, SOx, CO2, and trace halides), and particulate. The front end of the off-gas system is composed of a water spray quencher followed by two aqueous-based packed bed towers connected in series.

Periodically the scrubber sump solutions will be transferred to the existing SRS M Area Dilute Effluent Treatment Facility (DETF). This stream will be routed through one of the empty 35,000-gallon tanks in the IT/SF building, which in turn are coupled to an existing supernatant transfer line to the DETF. Analyses of these solutions will occur prior to transfer to the DETF to determine the necessary treatment required. Primarily this return stream will contain water, sodium nitrate and sodium nitrite. Other species will also be found in this stream in lesser quantities.

Following the packed bed scrubbers is a dry filtration process which assures the absence of particulate radionuclides (uranium) in the final process exhaust. These filtration units are baghouses and high-efficiency particulate air (HEPA) (99.5%) filters. Redundant parallel filtration trains are provided to enable servicing of the equipment without interruption of the process. A sampling port is provided after the HEPA filters for air monitoring. The final exhaust will contain steam, carbon dioxide, and NOx.

Vitrified Glass Waste Form

The waste glass produced is a borosilicate glass designed to meet the TCLP release rate requirements while maintaining a high waste loading (large waste volume reduction). Nominally, the ratio of final glass volume to initial waste sludge

volume is expected to be 0.176. Development glass formation studies will continue until processing begins to lower this ratio while maintaining the TCLP requirements. This will ultimately yield less final waste volume. Adjustments in glass composition do not impact the performance of the process or the total mass composition of the secondary waste streams (off-gas sump solutions and exhaust stack emissions). However, increasing the waste loading can increase the rate at which some species are generated.

The glass gems will be poured directly into 71-gallon steel drums. Any void space in the container resulting from the gems will be filled with inert sand (silicon dioxide). This mixture will provide structural integrity to the drum for stacking. The 71-gallon drum has a square cross-section which allows for a higher drum-packing density.

COMPARISON OF JOULE-HEATED AND PLASMA ARCVITRIFICATION OF ASBESTOS

Asbestos-containing materials (ACM) from abatement operations are highly heterogenous. Asbestos fibers are normally dispersed in composite matrices which may be inorganic, such as portland cement in pipe or boiler insulation, or organic, such as tar in mastics used for waterproofing and vinyl in floor tiles. Thus, ACM can contain substantial quantities of portland cement, gypsum, tar, plastics, metallic objects, a variety of other construction materials, and materials contaminated during abatement, such as workers' protective clothing. ACM is normally collected in standard double polyethylene bags in a wetted condition to insure safe removal, handling, and transport. The amount of water can vary from 10% to about 80% by weight.

Vitrification of asbestos-containing materials (ACM) has been explored in the past.(7) The crystalline structure of all known asbestos fibers is destroyed above 900o C to form benign oxide compounds, suitable for glass forming. Vitrification is a thermal process which converts waste into glassy materials. GTS Duratek and its developmental partner, the Vitreous State Laboratory of the Catholic University of America, made major (patented) advances in the vitrification of a variety of low-level radioactive and hazardous wastes for the U.S. Department of Energy. More recently, GTS Duratek has been working with Plasma Energy Corporation to commercialize plasma arc vitrification for a wide variety of hazardous and asbestos-containing wastes.

We have repeatedly shown that vitrification completely destroys asbestos fibers, eliminating their hazard, while producing a recyclable product. At elevated temperatures, the crystalline asbestos fibers melt and the chemically bound water of crystallization is evaporated, producing a molten pool of non-hazardous magnesia and silica oxides and salts. This destruction of the asbestos fibers eliminates their hazard and the need for special landfill disposal. With appropriate additives, the molten oxides and salts cool to form glass which can be recycled to the glass industry as cullet. Without additives, the mixture will cool to produce a glassy slag (if not glass) which can be used as construction aggregate or filler. Most importantly, the process eliminates the continuing liability associated with landfilling of ACM.

Vitrification not only destroys the asbestos fibers, it also melts the other components of ACM wastes. Non-combustible co-contaminants, such as heavy metals, are chemically bonded into the structure of the glass product and become non-leachable, meeting the requirements of the Resource Conservation and Recovery Act (RCRA) Toxicity Characteristic Leachate Procedure (TCLP) for non-hazardous material. Any combustible co-contaminants, such as organic compounds, in the ACM are also fully destroyed at the high temperatures at which vitrification occurs. Other components of the ACM wastes are incorporated into the glassy matrix. This is due to the powerful solvating of molten glass materials and their resulting ability to dissolve a wide range of inorganic materials.

Commercially, the benefits of ACM vitrification are significant. It competes effectively with the current method of landfill disposal, especially when a value-added end product is produced. The vitrified product can be used as high-quality, light-weight aggregate, inexpensive glass blocks, or fiberglass wool. More importantly, vitrification ends the liability associated with the current disposal techniques.

Durameltertm Vitrification Systems

GTS Duratek is working with both joule-heated and plasma arc vitrification systems, capitalizing on the advantages of each for specific waste treatment problems. Both

vitrification systems are capable of destroying asbestos fibers in waste materials.

Joule Heated Ceramic Melter

GTS Duratek's large-scale Joule Heated Ceramic Melter (JHCM), the patented DuraMelterTM 1000, is designed to produce glass at a rate of up to 3,000 kg per day. The main melting cavity in the DuraMelterTM is constructed of Monofrax K3 ceramic refractory. The melt chamber holds over 60 liters of molten glass. The glass pool is maintained at a temperature of 1,150oC. Heating is provided by passing electricity through the bath across two pairs of Inconel 690 alloy electrodes located on opposite faces of the melt chamber. 200 kW of power is available to the electrodes, if needed. Inconel 690 and Monofrax K3 were chosen as the molten glass contact materials due to their high chromium content and their resistance to glass attack. Electric plenum heaters (30 kW total heating capacity) are located above the melt pool for initial start-up of the melter and for supplemental heating of the feed during operation, if needed. Temperature monitoring is also performed from above, via thermowells inserted through the melter lid and submerged into the molten glass bath. The thermowells are made of Inconel alloy and contain standard commercial Type K thermocouples which are connected to electronic read-outs. The vitrification melter, including the feed chute, is maintained under a negative pressure to prevent the release of asbestos fibers or other contaminants from the melter. The off-gas treatment system incorporates a wet scrubber, mist eliminator, baghouse particle filters, and a HEPA filter, and can treat a wide range of exhaust contaminants.

Full bags of asbestos are dropped into the melt chamber through a double air lock chute located at the top of the melter. The inlet to the feed chute is located in an adjacent Asbestos Containment Room. The air from this room is continuously exhausted through HEPA filters to prevent the release of any asbestos fibers into the work area or the atmosphere. The bags of asbestos waste fed into the melter fall directly onto the molten glass pool. A separate port is used to feed a complementing blend of glass forming additives onto the molten glass. The plastic bag containing the asbestos waste rapidly burn upon encountering the elevated temperatures within the melter, releasing the asbestos wastes. Other organic materials also burn away rapidly. Upon contact with the molten glass, the chemically bound water in the asbestos fibers evaporates, inorganic salts are calcined to oxides, and the asbestos fibers are thermally destroyed through dehydroxylation. The resultant mixture melts into the molten bath of glass. Glass is discharged from the melt chamber via an air lift operating through a side exiting orifice (located near the floor of the melter), which connects to a riser and pour trough.

The DuraMelterTM incorporates a patented gas mixing device. This system injects air into the bottom of the molten glass pool to (a) more efficiently mix the glass components with the feed, (b) control the oxidation potential of the molten glass, and (c) increase the electrical resistivity of the molten glass for more efficient heating. The net result is a significantly higher throughput and lower cost than other joule heated melter systems.

Plasma Arc Vitrification System

Plasma arc vitrification is conducted by Plasma Energy Corporation (PEC) using one of their Model PT-250 Field Convertible Plasma Arc Torches. This will allow for plasma arc operation in both transferred and non-transferred modes of operation. Air and nitrogen are routinely used as the plasma gases. Direct current power for the plasma arc torch is supplied by a 1 KHz power source designed for operation from 10% to 100% of rated capacity while maintaining a 0.9 power factor for high efficiency performance. The plasma system is water-cooled in a closed-loop system. The torch's cooling water flow, pressure and temperature are continuously monitored. Interlocks are provided to automatically shut down the system if these safety parameters go out of range.

The plasma arc vitrification reactor is a metal shell approximately 6 feet in diameter with a refractory lining of insulating bricks (six inches thick) anchored to the water-cooled shell of the reactor. The roof is insulated with a central opening for the plasma torch. Commercial refractory crucibles are located inside the insulating brick to hold the molten asbestos waste materials. The roof provides additional openings for the bagged asbestos waste feed, and an observation port. The furnace exhaust exits through the upper side wall of the reactor.

The ACM is fed into the reactor inside unopened cardboard delivery drums containing the bags of asbestos-containing materials. The cardboard drums are loaded into a chute, passing through alternating slide gates (to prevent escape of reactor gases)

into the furnace just above the melt. This procedure eliminates any potential for escape of asbestos fibers into the environment.
The reactor is fitted with thermocouples strategically placed around the periphery of the reactor crucible to monitor the temperature in the molten glass pool. The molten glass pool will be maintained at a temperature of approximately 1,300oC to ensure complete destruction of the asbestos fibers. Optical pyrometer temperature measurements will also be taken through the view port. The molten vitrified product exits the furnace through a water-cooled underflow/overflow spout on a continuous basis at a rate of 250-900 kg/h. The molten product is then chilled on a water-cooled conveyor and stored.

ACM DESTRUCTION OPERATIONS

ACM from Commercial Sources and the White House

The commercial ACM was obtained from two asbestos removal job sites - insulation materials from a boiler and pipes at a school in Maryland, and ceiling and wall insulation from a hospital in Washington, D.C. In addition to the asbestos-containing materials, cement, other friable materials, plastics, pieces of metallic objects including remediation tools, and workers' protective gear were also present. The hospital ACM contained approximately 30% to 50% fibrous materials which were identified as amosite, chrysotile, anthophllite, and termolite. The school ACM contained approximately 20% to 40% fibrous materials which were identified as orthochrysotile and anthophyllite. The ACM from the White House was asphaltic roofing material containing 10-45% chrysotile asbestos fibers. The organic content of this material was approximately 50 percent.
The polyethylene bags were not presorted or opened except for random sampling. Whole bags, pre-weighed, were fed one at a time into the melter. Observation of bag contents came from random sampling and observation of the contents as they spilled from the burning bags onto the surface of the molten glass pool.
The details of commercial ACM tests were discussed earlier. (8) The temperature of the molten glass was maintained between 1080o C to 1155o C during the feeding period. The plenum temperature varied from 750o C to about 1000o C depending upon the extent of the cold cap and combustion of organics in the plenum area.
Approximately 1,230 kg of the ACM from the school and the hospital, contained in 100 bags, were fed to the melter. The mass of the bags varied from 5 kg to 20 kg with an average of about 12 kg. The White House ACM was contained in 60 polyethylene bags weighing just under one ton.
The bags of commercial ACM were fed at two different feeding rates. In the first campaign, the average feeding rate was about 50 kg/hr (1.2 tons/day) and increased to about 72 kg/hr (1.7 tons/day) in the second campaign. The residence time in the melter was about 40 hours at 1.2 tons/day feed rate and 28 hours at the 1.7 ton/day feed rate.
The bags fell directly onto the surface of the molten glass pool and immediately began to burn. The combustion process was usually completed in 1 to 5 minutes leaving inorganic materials on the surface of the molten glass which were consumed in an additional 5 to 10 minutes.
Significant amounts of wire of varying thicknesses were observed on top of the glass pool. On one occasion, a large metal screen (at least 6 in. x 6 in. x 18 in.) was seen. It was observed to simply dissolve into the glass pool. During operation, it was observed that the electrical power required to maintain temperature dropped at higher feed rates. This is largely due to the heat of combustion of the plastic bags. Table I summarizes the operational conditions of the DuraMelterTM 1000 during both campaigns.(8)
During all tests, samples were taken frequently from the discharged glass, scrubber sump solution, and off-gas stream. Glass samples were ground and analyzed in accordance with the EPA Transmission Electron Microscopy (TEM) procedure.[9] Several glass samples, including the last output of the melter during each campaign, were examined. No asbestos fibers were found in any of the samples.

DOE Savannah River Site

A series of test campaigns are being conducted with ACM from the Department of Energy's (DOE) Savannah River Site (SRS) to compare the performance of joule-heated and plasma arc vitrification systems. A total of five campaigns will be conducted with each vitrification system. The feeds presented to each vitrifier will be comparable, using characterized and pre-sorted bags of asbestos wastes. This series of campaigns will present a range of feed compositions encompassing the range of

asbestos waste materials available at the Savannah River Site, including:
- Naturally occurring asbestos fibers
- Transite
- Asbestos containing floor tile
- Pre-formed thermal insulation
- Spray-on insulation
- Asbestos containing mastic
- Other miscellaneous ACM construction products

These parallel campaigns are designed to provide the necessary data to assess and compare the two vitrification technologies on the basis of asbestos waste destruction performance, costs, and regulatory compliance.

A wide range of data will be collected during each campaign, including:
- Operating Conditions - temperatures, pressures, flows
- Operating Costs - electricity, additives, maintenance and repair
- Operability and Reliability - maintenance and repair actions
- Performance - any residual asbestos fibers in the glass or other outputs

The data obtained will provide a complete evaluation of the effectiveness and suitability of each vitrification system to treat asbestos wastes. They will also provide sufficient data to reliably scale the performance and costs (capital and operating) of future asbestos waste vitrification systems.

CONCLUSION

Two ongoing programs centered on the use of vitrification for the remediation of wastes at the Savannah River Site have been reviewed. The M area vendor treatment facility will convert nearly 700,000 gallons of mixed wastes into durable glass using a new plant collocated at the waste storage site, and wholly managed and operated by a subcontractor to the site's management and operating contractor. A vitrification development program is also underway, where the destruction of SRS asbestos containing materials will be compared using two melting technologies: joule heated and plasma heated. This second program will highlight the benefits of both technologies for use in further consideration of vitrification as a treatment method for these types of wastes.

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11-4

LOW- AND INTERMEDIATE-LEVEL WASTE VITRIFICATION: BASIC PRINCIPLES, PROCESS UNITS AND PRODUCT CHARACTERIZATION

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ABSTRACT

Since the middle of 1970-th a low- and intermediate-level waste vitrification process was being developed in Scientific & Industrial Association (SIA) "Radon". There were proposed to vitrify both liquid and solid wastes with formation of borosilicate and aluminosilicate-based glass or glass ceramics. An activity was being trained as follows: development of glass and glass ceramic compositions suitable for production by electric melting and final disposal; development of vitrification flow sheet; selection of melter type, melter design, development of batch preparation method and design; calculations of optimized electrical and heat process variables; development of process automated control system; experimental study of vitrification process on lab-scale and pilot plants; product testing under natural conditions; development of full-scale plants, their installation and testing. The process basic principles are based on maximum of waste volume reduction and ecological safety and were assigned as follows: waste oxide content in final product has to be as high as possible; minimum non-radioactive additives to waste must be utilized; joint vitrification of liquid and solid wastes is preferable; radionuclide losses during the processing and storage have to be minimized; a chemical durability of final product may be similar one to bituminized waste or higher; the leach rate of radionuclides from the glass or glass ceramic wasteforms has not to be increased by one order of magnitude after the gamma-irradiation to absorbed dose of 100 kGy (such dose will be accumulated by product for 300 years of storage). From these points of view total waste oxide content in final wasteform can be carried to 80-90 or even 100% maintaining its chemical durability on the level of bituminized waste (10-4-10-5 g/(cm2 day)). By this waste volume reduction factor at vitrification is higher than one at bituminization by factor of 5-10. Solid residue of liquid waste contain mainly sodium salts (nitrate, carbonate, ortho-phosphate, sulfate, chloride and metaborate in the case of VVER waste from Nuclear Power Station) as well as calcium-magnesium bicarbonates and corrosion products. Solid wastes as incinerator ash, contaminated soil, spent inorganic ion-exchangers (zeolites, clays), asbestos materials, spent lining and heat isolation contain mainly high fusible oxides as network-formers - SiO2, Al2O3, P2O5 and ceramic forming species - CaO, MgO, Cr2O3, ZrO2, TiO2 etc. Thus, common vitrification of liquid waste from regeneration of ion-exchangers and contaminated soil (loam, clay, sandstone) results in production of almost homogeneous glass on either aluminosilicate or aluminoborosilicate basis. No any non-radioactive additives are used. These glasses may be produced in both Joule-heated ceramic and coreless induction (cold crucible) melters. Melting temperature is ranged between 1000 and 2000oC depended on liquid to solid wastes ratio. Chemical durability of glass also depends on glass composition. Vitrification of flux free incinerator ash results in formation of glass-ceramics. The nature of basic crystalline phase, its content in material and crystal dimensions depend on initial ash composition. Melting of "Radon" incinerator ash containing phosphates yields apatite-based glass ceramics. As far as glass forming agents are added to ash the glass fraction is increased. A borosilicate or aluminosilicate glass produced from liquid waste salts and contaminated soil may be used as glass forming additive. In this case total waste oxide content remains 100%. Spent inorganic ion-exchangers, asbestos materials, spent lining and heat isolation can be melted themselves in cold crucible at temperatures over 1500oC yielding glass-crystalline or fully crystalline materials as well as vitrified in common with low melted liquid waste salts. Contaminated metal reinforcement to be consolidated by direct either electroslag or induction melting. Small metal parts may be either subjected to oxidizing following by incorporation of oxides in glass and glass ceramics or embedded in glass matrix by nesting in containers following by molten glass filling. The most intimate waste processing takes place in plasma-heated shaft furnace. High temperature treatment of various solid wastes (organic, inorganic and metallic) results in production of metal-slag composite. Waste column in shaft prevents the radionuclide losses. Total radioactive cesium loss does not exceed 1-2%. A chemical durability of this wasteform is approximately the same as glass. High temperature treatment of various types of low- and intermediate-level wastes gives rise to opportunity to replace the

bituminization and cementation processes by melting/vitrification. Volume of vitrified block is smaller than bituminized block by a factor of 3-6 and smaller than cemented block by a factor of 10-20 at the same incorporated activity.

INTRODUCTION

Moscow government organization SIA "Radon" is concerned with collection, transportation, processing and final disposal of low- and intermediate level wastes as well as spent radiation sources in Central Russia.

Until 1990 the basic methods of waste immobilization were cementation (for low salt low level liquid waste), bituminization (for high salt, low- and intermediate level liquid waste), incineration (for burnable waste) and burial (for spent radiation sources and contaminated metal). Incinerator ash as well as another types of inorganic solid wasted were also immobilized by cementation. Above-mentioned methods are not fully safe and do not ensure maximal waste volume reduction coefficient.

Since the middle of 1970-th a low- and intermediate-level waste vitrification process was being developed in SIA "Radon". A works were directed to develop a method of immobilization of liquid waste in borosilicate glass. Natural calcium borosilicate (datolite), sandstone and loam clay were used as glass forming agents (1,2). A Joule-heated ceramic melter was used as basic unit of the vitrification plant. Primarily only liquid waste vitrification was considered. In the beginning of 1980 incinerator ash was proposed to be vitrified (3). Simultaneously boron-free aluminosilicate glasses and glass-crystalline materials (GCM) was proposed to be applied (2,4). A development of the vitrification process was restrained by the absence of reliable, safe and remotely operated melters are being able to be operated at high temperatures with a great life time. In the middle of 1980 such type of melter was found and being under design. This is so-called "cold crucible" - induction melter not containing any refractories and electrodes contacting with the melt. The cold crucible design has been described in detail (5). Since effective melter for the processing of high temperature materials has become available a conception is being similar to minimum additive waste stabilization project (6) was proposed. According to this conception various types of solid inorganic radioactive wastes (contaminated soil, spent ion-exchangers, lining, heat-isolation (for example asbestos materials), incinerator ash etc. should be utilized as additives to liquid waste alkali-containing salts at the vitrification. Until the present time this conception is partially realized. The liquid waste vitrification process was developed and the installation of the industrial-scale plant is being finished now. The solid wastes melting/vitrification processed are under development primarily on the stage of pilot plants. Some processes are under investigation now and the lab-scale plants are under exploitation.

RADON WASTE PROCESSING CONCEPTION

Block diagram of integrated waste processing developed in Radon is shown on Fig.1. Inorganic and organic radioactive wastes coming to Radon may be conventionally classified as liquid inorganic (water solution or suspension of salts), solid inorganic (contaminated soil, spent inorganic ion-exchangers, spent lining and heat-isolation, filter materials), solid organic (wood, paper, plastics, biomaterials, coals, rubber etc.), liquid organic (spent oils and extractives) and metallic (spent radiation sources, contaminated metal) wastes.

Liquid inorganic waste with low salt content is purified by reagent free methods for example electro-dialysis. High-salt waste contains various sodium salts such as nitrate, carbonate, borate (VVER waste), chloride, sulfate, calcium-magnesium carbonates as well as minor quantities of potassium, aluminum, phosphorus, transition metal compounds and organic impurities (Table I). The main radionuclides in this waste are betha-gamma emitters such as Cs-137 and Cs-134 (~90%), Sr-90 (~8%), Co-60 (~1-2%), REE radionuclides (<1%). Alpha-emitters (U,Pu) content is insignificant (<<1%). An average specific gamma-activity is 10-1000 MBq/m3.

Solid inorganic waste contains a very wide set of components and its chemical composition is widely varied. These materials lose a chemically bound water, volatile anions and organic constituent at calcination. The residue consists of high-fusible oxide compounds (SiO_2, Al_2O_3, CaO, MgO, TiO_2, ZrO_2, Cr_2O_3, MnO, FeOn, NiO occurred as oxides, silicates, titanates, spinels, calcium-magnesium phosphates, aluminates, aluminosilicates, ferrosilicates etc. - Table II). These wastes may be contaminated by both betha-gamma- and alpha-emitting radionuclides in comparable quantities.

Solid organic radwaste consists of cellulose (wood, paper, cardboard, clothes),

polymers and plastics (polyethylene, polypropylene, polyvinylchloride, teflon, plymethylmetacrylate, rubber etc.), biomaterials (trial animals, human pathology waste), spent active carbon, graphite, spent ion-exchange resins. More-over, inorganic and metallic constituents (glass and ceramic pieces, cables, reinforcement) occur in solid waste packages. Liquid organic waste is represented by spent oil and contaminated organic solvents and extractives (tributyl phosphate). Common incineration of solid and liquid burnable wastes yields incinerator ash (slag) whose composition depends on inorganic impurities content. The main components are SiO2, Al2O3, CaO, MgO, P2O5 (Table III). Both betha-gamma- and alpha-emitters may be occurred in this waste. Sometimes alpha-emitters are significantly predominant. The specific activity of alpha-emitters in some ash lots may reach to 100 MBq/kg while specific activity of betha-gamma-emitters is only 0.1-1 MBq/kg.

Radioactive metal waste may be divided into spent radiation sources (cesium, strontium, cobalt) and contaminated metal reinforcement (ferrous and non-ferrous metals). Spent radiation sources is incorporated in low-fusible metal matrix. Usually, contaminated lead and tin are used as matrix (7). The ferrous and some non-ferrous (aluminum, zinc, titanium) metal scrap may be oxidized or dissolved in recycled nitric acid and oxides or salts produced may be admixed to liquid inorganic waste (see Fig. 1) to be vitrified. The level of contamination of metal waste may be widely varied.

THE BASIC PRINCIPLES

The process basic principles are based on maximum of waste volume reduction and ecological safety and were assigned as follows:

- the vitrification process must be as simple, safe and reliable as possible;
- the waste oxide content in final product has to be as high as possible;
- minimum non-radioactive additives to waste must be utilized;
- common vitrification of liquid and solid wastes is preferable;
- radionuclide losses during the processing and storage have to be minimized;
- a chemical durability of final product may be similar one to bituminized waste or higher;
- the leach rate of radionuclides from glass or glass-crystalline wasteforms has not to be increased by one order of magnitude after the gamma-irradiation to absorbed dose of 100 kGy (such dose will be accumulated by product for 300 years of storage).

PROCESS UNITS AND EQUIPMENT

Liquid waste vitrification.

Process involves the steps as follows: waste preparation, batch preparation, vitrification itself, glass blocks annealing and off-gas purification with nitric acid recycling.Flow sheet of liquid waste vitrification plant was described in (8,9) in detail. Liquid waste from the waste tank is pumped to interim storage tank to be processed. This waste portion is concentrated in rotary evaporator to salt content of 1000-1100 kg/m3 following by the concentrate transfer to the batch mixer. Contaminated soil (sandstone, loam clay) is used as glass-forming additive to the waste salts. The batch containing clay particles with moisture of 20-25% possess by the plasticity, tyxotropy and is able to be transported within significant distance and to be stored for the long time in sealed reservoirs without segregation. So, the separation of the batch preparation and melting/ vitrification stages may be realized. The batch is fed into the melter by continuous or semi-continuous mode. At the stage of process development Joule-heated ceramic melter was used as basic unit. But then a more advantageous type of the melter was developed. This is high frequency induction melter - "cold crucible" not involving refractories and internal electrodes. At the present time three streams operating at the frequency of 1.76 MHz and output power of 160 kW per each are under operation now. Molten glass is poured into containers located in the pouring zone of a tunnel furnace for the solidified glass annealing. The containers with glass annealed are headed for the final disposal. Off-gas is purified and discharged to atmosphere. From time to time when it is necessary a melt portion is discharged in granulator to be pelletized. Silicate or borosilicate glass granules are used as glass-formers (flux) at incinerator ash vitrification. The main process variables are given in Table IV. The same plant can be used for NPP liquid, for example VVER, waste vitrification. Some problems at batch preparation arising from crystallization of sodium hydroxylmetaborate were successfully resolved by installation of an additional heating of feed line.

Solid waste melting/vitrification.
Solid radwaste contaminated by mainly Sr-90 and alpha-emitters or chemically bonded cesium is suggested to be melted/vitrified in plasma melter. Various constituents of solid radwaste is fed into the plasma melter which is ceramic vessel cooled from outside. Arc torch plasmatrones are used as a source of heating. Waste constituents ratio is maintained so as melt and solidified product to be optimized for the properties. The melt is poured into canisters which then is sealed and headed for ultimate disposal. The plant is supplied by off-gas system. This method is suitable for processing of solid wastes not containing the great quantities of volatile components to avoid a volatilization of radionuclides with off-gas. This is considered as candidate method for processing of contaminated soil, spent inorganic ion-exchangers (zeolites, clay minerals etc.), spent lining (refractory materials) and heat isolation (asbestos materials, glass and ceramic fibers), spent filter materials (silica gel, keramzit etc.), materials formed at Nuclear Power Plant decommissioning (concrete, construction materials). These materials have a high melting points. Process variables are represented in Table IV.
Solid and liquid burnable wastes are subjected to incineration in chamber furnace with fuel heating. Waste decomposition results in gases release and formation of incinerator ash residue including inorganic residue of burnable waste and inorganic impurities finding way into the solid waste packages. Incinerator ash may be also melted/vitrified in the plasma melter. However, its melting point is not so high. Moreover, ash contains significant quantity of cesium radionuclides. The ash processing at very high temperatures commonly with another types of solid wastes can results in elevated cesium loss. A possibility of ash processing by this way is also considered. But at the present time other method of ash processing is employed now. Vitrification of incinerator ash is conducted in induction melter of cold crucible type supplied by dumping bottom for the discharge of collected metal. Borosilicate or aluminosilicate glass granules prepared from vitrified liquid waste are utilized as glass forming additives (fluxing agent) to ash. Total ash oxides content reaches 70-80 wt.%. The principal process variables are given in Table IV.
Before the vitrification in cold crucible, incinerator ash is subjected to separation on coarse and fine fractions. Fine fraction is directly headed for the melter. Coarse fraction is charged into containers and filled by melt from cold crucible. Metallic impurities are collected on the crucible bottom and periodically discharged in containers by a hinging bottom out. Incinerator ash vitrification process in cold crucible was described in detail in (8,9).
Solid waste processing in plasma-fuel heated shaft furnace with liquid slagging
This process offers common treatment of burnable, non-burnable and some fraction of inorganic wastes. Waste packages containing cellulose, biological, polymer, carbon materials and inorganic impurities (glass, ceramics, soil, metal etc.) are charged into the shaft furnace heated by plasma-fuel torches. Organic constituent is subjected to heating, melting, gasification, pyrolysis and decomposed yielding gaseous phase and coke. Inorganic constituent undergoes a complex transformation including solid phase reaction under reducing condition with mineral formation yielding a slag. Off-gas is reburned in an after-burning chamber heated by plasmatrone. Slag is partially melted on the bottom of the shaft at temperature of 1400-1500oC and poured into containers. The main process variables are shown in Table IV. A significant process advantages are the solid waste processing in single unit and very low radionuclide (particularly cesium) loss (1-2% and lower) (10).
PRODUCTS CHARACTERIZATION
As a result of waste processing in cold crucibles, plasma melter and plasma shaft furnace three types of wasteforms are produced: glass, glass-crystalline and predominantly crystalline materials (Table V).
Vitrification of liquid radioactive waste in cold crucible yields a vitreous materials not containing appreciable quantity of crystalline phases. Based on X-ray diffraction and electron microscopy techniques these materials may be characterized as glasses. Infra-red and electron paramagnetic resonance study showed that structural basis of glasses is random network of SiO_4 and AlO_4 or SiO_4 and BO_4 tetrahedra as well as BO_3 triangles for aluminosilicate and borosilicate glasses respectively. Waste elements occur mainly as cation-modifiers (Na, K, Cs, Mg, Ca, Sr, Fe, Co, Ce, Nd etc.). Actinides content in glass is deficient to conclude correctly about their state in glass structure.
Vitrification of incinerator ash in cold crucible yields a glass-crystalline

materials (GCM) where vitreous to crystalline phases ratio is widely varied depending on waste components ratio. An investigation of waste elements partitioning between the phases has shown that most of sodium, potassium and cesium as well as a some fraction of strontium and iron enter the vitreous phase but most of calcium, strontium, iron, cobalt, nickel, rare earth elements occur in various crystalline phases (silicates, silicophosphates, spinels, intermetallic compounds etc.). Alpha-emitting radionuclides (primarily U-238, Pu-239 and Am-241) enter both vitreous and crystalline phases.

Slag product from plasma shaft furnace is glass-crystalline. As in the case of ash vitrification, it contains both vitreous and crystalline (apatite-based) phases. Their ratio is depended on the waste composition. Waste elements partitioning between the phases of this product is similar to one in vitrified ash.

Plasma melting of solid inorganic waste results in formation predominantly crystalline material with minor quantity of vitreous phase. Mineral assemblage is strongly depended on the waste composition. Cesium radionuclides tends to be transferred to the vitreous phase.

The properties of the wasteforms produced are shown in Table V. The greatest losses of radionuclides during the process were observed from plasma melter due to the highest operating temperature and opened melt surface. The least losses are at liquid waste vitrification because the lowest process temperature (1100-1200oC) and the presence of the batch layer on the melt surface.

Cesium radionuclides are the most mobile in all cases. The leach rate of Cs-137 from alumino-silicate glass, containing 40 and 50 wt.% of liquid waste oxides is approximately 3 10-11 and 1 10-9 kg/(m2 s) respectively (measured by IAEA technique (11)). These values are lower by 1 to 3 orders of magnitude than found for bituminized waste with the same incorporated radioactivity. Gamma-irradiation of glass to absorbed dose of 100 kGy (such dose will be accumulated by glass for 300 years of the storage) did not increase the leach rate of cesium more than by a factor of 3-5. The leach rate of cesium from borosilicate glasses is found to be on the same level. The leach rates of strontium, cobalt and actinides were approximately by 1-2 orders of magnitude lower than cesium. They were almost not increased after the gamma-irradiation to dose of 100 kGy.

The leach rate of cesium from GCM produced at the incinerator ash vitrification in cold crucible as well as the waste processing in plasma shaft furnace is slightly lower compared to glass. It is probably connected to partial incorporation of cesium in slag minerals. The leach rates of strontium, cobalt and REE were very low due to their predominant accumulation by slag minerals with high chemical durability. An effect of gamma-irradiation on leach rate of Sr, Co and REE is negligible.

The leach rates of Sr, Co, Fe and REE from fully crystalline materials produced at solid inorganic waste processing are very low and often they are lower than detection limit. The final wasteform has a great radiation stability.

The highest waste volume reduction factor was reached at solid organic waste processing with liquid slagging or ash vitrification. It reaches of 300-500. The volume of vitrified incinerator ash compared to volume of initial ash is less by a factors of 3-5. Volume reduction factor at liquid inorganic waste vitrification is 3 to 4. It is approximately the same at the solid inorganic waste plasma melting. Thus the volume of glass or GCM block is smaller than the volume of waste-bitumen or waste-cement block with the same activity by a factors of 3-5 and 10-20 respectively.

CONCLUSION

The melting/vitrification of low- and intermediate-level radioactive wastes is more preferable method of their immobilization compared to bituminization or cementation. This method converts almost of any types of radioactive wastes in stable, compact and chemically durable form suitable for a long-term storage and final disposal. A significant volume reduction is reached. Common vitrification of the different types of radioactive wastes excludes as far as it is possible utilization of an inactive additives. In that case the waste oxide content reaches up to 100% and the waste volume reduction factor can reach of 3-5 to more than 500.

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THE EFFECT OF CHEMICAL COMPOSITION ON THE PCT DURABILITY OF MIXED WASTE GLASSES FROM WASTEWATER TREATMENT SLUDGES*

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ABSTRACT

An experimental program has been designed to examine the chemical durability of a large glass composition space derived from the vitrification of simulated wastewater treatment sludges. These sludges represent the majority of low-level mixed wastes currently in need of treatment by the U. S. Department of Energy. The major oxides in these model glasses included SiO_2, Al_2O_3, B_2O_3, Na_2O, CaO, and Fe_2O_3. In addition, three minor oxides, BaO, NiO, and PbO, were added as hazardous metals. FeO was formed during melting. The major oxides were each varied at two levels resulting in 32 experimental glasses. The compositions of these glasses represented the extreme vertices of a five-dimensional "hyperspace." The chemical durability was measured by the 7-Day Product Consistency Test and normalized sodium release rates (NRRNa) of these glasses ranged from 0.01 to 4.99 g.m-2.d-1. The molar ratio of the glass-former to glass-modifier (F/M) was found to have the greatest effect on durability. Glass-formers included SiO_2, Al_2O_3, and B_2O_3, while Na_2O, CaO, BaO, NiO, FeO, and PbO were glass-modifiers. As this ratio increased from 0.75 to 2.0, NRRNa was found to decrease between one and two orders of magnitude. Another important factor was the Na_2O/CaO ratio. As this ratio increased from 0.5 to 2.0, NRRNa increased up to two orders of magnitude for the glasses with the low F/M ratio but almost no effect was observed for the glasses with the high F/M ratio. Increasing the iron oxide content from 2 to 18 mole percent (4 to 35 weight percent) was found to decrease NRRNa about one order of magnitude for the glasses with low F/M but iron

had little effect on the glasses with the high F/M ratio,The durability also increased when 10 mole percent Al2O3 was included in low iron oxide glasses but no effect was observed with the high iron glasses. The addition of B2O3 at levels up to 9 percent had little effect on durability. The effects of other composition parameters on durability are discussed as well. In addition, a strong positive linear relationship between pH and log NRRNa was also observed when leachate pH was greater than 11.

BACKGROUND

The Mixed Waste Integrated Program (MWIP) at the U. S. Department of Energy (DOE) has taken a serious look at vitrification for treatment of low-level mixed waste streams. This was prompted, in part, by the fact that the EPA has declared that vitrification is the Best Demonstrated Available Technology for the disposal of high-level radioactive waste (1). Among the low-level mixed waste streams within the DOE complex, which are under consideration for vitrification, are wastewater treatment sludges at the Savannah River Site (SRS), Oak Ridge Reservation (ORR), Rocky Flats Plant (RFP), and Los Alamos National Laboratory (LANL). These sludges typically contain low levels of radioactive and hazardous metals, water, and additives from the waste treatment process itself. The treatment additives normally include precipitating and flocculating agents such as sodium, calcium, magnesium, and iron compounds or siliceous filter aids from the dewatering process. These treatment additives tend to be soluble in silicate glass and are thus amenable to vitrification.

The compositions of these sludges vary not only according to the original waste but also according to the wastewater treatment process. Stored in M-Area at SRS, is a wastewater treatment sludge from their aluminum and nickel cladding operations. Even though uranium and nickel are the chief radioactive and hazardous species in this waste, the sludge is primarily comprised of sodium hydroxide from the precipitation process and perlite and diatomaceous earth from the dewatering process (2). These two filter aids are comprised chiefly of silica and alumina. Vitrification has already been chosen as the treatment for this sludge (2). Another candidate mixed waste is stored at the West End Treatment Facility (WETF) at ORR. Species targeted for immobilization include uranium and various hazardous metals but the majority of the waste consists of calcium and aluminum hydroxides from the precipitation process (3). An example of a RFP waste is the aqueous sludge from the plutonium recovery operations. In addition to small amounts of uranium, plutonium, and americium and several hazardous metals, this waste consists primarily of sodium, calcium, magnesium, and iron hydroxides from the treatment processes and some siliceous filter aid material from dewatering. The process water from decontamination operations at LANL contains small amounts of uranium, plutonium, and americium. The resulting sludge from the wastewater treatment process is comprised chiefly of calcium hydroxide from precipitation and alumina and silica from the spent filter aids (4). A summary of the major oxide constituents in these mixed wastes is shown in Table I.

Pilot-scale vitrification testing of these wastes is currently being carried out at Clemson University (5). To date, tests have been carried out on surrogates of the M-Area sludge and the WETF sludges. This resulted in the production of glasses which passed the Toxicity Characteristic Leaching Procedure (TCLP) as per the Land Disposal Restrictions (6,7). Formulations for the glass compositions used in these tests included glass-forming siliceous materials such as perlite, diatomaceous earth, and precipitated silica and fluxes such as sodium tetraborate or sodium carbonate. Results, so far, indicate that these sludges may be good candidates for vitrification.

A large glass composition space has been developed in these studies which is intended to bracket most glass products that might result from the vitrification of wastewater treatment sludges currently in storage at DOE sites throughout the United States. Glasses have been prepared with simplified compositions and tested for chemical durability by the 7-Day Product Consistency Test. This should provide not only target composition ranges for each of these waste types but also a data base from which to predict the behavior of glassy products resulting from the vitrification of a wide variety of mixed wastes.

EXPERIMENTAL

Experimental Design

A glass composition space has been developed to serve as a simplified model for

studying the durability of the products which might result from the vitrification of wastewater treatment sludges. This mixture is composed of six variable components SiO2, Al2O3, B2O3, Fe2O3, Na2O, and CaO. Three hazardous species, BaO, PbO, and NiO were also included at fixed levels. These components are classified into three groups depending upon their role in the glass structure. The first group, glass-formers (F), includes SiO2, Al2O3, and B2O3. The second group, glass-modifiers (M), includes Na2O and CaO plus the hazardous species BaO, PbO, and NiO. Lastly, Fe2O3 is grouped by itself. The sum of the six variable oxides always totals 95 mole percent while the BaO, PbO, and NiO levels are fixed at 2, 1, and 2 mole percent, respectively.

With the aid of ternary phase diagrams and exploratory experiments, a large glass-forming region was identified which is illustrated in the F-M-Fe2O3 ternary phase diagram shown in Fig. 1. The four vertices of the region represent the compositions given in Table II, labeled 1, 2, 3, and 4. Note that Fe2O3 content is either 2 or 18 mole percent. These correspond to about 4 and 35 weight percent, respectively. At the low iron level, the ratio of glass-former to glass-modifier (F/M), is either 42/56 or 63/35. At the high iron level, F/M is either 35/47 or 56/26. These four vertices of the glass-forming region can then be represented by two independent compositional variables, the Fe2O3 content and the F/M ratio. This two-dimensional composition space is further expanded by the substitutions of Al2O3 and B2O3 for SiO2, giving rise to two additional compositional variables. B2O3 is added at 0 or 1/7 mole percent of the total glass-former content. Al2O3 is substituted for SiO2 at either 0 or 10 mole percent of the total glass composition. A final compositional variable, the Na2O/CaO ratio, is either 0.5 or 2.0. This increases the number of independent compositional variables to five, resulting in a five-dimensional space consisting of 32 vertices. This composition space is referred to as "Hyperspace." The target glass compositions of these 32 vertices are given in Table III, in oxide mole percent.

The scheme used to identify these glasses is comprised of a number followed by three letters. Two of the compositional variables are identified by the number and the remaining three variables are identified by the three letters. The number refers to either of the four vertices of the two-dimensional space in Fig. 1. But after expansion, these vertices become three-dimensional "subspaces." Subspaces 1 and 2 each contain 2 mole percent Fe2O3 while the compositions in Subspaces 3 and 4 contain 18 mole percent Fe2O3. Subspaces 1 and 3 contain the lower F/M ratios, 42/56 and 35/47, respectively, while Subspaces 2 and 4 contain the higher F/M ratios, 63/35 and 56/26, respectively. These four subspaces are arranged in Fig. 2 to illustrate the vertices of the five-dimensional Hyperspace.

Within each subspace, the Na2O/CaO ratio, the B2O3/F ratio, and the Al2O3 content vary similarly. The first letter is either an L or H depending on whether the Na2O/CaO mole ratio is low or high, i.e. 0.5 or 2.0. The second letter is either O or H depending on whether the B2O3/F mole ratio is 0 or 1/7. The last letter is either O or H depending on whether the Al2O3 content is 0 or 10 mole percent. This scheme is summarized in Table II and all 32 vertices of the Hyperspace are depicted in Fig. 2. To illustrate this scheme, the compositional variables of Glass 1LOO are given as follows: F/M, 42/56; Fe2O3, 2 mole %; Na2O/CaO, 0.5; B2O3/F, 0 mole %; and Al2O3, 0 mole %, while the hazardous metal oxides are always fixed at: BaO, 2 mole %; PbO, 1 mole %; and NiO, 2 mole %. The target oxide compositions for all the glasses are given in Table III. The glasses representing the compositions of all 32 vertices of this five-dimensional space were prepared in order to study their durability. Four of the compositions, 4LOH, 4LHH, 4HOH, and 4HHH, did not result in a homogeneous melt and their compositions had to be modified by lowering the iron content. The adjusted compositions for these glasses are indicated in Table III.

Glass Preparation

Glasses were prepared according to the target compositions in Table III. Reagent grade oxides or carbonates were mixed and then melted in high purity alumina crucibles for two hours at 1350C in a bottom-loaded glass melting furnace manufactured by Deltech, Inc. of Denver, CO. Afterwards, each melt was quenched on a stainless steel plate. The resulting glass was milled to a -35 mesh powder which yielded about 1 kg of glass. The resulting powders were then remelted in platinum crucibles at 1350C for 2 hours and cast into graphite molds to produce disks 40 mm in diameter. The glass disks were immediately placed in an annealing furnace at 450C and allowed to gradually cool to room temperature.

Glass Analysis

The elemental composition of each glass disk was determined by wavelength dispersive XRF spectrometry. Each glass disk was first polished to a 600 grit finish. This analysis was performed on a Rigaku Model 3271 Sequential XRF spectrometer utilizing a "standardless" fundamental parameters software routine developed by Rigaku. The instrument description and conditions have been previously described (8). Selected glasses were also analyzed by wet chemical techniques (9) to confirm the XRF results (10). The redox nature of the glass was estimated by the determination of the Fe2+/Fe3+ ratio (11). In this procedure, a powdered glass sample was mixed with ammonium vanadate to preserve the redox stoichiometry and then dissolved in hydrofluoric and hydrochloric acids. The Fe2+ content was determined colorimetrically after complexation with FerroZine iron reagent. Total iron content was then determined the same way following reduction of any Fe3+ with citric acid. Chemical durability testing was carried out by the 7-Day Product Consistency Test (12). This test measured the concentrations of all nine cations released from the crushed glass (75-150 m) into the leachate after heating in deionized water at 90oC for seven days. The normalized elemental release rate (NRRi), in g.m-2.d-1, was determined from Eq. 1,

Eq. (1)

where Ci is the concentration of element i in the leachate, in gm-3, VL is the volume of the leachate, fi is the weight fraction of element i in the original glass, SAg is the surface area of the glass, and t is leaching time. The SAg/VL ratio is assumed to be 1950 m-1. This PCT test was carried out in triplicate for each glass and the average pH and elemental release results are reported.

RESULTS AND DISCUSSION

The chemical oxide content of the glass products, representing the vertices of the composition space, were determined by XRF spectrometry and are presented in Table IV. These results were compared to results from the analysis of several glasses by wet chemical dissolution followed by ICP and AA spectroscopy (10). This comparison showed no statistical difference in the results between the two analytical methods which confirmed the accuracy of the XRF method. This appears to be one of the first reported uses of XRF spectrometry for an extensive quantitative chemical analysis of waste glasses. The simplicity of this technique over the more tedious wet chemical methods greatly facilitated waste glass analysis and should be exploited more in the future.

The measured compositions are in close agreement with the intended target values from the experimental design. One minor exception is the higher than expected alumina content which is presumably due to corrosion of the alumina crucibles during the first melting step. Another difference is the presence of FeO in the glasses which resulted from reduction of the Fe2O3 during melting. Nevertheless, since the actual glass analyses are similar to the target values, the original target compositional variables will be used in the following discussions. These discussions on durability will focus on the following five composition variables.

- glass-former to glass-modifier ratio (F/M)
- iron oxide content Fe2O3
- Na2O/CaO ratio
- B2O3 to glass-former ratio (B2O3 /F)
- Al2O3 content

The results from the 7-Day Product Consistency Test are presented in Table V as normalized elemental release rate, in g.m-2.d-1. The variability in the individual release rates (NRRi) is due either to selective leaching of one element over another or reprecipitation. However, in phase separated glasses or glasses with crystalline phases, leaching variability can also result from the difference in durability of the different phases present. As expected, the highest release rates are for sodium (NRRNa) and it is these values which probably best reflect the relative durabilities of the various glasses. The NRRNa is presented again in Table VI.

There is no PCT benchmark by which low-level mixed waste glasses are judged for chemical durability. In the absence of a durability criteria, comparisons can be made with high-level waste glasses, for which the PCT test was developed. The durability of several high-level waste glasses, including SRL 131 and SRL 165, have been studied extensively and appear to be quite durable. The NRRNa for these glasses is 0.04 and 0.02 g.m-2.d-1, respectively (13). The Environmental Assessment (EA) glass, which has a sodium release rate of about 1 g.m-2.d-1 represents the minimum

acceptable durability for HLW glass at the Defense Waste Processing Facility at the Savannah River Site (14).

Relationship between pH and Normalized Sodium Release Rate

The relationship between leachate pH and log NRRNa is plotted in Fig. 3. Above a pH of 11, this glass system exhibits a strong positive linear relationship between these two parameters. Below a pH of 11, no such correlation existed and the line in the figure is not fitted to these points. This correlation is not surprising. The leachate pH is expected to increase upon dissolution of silicate glasses because the dissolution mechanism involves the exchange of protons from the leachate for sodium ions in the glass. This results in an increase in pH and an increase in the sodium content of the leachate.

Effect of Glass-Former to Glass-Modifier Ratio on Durability

There are four F/M ratios examined in this composition space. For the low iron glasses, the glass-former to glass-modifier ratios are 42/56 and 66/35. The F/M ratios for the high iron glasses are 35/47 and 56/26. When the corresponding sodium releases for low iron glasses are compared, one finds that the NRRNa increases by up to two orders of magnitude as F/M decreases. Therefore, with low iron glasses, durability is quite sensitive to the F/M ratio and decreases significantly as F/M decreases, regardless of the other compositional variables. Examination of NRRNa reveal that all compositions with low iron and high F/M have comparable durability to the HLW glasses such as SRL 131 and SRL 165 (13). The NRRNa values are all below 0.30 g.m-2.d-1. However, five of the eight glasses with low iron and low F/M ratio have release rates in excess of 1.0 g.m-2.d-1. Therefore, in the vitrification of low iron oxide-containing glasses, special attention must be paid to the F/M ratio in order to ensure a consistently durable product.

For the high iron-containing glasses, the effect that F/M has on durability depends upon additional compositional variables, especially the Na2O/CaO ratio. The F/M ratio significantly affects NRRNa when the Na2O/CaO ratio is high, the durability, however, does not appear to be affected when the ratio is low. This suggests that sodium release rates would not be as sensitive to composition fluctuations in the F/M ratio if the glasses had both a higher iron oxide content and lower Na2O/CaO ratios.

The sodium release rates for all glasses with the high F/M ratios are all less than 0.30 and most are below 0.10 g.m-2.d-1. The durability of these glasses is comparable to that of the HLW glasses. The NRRNa for glasses with low F/M ratio and high Na2O/CaO ratios range between 0.96 and 4.99 g.m-2.d-1.

Effect of Iron Oxides (FexOy) on Glass Durability

The majority of glasses in this composition space had iron oxide contents of either 2 or 18 mole percent. The FeO and Fe2O3 content in these glasses are given in Table IV. The resulting FeO to FexOy ratio in these glasses ranged from 0.08 to 0.49 indicating that the glasses were mostly oxidized. No attempt was made to examine the effect of iron redox state on glass durability.

The NRRNa values for the glasses with high iron content were all below 0.10 g.m-2.d-1, except for the glasses that had both low F/M and high Na2O/CaO ratios. These four glasses had sodium release rates near or above 1.0 g.m-2.d-1. The low iron glasses displayed a similar trend; the majority of the release rates were below 0.29 gm-2d-1, again however, the glasses with both a low F/M ratio and a high= Na2O/CaO ratio all had NRRNa values above 2.5. In addition, the low iron glasses, 1LOO and 1 LHO, had NRRNa values of 1.16 and 0.65, respectively.

The effect of increasing the iron content on release rates is discussed. For the glasses with a high F/M ratio, i.e. Subspaces 2 and 4, the iron content has little affect on NRRNa. The only measurable effect is with the high Na2O/CaO ratio glasses, but the effect is minimal. For the glasses with a low F/M ratio, i.e. Subspaces 1 and 3, the durability does show a significant dependency on iron content, the NRRNa decreases as the iron content increases. So increasing the iron content from 2 to 18 mole percent does significantly reduces leaching if the F/M is 0.75; but there is little effect if the F/M ratio is up around 2.0.

Effect of Na2O/CaO Ratio

The Na2O content in the 32 glasses of this study ranges from about 6 to 30 weight percent. The glasses with the Na2O content between 6 and 12 weight percent, all have release rates below 0.1 gm-2d-1 while the glasses with higher soda content tend to have considerably higher release rates.

The glasses all have Na2O/CaO mole ratios of either 0.5 or 2.0. Glasses with the

lower ratio tend to have lower release rates, as expected. The effect of the Na2O/CaO ratio on the NRRNa is variable and depends upon the F/M ratio in the glass. When the F/M ratio is high, the Na2O/CaO ratio has little effect upon the release rate, i.e. all the glasses have relatively low NRRNa. But when the glass-former content is low, the Na2O/CaO ratio has a strong influence on durability as discussed previously. Values for NRRNa tend to increase one to two orders of magnitude as the Na2O/CaO ratio increases from 0.5 to 2.0.

Effect of B2O3

In half of the glasses in this study, B2O3 was substituted for SiO2 at levels between 6 and 10 weight percent; the other half contained no B2O3. The addition of B2O3 appears to have no significant effect on durability. Since B2O3 has a tendency to lower glass viscosity, advantage can be taken of this to improve melt processing without sacrificing durability. The effect of adding B2O3 at levels above 10 weight percent were not examined but are expected to decrease glass durability.

Effect of Al2O3

Aluminum oxide has been substituted for SiO2 at levels between 11 and 18 weight percent. The glasses without added Al2O3 typically contain less than 2 percent, resulting from corrosion of the alumina crucible during the first melting procedure. The effect of added Al2O3 on release rates is dependent upon the iron content in the glass. The durability is improved when Al2O3 is added to the low iron-containing glasses but showed no effect when added to the high iron-containing glasses.

CONCLUSIONS

A large glass composition space has been developed which is intended to bracket most glass products that might result from the vitrification of wastewater treatment sludges currently in storage at DOE sites throughout the United States. These surrogate glasses attempt to simulate the actual waste glasses with nine of the more representative oxides. The major oxides include SiO2, Al2O3, B2O3, Fe2O3, Na2O, and CaO. In addition, three minor oxides, BaO, NiO, and PbO, are included to represent hazardous metals. The levels of the six major oxides were varied in this mixture and, as a result, five independent variables were selected. These five composition variables included the glass-former to glass-modifier ratio (F/M), the iron oxide content, the Na2O/CaO ratio, the B2O3/F ratio, and the Al2O3 content. Glasses were made whose compositions corresponded to the 32 vertices of the resulting five-dimensional "hyperspace." The chemical durability of these glasses were determined by measuring the sodium release rates (NRRNa) by the 7-Day Product Consistency Test.

A comparison of the normalized sodium release rates for these glasses showed that their chemical durability can be quite sensitive to changes in the F/M ratio, the iron oxide content, and the Na2O/CaO ratio. The F/M ratio had the greatest effect on durability for the glasses examined in this study. As F/M increased, the durability of all the glasses improved by one to two orders of magnitude. Furthermore, among the glasses with a low F/M ratio, it was found that raising the Fe2O3 content from 2 to 18 mole percent (4 to 35 weight percent) increased the durability by about one order of magnitude. On the other hand, increasing the Na2O/CaO ratio from 0.5 to 2.0 decreased the durability about one order of magnitude. The B2O3 content had very little effect, while the effect of Al2O3 was mixed, with some glasses showing improvement in durability and others showing no change.

All of the glasses with the F/M ratios greater than one, had normalized sodium release rates below 0.3 and most were below 0.1 g.m-2.d-1. These release rates were comparable to the durable SRL 131 and SRL165 HLW glasses from the Savannah River Site. Glasses with the lower F/M ratios were much less durable, however, one durable composition region was detected for glasses with a low F/M ratio. It was found that relatively durable glasses could be produced with a low F/M ratio if they also had a low Na2O/CaO ratio and contained 10 mole percent Al2O3 (~14 weight percent).

This sensitivity in glass durability to composition fluctuations necessitates the need for careful control of the glass composition during vitrification. Furthermore, these waste streams can be quite inhomogeneous. And this inhomogeneity can lead to significant deviations from the target glass compositions resulting in production of a glass with poor durability. And so, implementation of a strategy to control glass composition is mandatory, if glass durability is to be consistently maintained during vitrification.

This paper reports the first in a series of studies intending to fully characterize the durability of a large range of glassy products which might result from the

vitrification of wastewater treatment sludges. Work is presently underway to prepare and test additional glass compositions within this space and to test these glasses after they have been heat treated to simulate slow cooling rates and greater devitrification. All of these glasses are being further analyzed for phase composition in an effort to characterize the effect of cooling rates and devitrification on durability. Future work will also report on the TCLP results of these glasses. Finally, an attempt will be made to develop an empirical model to predict durability from glass composition and cooling rate.

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11-6

AN EVALUATION OF GLASS-CRYSTAL COMPOSITES FOR THE DISPOSAL OF NUCLEAR AND HAZARDOUS WASTE MATERIALS

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ABSTRACT

This program demonstrates that waste forms composed of a glass-crystal composite can be applied to a much wider range of waste streams than those amenable to homogeneous glass production alone. With this approach, radioactive constituents (U and Ce, the latter as an analogue for Pu) and hazardous metals (such as Cr, Cd, and Ni) have been strongly partitioned into corrosion-resistant mineral phases, with Pb also showing limited incorporation in minerals as well. A variety of corrosion tests have shown the glass-crystal composite to be more durable than high-level nuclear waste glasses, such as SRL 202.

INTRODUCTION

Waste forms made of a glass-crystal composite (GCC) are being evaluated at Argonne National Laboratory for their potential use in the disposal of low-level nuclear and hazardous waste materials. This waste form is being developed within the framework strategy of DOE's Minimum Additive Waste Stabilization (MAWS) Program. The MAWS protocol involves the blending of multiple waste streams to achieve an optimal feed composition, which eliminates the need to use large amounts of additives to produce an acceptable waste form. The GCCs have a particularly useful utility in their ability to incorporate waste streams with high metal contents, including those that contain large amounts of scrap metals, and in their potential for sequestering radionuclide and hazardous constituents in corrosion-resistant mineral phases.

This paper reports the results from tests conducted with simulated feeds representative of potential DOE and industry waste streams. Topics addressed include the partitioning of various radioactive and hazardous constituents between the glass and crystalline portions of the waste form, the development of secondary phases on the altered sample surfaces during corrosion testing, and the fate of waste components during corrosion testing, as indicated by elements released to solution and microanalysis of the reacted solid samples.

EXPERIMENTAL

Sample Casting

Eighteen samples representing a wide range of potential waste streams and contaminant compositions have been produced in crucible melts thus far. Test samples were formed by blending pre-dried (at 500C) Idaho National Engineering Laboratory (INEL) soil with various chemical additives to derive a desired sample composition (Table I). Simulated amounts of EPA listed toxic and carcinogenic metals (Pb, Cr, Ni, Cd, and Cu) and the radioactive elements U and Ce (the latter as a surrogate for Pu) were also added to the samples.

The sample mixtures were powdered in a ball mill and then added to a 99.8% Al_2O_3 crucible for melting. Samples were heated to temperatures of 1300 to 1500C for 40 min, cooled to 1000 to 1200C, held at this temperature for three hours, and then allowed to cool in the test furnace to room temperature. The 3-h temperature hold saturated the melt with respect to certain crystal phases, allowing crystal formation to proceed at an accelerated rate. Times and temperatures used for individual melts are indicated in Table I. All samples were subsequently annealed at 500C for one hour to remove thermal stresses that built up during solidification. Volatilization of materials was noted by comparing the weight differences of the starting powder and final sample weights. Results from chemical analysis of the solidified samples will be compiled in the future to determine which constituents were lost and in what proportions.

Test Procedures

The GCCs have been subjected to vapor hydration tests at 200C and Product Consistency Tests (PCT) at 90C so that the GCC durabilities could be compared against other waste forms. Toxicity Characteristic Leaching Procedure (TCLP) tests were also performed to determine the applicability of GCCs for EPA licensing as a

nonhazardous waste.
Disk-shaped monoliths (1-mm thick, 10-mm dia), polished to a 600 grit finish, were used in the vapor hydration tests. Tests were initiated by suspending two monoliths by a Teflon wire inside a 304L stainless steel Parr reaction vessel (22 mL capacity) held closed by a compression fitting. High-purity deionized water was added to each vessel in an amount (0.25 mL) necessary to achieve 100% relative humidity at test temperature, but limited enough to prevent dripping of condensed fluid from the samples. This latter aspect was deemed necessary to prevent the dripping of leachate from the samples and a resultant loss in corrosion products. After completion of the prescribed test interval, the sample vessels were removed from the oven, cooled in an ice bath, and opened. Reacted sample monoliths were examined first by optical microscopy, followed by detailed scanning electron microscopy/energy dispersive X-ray spectroscopy (SEM/EDS), X-ray diffraction (XRD), and analytical electron microscopy (AEM) analysis. Surfaces of the altered samples were characterized with respect to both the fate of the primary crystalline and glassy components of the waste form and the formation of alteration phases.
The 200C temperature and high ratio of sample geometric surface area/liquid volume (SA/V) (~106 m-1) used in these tests accelerate the reactions occurring at the sample surface and thereby induce the formation of a long-term corrosion sequence of minerals during a relatively short interval. Previous studies have shown that 50,000- to 140,000-year alteration trends produced during volcanic glass alteration in near-surface (~25C) geologic environments may be replicated in 200C vapor hydration tests in as little as 21 days (1). This similarity also suggests that the reaction mechanism controlling the corrosion of glass remains unchanged within the temperature range of 25-200C.
Static corrosion tests used in this study followed a modified format of the PCT. Sample fractions of 100 to 200 mesh size (75 to 150 mm) were reacted in deionized water, in sealed Teflon vessels, at a SA/V of 2000 m-1, a temperature of 90C, and for periods of 7, 28, and 91 days. The sample size fraction used in these tests produced particles with mean diameters that were often less than those of the primary melt crystals. Under these circumstances, the corrosion rates of the waste form, as determined during the PCTs, may be artificially accelerated by increasing the exposed SA/V ratio of the crystal. For this reason, the effect of crystal size on corrosion rates will be determined largely from the data generated in the vapor hydration tests.
After completion of the prescribed PCT intervals, sample aliquots were withdrawn, at test temperature, for various solution analyses. Carbon content was measured with a Dohrman Total Carbon Analyzer, cations by inductively coupled plasma/mass spectroscopy (ICP/MS), and pH with a combination electrode. All analytical measurements, including pH, were made at room temperature. Cation accuracy and precision are both <10%, carbon accuracy is <5%, while analytical drift over the time period required for pH analyses is typically less than 0.02 pH units. Reacted surfaces of a limited number of PCT solids were also examined by SEM/EDS and AEM analysis.
The TCLP tests were conducted at room temperature following established EPA testing procedures (2). This test measures the leachability of eight toxic metals (Ag, As, Ba, Cd, Cr, Hg, Pb, and Se) after reaction in an agitated sodium acetate buffered solution for 18 hours. The TCLP tests are used to determine whether a sample meets EPA regulatory release requirements for hazardous wastes, which are set at a solution concentration equal to 100X the established drinking water limits for each hazardous metal. The leaching characteristics of nickel were also examined, with the land-based disposal limit for this element set as the targeted leachate level.
RESULTS
Waste Form Composition and Mineralogy
The unaltered GCC samples can best be described as porphyritic, with relatively coarse-grained crystalline phenocrysts suspended in a predominantly glassy matrix. The mineral phases exhibited a variety of shapes, including blocky, trigonal, lath, and round-shaped crystals. The distribution between glassy and crystalline regions was generally nonuniform, with both glass- and crystalline-rich regions occurring in most samples. Sample compositions with very high proportions of crystalline phases sometimes resulted in samples that fractured during sample preparation.
The crystalline phases that formed were dominated by the spinel group minerals. The SEM/EDS compositional analysis suggests that magnetite (Fe3O4) was the most common

spinel formed, with this phase incorporating variable amounts of Cr, Ni, Al, Ca, Zn, Cd, and Ti in its lattice structure. Other types of spinels formed when the melt composition was varied. These additional spinel phases included hercynite ($FeAl_2O_4$) in Fe- and Al-rich melts, and franklinite ($ZnFe_2O_4$) in Fe- and Zn-rich melts. Zirconolite ($CaZrTi_2O_7$) and a Zr-U oxide phase were formed in melts to which Zr and Ti were added. Zirconolite was able to incorporate up to 11 elemental wt.% uranium and 7 wt.% cerium in its mineral structure, while the Zr-U phase contained up to 10 wt.% uranium and 3 wt.% cerium. In melts where zirconium was not added, uranium was incorporated in unidentified uranium oxide, U-Ca-oxide, or U-Na-Ca-oxide phases. In addition to zirconolite, other titaniferous phases were formed, including rutile (TiO_2), titanite ($CaTiSiO_5$), perovskite ($CaTiO_3$), an unidentified Fe-Ti-oxide [possibly pseudobrookite (Fe_2TiO_5)], and unidentified Ca-Fe-Si-Ti and Ti-Ca-Ce-Fe phases. Perovskite also contained up to 9 wt.% cerium and trace amounts of uranium. Anorthite [a calcium-rich feldspar ($CaAl_2Si_2O_8$)] was formed in melts with high Ca-Al contents. Trace amounts of lead have been detected in anorthite, while the feldspar minerals also have the potential to incorporate barium, a hazardous element; and sodium and potassium, two elements that generally reduce the durability of waste glasses through the formation of nonbridging oxygen bonds. Augite [$(Ca,Mg,Fe,Ti,Al)_2(Si,Al)_2O_6$)] was also detected in some samples, occurring as elongated crystals with a density and composition not too unlike the surrounding glass matrix material.

All glass matrix regions were dominated by the presence of Si, with lesser amounts of Al, Ca, Fe, and K. The residual glass matrix was enriched in Si and Al relative to the compositions of many high-level waste glasses, a feature that is expected to result in the formation of a relatively durable glass matrix. The following constituents were detected in trace amounts: Ti, Mg, Cr, Ni, Mn, Zr, Na, Zn, Cd, U, Pb, and rare earth elements (including Ce). It is not known for certain if all of these elements were actually dissolved in the glass network or were present as crystals that were too small to be detected during SEM/EDS analysis (<100 nm). The AEM examinations of glass-rich regions from some samples revealed the presence of tiny crystals from regions of samples that appeared to be completely vitreous during the preceding SEM examinations. These crystals probably formed as the melt was allowed to cool to room temperature.

Bulk distribution ratios were calculated for Cr, Ni, Fe, U, and Ce, as determined by using SEM/EDS analysis of the crystalline and glassy regions of the samples. These results indicate that Fe, Ni, and especially Cr are strongly partitioned into the spinel crystals (Table I). Uranium and cerium were also strongly partitioned into crystals, especially when zirconolite formation was induced by the addition of Zr and Ti to the melt.

Durability Testing

The release of selected components from the GCC samples during PCT runs are presented as normalized elemental release values (NLi) in Fig. 1. These values reflect the release of a specific element i normalized to both the SA/V ratio of the test and the proportion of element i contained in the solid material. Results can thus be used to directly compare release rates from waste materials with different compositions and tests conducted at different SA/V ratios. Note that the NLi values can be influenced by several parameters including the release rate of an element from the waste form, precipitation of alteration minerals containing element i, and/or differential release rates from primary phases contained in the GCC wastes. The PCT results indicate that the GCC waste has favorable NLi rates when compared to simulated nuclear waste materials, such as the SRL 202U high-level nuclear waste (HLW) borosilicate glass. The NLi rates for most alkali and alkaline earth elements overlap with those of the SRL 202U glass. For example, NLCa data from the GCCs are nearly identical to those of the SRL 202U glass (Fig. 1a), while NLK values also overlap with the SRL 202U glass but display more variability among the different samples (Fig. 1b). Release rates for both Si and U are significantly lower than those of the SRL 202U glass (Figs. 1c and 1d). Silicon is primarily contained within the glass matrix, so its reduced NLi reflects a more durable glassy material in the GCCs relative to the SRL 202U glass. The low NLSi exhibited by the GCC tests can be attributed to the solution pH values that result from the leachate being in contact with the samples. The leachate attains pH values of 8.9 to 9.9 for most GCC samples during test periods of 7 to 182 days. A pH of ~9 represents a minimum for the solubility of silicic acid in these systems. Thus, the lower pH values in these

tests is reflected by lower silicon release rate. By contrast, the pH for the SRL 202U glasses averages between 10.0 and 10.4 for test periods of 7 to 540 days. These slightly higher pH values have resulted in a greater degree of glass dissolution relative to the tests with the GCCs.

The NLU values from the GCC samples are also significantly reduced relative to the rate of the SRL 202U glasses (Fig. 1d). This trend may reflect the reduced solubility of the silicon-rich glass matrix as discussed above or, alternatively, the incorporation of uranium in corrosion-resistant mineral phases. This latter hypothesis was further examined by comparing NLU values from two samples with a similar overall composition, except that one sample (G2) had Zr and Ti added to induce the formation of zirconolite, while the other (G4) did not. A comparison of the NLU patterns for the two samples indicates that uranium release was reduced by a factor of 5 to 15 when the uranium was preferentially partitioned into the zirconolite phase (Fig. 2). The release of Ca and Ti was also significantly reduced, while cerium was only slightly reduced as a consequence of its incorporation in zirconolite. Both the G2 and G4 samples also have NLU values that are still lower than those of the SRL 202U glass doped with a similar concentration of uranium.

Results from vapor hydration tests also suggest that the GCCs are a durable waste form. Overall appearances of the reacted sample surfaces after testing and the compositions of the alteration phases suggest that the iron-rich crystalline phases are relatively inert to chemical attack when compared to the glassy matrix. On most samples, the slightly corroded crystals are surrounded by depressed surface regions that mark the locations of more rapidly corroded glass matrix regions. The minor alteration of the spinel phases that does occur probably results in the formation of iron-oxide precipitates on the sample surface.

The SEM/EDS investigations revealed only minimum development of secondary reaction products on the G2 samples at 7, 28, and 91 days, with most of the phases being composed of clays, iron oxides, and calcite ($CaCO_3$). Radionuclide and hazardous element contents in these alteration phases are also minimal, indicating that these components are being retained in the GCC waste form. Lead was present in trace quantities in the alteration phases deposited on samples in which lead was added, a feature that probably reflects its release from the glass regions during corrosion. Cerium was also present in alteration phases on samples where the glass matrix was the primary host for this element. When crystalline phases such as zirconolite and perovskite hosted cerium, its release was diminished to the point where it was no longer detected on the sample surfaces. Similarly, uranium was not detected in any of the alteration phases observed. This suggests that uranium was retained in the primary waste form, despite the very corrosive conditions of the tests.

The TCLP results indicate that release rates for hazardous elements are generally below EPA established limits for hazardous waste declassification (see Fig. 3 for Ni, Pb, Cd, and Cr results). Exceptions occur when waste loadings of specific elements are very high, for example, 16% NiO (sample G19) and 20 to 25% CdO (samples G14 and G15). These high metal concentrations were included in the study to aid in identifying host crystals and to bracket the maximum waste loadings allowed for the respective hazardous elements. These concentrations are not necessarily representative of the compositions expected for DOE waste streams.

DISCUSSION

Experiments to date have shown that simulated waste streams containing Cr, Ni, Pb, Cd, U, and Ce (as a surrogate for Pu) can be incorporated into GCCs, with all elements except lead being strongly partitioned into the crystalline phases. The results from this study indicate that Cr and Ni are partitioned almost exclusively into spinel crystals that are resistant to chemical attack, resulting in these elements being immobilized in the waste form. Similarly, U and Ce can be isolated in crystalline phases such as zirconolite and perovskite. Crystal phases also have the potential to incorporate a wide array of other hazardous elements into their structures, including Ag, As, Ba, and Th, and glass network-breaking elements, such as Na, F, S, and P. With homogeneous glass waste forms, the addition of significant quantities of metal and network-breaking elements may significantly diminish the waste form durability. With the GCCs, however, these elements are incorporated into a variety of crystalline materials, thereby leaving the residual glass material enriched in silicon. High-silicon glasses have been shown to be resistant to corrosive processes.

Of key interest to this study is the fate of the various hazardous and radioactive

components during exposure of the GCC waste form to corrosive fluids. Results from durability tests indicate that the GCC has corrosion- resistant properties that are similar to, or better than, those of simulated HLW glasses. Solution results from static PCTs show NLi rates for Al, Na, K, and Ca that are comparable with those of SRL 202U glass reacted under similar test conditions, while NLi values for Si and U are significantly lower than those of the SRL glass. Samples that have incorporated uranium into crystalline phases, such as zirconolite, have the lowest overall NLU levels. This reflects the high durability of the crystalline phases relative to predominantly glassy materials (Fig. 2).

Sample reactions on the vapor hydration tests offer an interesting insight into the mobility of the GCC components during corrosion testing. Elements that are released from the altered samples in these tests rapidly accumulate in the thin film of water that is present on the sample surface. Various minerals will precipitate in this film of water, as their respective solution concentrations increase, thus recording the preceding release of elements from the solid samples. Hazardous and radioactive elements that are partitioned into the crystalline phases during melt solidification are nearly absent from the altered sample surfaces, and this indicates a strong retention by the primary crystal hosts. Elements that are present largely in the glass matrix occur in greater abundance in secondary alteration products on the sample surface but still are relatively uncommon, and this indicates a high degree of chemical durability for the glass matrix as well.

CONCLUSIONS

The test results indicate an exceptional potential for the development of a low-cost MAWS process for disposing of mixed waste streams with high metal content. This process would form a highly durable waste product that should easily exceed the minimum disposal performance criteria. The production of a durable tailored GCC waste form depends on proper processing parameters, such as waste stream composition, melt temperature, and cooling rate. However, tests conducted thus far suggest that these processing parameters are relatively flexible and allow GCCs with acceptable durabilities to be produced with a minimum of processing requirements and additives.

Crystalline phases formed in GCC wastes include spinels, zirconolite, perovskite feldspars, and Fe-Ti oxides. Most of these phases are also resistant to aqueous corrosion; thus, their formation is responsible for the low amounts of hazardous and radioactive elements released from these waste forms during sample alteration. Results from durability tests indicate that the GCC waste form has corrosion-resistant properties that are similar to, or better than, those of simulated HLW glasses such as SRL 202U.

The glass fraction of the GCC serves as a binder for the crystalline phases and also incorporates elements excluded from the crystal structures. Minor amounts of U and Ce, as well as the majority of the lead present, have been detected in the glass matrix. The release rates of glass matrix components to solution also appear to have been lowered relative to tests with high-level waste glass, presumably because of fractional crystallization processes that increased the Al and Si concentrations of the residual glass matrix material.

The reduction in waste volumes associated with the elimination of the need for glass additives during GCC formation will be a great economic benefit during the disposal of large volumes of low-level radioactive waste materials. The high durability of the GCC waste form also indicates excellent potential for its application in the processing to other DOE waste streams, including intermediate- and high-level nuclear waste materials; DOE wastes with high levels of Na, P, S, Ba, and As; and plutonium-laden defense wastes associated with weapons production and dismantling activities. The use of a GCC waste form offers an attractive and economical alternative to waste disposal by using homogeneous vitreous materials.

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FIGURE CAPTIONS

Fig. 1. Normalized elemental release (in g/m2) for glass-crystal composites and high-level waste (HLW) borosilicate glass samples from Product Consistency Tests. All tests were conducted in deionized water, at 90C, at a glass surface area/liquid volume ratio of 2000 m-1. The sample compositions are given in Table I, while the composition and solution release data for the SRL 202U HLW glass are from Ebert (3): (a) Ca, (b) K, (c) Si, and (d) U.

Fig. 2. Log normalized elemental release of two samples with nearly identical compositions. For the G2 samples, Zr and Ti have been added to induce the precipitation of zirconolite, while zirconolite is absent in the G4 sample. Test conditions, sample key, and source of SRL 202U data are the same as in Fig. 1.

Fig. 3. Solution concentrations from Toxicity Characteristic Leaching Procedure tests compared to established EPA release limits for the delisting of a hazardous waste.

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VENDOR VITRIFICATION PROGRAMS AT THE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT

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ABSTRACT

During the past three years, two separate and distinct low-level waste vitrification projects have been initiated at the Fernald Environmental Management Project: The Minimum Additive Waste Stabilization (MAWS) demonstration project and the CRU 4 Pilot Plant program. The projects are technically and contractually distinguishable and serve two very different needs in the DOE complex. This paper contrasts the technical, operational, and contractual differences associated with each project and highlights some operational and managerial lessons learned during the MAWS project which may be applied to other DOE vitrification projects.

INTRODUCTION

Historically, vitrification's acceptance as a solution to stabilizing nuclear waste has been limited to high level programs because of the costs associated with fully remote operations and chemical additives required to produce a stable waste form. These high level waste glasses characteristically contain small fractions of waste and large fractions of additives. Where the overriding concern is stabilization, as it is with high level nuclear waste, the costs associated with this relatively inefficient process are acceptable. In comparison, the sheer volume of the low level nuclear and hazardous waste in the DOE complex makes the costs associated with the high level approach of "additive" vitrification impractical, especially in times of decreasing federal agency budgets.

The MAWS project at Fernald was initiated to demonstrate that one of the limiting aspects of vitrification for low level wastes, additive costs, could be overcome through an innovative blending of existing site waste streams where the glass forming potential of various wastes is exploited. For GTS Duratek, this project included the installation, startup, and operation of the DuraMelterTM 300 system which is comprised of feed, melter (operating nominally at 1100oC), and off-gas sub-systems. The blending theory increased the waste fraction of the final glass product to an excess of eighty percent, thus dramatically reducing the costs not only for purchased additives, but also the variable costs for operations and disposal. In addition to the minimum additive savings, further economies were realized through the large volume reduction typical of vitrification. The other element of the program which increased efficiency was the integration of an aggressive pre-treatment of the soil waste stream provided by Lockheed Environmental Services that concentrated the contaminants in a vitrifiable soil fraction and returned a large percentage of the initial waste stream back to the site as clean soil.

MAWS is the synergistic use of multiple remediation technologies and waste stream combinations to produce the smallest amount of final vitrified waste form practical thereby significantly reducing the life-cycle cost of the process. GTS Duratek believes it is a concept that has significant potential for other sites within the DOE. That complex-wide potential is currently being quantified at the Catholic University of America's Vitreous State Laboratory through a separate contract from Argonne National Laboratory.

In contrast to MAWS, the CRU 4 Pilot Plant program was initiated to remediate a single identified waste stream of concern to the stakeholders and regulators. The silo wastes at Fernald contain large amounts of sulfates, heavy metals, and radioactive isotopes emanating Radon which make higher temperature melting desirable. Not only are the melter design assumptions different for the silo waste, GTS Duratek's scope is also different for this project with the company currently contracted to provide only the melter and glass handling system with a limited startup period following construction. Operation of the Pilot Plant will be the responsibility of the site management contractor, the Fernald Environmental Restoration Management Corporation (FERMCO). The higher design temperatures specified for this melter introduced many engineering and materials challenges that had to be solved. The differences between the two melter systems are highlighted in the following paragraphs.

TECHNICAL DIFFERENCES

The MAWS DuraMelterTM 300 System

As mentioned above, the DuraMelterTM 300 system at Fernald consists of a feed system, melter, and off-gas sub-systems. The feed system for the melter includes two 4,000 gallon feed tanks that are independently mixed by both propeller mixing (located at the top of the feed tank) and recirculation pumping at a rate of about 85 gallons per minute. For the recirculation train, feed is removed from a port at the bottom of the feed tank and returned to the top of the feed tank. These two sources of mixing action decrease settling in the tank and allows one of the mixing sources to be removed from the system for maintenance, if necessary. A small fraction of the feed is drawn off the recirculation line by the main feed pump which pumps the feed slurry into the melter. The slurry is fed into the melter through a water cooled feed tube at a rate of about 600-800 ml/minute.

The feed tanks have been scaled up from a 55-gallon drum for the laboratory version, the DuraMelterTM 100 to two 4,000 gallon tanks for the DuraMelterTM 300. Each tank provides sufficient feed for 8 to 10 days of operation enabling one tank to be prepared while the second supports processing. The challenges encountered include recirculating an abrasive slurry, settling in the feed lines, and batching-up such large volumes of slurry.(1)

The DuraMelterTM 300 is sealed, joule-heated melter operating at a nominal temperature of 1100oC and is equipped with a glass forming gem machine installed under the discharge chamber. Gems resemble flattened marbles and are about 0.5" in diameter. The gem machine cuts the discharged molten glass stream with a gear-like cam and deposits the glass on a cooling disc. The DuraGemsTM rotate about 270 degrees on the cooling disc where they are then scrapped off the disc and fall into a 35-gallon sealed drum. The purpose for the gem machine is to provide for rapid cooling of the discharged glass. This is essential for the glass to cool quickly due to the high concentrations of fluoride, found in the pit wastes, which crystallize when allowed to cool slowly (bulk discharged). This was demonstrated in a fluoride run that was performed in early January. The glass that was discharged directly into the 35-gallon drum crystallized and the glass that was discharged into the gem machine produced glass DuraGemsTM which showed no evidence of crystallization.

Several campaigns were performed on the DuraMelterTM 300. The first slurry feeding of the melter began in September 1993. The feed composition was a borosilicate recipe that did not require the full use of the off-gas treatment system. This campaign was the first shakedown run. During the campaign, the first production DuraGemsTM were made. After the run, modifications to the system were made, including modifying the feed system, gem machine, and off-gas system. The first campaign using the off-gas treatment system occurred in January 1994. During that run, a fluoride containing feed, similar to the radioactive composition without the uranium or thorium, was fed into the melter. This glass was bulk discharged during compositional turnover; fluoride gems were produced after 72 hours of continuous feeding.

The results of the first fluoride run were informative and impressive. Two objectives of the run were to 1) provide FERMCO with information to perform a safety assessment on the melter system and 2) to perform a shakedown with the operation of the full off-gas system. The run was 75 hours long with continuous feeding for the entire time. There were no major complications during the run. The off-gas system operated well within the design specifications. Feeding was maintained at a rate between 600-800 ml/min (0.16-0.21 gal/min). The cold cap coverage was about 85-95%

throughout the run and was maintained by controlling the feed rate, bubbling rate, and glass pool temperature. The quantity of fluoride retained in the glass was the highest ever for any DuraMelterTM run: 76.6% of the fluoride fed into the melter was retained in the glass. At steady state conditions, this number is expected to be about 74%.

A significant factor contributing to the high fluoride retention was the constant cold cap of 85-95%. Data gathered during an earlier DuraMelterTM 10 run showed that the larger cold cap helped to maintain the fluoride in the glass. The fluoride composition was about 97% turned over from the initial borosilicate composition after 57 hours of continuous operation.

The glass pool temperature was maintained at about 1090C with the plenum temperature at about 630C. The latter temperature is controlled by the lid heaters but is also affected by the size of the cold cap. The off-gas system which scrubs the acid gases and traps particulates also reduces the temperature of the gas stream from the melter. For example, the off-gas entering the quencher was approximately 240oC while the air entering the scrubber was reduced in temperature to about 185oC. The moist air exiting the scrubber was about 38C.

Analysis of the composition of the quencher and scrubber solutions was performed on samples taken throughout the run. Once the quencher solution reaches saturation with respect to fluoride, the remaining 26% fluoride can be recovered as sodium fluoride and recycled into the subsequent feed batch. After 20 hours into the run, the concentration of fluoride averaged 0.7 molar in the quencher and 0.12 molar in the scrubber.

During the period of August through November 1994 the melter successfully processed over 14,000 liters of actual high fluoride, low level radioactive and hazardous mixed waste from Pit 5 and 3,300 kilograms of uranium-contaminated soil from the site in four radioactive campaigns. The analytical results from the radioactive campaigns are not yet available.

Part of the proof of the MAWS concept is in the results of the leach testing of the glass and of the emissions from the melter during processing of radioactive wastes. The Toxicity Characteristic Leaching Procedure (TCLP) results from processing actual FEMP wastes on the laboratory located DuraMelterTM 100 show that the eight (8) RCRA metals were well below the EPA Limit (Table I).

Retained in the glass are 99.8% of the lead, 99.1% of the barium, and 99.8% of the uranium and thorium. Chromium detected in the off-gas system is largely due to the chromium in the refractory bricks and Inconel used in the construction of the melter. Over 70% of the chromium is retained in the glass and all but 0.06% is captured for recycling into the feed. Emissions measured during the radioactive campaign on the DuraMelterTM 100 show that the concentration of the RCRA metals, fluoride, uranium and thorium are below 25 ppb and most are less than 1 ppb (Table II).

The fluorine was measured to be less than 25 ppb; for comparison, the OSHA limit is 3 ppm. The analysis of samples from the DuraMelterTM 300 radioactive campaigns is expected to yield similar results.

The CRU-4 Pilot Plant Melter and Gem Machine

The scope for the CRU 4 Pilot Plant program is limited for GTS Duratek to designing and delivering a melter and gem machine, supervising the construction, and providing startup testing oversight. This melter will be a compliment to the balance of the Pilot Plant design which is currently under construction at the Fernald site and will provide the operating contractor with relevant data requisite for full production scale facility design.

Nominally, the DuraMelterTM 1000-HT is capable of producing one ton of glass per day, using a similar patented bubbling/agitation device as the MAWS DuraMelterTM 300. The primary difference between the design of these two melters is temperature of operation. The smaller unit processes glass at a nominal temperature of 1100oC, while the larger unit is rated at 1350oC.

The DuraMelterTM 1000-HT has been specifically developed to meet the design requirements for the K-65 silo wastes. Glass formulation tests (2) have recommended a design temperature for processing of 1350oC. This requirement coupled with the complex chemistry of the silo wastes has necessitated a novel design approach to joule heating the waste glass bath. GTS Duratek has developed a composite electrode consisting of three materials: a high temperature liquid, a refractory metal, and a refractory ceramic. Combined, these materials permit a conventional joule heated

melter electrode arrangement, which allows the aggressive glass chemistry to contact only suitable materials.
The glass discharged from this melter will be channeled to a DuraGemTM machine, which will convert the glass stream into gems similar to those produced in the MAWS system.

CONTRACTUAL DIFFERENCES

The distinctions between MAWS and the CRU 4 Pilot Plant program extend to the contractual mechanisms employed as well. The MAWS program evolved out of a Program Research and Development Activity (PRDA) sponsored by DOE's Office of Technology Development (OTD) (EM-50) and administered by the Argonne National Laboratory. When the project moved from the PRDA stage to the demonstration stage, it was natural and logical for ANL to continue the overall program management. This was accomplished through a series of three contracts with GTS Duratek for Phases I and II of the project with Fernald designated as the host site.
Phase I covered the laboratory work at the VSL and the construction and initial startup of the site located DuraMelterTM 300 system. Phase II covered the remaining pre-radioactive testing and the processing of actual site wastes from Pit 5 and the soil concentrates provided by the Lockheed soil washing system. The Phase I contract was fixed price and the two contracts that comprised Phase II were Cost Plus Fixed Fee. Both Phase I and Phase II were jointly funded by OTD and Environmental Restoration (EM-40). The CRU-4 Pilot Plant program contract, on the other hand, is strictly a fixed price contract between GTS Duratek and FERMCO for design and delivery of a melter and gem machine with ancillary requirements for construction and startup oversight.
The obvious complexity of the MAWS contractual framework, which involved project management from EM-50, EM-40, DOE Fernald, Argonne, FERMCO, Lockheed, The Catholic University of America, and GTS Duratek, developed as the project developed and represented the real needs of the project at the time. However, as the technology integrator, GTS Duratek found this framework to be very complex and somewhat inefficient simply because of the number of different decision makers involved. Project management of the CRU 4 Pilot Plant program in contrast, is completely contained within two organizations, GTS Duratek and FERMCO. Technological issues aside, the managerial logistics for this project are much simpler.

LESSONS LEARNED FROM THE MAWS PROJECT

One of the objectives of a demonstration project is to identify the strengths and weaknesses of the process. As would be expected in a full-scope demonstration project such as MAWS, many lessons were learned. With several months elapsed since the radioactive campaigns, two items stand out, one operational and one contractual or managerial.

Operations

One of the objectives of the MAWS demonstration was to achieve the maximum waste loading (waste as a percentage of the total volume) in the final glass product because every kilogram of additives not used reduces the cost for chemicals, processing, and disposal. Laboratory results using actual waste showed that waste loadings of greater than 95% could be achieved. These tests were conducted for short periods (1-3 days in duration) on a system that while functionally identical was simpler in mechanical configuration. One of the unique characteristics of the Fernald pit waste is a fairly high concentration of magnesium fluoride. While this constituent presents significant challenges in the off-gas system, it also is attractive because the fluoride acts as a glass fluxing agent thereby permitting processing at reduced temperatures. The Fernald MAWS recipe counts on the fluoride and its loss through volatilization contributes to crystalline formations in the glass. This migration of the fluoride into the off-gas occurs almost exclusively during idling periods when feed to the melter is stopped and the cold cap dissolves. During the first radioactive campaign, for which the feed stock was actually 97 wt% waste, equipment maintenance requirements necessitated prolonged idling periods resulting in some crystal formations in the discharge chamber. The "down time" required to clean these deposits added significantly to the cost of operations. The lesson to be learned is that there is a clear trade-off between waste loading and operational flexibility where a volatile waste constituent is relied upon for acceptable glass formation. In hindsight, a somewhat reduced waste loading could have provided the needed operational margin at slightly increased cost for chemicals and prevented the need to shut down the system to remove the crystalline deposits

and thereby expend significant funds for idle operator labor.
Contractual/Managerial
As discussed above, the contractual and project management mechanisms which developed for the MAWS program were relatively complex whereas those same mechanisms for the CRU-4 Pilot Plant program are relatively simple. While there is clearly a difference is scope for the two projects which, in part, explains the difference in management approach, to the greatest extent possible "simple is better." Other technology projects in their formative stages would be well served to take the time necessary to simplify the management structure of those projects as much as is reasonably practical. In the long run, the time spent will be well worth the cost.
CONCLUSION
Two vendor programs for the vitrification of Fernald wastes have been reviewed. Each of these programs has been contrasted from a technical and contractual perspective. One of these focused on the MAWS process aimed at minimizing the costs of vitrification. The second centers on the process equipment which completes the pilot process plant for the remediation of K-65 silo (CRU-4) wastes. Through vitrification, GTS Duratek is able to take Fernald wastes, in a variety of forms, and convert it into a stable product, safe for long term disposal.
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Session 12 -- Robotics and Remote Technologies
Co-chairs: Joe Angelo, Consultant;
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AN INTEGRATED SYSTEMS APPROACH TO REMOTE RETRIEVAL OF BURIED TRANSURANIC WASTE USING A TELEROBOTIC TRANSPORT VEHICLE, INNOVATIVE END EFFECTOR, AND REMOTE EXCAVATOR*
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ABSTRACT
Between 1950 and 1972, over 65,000 m3 of transuranic mixed waste was buried in shallow pits and trenches in the Subsurface Disposal Area at the Idaho National Engineering Laboratory,s (INEL's) Radioactive Waste Management Complex. Commingled with this waste is up to 283,000 m3 of fill soil. The pits and trenches were constructed similarly to municipal landfills with both stacked and randomly dumped waste forms such as barrels and boxes. The main contaminants are micron-sized particles of plutonium and americium oxides, chlorides, and hydroxides. Retrieval, treatment, and disposal is one of the options being considered for the waste. This report describes the results of a field demonstration conducted to evaluate technologies for excavating and transporting buried transuranic wastes at the INEL and other hazardous or radioactive waste sites throughout the Department of Energy complex.
The full-scale demonstration, conducted at RAHCO International,s facilities in Spokane, Washington, in the summer of 1994 evaluated equipment performance and techniques for digging, dumping, and transporting buried waste. Three technologies were evaluated in the demonstration: an innovative end effector for dust-free

dumping, a telerobotic transport vehicle to convey retrieved waste from the digface, and a remotely operated excavator to deploy the innovative end effector and perform waste retrieval operations. Data were gathered and analyzed to evaluate retrieval performance parameters such as retrieval rates, transportation rates, human factors design, and the equipment,s capability to control airborne contamination spread.

INTRODUCTION

From 1950 to 1972 over 65,000 m3 of transuranic (TRU) waste was disposed at the Subsurface Disposal Area (SDA) in shallow land-filled pits and trenches at the Idaho National Engineering Laboratory,s (INEL,s) Radioactive Waste Management Complex (RWMC) (1). These wastes consist of barrels, boxes, and large objects contaminated with TRU and organic materials. The primary hazards associated with these wastes are plutonium and americium compounds and organic solvents from weapons production activities at the Rocky Flats Plant. These wastes were placed in pits and trenches and covered with an average of 1 to 2 m of soil as a cap to seal the pit or trench. It is probable that during storage many of the waste containers may have lost their integrity, and the TRU materials may have leached into the surrounding soil.

In 1989, the RWMC became an Environmental Protection Agency (EPA) Superfund site. Current U.S. Department of Energy (DOE) plans to remediate the site include removing the wastes from these pits and trenches for further processing. Because of the possibility of contamination spread, retrieval of buried waste must be accomplished in a manner that will minimize the generation and spread of contaminated soils and significantly reduce the risk of accident or injury to workers. The dumping of dry soil into a funnel/dumpster arrangement has been found to be the primary mechanism for dust generation during the retrieval of buried TRU waste. The primary goal of this technology development is to reduce dust generation and the potential spread of airborne contaminants during retrieval and dumping operations and to remove the workers from the hazardous area by performing operations remotely.

In addition to excavation and dumping, exhumed waste must be conveyed away from the retrieval area to a packaging area or directly to a treatment facility. A robotic conveyance system was designed to remotely convey waste from the digging area, thus removing humans from the hazardous zone.

PROJECT DESCRIPTION AND OBJECTIVE

The development, demonstration, testing, and evaluation of a system for retrieval, conveyance, and dust-free dumping of buried waste was funded by the DOE Office of Technology Development through the Buried Waste Integrated Demonstration Program. The concept and design of the system was developed for the Buried Waste Integrated Demonstration program by RAHCO International in conjunction with Spar Aerospace Ltd. and RSI Research Ltd. A full-scale, non-radioactive demonstration was conducted at RAHCO International,s facilities in Spokane, Washington, in the summer of 1994 to evaluate equipment performance and techniques for digging, dumping, and transporting buried waste.

The objective of the demonstration was to explore the effectiveness of methods and equipment to remotely retrieve and transport buried waste away from the digface and control dust generation and contamination spread while dumping retrieved waste (2). Three technologies were evaluated in the demonstration: an innovative end effector (IEE) for dust-free dumping, a telerobotic transport vehicle (TTV) to convey retrieved waste from the digface, and a remotely operated excavator (REMEX) to deploy the IEE and perform waste retrieval operations. The TTV and IEE were developed and demonstrated by RAHCO International with Spar Aerospace and RSI Research. The REMEX is an existing remotely operated robotic excavator provided by Spar and modified to accept the IEE. These systems were developed or modified in close coordination with one another to ensure a systems design approach that maximized the effectiveness of the combined technologies.

EQUIPMENT

Remotely Operated Excavator (REMEX)

Spar and RSI have modified a conventional Hitachi EX200LC excavator (see Fig. 1) for remote operation. The modifications include the use of coordinated control technology to control the remote operation of the excavator (3). The implementation of coordinated control technology has been supported by the University of British Columbia, which has developed this technology for use with heavy equipment operation. A four degree of freedom joystick that replaces the arm of the operator,s chair controls the excavator digging functions (see Fig. 1). The conventional track control foot pedals are

Figure 1. The REMEX is a conventional excavator that has been modified with a sophisticated coordinated control system. It is operated telerobotically via a four degree of freedom joystick and operator,s control station.
maintained. The primary mode of operation is from a remote control operating station that can be located up to 2,000 ft from the REMEX and is linked to the REMEX by a telemetry cable. At the remote control station, the operator controls engine shutoff, travel, and excavator digging functions. All REMEX functions that could be performed from the cab can be performed from the remote control station.
The use of coordinated control technology allows the REMEX to be operated in a dexterous and precise manner. Conventional, manually-operated excavator systems are typically operated by two joysticks, each with two degrees of freedom. Deflecting the joysticks corresponds directly to opening and closing the valves that move the excavator arm. With the new technology of coordinated motion control, the operator uses a single joystick to control the excavator arm. There is one-to-one correspondence between the motion of the master controller and motion of the bucket. For example, a deflection of the master controller in the x-direction will produce a motion of the bucket in the x-direction with respect to the operator's frame of reference, regardless of the current configuration of the excavator arm. The speed of motion of the bucket over the ground will be proportional to the deflection of the master controller in that direction.
Innovative End Effector (IEE)
The REMEX has been fitted with an IEE, as shown in Fig. 2, to reduce dust generation during digging and dumping operations. The IEE consists of three structural assemblies - an inverted thumb, an attachable/detachable integrated transfer module (ITM), and a shovel assembly - as well as the necessary actuating linkage and hydraulic cylinders. The thumb has proven to be extremely versatile, dexterous, and productive in handling the expected waste forms. The front shovel configuration with the thumb on top was chosen to increase digging efficiency and reduce dust generation when the shovel digs into the waste. The ITM acts as a detachable loading, handling, and conveyance bucket when digging, handling, and conveying 55-gal drums, dirt, and small debris. It is a rectangular, welded steel structure and holds approximately 1 m3 of soil and waste, or two 55-gal drums. The lower shovel assembly, consisting of short digging teeth, a guide skirt, an overflow opening, and a latching mechanism, has multiple functions. When used with the thumb, it can retrieve large items and guide 55-gal drums into the ITM. It also acts as a shovel to scoop dirt and debris into the ITM. To prevent the ITM from overfilling, the shovel assembly has a small overflow opening between the teeth and top of the ITM. The shovel assembly also has a flared skirt to accommodate easy attachment of an empty ITM to the shovel assembly. A latch mechanism on each side of the shovel assembly secures the ITM to the assembly.
Figure 2. The multipurpose end effector integrates innovative features with proven excavation and retrieval technology.
Telerobotic Transport Vehicle (TTV)
The remote conveyance system (Fig. 3) was designed and provided by RAHCO International and Spar Aerospace. It consists of a track-mounted TTV, remote operator control station, and waste transport container (WTC). The TTV transports the loaded/unloaded transport containers and ITMs. The TTV is fully integrated with the end effector/waste-handling system to provide a waste conveyance solution that eliminates dust generation during conveyance operations. The TTV is designed as a multiple function vehicle. First, it conveys the empty WTCs and ITMs from the storage area to the excavation site. Second, it serves as a transfer station at the excavation site by a) supporting the transport container, b) opening, closing, and locking the container lid, and c) providing a staging platform for the empty ITM. Third, it conveys full containers and ITMs from the excavation location to the dumping/unloading location.
Fig. 3. The TTV mechanical subsystems consist of a track assembly, main frame, diesel power plant, hydraulics system, and ITM cradle. The WTC carries the ITMs and can be used to transport large objects. All of the controls necessary to operate the TTV in any operating mode are provided by the TTV operator,s control station. Co-locating the TTV control station and the REMEX control station allows for close coordination between the operators and greatly facilitates communications.
A main frame connects the tracks and supports the loads. One track is mounted rigidly to the frame while the second track is connected by horizontal pivot pin.

This minimizes racking of the frame as the TTV travels over irregular terrain. The TTV is propelled by a set of hydraulic motor-powered crawler tracks. The tracks provide stability on slopes and excellent maneuverability. Employing skid steering, the TTV can turn 360 degrees on its own center on a 4.5-m-wide path.
The 4,100-kg capacity WTC receives retrieved waste in the form of filled ITMs, 1.25 x 1.25 x 2.4-m boxes and many other expected, irregularly shaped, large waste items. After receiving the waste items directly or in an ITM, the container is closed and automatically sealed and locked to contain any dust generated during conveyance. The WTC is a steel structure approximately 1.5-m wide, 1.5-m high, and 2.8-m long, weighing approximately 1,270 kg empty. The lid structure is opened by a lift mechanism to provide frontal access, allowing easy placement of the ITM or other waste items by the operator. A pocket, located in the bottom of the container, is configured and sized to provide an easy target for the operator and self-aligns the pocket and the ITM during loading. Dust seals, located on all sealing surfaces, are provided to contain the dust generated during conveyance. A spring-loaded, latching mechanism engages the lift bar on the lid to lock and secure the lid during handling, preventing inadvertent opening of the container.
In operation, the IEE selectively retrieves the expected waste forms and places the collected waste in the WTC. For large waste items, the IEE places the item directly onto the container floor. For dirt, debris, and 55-gal drums, the excavator scoops the materials into the ITM, swings the filled ITM over the transport container, and lowers the ITM into the container pocket where it is detached. An alternate retrieval method is to detach the ITM, use the IEE to place objects into the ITM, and reattach the ITM to the IEE for placement in the WTC. An empty ITM, staged on the TTV, is then attached to the IEE and swung back to the digface (see Fig. 4). Simultaneously, the container lid is closed, securing the filled ITM in the container. The ITM is then carried by the TTV to a storage area receiving station where the filled container and enclosed ITM are off-loaded. An empty ITM and an empty container are loaded onto the TTV. The TTV then travels back to the digface.
Figure 4. A filled ITM is shown being placed inside the WTC on the TTV. After releasing the ITM, the lid on the TTV is closed and the TTV is repositioned to allow the REMEX to pick up the empty ITM staged on the back of the TTV. The REMEX continues retrieval operations while the TTV transports the waste to a processing or storage facility.
Spar Aerospace provided guidance and control systems for the TTV, consisting of the TTV control system, operator control station, and three radio frequency (RF) links between the TTV control system. The TTV control system has three separate microprocessor controllers to provide dedicated track control, waste container operation/control, and selected control of the forward and rear-mounted vehicle camera/pan and tilt units. Each microcontroller has a resident INTEL 80C196K microprocessor running proprietary software and shares a common RF link to the operator control station. The TTV also has onboard sensing systems to provide track speed transduction, a rate gyro for heading and noise filtering around the speed transducers, waste-handling logic function switches, and a vehicle acoustic ranging system array (VARSA).
The video system, comprised of forward and rear-mounted camera/pan and tilt units, gives the remote operator visibility in forward and reverse directions. The video system is used for steering control and tracking of specially designed acousto-video targets. The targets provide the TTV operator vision tracking to the REMEX to allow execution of the 90-degree turn required to align the transport container with the REMEX and receiving station.
Three vehicle control modes are provided: 1) open loop track control, 2) telerobotic control, and 3) waste container operation. In open loop track control (Mode 1), the tracks are operated from the vehicle-mounted manual control (pendant) or from the remote operator's joystick if the normal telerobotic control function either fails or is not required. Telerobotic control (Mode 2) provides resolved control of tracks through a single joystick at the remote operator's station. The joystick commands provide proportional forward/reverse and left/right directional control. The microprocessor at the operator's station resolves joystick commands into track speed commands that are sent by the RF link to the vehicle track microcontroller. The track microcontroller integrates the track speed commands with speed transducer and rate gyro feedback to control the tracks. Waste container operation (Mode 3) provides for loading and unloading while the TTV is stationary. Operator commands

from the remote control panel initiate individual control functions such as lid unlatch, open/close, and latch. The onboard control system also monitors limit switches to ensure each function has also been completed.
For teleoperated control, the TTV is operated from the remote operator control station. The control station consists of a microprocessor, control panel, RF modem, video monitor/controller, and graphical overlay controller (see Fig. 3). Operation of the TTV can be carried out using Modes 1 and 2 at the remote operator's station or independently at the TTV itself using a pendant mounted controller. An emergency stop button is also provided on the TTV and on the remote control station.
For this demonstration, the operator controlled the vehicle in the telerobotic Mode 2 using the single joystick for travel control and the forward or rearward cameras for visibility. As the TTV approached the REMEX, the operator tracked to an acousto-video target set on free-standing targets. Within 9 m of the REMEX, the VARSA was able to transduce range to the target and attitude. A graphical overlay was displayed on the operator's control station monitor. The overlay provided the operator with an image to align with the target. Boundary conditions on the overlay provide the operator with alignment and approximate ranging data, confirming the data provided by VARSA that was displayed on the panel. From this, the operator determined the distance and attitude of the vehicle with respect to the targets. Once the TTV reached a designated point near the REMEX, the operator commanded the TTV to pivot 90 degrees and align to a second free-standing target. This positioned the TTV perpendicular to the REMEX so the ITM could be placed in the transport container. The TTV was then moved to place the empty ITM at the correct location for attachment to the REMEX using the IEE. Then the TTV returned to the receiving station where the vehicle was docked to a single, free-standing, acousto-video target to unload the filled WTC.

DEMONSTRATION RESULTS

The 1994 demonstration was divided into two phases. Phase 1 consisted of a full-scale test to dig and dump dry, INEL soils mixed with simulated waste to assess the ability of the IEE to control dust generation and the potential spread of contamination during dumping operations. Phase 1 of the demonstration was performed inside a building where conditions could be sufficiently controlled to simulate a retrieval environment. Dust and rare earth tracer spread data were collected using air monitors. The demonstration included both digging and dumping operations; however, the primary objective was to test the ability of the system to reduce the generation and spread of dust during the dumping operation. Phase 2 consisted of a full-scale retrieval of simulated buried waste using the TTV, IEE, and REMEX. The purposes of Phase 2 were to test the capabilities of the TTV and to assess the ability of the IEE, TTV, and REMEX to effectively handle, transfer, and transport the waste from the digface as an integrated system.
Results from the Phase 1 tests show the IEE to be a very effective and reliable system for retrieving and transporting buried waste through the excavating process. The IEE is rugged and well suited for digging heterogeneous buried waste. Retrieval rates of 60 m3/day (80 yd3/day) or more can easily be achieved. The IEE was very effective at handling the various waste forms present in the demonstration. In particular, the flexibility of being able to release the ITM and work with the IEE thumb to pick up objects and place them into the ITM or to move items around on the digface added to the ability of the system to handle a large variety of waste forms. The IEE and thumb were found to be dexterous and capable of handling waste forms ranging from 55-gal drums to loose paper and soil. The overall system integration of the REMEX, IEE, ITM, and WTC was excellent and added to the smooth and successful completion of the Phase 1 demonstration.
The process of using the IEE resulted in dust spread that was two orders of magnitude below the test goal. The transfer of the ITM from the excavator to the transfer point was accomplished with virtually no visible dust or tracer spread as expected. The Phase 1 demonstration successfully proved that the concept of eliminating the dumping at the digface is a viable, immensely practical approach to limiting potential contamination spread during buried waste retrieval. Controlling the dust generation at the source of digging combined with the IEE,s inherently dust-free dumping operation should result in substantially greater than 98% reduction in dust spread over cases without regard for contamination control. Following transport to a processing facility, the ITMs may be dumped in a glovebox environment where contamination control systems are more easily implemented.

The Phase 2 test combined the REMEX, TTV, and ITM system to demonstrate the integrated systems approach of the design in a full-scale retrieval. The TTV and IEE were designed as a system for enhanced performance. The spring bed on the TTV assisted with easy alignment and transfer of empty ITMs from the TTV to the REMEX, while the tapered recess in the WTC assisted with alignment during depositing full ITMs. The systems approach provided for smooth and easy integration between the TTV, IEE, and REMEX.

The throughput data show that the system had a calculated production rate of 34.75 m3/day during the demonstration. With only slight modifications, a single TTV/REMEX system with multiple ITMs would easily achieve 60-100 m3/day and multiple TTVs could greatly increase throughput rates. Lack of sufficient horsepower on the TTV was the single factor that most limited throughput during this demonstration; this is also one of the easiest items to correct.

The capability of the system to handle the various waste forms is excellent. The IEE and TTV integrated well and there was virtually no spillage when loading and transporting the waste. Operator procedures and approaches such as detaching the ITM to see the digface and placing objects directly into the ITM were facilitated by the systems flexibility and enhanced the IEE,s ability to handle waste forms and reduce dust generation during retrieval operations. The IEE and thumb were extremely effective in handling and transporting the large objects from the digface. This operation was completed without any drops or slippage of the large objects.

Piloting the TTV was impaired by the lack of adequate horsepower and uneven power application to the tracks. Work arounds through equipment modifications and modified piloting methods were used to successfully complete the demonstration. The primary effect of this was that the vehicle speed had to be reduced significantly to maintain sufficient horsepower output to easily maneuver. Other than persistent minor engine problems, TTV reliability was excellent.

The TTV control system functioned well throughout the tests. Telemetry tests showed that the control system range easily met the 150-m goal. The wireless video worked well and provided a clear video signal to the TTV operator, although there was some interference from the antenna mast in specific orientations and range was limited to less than 75 m. The RF emergency stop malfunctioned and repeatedly shut the TTV engine down, so one relay was bypassed to continue the demonstration. Ultrasonic sensors mounted on the TTV provided information to the TTV operator to dock at the targets. Some crosstalk between two ultrasonic sensors gave erroneous readings to the operator. These readings were far enough from the actual readings so that the two could be easily distinguished.

The data collected for the maneuverability tests show that the TTV can repeat its placement to the targets within the specified 30 radial cm 77% of the time. Crosstalk between ultrasonic sensors contributed to the less-than-expected results. A minor software change would easily eliminate sensor crosstalk, but was not readily field implementable. The TTV had no significant problems negotiating the terrain or the 10-degree sloped hill after the horsepower output was tuned.

The equipment used for this demonstration was designed to investigate concepts for retrieval, dust-free dumping, and conveyance of buried waste. Prototype hardware and low-cost components were adapted to facilitate an inexpensive, short schedule, proof-of-concept demonstration. The hardware concept and design approach is to support a TRU environment retrieval; however, due to cost and schedule constraints, no attempt was made to provide fully industrialized equipment. There were no aspects of the equipment design that would make it difficult or prevent it from being upgraded to operate in a TRU buried waste retrieval environment.

FUTURE DEVELOPMENT AND DEMONSTRATION ACTIVITIES

The systems developed in FY-94 will be enhanced in FY-95 in preparation for a follow-on demonstration. During the summer of 1995, the Buried Waste Integrated Demonstration (BWID) program will perform an integrated field demonstration at the INEL. The integrated tests will be performed on a simulated buried waste pit. The scope of the demonstration includes the characterization, retrieval, and removal of the simulated waste from the site. Primary equipment for the retrieval and removal will include the REMEX with IEE, a remotely-controlled gantry crane deploying a cooperative telerobotic retrieval system that will deploy digface characterization, multiple manipulators and soil vacuum systems, and an upgraded TTV. These devices will be used to remove overburden soil, place waste material in transport containers, and convey waste from the dig site. All aspects of this process will be

operated remotely from a control station housed in a separate building.
The FY-95 demonstration calls for removal of a limited portion of a buried waste pit known as a hot spot. Removal of the hot spot without disturbing the surrounding waste requires that the excavation be accomplished from and above grade position. This requires that the IEE design (including ITM and the WTC) be modified from its current front shovel configuration to a backhoe configuration. This will allow the REMEX to operate from the above-grade position and dig down into the waste while maintaining the functionality and dust-free dumping capability of the current IEE design.
The REMEX will be modified to accept the backhoe configured IEE and an adaptive tuning system to maintain dynamic stability with varying end effector loads will be implemented. Other upgrades such as a stereo vision system, ruggedized sensors, and addition of coordinated control of the thumb will be added to enhance the REMEX performance and improve the human-machine interface.
From its inception, the conveyance vehicle was intended to be capable of semi-autonomous operations. The TTV developed in FY-94 was the first step toward this capability. It was designed to prove the hardware and operational concepts for buried waste conveyance and to demonstrate integration with the IEE and REMEX. The second step to be taken in FY-95 is to upgrade the control system to provide this capability. The TTV control system will be upgraded to a self-guided transport vehicle (SGTV). The SGTV will use a state-of-the-art video tracking system along with mission planning software to preplan and then semi-autonomously perform the conveyance operations. This will greatly enhance the operational flexibility of the system and relieve the operator of the highly demanding task of piloting the vehicle in a teleoperated mode. Other system upgrades such as an improved engine, hydraulic system, and collision avoidance detection will be deployed on the SGTV.
SUMMARY
The IEE proved that dust-free dumping is an efficient, effective means for reducing contamination spread during retrieval operations. The IEE design can provide the same capability as standard end effectors for retrieval operations without the inherent problem of contamination caused by a dusty dumping process. The integrated systems design approach for the development of the REMEX, IEE, and TTV provides an effective solution for remote retrieval and conveyance of buried waste. This demonstration successfully showed that buried waste retrieval can be performed remotely at low cost, at reasonable throughput rates, and with greatly improved safety for the worker and environment.
The data obtained in this demonstration will eventually be used by Environmental Restoration during the feasibility study for parts of the RWMC and, possibly, by additional programs in other locations. The data will also be used by Environmental Restoration to project additional technology needs for BWID out-year planning.
The DOE Office of Technology Development is developing and demonstrating technology to allow remote retrieval of buried waste. The BWID program is leading this effort and is planning for a retrieval demonstration to be held in the summer of 1995. The demonstration will use three primary retrieval components: a remote excavator, a robotic gantry crane, and an autonomous waste conveyance vehicle. Each of these systems has had development and demonstration activities in the past, and will be further developed for inclusion in the 1995 demonstration.
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12-3
REHABILITATION OF THE SOIL USING THE MOBILE ROBOT VIRGILE

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ABSTRACT
For three years, the Robotics and Teleoperation Service of the CEA (the French Atomic Energy Commission) has been working on VIRGILE. This teleoperated vehicle is based on a four wheel drive Mercedes vehicle. It is adapted to be totally controlled by its embedded computer. The steering wheel, the throttle, the gearbox and the brakes are connected to the input/output boards of the VME computer. In a distant control room, the operator is able to drive the Mercedes using a joystick connected to an other VME computer. A Sun workstation controls two screens. On the first screen, the dashboard of the vehicle is displayed. On the second one, a top view represents the motions of the robot. The whole software is written in ADA language. The embedded computer and the fix one are connected together by a radio link relaying a 39400 b/s serial link. Furthermore, the pilot can see the environment of the robot on three video screens displaying pictures from the four embedded cameras.
The first application developed with VIRGILE is a part of the RESSAC project. The aim of this project is to rehabilitate the soil after a contamination due to a nuclear incident. The principle of the project is to pick up the grass on the soil, considering that the roots of the grass keeps the contamination at the surface of the ground. VIRGILE takes place in this scenario when there is no grass on the floor. Just after the incident, it can be teleoperated to sow grass of which roots will retain the contamination.
Two huge ramps has been installed at the rear of VIRGILE. On these ramps, sprays spreads the suitable liquid. The operator can control on the top-view interface that the whole area has been treated. The embedded computer is able to control the motion of the robot during short transmission interruptions. As soon as the transmission problem persists, the computer stops the vehicle.
The performed tests has been very satisfactory. The VIRGILE system is reliable and the vehicle itself is easy to control. If the dimensions of the vehicle will probably allow to efficiently protect the embedded computer against radiations, the problem of the safety of the transmission could be critical. But the power of the embedded computer will be used to increase the autonomy of the vehicle and therefore to decrease the need of high quality transmission. VIRGILE will be ready for the RESSAC project.

INTRODUCTION
The RESSAC project (French acronym for Soils and Surface Rehabilitation after Accident) has been initiated by IPSN (Nuclear Protection and Safety Institute). The aim of this project is to rehabilitate the soil after a contamination due to a nuclear incident. The principle of the rehabilitation is to pick up the grass on the soil, considering that the roots of the grass keeps the contamination at the surface of the soil. When there is no grass on the ground, a first step would be to sow a "fast growing" lawn of which roots will retain the contamination.
In order to efficiently and rapidly act after the contamination, the use of teleoperated vehicles is recommended. The work of STR (Teleoperation and Robotics Service) of CEA was to demonstrate the feasibility of a teleoperated vehicle able to perform the sowing task.
A first phase in the STR work was to realize the interface between a standard vehicle and an embedded computer. Then the embedded computer just had to communicate with a remote control station. During the second phase, the system performances have been evaluated in terms of ergonomy and driving accuracy. One of the most important criteria was the easiness in performing a systematic sowing task.

PRESENTATION OF THE SYSTEM
For the RESSAC project, STR has developed VIRGILE. This unmanned vehicle is based on a four wheel drive Mercedes. An operator can totally control VIRGILE from a remote control station.

The Vehicle
The used vehicle is a 300GD Mercedes. It is 5 m long and 1,70 m wide. The engine is a diesel one. The gear box is automatic and the steering is powered. These features have made the adaptations easier. The gear lever is controlled by a hydraulic jack. A small hydraulic motor is able to drive the steering wheel. The length of the vehicle allowed an hydraulic power plant, two extra batteries, an analog controller and a VME computer to be embedded.
Four fixed video cameras has been installed inside the vehicle to transmit pictures

of the vehicle environment to the operator. Three radio transmitters and one receiver link the vehicle to the remote control station.
During teleoperation phases, the embedded computer receives commands through the radio. It interprets the commands and sends them to the analog controller that controls the hydraulic actuators of the steering wheel, the throttle, the brakes and the gear lever.
VIRGILE can also be controlled in a usual way. Pressing a button on the dashboard gives the control back to an embedded pilot.
Fig.1 The VIRGILE Robot
The Remote Control Station
The remote control station includes three video monitors, deux computer screens, a SUN workstation, a real time computer, three radio receivers and one radio transmitter. On a desk in front of the screens, a joystick and four buttons are the input devices of the system for the operator.
A keyboard and a track ball are also available for operations on the computer.
The remote control station stays on castors. It can be easily moved into a van and be used in any suitable place.
The Radio Link
A full duplex radio link allows the connection between the computer of the remote control station and the embedded one. This link carries a RS422 serial link at 39400 b/s.
Two video pictures are sent from the vehicle to the control station by two transmitters developed by CEA/LETI.
The Sowing Tool
To perform the sowing task, a tool has been fixed at the rear of the vehicle. In order to avoid driving problems due to a trailer, the two six meter ramps have been fixed directly on the chassis of the vehicle. The length of the ramps has been chosen in such way that a U-turn allows the ramps to cover continuously a surface. A wheel is fixed at the end of each ramp. And the ramps are articulated on the chassis in order to deal with a non perfectly flat field. Shrouds from front and from rear keep the ramps perpendicular to the vehicle.
A compressed air bottle provides pressure in the container of product to be sowed. The sowing is controlled by two electrovalves connected to the embedded computer.
Fig. 2 Top View of the Sowing Tool
THE FUNCTIONS OF THE SYSTEM
The Driving
The pilot in the remote control station can control the direction and the speed of the vehicle using the joystick. He drives according to the video pictures displayed on the three video screens.
He can select the drive, rear or neutral position of the gear by pressing the buttons at the top of the joystick. Pushing the joystick controls the acceleration, pulling the joystick controls the braking. When VIRGILE reverses, the rear camera is selected and displayed on the main video screen and the the direction commands are flipped. So the pilot has the same feeling in driving backwards as in driving forwards.
On the left hand computer screen a dashboard is represented. On the main window, the driver can check the speed, the rolling and pitching angles of the vehicle, the selected gear position. On other pop-up windows, the state and the control of the devices (lights, windscreen wipers, sowing tool, horn...) are available. In the upper left area of the screen, warning messages are displayed and an emergency window automatically pops up in case of serious problem: loss of transmission, motion with the handbrake locked or with an opened door for instance.
The Mission following
On the right hand computer screen a top view shows the global trajectory of the robot. A graphic toolkit allows the operator to draw some landmarks as buildings, roads or trees. A planned trajectory can also be represented on the screen. So the pilot can check he is driving on the planned trajectory. If a map of the environment was available, it could be represented on this screen.
It is also possible to draw the surface covered by the ramps behind the vehicle. The operator can check that the vehicle has sowed every where on the field.
THE TESTS
The Evaluation Site
The tests have been performed at CEA center of Saclay in the south of Paris. The

movements of VIRGILE took place on a more or less rectangular shaped field. The field is 200 meters long and 40 meters wide. The ground is horizontal but irregular. A 50 cm high bump allows to test the articulation principle of the ramps and the measurement of the rolling and pitching angles. The ground is partly grassy.
Fig. 3 the Sowing Tool Mechanism
The Test Courses
The operators have freely driven on the field trying short turns and high speed straight lines. The performances have been particularly evaluated during three kinds of courses:
- Slow slaloms: the speed is about 5 km/h. The gates are 2.5 meters wide. Two consecutive gates are in opposite corners of a 5 meters square.
- Fast slaloms: the speed is about 15 km/h. The gates are 4 meters wide. Two consecutive gates are in opposite corners of a 10 meters square.
- Systematic covering: the vehicle covers the field making 20 meters long straight lines 12 meter apart.
The Pilots
Four pilots have regularly driven VIRGILE. Three of them were engineers working on the project. The fourth was a mechanics technician not particularly involved in the project. The latest pilot has tested the performances of the vehicle on the three courses described above.
THE RESULTS
Ergonomy
After set up and configuration, driving the vehicle seemed very easy to the operator. Thanks to the human factor specialist's advice and after some corrections during the tests, it is possible to drive VIRGILE with just one hand. Handling the joystick, the operator can control the gear box by pressing buttons of the top of the joystick with his thumb. The automatic selection of the rear camera makes the driving easier: the driver does not need to let the joystick to select the suitable camera.
Flipping the direction controls when the vehicle drive backwards is surprising for the driver during the first few seconds. One can thinks it is because of the operator is used to drive cars and instinctively turns the steering wheel on the right for going backwards left. But after the first hesitation, the new driving way is comfortable. Maybe too comfortable and sometimes the operator even forget he his driving backwards.
We have noticed that the operator does not look very often at the computer screens. Actually he is concentrated on the video screens and does not loose time in looking at computer screens. As long as everything is fine, he actually does not have to. The main information he could need from the computer screen is the actual speed of the engine. Looking the video screens does not provide a good estimation of the speed. It could be useful to provide speed information either by mixing numeric information on the video screens or by using a sound feed-back of what happens inside the car.
Reliability
The weakest point is the transmission system. During the three days of intensive evaluation we had many contact losses between the vehicle and the remote control station. Of course these control losses are handled by the computers. When the embedded computer has not received information for a second, it stops VIRGILE waiting for a communication improvement. Fortunately the radio link comes back after a couple of seconds but if it had not been the case, the vehicle would have been definitely lost. We have used standard radio system for the VIRGILE system but for a real application it will be necessary to use high performance transmission system. Because the loss of the control radio link is detected and handled by the computers, this problem is not too serious. On the other hand the loss of the video transmission link is much more serious. When the operator drives at 40 km/h, he does not like to loose the visual control of his trajectory. Today, the embedded computer is not able to detect video link problems, so the operator has to brake the vehicle by himself when he does not see precisely enough where he is driving. The transmission system of the video pictures has to be even more highly efficient than the control transmission system.
We have not been faced to the problem of radiation but we know that it is something we would have to handle before real application of such a system. After a nuclear incident, the radiation level could disturb the working of computers and cameras.

There is still some work for making VIRGILE as reliable in this kind of environment as in a "pure" one.

Performance

In terms of accuracy in driving, operators have been astonished at the efficiency of the system. Of course the fix cameras inside the vehicle does not allow the operator to anticipate the needed motions but the speed of the steering wheel control makes up for this problem. With a short wrist movement on the joystick, the operator is able to turn the wheels as far as using more than one steering wheel turn. During the slaloms, very few gates has been touched.

The systematic covering task does not raise any particular problem except checking ramps for collision. It will be necessary to use two more cameras checking for the trajectory of the ramps.

CONCLUSION

Today VIRGILE works in accordance with the specifications. The main limitations before performing a real rehabilitation mission are the transmission reliability and the resistance to radiations. If dedicated sub-systems are used, these limitations will be suppressed.

In order to make the vehicle easier to be driven, the future developments will concern the autonomy of the vehicle and mainly improvements of the remote control station.

The autonomy can be used to perform systematic covering tasks. The operator will just check that everything is going fine. Autonomy can also be useful to solve transmission loss. The vehicle could come back autonomously to get the radio contact back.

Experiment shows that, as soon as the environment becomes highly unstructured, the operator does not have enough information to drive efficiently. But it is also shows that he already can not deal with all the available information. We have to find a new way of driving teleoperated vehicles, of giving information to the operator in order to make an unmanned vehicle even easier to drive than a usual car.

We can not wish that soil rehabilitation task becomes an every day task but we have to be ready to perform it as efficiently as an every day task.

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PERFORMANCE TESTING OF A SYSTEM FOR REMOTE ULTRASONIC EXAMINATION OF THE HANFORD DOUBLE SHELL WASTE STORAGE TANKS

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ABSTRACT

A mobile robotic inspection system is being developed for remote ultrasonic examination of the double wall waste storage tanks at Hanford. Performance testing of the system includes demonstrating robot mobility within the tank annulus, evaluating the accuracy of the vision based navigation process, and verifying ultrasonic and video system performance. This paper briefly describes the system and presents a summary of the plan for performance testing of the ultrasonic testing system. Performance test results will be presented at the conference.

INTRODUCTION

Raytheon Service Company and RedZone Robotics have been contracted by Westinghouse Hanford Company to develop a system for inspection of the double shell waste storage tanks at the Department of Energy Hanford Site. The system must be able to perform Ultrasonic Testing (UT) and over 100% of the primary and secondary tank walls and

the primary tank knuckle as well as over a small percentage of the tank bottom which is accessible via air slots under the tank The system must also be capable of executing a specified initial inspection sequence in a 36 hour period, including cleaning, visual verification of surface preparation and ultrasonic inspection.
The Double Shell Tank Inspection system consists of a Mobile Control Center, Deployment Module, Cable Management Assembly, Robotic Mechanism, Ultrasonic Testing (UT) System, Visual Testing (VT) System, Robot Control System, Pneumatic Distribution System and Electrical Distribution System. An artists rendition of the system is shown in Fig. 1. A rigorous testing program is necessary to ensure that operations in the actual tank farm are smooth and trouble free and that the system will meet the requirements of the intended application. This paper focuses on performance testing of the Ultrasonic Testing system.
Fig. 1. Double Shell ank Inspection System.
TEST FACILITY
Complete and safe testing of the system requires a dedicated test facility. A full scale, partial section of a double-shell tank has been constructed in RedZone Robotics' Pittsburgh facility. The mockup is approximately 35 feet long and 12 feet nigh, constructed of three courses of four by eight foot sheets of rolled steel. A section of the mockup includes simulated haunch and knuckle regions, along with air slots. A platform mounted above the tank section represents ground level, and is used for deployment of the system through a riser. Westinghouse Hanford Company has constructed a similar, but somewhat larger facility at the Hartford site for testing and training. The performance test will be conducted at the RedZone facility.
ROBOT MECHANISM
The robotic mechanism supports deployment of the UT and VT systems. The mechanism consists of a cable management assembly, a storage enclosure, a mobile vehicle called the scan carriage and a cleaning system to prepare the tank surface for inspection. To the extent possible, the scan carriage is constructed of materials that can withstand a total radiation dose up to 1 x 106 Reds. Electronics are kept to a bare minimum, and no on-board electronics me used for components needed for recovery. An artists rendition of the scan carriage is shown in Fig. 2.
Fig. 2. Scan Carriage.
Scan Carriage
The scan carriage vehicle is a walking platform that carries cleaning and inspection hardware between the wails of the annulus of the waste tanks. The vehicle supports inspection of the primary and secondary tank wails and the floor of the primary tank via the air slots. From any location, a rectangular scan area of 64 in. nigh by 22 in. wide can be accessed. In order to inspect the secondary tank wall, the scan carriage must be relieved and rotated 180 degrees. The scan carriage is able to negotiate obstacles (air supply pipes) that are up to 15 in. from the primary tank wall, and 15 in. from the secondary tank wall. A separate tool package is used to access the air slots under the tank. The vehicle is sized to fit through a 24 in. access riser. The scan carriage vehicle navigates with the aid of an industrial machine vision system which tracks weld seams and identifies weld intersections using edge detection and line fitting algorithms,
The performance of the scan carriage will be tested by executing a series of maneuvers, including deployment. negotiation of obstacles, carrying out cleaning and inspection tasks and fail safe recovery.
Cleaning System
Effective ultrasonic examination requires that the weld, the heat-affected zone and surrounding area be free of excessive rest build-up, scale and debris. To meet this requirement, each inspection tool is outfitted with a maneuverable cleaning head. Each cleaning system consists of a cleaning head, power sources (air and electric) and attachment hardware for the corresponding payload delivery device. The cleaning system removes loose material from the base metal, but leaves tightly adhering mill scale.
An application based test will be used to demonstrate the performance of these tools. In operation, the cleaning tools are used to prepare the surface for inspection. So, the test will start with realistic surface conditions. The robot will clean the surface, and a UT inspection will be performed. Success is demonstrated by good ultrasonic coupling.
Visual Testing System
Preparation of the tank surface for ultrasonic inspection is verified by the Visual

Testing system. Additional functions of the VT system are to monitor cleaning and ultrasonic operations, and to provide video image feedback to the robot navigation system and the operator. The VT functions are performed by four cameras, one (mounted on a pan and tilt on the scan carriage) to provide a view of the general area around the scan carriage, one to perform wall inspection, one to perform air slot inspection and one (deployed through a 3 or 4 inch tank riser) to provide an overview of the scan carriage as it enters, leaves and moves within the annulus.

DESCRIPTION OF THE ULTRASONIC TESTING SYSTEM

The UT examination is designed to determine tank wall thickness, and detect and size corrosion induced pitting and cracking. The P-Scan Ultrasonic testing system, manufactured by the Force Institute in Denmark, will be used for data collection and analysis. The multi-channel data acquisition capabilities of the P-Scan system allow for simultaneous and continuous acquisition of data from four transducers. There are two transducer array modules in die DSTI system.

Tank Wall Transducer Array Module

The tank wall transducer array module contains transducers oriented at a variety of angles, using various wave modes, and frequencies to provide detection and sizing of planar reflectors and wall thickness measurements within the specified examination volume. The module can be moved to and from the weld centerline and along the weld axis. It is also capable of +180 degree rotation to provide maximization of echoes for detection, discrimination, and sizing of indications in the specified examination area regardless of their orientation to the weld centerline.

The scan pattern begins at the uppermost point of the examination area and proceeds down the tank wall, following the shortest or most convenient combination of vertical and horizontal welds. Scanning will be from both sides of die weld where access allows. Scan motion is provided by the Scan Carriage. The nominal scanning speed is 3 inches per second.

Knuckle and Air Slot Transducer Array Module

The knuckle and air slot transducer away is capable of movement in a straight line beginning at the upper most examination point of the knuckle and extending 24" into die air slot. Data is also obtained while retracting die head from the air slot. Due to the restricted area in the air slot, indexing of the transducer array module is not possible. Required examination volume coverage is provided by the orientation of the transducers within the module. The nominal scanning speed is 3 inches per second. The knuckle and slot UT transducer module mounts to the scan carriage on die output of one of the payload reels mounted on die secondary slide of the y-member. Vertical and horizontal motion are provided by the scan carriage.

UT SYSTEM PERFORMANCE TESTING

The flaw detection and sizing methodology for planar flaws evolves from techniques proven in the nuclear power industry for the detection and sizing of intergranular stress corrosion cracking. These techniques provide accuracy meeting the specification, but are dependent on surface condition. Testing of the UT System will verify the accuracy of these techniques by examining test plates with known flaws under expected operating conditions. The mockup tank surface will be heated to test the effect of heating of the couplant on UT scanning. Testing of the VT system will be conducted to confirm the ability of the system to provide the required supper for ultrasonic testing and the robot navigation and tele-operation functions. A dusty environment will be created inside the mockup annulus to confirm the ability of the VT system to function under realistic conditions.

The UT system performance test is to be conducted in a manner that simulates the way in which examination is conducted in the field and shall be performed in accordance with a written procedure provided by the contractor.

Performance Demonstration Samples

Performance demonstration specimens shall be fabricated from the same DST wall material or equivalent materials and shall have surface condition representative of the general condition of the DST scanning surface. Performance demonstration specimens shall have sufficient area to minimize spurious reflections and shall include thickness, pitting, and crack samples. The ultrasonic reflectors will be distributed to cover the through-wall thickness (t) in a minimum range from 0.25t to 1.0t

Performance demonstration specimens shall be divided into grading units. A grading unit is deemed as a minimum of 3-inches of continuous vertical or horizontal length. If a grading unit is designed to be flawed, then one (1) inch of unflawed material

must exist on either side of the grading unit The specific segment of length used in one grading unit may not be used in another grading unit.
Orientation of the cracks shall be either perpendicular or parallel to the major axis of the performance demonstration specimen. Flawed and unflawed grading units shall be randomly mixed. Performance demonstration specimens shall be embedded in the primary tank wall with their sound entry surface facing the annulus; sound entry surface of the samples shall be flush with the surrounding wall surface; exposed back surface of the samples shall be masked. The same performance demonstration specimens may be used for both detection and through-wall sizing.
Calibration
An initial calibration shall be performed and documented at the start of the performance test. The examination methodology requires multiple calibrations corresponding to 25F temperature increments. To verify the function of the temperature detectors in the UT modules at least two widely different readings shall be taken at location where the temperature can be independently measured. The high reading should be about 200 10F.
Examination Procedure
All ultrasonic indications which exceed 20% reference shall be investigated to determine their maximum amplitude. All ultrasonic indications which exceed 50% reference shall be recorded on the indication report form. Any indication determined to be originating from a surface connected planar flaw shall be tagged and reported, regardless of amplitude.
Wall thickness measurements less than 87.5% of specified original wall thickness will be tagged and reported. Stress corrosion cracks greater than 0.5 in. in length will be tagged and reported. Corrosion pitting greater than 0.35 times specified original wall thickness in depth will be tagged and reported.
For reportable indications, date shall indicate the location, area extent, and maximum amplitude for each indication. For thin areas, the reported date shall summarize the thickness measurement results.
Reports
An Indication Report Form shall be prepared for each Performance Test Plate. Location, orientation and size of cracks shall be reported on the Indication Report Form and marked on the P-Scan printouts.A detailed Tank Inspection Report shall also be prepared summarizing findings during the entire performance test.
All P-Scan examination date shall be printed in hard copy form identifying any areas with shear wave amplitudes exceeding 50% reference. The report shall note the maximum amplitude location in the X and Y axis, and the global robot coordinates. Any area that exhibits ultrasonic echoes exceeding 50% reference, hat which are determined to be geometric in nature, shall have a brief written statement outlining the basis for the conclusion. Any area determined to contain planar oriented flaws shall be identified on the hard copy printout and in a written report describing the techniques utilized and basis for the conclusion.
Performance Criteria
To complete the detection performance test successfully, the system must perform in accordance with Table I.
N- 1 crack depth from a total of N shall be reported. A depth result on one (1) crack is allowed to be left blank. If depth of all N cracks is reported, the fast N consecutive samples will be evaluated. A false call is defined as calling an uncracked grading unit cracked. A crack parallel to the major axis direction is detected successfully if a minimum of .50% of its actual length has been reported. An allowance of 0.5 inch is permitted in one direction or the other from the actual crack location.
All detected flaws shall be sized to the following requirements:
a. For circumferential flaws the minimum detected flaw shall be t in length +/- 0.5 inches and 0.5t in depth +/- 0.050 inch;
b. For longitudinal flaws the minimum detected flaw shall be 12 inches in length +/- 0.5 inches and 0.2t in depth +/- 0.050 inch;
c. For Pitting the minimum detectable pit shall be 0.7t in diameter and 0.35t in depth +/0.02 inch. The following criteria apply to depth sizing and wall thickness measurement:
 d. Each successfully reported depth-sizing measurement shall be within +/- 0.050 inch of the true depth.
 e. Wall thickness measurement shall be accurate to within +/- 0.0025 inch.

Timing Documentation
The time required for data acquisition, analysis and reporting on each sample shall be documented. The following tasks must be performed during this time:
a. Repositioning the scan carriage within one performance demonstration specimen.
b. Scanning one performance demonstration specimen for reflectors oriented perpendicularly and parallel to the major axis. This activity will consist of some number of complete scans.
c. Data analysis and generation of required documentation.
The following items are not included in the acquisition time:
d. Initial UT system calibration or calibration check.
e. Positioning scan carriage at the scan start on the performance demonstration specimens.
PROJECT STATUS
The fabrication of the Double Shell Tank Inspection System is complete and testing will have been completed by the time this paper is presented. Results of performance testing will be presented at the conference. Tank inspection operations are expected to begin in June 1995.
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DEVELOPMENT OF THE STORED WASTE AUTONOMOUS MOBILE INSPECTOR (SWAMI II)
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ABSTRACT
A mobile robot system called the Stored Waste Autonomous Mobile Inspector (SWAMI) is under development by the Savannah River Technology Center (SRTC) Robotics Group of Westinghouse Savannah River Company (WSRC) to perform mandated inspections of waste drums stored in warehouse facilities. The system will reduce personnel exposure to potential hazards and create accurate, high-quality documentation to ensure regulatory compliance and enhance waste management operations. Development work is coordinated among several Department of Energy (DOE), academic, and commercial entities in accordance with DOE's technology transfer initiative. The prototype system, SWAMI I, was demonstrated at Savannah River Site (SRS) in November, 1993. SWAMI II is now under development for field trials at the Fernald site.
INTRODUCTION
Thousands of drums of radioactive, hazardous, and mixed waste are currently stored at DOE sites throughout the United States. These drums are stored in warehouse-like facilities on an interim basis, pending final disposition. Recent emphasis on anticipated decommissioning of facilities indicates that many more drums of waste will be generated, requiring additional storage. Federal and state regulations dictate that hazardous waste covered by the Resource Conservation and Recovery Act (RCRA) be inspected periodically for container degradation and to verify inventories. All known DOE waste storage facilities are currently inspected manually. A system to perform robotic inspection of waste drums is under development by the SRTC Robotics Group of WSRC. The robotic system is the Stored Waste Autonomous Mobile Inspector (SWAMI).
The DOE Office of Technology Development (OTD) is directing this effort through the Robotics Technology Development Program (RTDP) and through Morgantown Energy Technical Center (METC). In accordance with DOE's technology transfer initiative, development work is coordinated among several DOE, academic, and commercial entities. Along with WSRC, participants include the Lawrence Livermore National Laboratory (LLNL), the University of Michigan, the University of South Carolina, the Georgia Institute of Technology, Martin Marietta Astronautics Division, and Fernald Environmental Restoration Management Corporation (FERMCO).
OBJECTIVES AND BENEFITS
Program objectives fall into several categories: reducing personnel hazards, increasing cost effectiveness, improving inspection data quality, and providing

additional information to enhance waste management operations.
Manual inspections of waste drums require personnel to be in close proximity to waste materials for extended periods of time, creating a potential for exposure to hazardous or radioactive materials. Robotic inspections will drastically reduce personnel exposure to these hazards. Significant cost savings can also be realized by reducing the personnel time required for inspections.
Since facilities of this type are typically capable of storing thousands of waste drums at a time, manual inspection is extremely tedious, and significant variations in inspection quality can be expected. Additional inconsistencies are introduced by the variations in the competence and experience level of inspectors. Robotic inspection of waste containers will not exhibit the degradation of quality or the variability associated with human performance in highly repetitive tasks.
High-quality, consistent inspection data will contribute directly to prevention of waste containment failures. The vehicle's sensor mast will also allow SWAMI to inspect the topmost level of drums (up to four drums high) as competently as the first level, a feat not easily achieved with manual inspections.
SWAMI also provides additional information not currently available with manual inspections. During the inspection, SWAMI will store images of each drum and data associated with it, including geometric data, the drum's location, bar code number, and a time stamp. After downloading, this information can be accessed directly through the site database for inventory verification, drum condition trending and other operations support. SWAMI will also conduct a rigorous survey of the facility floor for potential radioactive contamination. Performing a radiation survey during every inspection serves two functions: it provides an early alert to a radioactive spill, and also prevents the vehicle from unwittingly picking up the contamination and tracking it throughout the facility, as might conceivably occur during a manual inspection.
PROGRAM
The project is divided into four phases: prototype development and demonstration (SWAMI I), system development (SWAMI II), demonstration and testing of SWAMI II at Fernald, and transfer of technology to industry. Prototype development was initiated in 1992 and culminated in a series of SWAMI I demonstrations during the RTDP Robotics for Mixed Waste Operations Demonstration at Savannah River Site in November 1993 (See Fig. 1.). SWAMI II is currently being developed, and will incorporate lessons learned during prototype development, include several features not found on SWAMI I, and address Fernald-specific requirements. SWAMI II is currently scheduled to begin approximately 17 weeks of field demonstrations and testing at Fernald in June of 1995. Following the demonstration, an offering of SWAMI II technology will be made to industry through the SRS Technology Transfer Department.
Fig. 1. SWAMI I at the November, 1993 RTDP Robotics for Mixed Waste Operations Demonstration.
Since technology transfer to industry is a fundamental program objective, developing a system that performs its function in a cost-effective manner is of major import. Development strategy is to utilize known commercial technologies, then apply and integrate them in an innovative way. Vehicle subsystems are being developed independently and then integrated on the vehicle platform. This modularization of subsystems is also reflected in the system software.
PROJECT CONSTRAINTS
Although the parameters used for developing the vehicle are a composite of the general features of existing and planned storage facilities at the target sites, SWAMI II will initially be deployed at the Fernald Site for testing and demonstrations. As a result, Fernald waste storage specifications are predominately being used to configure SWAMI II. Waste storage facilities at Fernald are a combination of previously existing process facilities and dedicated storage facilities. Although other containers are present in relatively small numbers, SWAMI II will be targeted to inspect 55- and 85-gallon drums. These containers are stacked on pallets up to four levels high. Each drum is bar coded with a unique number for identification. Bar code label placement is relatively uniform. Aisles are a minimum of 0.91 m (36 in.) wide, many of which allow vehicle access from only one end, requiring
SWAMI II to back out of aisles. Storage facilities typically do not have temperature control. Several storage facilities have a drum capacity of approximately 12,000 drums.

SWAMI I DESCRIPTION

The SWAMI I systems fall into four general groups: the operator computer, the vehicle supervisory computer, the base vehicle, and vehicle subsystems. There are four main vehicle subsystems: image capture/compression/storage, bar code scanners, the radiation monitor, and the position determination system.

Operator Computer

The operator computer uses two programs to interact with SWAMI I: the SWAMI Operator Interface and RADMAP. Both were developed by SRTC. The SWAMI Operator Interface provides remote vehicle controls, an inspection interface, and a configuration interface. RADMAP provides a facility map on the operator screen with the vehicle's current position and status indicated with an icon. Radiation data can also be plotted as it is received from SWAMI I, or at a later time. Other information, such as alarms, can also be displayed. RADMAP has been submitted for copyright by SRTC and will be available for technology transfer. The operator computer communicates with the vehicle supervisory computer over a spread spectrum radio ethernet.

Vehicle Supervisory Computer

The vehicle supervisory computer consists of three microcomputers on a half-height VME backplane and other peripherals, including a LAN board, and digital and serial interfaces. It communicates with the operator computer over a spread-spectrum radio ethernet, and commands the base vehicle and onboard subsystems through RS-232 and RS-485 (multidrop) serial ports. The vehicle supervisory computer software was developed by the University of South Carolina. SWAMI I is the first commercial mobile robot to utilize GENeral Interface for Supervisor And Subsystems (GENISAS) software from the Generic Intelligent System Control (GISC) library created by the OTD Robotics Program. "GENISAS is a GISC-Kit package that provides general communication software interface capabilities (such as command processing and event handling) between the supervisory control system and subsystems." (1)

Base Vehicle

The base vehicle is a modified Transitions Research Corporation (TRC) HelpMate. Several modifications, such as pallet-sensing ultrasound transducers and vehicle software enhancements, were added by TRC. Other modifications were made by SRTC, including the addition and integration of subsystems. The vehicle is equipped with several systems to perform navigation, collision avoidance and collision recovery. Basic navigation is performed by dead reckoning, using encoders on the drive wheels and a gyroscope. An ultrasonic array faces forward and to each side of the vehicle. These sensors register physical features and update navigation. Collision avoidance is also accomplished using these sensors. Additional ultrasound transducer "blossoms" are mounted on the front bumper to register off of drum pallets. A forward-looking structured light system provides additional collision avoidance capability. Should a collision occur, pressure-sensitive strips located around the vehicle's periphery and compliant bumpers with deflection sensors are used to detect the impact location to aid in recovery. Emergency stop switches are also provided. The vehicle's 0.71 m (28 in.) overall width will allow it to successfully navigate within the 0.91 m (36 in.) wide aisles at Fernald.

Image Capture/Compression/Storage

The SWAMI I image capture/compression/storage system uses fixed-mount CCD color video cameras and light strobes for each drum level and aisle side. The strobes provide consistent lighting levels for image capture while minimizing power consumption. A PC-DOS (486/33 MHz) computer performs the actual image collection functions. At a command from the vehicle supervisory computer, the image system captures the digital drum images, compresses them, and attaches other data to the compressed file, including each drum's bar code number, drum location and a time stamp. The entire file is then stored to an onboard optical disk. Image capture and compression functions are performed on commercially available PC-DOS boards.

Bar Code Scanners

Fixed-mount rastering bar code scanners read the unique bar code number on each drum. A separate scanner is used for each drum level and aisle side. The scanners use a 670 nm visible laser diode light source, which is eye-safe. Since the bar codes are oriented "picket fence" style, the laser scans horizontally. A raster feature indexes the horizontal laser line vertically, creating a "scan window." This window allows the scanner to read bar codes whose location and orientation are not precisely controlled. The simple raster feature avoids the complexity, expense, weight, and power requirements of omnidirectional units. The scanners collectively

communicate with the vehicle supervisory computer through a single RS-485 multidrop port.

Radiation Monitor

The radiation subsystem monitors the floor for potential alpha and beta-gamma radioactive contamination as SWAMI I performs its inspection. The system uses gas proportional detectors, located in front of the vehicle, to detect contamination before the vehicle passes over it. Onboard P-10 gas cylinders are required to continuously purge the detectors. The system is the same as that used on SIMON, another SRTC mobile robot. SIMON has demonstrated that robotic surveys are superior to manual surveys in detecting low levels of radioactive contamination. As SWAMI I performs the inspection, data from the radiation subsystem is sent to the operator interface computer in real time over the radio ethernet and displayed on RADMAP.

Position Determination

Due to the large area SWAMI I must navigate within, accumulated errors in the base vehicle's dead reckoning navigation system will become significant. The position determination subsystem updates the vehicle's dead-reckoned position and maintains the vehicle's odometry within acceptable accuracy. Accuracy is required not only for vehicle navigation, but also so that drum images are taken from the proper vehicle position. The system uses a 360-degree scanning laser to read bar coded retroreflective fiducials placed on the facility walls. This information is then used to periodically refine the HelpMate's dead-reckoned position.

SWAMI II DEVELOPMENT

The SWAMI II system to be tested at Fernald will utilize the same base vehicle as SWAMI I, but will differ in several aspects from the prototype already demonstrated. Additional subsystems will enhance SWAMI II's ability to perform a comprehensive inspection and reduce the operator workload. Several changes are also being made to address Fernald's specific site requirements.

Geometric Inspection

A significant addition to SWAMI II will be a geometric inspection subsystem, which uses structured light to detect drum dents and blisters. Since the storage facilities at Fernald typically experience significant turnover, damage to drums (such as dents) from handling operations can occur. Another potential failure mode is corrosion of drums from the inside. This type of failure is difficult to detect visually, until it is manifested by a blister on the drum's exterior. The geometric inspection system will detect these blisters to identify an imminent containment failure. It utilizes two arrays of laser line generators and CCD cameras to obtain range-to-surface data.

Waste storage drums at Fernald are painted glossy black, a surface which makes geometric inspection with structured light difficult. Structured light systems rely on a diffuse reflection of the laser line for detection by the camera. The curved glossy black drum surfaces absorb some of the light, and reflect most of the remaining light specularly. As a consequence, very little of the impinging laser light is visible to the imaging cameras. To obtain an adequate return from the projected laser line under these conditions (while remaining in the Class II eyesafe laser realm), the orientation, laser intensity and standoff distance of the sensor package must be maintained as the sensor passes each drum.

In order to provide the required geometry, inspection platforms will be built so as to position two arrays of laser line projectors to generate two contiguous horizontal laser lines on the drum (See Fig. 2.). Two laser lines are required to fully inspect each drum's visible surface, since the cameras' view of each laser line will be periodically blocked at the top or bottom of each drum. An interlaced array of CCD cameras will sequentially capture the laser line image as the inspection array moves vertically, parallel to the drum axis. Data from the geometric inspection will be transmitted to the operator computer during the inspection mission for subsequent computer analysis.

Drum Locating

Since the structured light system requires its lasers and cameras to be aimed accurately relative to the drum axis, a drum locating subsystem will use ultrasound sensors to locate each drum during the inspection. One pair of ultrasounds mounted on the vehicle will locate the bottom most drum to position the vehicle for inspection of a drum column. As the inspection array moves vertically and transitions to subsequent drum levels in the column, another set of sensors mounted on the inspection array will provide drum location information so that variations in

drum location within the column can be accommodated.

Rust Spot/Streak Analysis

As drums are inspected, SWAMI II will also capture and transmit color drum images to the operator computer. In order to achieve the required resolution, each drum will require several piecemeal images to be taken. The operator computer will then perform an image analysis for rust spots and streaks, and will present only those drum images which indicate a potentially unacceptable corrosion condition to the operator. Drum images and associated data will be archived in a mass storage medium accessible to the operator and other site organizations. A prototype for the rust analysis system was developed by Martin Marietta Astronautics. Lawrence Livermore National Laboratory is developing the rust analysis subsystem for SWAMI II.

Site Database Interface

Fernald is implementing a Sitewide Waste Information Forecasting and Tracking System (SWIFTS). SWAMI II will interface with SWIFTS to obtain current information on the waste container inventory in the facility to be inspected, including each container's size and location within the facility. After an inspection, SWAMI II will report back to the database any discrepancies between the expected and discovered inventory. Drum images, geometric inspection data and other SWAMI II file data can also be accessed via the SWAMI II database.

Autonomous Backing

Since many of Fernald's aisles allow vehicle access from only one end, the base vehicle will be modified to allow autonomous backing with sensors to prevent collisions. Although the HelpMate is capable of backing to recover from a collision, its path planning software does not accommodate following a path while backing. Furthermore, the HelpMate's sensor array is configured for forward motion. The University of Michigan is developing an independent ultrasound subsystem which will use its own sensor array to navigate and control the base vehicle drive system during backing. It will also detect objects in its path to prevent collisions.

Scintillation Type Radiation Monitor

Scintillation-type radiation detectors are being implemented for SWAMI II. SWAMI I uses gas proportional detectors, which must be purged with P-10 gas before and during operation. Scintillation detectors do not require P-10 gas, so the onboard space requirements and mass of gas cylinders are eliminated. Lengthy preinspection purge times and gas management equipment are also eliminated. The computer and detector system is being developed by National Nuclear Corporation (NNC).

Position Determination

SWAMI II will also feature a position determination system being developed by the Georgia Institute of Technology. The system utilizes two CCD cameras mounted on a pan unit with strobes to illuminate retroreflective fiducials placed on the facility walls. Each fiducial has a pattern which the system uses to calculate both the direction and range to the target. The availability of range data will enable the system to calculate SWAMI II's position and orientation more accurately than the system used on SWAMI I. The Georgia Tech system also compares well with the SWAMI I system with respect to operating range: approximately 200 feet versus 100 feet.

SUMMARY

The SWAMI system will provide a safe, cost-effective and comprehensive method to perform mandated inspections of waste drums and obtain useful data for facility and waste management. Because it does not rely on extensive facility modifications, SWAMI will be able to be deployed in a specific facility with a minimum investment of time and capital. By automatically culling out those drums whose condition is acceptable, facility personnel can allocate more time to investigation and disposition of suspect drums.

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A PROCESS FOR DOCUMENTING AND TRACKING STAKEHOLDERS AND ISSUES ASSOCIATED WITH THE MANAGEMENT AND TRANSPORTATION OF SPENT NUCLEAR FUEL (SNF)

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ABSTRACT

In previously published work (Waste Management '94), the authors presented an analytic model which was developed to assist in performing site-specific assessments of both physical, and regulatory and institutional information in support of decision-making with regard to developing civilian, at-reactor SNF storage facilities (Reactor-Specific Interface, Storage, Modal Operations, and Regulatory Considerations Model).

This paper describes a Stakeholders and Issues Model which was developed as an outgrowth of the authors' previous work, and which represents an extension and expansion of the regulatory and institutional data base developed as part of the prior model. The paper focuses on the identification and characterization of parties (Stakeholders) that have historically intervened, and/or have been granted the right to intervene, in judicial, administrative, or legislative proceedings relating to the at-reactor storage and transportation of civilian SNF. The paper describes a process of documentation and characterization of Stakeholders, issues, and Stakeholder positions relevant to civilian SNF management over time. We conclude that such a process of data collection and characterization may be a useful tool to support waste management issues identification and resolution activities.

THE ROLE OF STAKEHOLDERS IN NUCLEAR WASTE MANAGEMENT

Spent fuel and high-level radioactive waste (HLW) storage and transportation activities have historically engendered heated controversy, political reaction, and extensive litigation. The debate over such activities has involved several Federal and private sector shipping campaigns and related actions, and many Stakeholders, interrelated issues, and conflicting objectives over a substantial period of time. The experience of both responsible Federal agencies and private sector owners/shippers of SNF and HLW derived from such campaigns and such actions has consistently demonstrated that Stakeholders play a significant role, both directly and indirectly, in shaping public policy with regard to the management and transportation of these materials. Stakeholder activities, including exercise of applicable regulatory authority; intervention in Federal, State and local judicial, regulatory, and administrative proceedings; utilization of existing Stakeholder outreach/participation mechanisms and other public forums; and other demonstrations of interest have been, and will continue to be, an essential and intricate part of the process involved in identifying, shaping, prioritizing, and resolving issues relevant to the storage and transportation of SNF and HLW.

This important public policy role of Stakeholders was explicitly acknowledged in the drafting of the Nuclear Waste Policy Act of 1982, as amended (NWPA). In enacting NWPA, Congress recognized that spent fuel and high-level waste management and transportation had become important subjects of public concern, and that public and private sector participation in the planning and development of the nuclear waste management system was essential to promote better understanding and to foster public confidence in the safety of the program. As a consequence, provisions of NWPA went to unusual lengths to specify rights and to provide mechanisms and funding for

affected governments to oversee and participate in the program, and to provide extensive opportunities for public participation in, and comment on, the program.

DEVELOPMENT OF A MODEL TO TRACK STAKEHOLDERS AND STAKEHOLDER CONCERNS AND POSITIONS

Conflict over spent fuel and radioactive waste management activities has been extensive and often very complex. There have been a significant number of administrative and judicial "events" related to spent fuel and high-level radioactive waste transportation and other activities, and a correspondingly large number of identifiable Stakeholders and interrelated issues. We have found that the volume of such information and other factors make analysis and utilization of such information in decision-making and issues resolution processes difficult. These other factors which we have identified include:

Complexity of interests across events. Stakeholders may often have an interest in, and be active regarding, more than one relevant activity (e.g., transportation; storage; disposal) or program (e.g., commercial spent fuel management; Defense Complex site cleanup; foreign research reactor spent fuel acceptance). An identification of these various involvements may be necessary to understanding underlying or overall Stakeholders concerns.

Diversity and scope of Stakeholder representation. Coalitions of organizations are often formed to engage in relevant intervention activities. Also, we have found that the identification of national-level Stakeholder organizations with relevant interests/concerns may result from the investigation of regional/local members which they support. Such national organizations include the Union of Concerned Scientists, which has provided technical support to the administrative intervenors in proposed Calvert Cliffs (MD) and Point Beach (WI) at-reactor storage activities.

Interrelationship of issues. Issues relevant to the transportation of SNF and HLW may appear to be different from issues raised relevant to related activities (e.g. storage) or programs (e.g. Defense Complex site cleanup). However, such issues are often interrelated, as resolution of a storage issue may require or impact resolution of a transportation issue. A historical example is the suit brought by the NY Energy Research and Development Authority (May, 1982) for the return to certain utilities of spent fuel stored in West Valley, NY: this litigation triggered additional intervention by the proposed transit and receiving States.

Given the complexities involved, and the historically important role of Stakeholders in identifying, shaping, and prioritizing issues relevant to SNF and HLW management, the authors perceived a strong need for an analytic tool with which to identify and track Stakeholders and relevant issues and positions in a comprehensive and systematic manner.

The Stakeholders/Issues Data Base Model was therefore conceived and developed by the authors to assist in the identification, characterization, tracking, and correlation of Stakeholders, issues, and Stakeholder positions across relevant events and time ("relevant events" are defined as documentable judicial, administrative, and/or legislative proceedings). The authors believe that the Model can be used as a tool to aid in decision-making and in facilitating issues resolution by providing both historic and current information regarding Stakeholders and Stakeholder concerns and positions over a wide range of SNF and HLW management activities.

DESCRIPTION OF STAKEHOLDERS AND ISSUES MODEL AND COMPONENTS

The Stakeholders/Issues Model focuses identification, characterization, tracking, and correlation of Stakeholders, issues, and Stakeholder positions utilizing documented participation in particular events (defined in the preceding paragraph) relevant to the management and transportation of SNF and HLW. The Model's key components are Event Modules which record and organize, by event, data considered by the authors to be essential to Stakeholder characterization across relevant events and time.

The Event Modules consist of two sub-components of important data: 1) Stakeholder identification and characterization and 2) Issue identification and characterization. The Stakeholder identification and characterization sub-component includes important information regarding a Stakeholder's organization, size, membership/constituency, etc. (as relevant) for each Stakeholder participant in the specific event. The Issue identification and characterization sub-component identifies the specific issues raised by each participating Stakeholder in order to characterize the relationships among the identified issues (e.g., nature, frequency, constituency, etc.).

The Stakeholders/Issues Model, with its component Event Modules and sub-components

and their relationships, is depicted in Fig. 1. Event Modules are separately categorized for use in Stakeholders/Issues analyses relevant to specific types of activities. For example, Fig. 1 depicts three separate categories of Event Modules: 1) at-reactor (ISFSI) licensing and deployment events, 2) transportation campaign events, and 3) EIS-related events. A comparison of Stakeholders and of issues can be developed among these events.

The data contained in the Event Modules has been developed from several sources, including literature searches and reviews (e.g., relevant judicial, regulatory, and administrative case files and published reports regarding the specific events), supplemented by direct surveys, interviews, and correspondence with party participants in the events.

The Model can be utilized to provide outputs of information identifying, characterizing, tracking, and correlating relevant events, Stakeholders, and issues in a variety of ways. The next Section of this paper provides some examples (a "walk through") of the potential outputs of the information from the Model.

CHARACTERIZATION OF WASTE MANAGEMENT EVENTS, STAKEHOLDERS, AND ISSUES

We have mentioned that this paper focuses on relevant judicial, administrative, and legislative proceedings or "events". By way of illustration, Table I presents a description of selected, at-reactor storage (ISFSI) licensing and deployment events for which documentation of Stakeholder participation is available and for which Stakeholder participation may be characterized through the use of the Model.

The above-cited efforts to license and deploy ISFSI technologies have generated significant public interest from diverse geographic areas, economic segments, and political/ideological positions from around the country. Table II characterizes 100 participants or Stakeholders in the six selected ISFSI events (note that a Stakeholder has been counted once for each of the presented events in which it has participated). The data indicates the diversity and number of governmental and private entities which hold that their jurisdiction, geographic area, or economic segment may be directly affected by the deployment of proposed at-reactor storage technologies. As a further example of this diversity, Table III depicts the specific organizations which are included on Table II in the category of Regional, State, and Local Energy and Environmental Concerns.

As a means of further characterizing Stakeholders, the Data Base Model enables us to compare Stakeholders across events in order to gain a perspective of the relationships (if any) among the issues associated with the different aspects of SNF and HLW management (storage, transportation, disposal). The model currently focuses on storage and transportation. Table IV presents examples of Stakeholders which have historically taken positions in both at-reactor storage events and transportation campaigns, i.e., events.

One of the findings of this correlation of data is that Stakeholders can be identified which have historically played a role in different aspects of SNF management (this finding is discussed again below with particular reference to transportation issues). This appears to result from the fact that certain SNF storage facilities have generated (or have been perceived as having the potential for generating) intense transportation activities. By way of contrast, the current public debate over at-reactor storage includes the issue of "permanency", i.e., the issue of whether it is feasible to expect dry-stored SNF to be removed to a centralized storage location (MRS) or to a disposal site in the near-term.

Transportation Issues

Table IV also depicts examples of the variety of intervention activities in which State, local, and non-governmental transportation Stakeholders have historically engaged, which have included (but have not been limited to): the attempted promulgation of regulations restricting (e.g. requiring special permits) or banning (effectively re-routing) SNF shipments; the attempted imposition on the shipments of State environmental requirements (i.e., State of Wisconsin, spill prevention and clean-up plan, 1985) and of NEPA requirements (twice successful, e.g. Sierra Club in 1988 and State of Idaho in 1989); and formally requesting that the U.S. Nuclear Regulatory Commission (NRC) impose extra-regulatory transportation safety/safeguard measures on the shipments (State of Illinois, 1983). Transportation Stakeholder activities have also included convening Public Hearings and other attempts (e.g. petitioning NRC) to participate in SNF transportation decisions and activities.

The transportation issues represented by these Stakeholder activities, as interpreted by the authors with reference to the campaign Data Base, fall into six

categories which are depicted on Table V; Table V also depicts the percentage of Stakeholders which have raised an issue. As noted on the Table, the available data may be biased toward those issues for which SNF transportation Stakeholders perceive a judicial, administrative, or legislative recourse. In this regard, we note that many such Stakeholder actions have included making various petitions to NRC which have been denied, and imposing route-related transport restrictions which have been preempted by the Federal Hazardous Materials Transportation Act (HMTA).

Earlier in this paper we demonstrated that certain Stakeholders have played a role in both SNF storage and transportation events. As depicted on Table V, our analysis of transportation issues confirms the earlier-described finding: one frequently recurring transportation issue has historically been whether or not the transportation campaign has been necessary and/or desirable, as compared to continued, in-place SNF storage. This controversy appears likely to escalate: as the National debate over at-reactor vs. centralized storage continues, it can be expected to be perceived as directly impacting additional Stakeholders.

CONCLUSIONS

Stakeholders have played, and will continue to play, an important role in the identification, shaping, prioritizing, and resolution of issues relevant to the management of high-level radioactive materials. Stakeholders are both numerous and diverse, and represent multiple segments of the U.S. population.

The Stakeholders and Issues Model developed by the authors appears to be a useful tool to assist in the identification and tracking of Stakeholders, issues, and Stakeholder positions relevant to the transportation and storage of high-level radioactive materials across relevant events and over time. This capability appears to be useful to aid in decision-making and in facilitating issues resolution by providing both historic and current information regarding Stakeholders and Stakeholder concerns and positions over a wide range of high-level radioactive waste management activities.

13-2

RAIL TRANSPORTATION OF FERNALD REMEDIATION WASTE

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ABSTRACT

Remediation of the Department Of Energy (DOE) Fernald site located north of Cincinnati will generate large quantities of low-level radwaste. This volume includes approximately 1,050,000 tons of material to be removed from eight waste pits comprising Operable Unit 1 (OU-1). The remedial alternative selected includes waste material excavation, drying and transportation by rail to a burial site in the arid west for disposal.

Rail transportation was selected not only because rail transportation is safer than truck transportation, but also because of the sheer magnitude of the project and the availability of bulk rail car unloading facilities at a representative disposal site. Based upon current waste quantity estimates as presented in the Feasibility Study for OU1, a fully-loaded 47-car unit train would depart the Fernald site weekly for five years.

This paper illustrates the steps taken to obtain agency and public acceptance of the Record of Decision for the remedy which hinged on rail transportation. A preliminary, but detailed, rail transportation plan was prepared for the project to support a series of CERCLA public meetings conducted in late 1994. Some of the major issues addressed in the plan included the following:

1. Scope of project leading to selection of rail transportation
2. Waste classification

3. Rail Company overview
4. Train configuration and rail car selection
5. Routing
6. Safety
7. Prior Notification Requirements
8. Emergency Response

A series of three public meetings identified a number of issues of prime concern to Fernald stakeholders. Following resolution of these issues during the public comment period, a Record of Decision (ROD) approving implementation of the rail transportation strategy was approved pending incorporation of EPA and State of Ohio comments on December 22, 1994.

INTRODUCTION

The Fernald Environmental Management Project (FEMP), formerly the Feed Materials Production Center (FMPC), is a Department of Energy (DOE) site which produced high-quality uranium for military defense beginning in 1951. Production at the FEMP was halted in July 1989. Later that year, the facility was placed on the National Priorities List (NPL) and designated as a Superfund site under CERCLA. The Fernald Environmental Restoration Management Corporation (FERMCO) assumed cleanup responsibilities from Westinghouse Environmental Management Company of Ohio (WEMCO) in December 1992 as the DOE's first environmental restoration management contractor (ERMC). The site is now being remediated under terms of a Consent Agreement with the U.S. Environmental Protection Agency (USEPA) and a Stipulated Amendment to the Consent Decree with the State of Ohio.

Remediation of Operable Unit 1, one of the five Fernald CERCLA Operable Units, will generate over 780,000 cubic yards of material from Waste Pits 1 through 6, a clearwell and a burnpit. The waste material in these pits were generated by chemical and metallurgical processes associated with the Fernald production mission and included slags, sludges, precipitates and filter cakes. The CERCLA Feasibility Study for Operable Unit 1 identified excavation, drying and waste transportation by rail to a permitted commercial disposal facility (such as the site owned by Envirocare of Utah, Inc.) located in the arid west as the recommended leading remedial alternative.

Although Fernald has rail access and currently uses rail transportation for coal delivery, the magnitude of the project made it clear that local Fernald stakeholders would have considerable interest in all aspects of rail movement of waste. To support this anticipated level of public interest, two public meetings were scheduled prior to actual presentation of the Operable Unit One Proposed Plan in a formal public meeting as required by CERCLA. Specifically, these meetings included a rail transportation workshop on August 9, 1994, and an "Availability Session" a week later between the public and CSX railroad, the initiating rail carrier. The meetings were successful in not only communicating Fernald transportation plans but also in obtaining stakeholder input into detailed transportation plans to be developed during implementation of the project.

These meetings were themselves preceded in early 1994 by a series of interactions in the form of workshops focused on the progress of the Remedial Investigation/Feasibility Study (RI/FS) and targeted individual communications and meetings with key stakeholders. At these workshops and meetings, DOE provided a definition of the evolving scope of the cleanup and reviewed critical issues. This early work allowed DOE to ascertain many of the specific issues of concern well in advance of the transportation focused meetings held in August. More importantly, the advance work enabled DOE to identify which stakeholders required more detailed information and which issues would be their focus of concern.

THE RAIL TRANSPORTATION OPTION

The formal presentations delivered to the public beginning in August, 1994 covered the overall rail transportation plan and addressed specific concerns that had surfaced during earlier dialogue with the public in numerous forums. The presentations were outlined as follows, with the following objectives:♀

Scope of the Project

It was pointed out that over 1,000,000 tons of material would require transportation to the Envirocare site located over 1900 miles from Fernald. To illustrate the magnitude of the project, an illustration was prepared showing the anticipated volume of material in the Rose Bowl football playing field (including end zones) stacked to a height of 122 yards.

PURPOSE
The intent was to indicate that the magnitude of the project was so great that transportation by truck could not be considered a reasonable alternative to rail transportation from the perspective of local traffic and project risk management.
Waste Classification
It was pointed out that the Operable Unit 1 material was classified as Class A waste, lowest on the scale including Class A, Class B, Class C, Greater than Class C and high level waste. It was also pointed out that the material was classified as Hazard Class 7 under DOT hazardous material transportation requirements.
PURPOSE
The intent was to illustrate that the material to be transported was relatively innocuous as compared to other nuclear material routinely transported by railroad and that standard procedures existed within DOT to transport the material.
Rail Company Overview
Information was presented on safety records, quantities of material transported, revenues, etc. for both CSX and Union Pacific, the two railroads involved in the movement.
PURPOSE
The purpose was to illustrate that two of the country's largest and most reputable rail transporters would be involved in the project. These companies have adequate infrastructure and hazardous material transportation experience to handle the project.
Train Configuration and Rail Car Selection
It was stated that in order to complete the project within the conceptually proposed five-year window presented in the Feasibility Study, a 47-car unit train would leave the Fernald site every eight or nine days. In essence, three separate trains would cycle continuously between Fernald and the Envirocare site. It was emphasized that the unit train concept was adopted based on stakeholder desire to minimize risk and maximize control over waste shipments.
The project would involve almost 11,000 car loads of material and 230 train trips. One hundred ton gondola cars were presented as the rail car of choice because of their efficiency in transporting bulk material and compatibility with existing off-loading facilities. The material would be wrapped in a flexible membrane within the gondola car and the rail car itself would have a hard cover attached.
PURPOSE
The presentation of the three unit train concept was intended to make the scope of the project very clear and understandable. The numerous car control and transit time advantages of unit trains over regularly-scheduled freight service were emphasized. Gondola cars were presented as being a proven vehicle for this type of service. The flexible membrane and hard covers illustrated the intent to go beyond normal DOT transportation requirements.
Routing
Local, regional and national routing plans were presented using detailed maps for each portion of the movement. The anticipated number of Fernald shipment carloads were then compared to the number of carloads of hazardous material passing through Cincinnati annually.
PURPOSE
The intent was to pull no punches and explain exactly where trains would be passing. Each individual was able to assess the impact these movements would have on their personal circumstances. The communities where Operable Unit 1 project personnel reside were clearly labeled on the local and regional maps.
The comparison of Fernald shipments to other hazardous shipments passing through Cincinnati were intended to illustrate that the impact of Fernald shipments would be minimal and, moreover, that emergency response controls were clearly in place.
Safety
The safety statistics of transportation between rail and truck were compared.
PURPOSE
The logistic of transportation between rail and truck were compared from their perspective of community traffic impact. It was graphically illustrated that one 47-car unit train could transport waste equivalent to that contained in 219 truck shipments. Statistics, from the Association of America Railroads, were used to demonstrate that shipment of hazardous waste is safer per ton mile than by truck.
Prior Notification Requirements

Legal requirements for prior notification to communities along the rail transportation route were presented.
PURPOSE
It was pointed out that, although there were no legal requirements for prior notification, Fernald had plans to go beyond these requirements with total openness with both local stakeholders and transited states.
Emergency Response
The six levels of rail emergency response available in the event of an emergency were presented in detail including:
- train crew
- railroad emergency response organization/emergency response subcontractors
- local authorities - on scene commander
- state emergency response organizations
- DOE regional radiological assistance teams
- Fernald emergency response organization
PURPOSE
The exhaustive presentation illustrated that the infrastructure to handle rail emergencies is 1) substantial and 2) already in place.
MAJOR ISSUES
Attendance at the rail transportation workshop held on August 9, 1994 indicated that local stakeholders had an intense interest in Fernald rail transportation issues. Although lively discussion occurred in virtually every aspect of the project, three issues elicited the most vocal response at the meeting and subsequent public comments.
Use of Shandon Switchyard
The Shandon Switchyard is a small switchyard located west of Fernald adjacent to DOE property. In order to reduce the amount of new rail infrastructure to be constructed on the Fernald site for rail car staging and storage, the Shandon Switchyard was considered for this purpose. Through community involvement, discussions with rail companies and field observations by DOE and FERMCO, it was determined that upgrades would be required for the Shandon Switchyard. These upgrades would require track replacement and lighting and security installation at minimum before use of the Shandon Switchyard could be implemented.
In general, the public reaction to the use of the switchyard was negative for a number of reasons, including: 1) potential spread of contamination and 2) stakeholder fears that safeguards and controls at the offsite Shandon Switchyard location would be difficult to administer thus subjecting the yard to intrusion.
A subsequent detailed evaluation of the cost associated with the use of Shandon Switchyard for storage of either only empty, or both empty and filled cars was performed. This study compared the use of the Shandon Switchyard with on-site rail upgrades indicating that the expected cost savings associated with the use of Shandon Switchyard did not justify pursuit of this option, particularly in view of public opposition. As a result, current plans are to construct the rail upgrades necessary to accommodate rail car storage and staging within the current perimeter of the Fernald site.
Fernald-Cottage Grove Branch Line
The Fernald plant and several local businesses access national rail service via the 24 mile "Fernald Branch Line" connecting Fernald to the CSXT main line in Cottage Grove, Indiana. The branch line has received little use in recent years. As a result, repairs and upgrades to the branch line have been minimal. The branch line has five trestles and 24 road crossings. Substantial upgrades would be required to the branch line before Fernald unit train shipments could be initiated.
Local stakeholder concerns over the condition of this branch line focused on the condition of the track, the trestles (one of which is almost 900 feet long) and the absence of protected road crossings. There was much discussion over the jurisdiction of road crossings (the State DOT rather than DOE or CSX). The public was reassured that the branch line would meet all applicable Federal Railroad Administration requirements before initiating Fernald rail shipments. The public clearly indicated a desire to remain involved in the development of the rail transportation plan.
Relationship with CSX
The lack of use of the Fernald Branch Line in recent years resulted in minimal branch line maintenance. As a result, many landowners with property along the Fernald Branch line developed considerable animosity against CSX over issues such as

track conditions, drainage, shrub cutting, fencing, etc. It was clear to DOE that these concerns would have to be addressed. As a result, an "availability session" was scheduled one week after the rail transportation workshop to permit local stakeholders to meet with CSX to voice their concerns. CSX sent two representatives to represent the company at the meeting. Some frustration was vented by the public and CSX realized that they needed to re-establish themselves as a good corporate citizen and ensure that the branch line was maintained properly.

THE RESULT

On August 23, 1994, the official proposed plan for the remediation of Fernald Operable Unit One was presented to the public. It was clear that the earlier rail transportation workshop and CSX availability session did much to address the public concerns over rail transportation. Comments received by DOE during the subsequent public comment period indicated that the public supported the rail transportation option. An aggressive campaign of community involvement lead to the acceptance of this new waste transportation option, a robust program of rail shipments from the Fernald site. On December 22, 1994, the Record of Decision for the remediation of Operable Unit One was approved by the USEPA.

13-3

SUMMARY OF NEEDS ASSESSMENT FOR LONG-TERM PLANNING OF DOE'S TRANSPORTATION AND PACKAGING ACTIVITIES

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ABSTRACT

The U.S. Department of Energy (DOE) has performed a global scoping of packaging and transportation needs. The assessment will allow DOE to prepare for changes in its transportation requirements in the future. This needs assessment, which was initiated in mid-August 1994 and completed in December 1994, provides a global look at the types and quantities of materials DOE will be required to package and transport from 1995 through 2030.

The results of the assessment indicate that DOE can expect a rapid increase in the shipment of radioactive and other hazardous materials during the late 1990s and on into the first decades of the 21st century. From a waste management standpoint, a significant increase in DOE-related packaging and transportation activities is expected as extensive remediation is undertaken at various DOE sites. For example, some 1400 facilities are expected to be transferred to DOE EM for decommissioning, decontamination, or other similar actions. As this occurs, the quantities of hazardous materials destined for long-term storage and/or disposal are expected to grow significantly. DOE must be prepared to accommodate these shipping needs relative to human resources, packaging, systems interfacing, logistics, regulatory compliance, training, and operations. The results of the needs assessment are expected to be used by DOE planners and managers to guide future efforts to most cost-effectively accomplish this changing mission.

Summary results of the Transportation Needs Assessment are outlined, presenting the findings in terms of both the projected growth of shipping patterns and the

resulting identified human resource, software, and hardware needs and their associated costs that DOE needs to prepare to satisfy in the future.

INTRODUCTION

The U.S. Department of Energy (DOE) has performed a global scoping of packaging and transportation needs. The assessment, which was initiated in mid-August 1994 and completed in December, 1994, will allow DOE to prepare for changes in its transportation requirements in the future. It provides a rational basis for DOE to plan, schedule, budget, and staff its support for future transportation.

This assessment is different from previous efforts in that it focused, to the extent possible, on projected, quantified shipping needs based on forecasts of inventories of materials which will ultimately require transport by the Department. In addition, experts provided inputs to the assessment on the growing needs throughout DOE resulting from changes in regulations, in DOE's mission, and in the sociopolitical structure of the country. Through the assessment, DOE's transportation needs have been identified for a time period extending from the present through the first three decades of the 21st century.

The needs assessment was accomplished in three phases that identified and assessed the following:

- the packaging, shipping, resource utilization, and method of managing packaging and transportation activities as they currently occur within DOE;
- the inventory of materials which DOE will need to transport on into the next century, as well as scenarios which project when, from where, and to where these materials will need to be transported;
- requirements (i.e., needs) and expected changes for DOE to accomplish the necessary transport safely and economically, based upon the current situation and the projections.

This paper summarizes some of the findings from the needs assessment. Included are summary discussions of the assessment's background, scope, and data sources. The projections of quantities of commodities to be shipped by DOE over the next few decades are summarized. A brief summary of some of the projected transportation requirements is then provided. Finally, an estimate of costs associated with satisfying these requirements is discussed.

BACKGROUND

The DOE has historically shipped various forms of materials into, out of, and between its sites. Data from the Shipment Mobility/Accountability Collection (SMAC) database for 1993 were used to establish a basis for comparing current packaging and shipping activities with projections for the future.

Exclusive of the large number of shipments arising from the Uranium Mill Tailing Remedial Action (UMTRA) program, which are handled separately in this assessment, most of the shipments made by DOE in 1993 were general commodities shipped by air or by motor freight in less-than-truckload (LTL) quantities. These shipments involved various types of

- general commodities (nonradioactive, nonhazardous materials shipment): 590,516 shipments comprising 95% of the shipments made by DOE;
- radioactive materials: 18,434 shipments comprising 3% of the shipments made by DOE; and
- hazardous nonradioactive materials: 14,735 shipments comprising 2% of the shipments made by DOE.

These data show that (in addition to some 200,000 UMTRA shipments) DOE was responsible for over 620,000 shipments in 1993, with a gross weight of approximately 495,000 metric tons. Though this is a small fraction of the 500 billion total commercial shipments made each year in the United States, it is comparable to the number of shipments made by large private U.S. companies, such as Dupont or Johnson & Johnson.

The needs assessment considered current and future projected numbers of shipments by commodity category and whether they were shipped

- inbound (from outside the DOE complex to a DOE site);
- intersite (shipment between two DOE sites); or
- outbound (to a site outside the DOE complex).

In 1993, 46% of shipments were inbound, 48% were outbound, and 6% were intersite.

SCOPE OF THE ASSESSMENT

The Transportation Needs Assessment report addressed, where possible, all significant domestic DOE transportation requirements for hazardous (both radioactive

and nonradioactive) and nonhazardous materials, including selected international shipments of radioisotopes and spent fuel. This included materials shipped by various DOE programs which are not shipped using the Transportation Safeguards System. The study contrasts the current system with future needs to the extent possible.

Relative costs and benefits of meeting future needs through changes in DOE's method of doing business are assessed within needs categories only. Data and study resources were insufficient to conduct a rigorous cost-benefit analysis.

SOURCES OF DATA FOR THE ASSESSMENT

In addition to using the SMAC database to establish current shipping requirements, future shipping needs were projected using the materials inventory data in the newly developed Baseline Inventory Report (BIR) and other individual sources as required. For example, specific program contacts provided inputs on laboratory samples (radioactive, mixed or hazardous), cesium and strontium capsules, high-enriched and low-enriched uranium, plutonium, and depleted uranium.

Disposition scenarios were selected by the team using DOE program sources such as Programmatic Environmental Impact Statements (PEISs) and other "official" sources. Underlying assumptions are stated.

The results of a survey of human resources utilized by DOE transportation activities, which was conducted by DOE in November 1994, were used to define the level of labor resources currently required by the Department to accomplish its mission and to serve as a basis for future projections of human resource requirements under different operating scenarios.

PROJECTIONS OF COMMODITIES TO BE SHIPPED

The assessment considered the following categories of materials:

1. General Commodities;
2. NonRadioactive Hazardous Commodities;
3. Miscellaneous Radioactive Commodities;
4. High-level Radioactive Waste;
5. Spent Nuclear Fuel;
6. Low-level, Mixed, and Transuranic Waste;
7. Contaminated Soil and Debris; and
8. Special Case Radioactive Materials.

Details concerning projected shipping rates for categories 6 through 8 will be provided in this paper, although all categories were included in the global shipment requirements and cost estimates which are discussed.

Since the sources of data for projecting quantities of materials to be shipped in the future generally provided only inventory and generation rate data, it was necessary for the study team to develop or assume site treatment processes and disposition paths based on information derived from other sources. For example, for several special cases, projections (scenarios) were developed based on data provided by experts in each field. These special cases included analytical laboratory samples, cesium capsules, highly enriched uranium, low-enriched and natural uranium, plutonium, and depleted uranium hexafluoride. The resulting projections therefore include estimates of the time period in which these materials will be shipped and whether the shipments will be inbound, outbound, intersite, and/or intrasite; however, these projections do not infer, in any way, that decisions have been made on any Federal actions, they are simply estimates of what could happen.

For the nonradioactive hazardous materials and miscellaneous radioactive commodities (e.g., instruments, medical isotopes, and radioactive materials, limited quantity), the 1993 SMAC annual shipping quantities were assumed to remain constant over the assessment time horizon.

The projections were developed in terms of the units generally used to quantify amounts of each commodity: for example, cubic meters of contaminated soil, low-level waste (LLW), mixed waste, or transuranic (TRU) waste; metric tons of initial heavy metal (MTIHM) of spent nuclear fuel (SNF); number of items such as high-level waste (HLW) canisters or special form capsules of radionuclides; or metric tons of hazardous materials or general commodities.

Table I summarizes the specific projections developed for the commodities in categories 6 through 8. This summary includes projections of total quantities shipped, type of shipment, and time period of shipments. The basic commodity categories and key underlying assumptions are as follows:

Commodity Category 6: Low-level, Mixed, and TRU Waste These materials will be

collected, treated and consolidated, and shipped for final disposition. Uncertainties for low-level and mixed waste are the selection of treatment, storage, and disposal sites and the degree of centralization of these activities. Uncertainties for TRU waste are the opening date and acceptance rate for the Waste Isolation Pilot Plant (WIPP).

Commodity Category 7: Contaminated Soil and Debris Treatment and disposition is already under way for contaminated soil and debris at UMTRA sites and for other DOE environmental remediation activities. Uncertainties include the total volume of materials and reduction in transportation requirements resulting from in situ treatment. Much of this transportation demand will be to move these materials for relatively short distances to consolidate contaminated material for final disposal. The underlying assumptions for the projections are that only 6% and 8% of the total non-UMTRA contaminated soil and debris is transported intersite and intrasite, respectively. Amounts will vary from site to site depending upon the nature of the contamination, characteristics of the soils, and the site's management capability.

Commodity Category 8: Special Case Radioactive This broad category includes the following:

- analytical samples ranging from no or low radioactivity to highly radioactive materials from HLW tanks and other sources;
- cesium capsules which are currently being returned to Hanford;
- uranium ranging from highly enriched to natural that will likely be transported to the Oak Ridge Y-12 site for permanent storage;
- plutonium, for which numerous disposition options are currently under evaluation by DOE (an option based on mixed oxide fuel production and use in commercial reactors was selected for this study); and
- depleted UF6, which was assumed to be primarily converted to U3O8 LLW at a commercial facility and returned to a DOE site for final disposition.

PROJECTED TRANSPORTATION REQUIREMENTS

Shipments of radioactive and other hazardous materials, primarily waste from site cleanup activity, will create the primary increase in transportation demand, particularly for intersite and intrasite transportation. With two exceptions, it is anticipated that DOE will generally meet major new transportation requirements by packaging the commodities and shipping them on fully loaded legal weight trucks (LWT) or rail cars. The exceptions are as follows: 1) no-, low-, or medium-radioactivity laboratory samples, and 2) radioactive isotopes which will generally be shipped in individual packages either by motor freight in LTL quantities or by air.

During the needs assessment, once the commodity quantity projections were developed, the number of truck and rail car loads of commodities requiring shipment and the individual packaging requirements were then projected.

It was not possible to specifically address the shipment of individual packages by LTL motor freight or air carriers. These shipments are currently a significant quantity and are expected to remain relatively constant over the next few years. Table II provides an abbreviated summary of the projected shipment history, in terms of combined full truck load and rail car load shipments (per year), for three aggregated categories of radioactive materials.

The low radioactivity materials were placed into two categories: UMTRA wastes, and non-UMTRA wastes. Most projected LWT shipments are intersite and intrasite or other short-haul shipments for UMTRA and other contaminated soil. These shipping projections are based upon the assumption that DOE will only ship approximately 50% of the UMTRA wastes and will ship only about 14% of the other contaminated soils and debris. If the amounts of UMTRA or non-UMTRA materials requiring transport were to increase, there would be a commensurate increase in the number of shipments required.

In the non-UMTRA low radioactive category, the outbound transport of depleted UF6, the intersite transport of LEU and LLW, and the inbound transport of depleted U3O8 accounts for most of the projected rail shipments. These shipments are projected to grow from a current level of less than 5000 shipments in 1993 and to peak at over 73,000 per year in the early 21st century. This is more than a factor of 12 growth in the numbers of these types of shipments.

Note that the estimated number of UMTRA shipments, which are typically only for a distance of a few miles. These shipments account for more than 200,000 shipments per year in the early to mid-1990s. The number of UMTRA shipments is projected to

decline to one-third that value in about 2010.
The number of truck/rail car loads of high/medium-radioactive commodities is projected to grow from 21 in 1993 to over 1,500 per year by the beginning of the next century. In the near-term (next 5 years), high-radioactivity laboratory samples, and SNF from naval, DOE, and university reactors are the primary sources of this projected growth. The LWT transport of TRU waste to WIPP in 2000 is projected to require 1000 truck shipments per year until 2020. From 2010 on, rail transport of HLW canisters and commercial reactor SNF to a repository is projected to be an increasing source of transportation demand, especially when compared to the single rail car shipment of HLW waste which occurred in 1993. Sources of other, smaller shipments are cesium capsules and highly enriched uranium. This projected annual volume of high- and medium-radioactive material shipments represents a major change which DOE needs to be prepared to address.

PROJECTED TRAVEL DISTANCE

The travel distance for all full rail car and truck load shipments was projected to increase in a fashion similar to the increases shown for number of shipments. It was projected that the total travel distance will rise from 10 million miles per year (excluding UMTRA) in 1993 to over 37 million in the early 21st century. The estimate that over half of the new shipments will be intrasite limits the increases in travel mileage. Should more materials move off of DOE sites, these estimates will be low.

PROJECTED HUMAN RESOURCE NEEDS

It was quickly determined that one of the major costs associated with DOE's packaging and transportation activities is the cost of human resources. Part of the needs assessment therefore focused on defining the current and project human resource requirements and the costs associated with them.
Following a brief review of some of the DOE's field offices and the contractors associated with the field offices, it was estimated that approximately 3200 full-time equivalent (FTE) staff are currently required to operate the current DOE transportation system. This number consists of approximately 800 managerial and administrative personnel and an estimated 2400 FTE hourly personnel performing shipping, receiving, and local transportation functions. The assessment team estimated that three times as many additional hourly personnel perform other DOE transportation activities, and these figures were the basis for projecting human resource needs into the future. In turn, these estimated resources were then used to estimate costs.

PROJECTED OTHER NEEDS

Other needs were quantified in the packaging and carrier area, including the need for over 1.7 million 55-gallon steel drums per year for packaging low-level, mixed, and TRU waste for shipping in the early 21st century. Also, there will be a need for over 150 reusable shipping casks for spent nuclear fuel, canisters of high-level waste, and TRU shipments.
Added transportation requirements for various radioactive materials and wastes will require the full-time, continuous availability of the equivalent of at least 750 rail cars and truck trailers.

PROJECTED COSTS AND COST SAVINGS

The data which have been only briefly summarized here were combined with packaging and infrastructure data to provide a "top-level," rough order of magnitude (ROM) projection of costs for accomplishing DOE's shipping mission. All cost figures were provided in FY 1995 constant dollars.

"Base Case" System Costs

Through 2030, it was estimated that under the current method of performing packaging and transportation activities, DOE will spend on the order of $30 billion to meet its projected shipping needs if it continues to operate in its current fashion. These costs include all identified human resource, packaging, carrier, and vehicle and site maintenance infrastructure costs. Table III summarizes these estimates for 1993 with projections at 10-year intervals starting in 2000. As noted above, the costs are only rough approximations because of the "broad brush" nature of this assessment and limited data availability. In particular, the DOE does not currently collect the cost and activity data needed to precisely quantify its costs or project its transportation resource needs.
It is felt that these projections establish a starting point for defining future funding needs and for assessing potential areas for significant cost savings. Costs estimates shown are those associated with the actual packaging of materials for

shipment and the performance of the shipment activities. These costs are borne by the individual DOE sites and programs.

Improving Transportation System Costs

The ROM cost analysis was used to identify potential areas for cost reductions and to approximately quantify such cost reductions. The cost analysis for an improved system often used the best judgement of experienced personnel to establish potential cost savings that would arise in each area associated with an integrated systems approach. An integrated systems approach to transportation resource management was assumed for estimating these cost savings. In this assumed approach the primary emphasis was on developing tools and methodologies and implementing them within the DOE complex in a manner such that:

personnel could become more efficient in performing their tasks (improved productivity);

greater efficiencies would result from aggregating shipments, improved utilization of carriers and improved carrier-service procurements;

packaging costs would be reduced through consolidating procurements and standardizing package designs across the DOE complex;

better tariffs would be available through negotiations and by taking advantage of multiple-site shipments; and

more efficient use would be made of equipment, facility, and other hardware-related resources.

It was estimated that if these steps were taken, an improved system could result which would have the potential for significant cost savings as illustrated in Table IV. Such steps should not only reduce costs, but might also enhance effectiveness and safety. However, it must be stressed that an integrated approach to transportation and the assumed development of the needed tools and methodologies and their implementation in the field, or a similar approach to enhanced efficiency, must be realized if these cost savings are to be achieved.

Potential Cost Savings

The data presented in Tables III and IV show specifically that, based upon the ROM cost estimates performed, the DOE's packaging and transportation costs might be reduced by about $7 billion through 2030. The cost reductions were assumed to be achieved through the application of an integrated systems approach to transportation resource management. The analyses summarized in these two tables illustrate that human resource costs may be expected to drive overall DOE transportation costs over the next few decades. This is illustrated in Table V, which shows that human resource costs account for just under $20 billion (65%) of the estimated $30 billion in total transportation costs for the base case. Of the $7 billion savings in the improved system case, $5 billion is due to more efficient utilization of human resources. These savings result from elimination of redundant functions and continuous upgrade of various tools and staff training.

CONCLUSION

Based upon the Transportation Needs Assessment, which has been briefly summarized in this paper, DOE is facing orders of magnitude increases in shipping rates of some radioactive and other hazardous materials by the 21st century. Total transportation costs over the period from 1993 to 2030 may reach $30 billion. However, there are a number of measures offering the potential for significant cost reductions. It was estimated as much as $7 billion might be saved over that time period. This estimated cost reduction is based upon a ROM assessment, assuming that an integrated systems approach to transportation resource management is implemented. This approach emphasizes developing tools and methodologies and implementing them within the DOE complex to enhance the field's capabilities and efficiencies. It was noted in the study that achieving these savings will require DOE to make near-term investments to accomplish these goals.

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DEVELOPMENT OF AN OPTIMIZED PACKAGE FOR LOW AND MEDIUM LEVEL RADIOACTIVE WASTE

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ABSTRACT

The paper reports on the development of an optimised package for the final repository KONRAD in Germany. The basic idea of this concept is to produce a container by using radioactively contaminated steel scrap coming from dismantling.

In a second step, highly radioactive waste is inserted into this container and finally, in a third step, residual space between the scrap and the container walls is filled by pouring in radioactively contaminated liquid steel.
This method optimizes the package volume and saves space in the final repository.
The project is sponsored by the Commission of the European Communities.

THE MELTING PROCESS FOR RADIOACTIVELY CONTAMINATED STEEL SCRAP AT CARLA PLANT IN GERMANY

Melting of radioactively contaminated steel scrap is commonly known as a practicable, ecological and economic way for treating metallic material coming from the dismantling of nuclear power plants and installations.
This process, resulting from parallel developments in Great Britain and Germany in the early eighties, is practiced today in Sweden, France, the United States and Japan as well. Further to the melting process, a method for the re-use of this steel for the production of components to be used within nuclear facilities was developed in Germany. Especially the production of packages for transport and storage of radioactive material led to satisfying solutions. A complete survey about the melting process, regarding all items from licensing to re-use, from legislation to melt materials, is given in the Report of the European Commission on the "Technical seminar on melting and recycling of metallic waste materials from decommissioning of nuclear installations" (5).
Thus, several thousands of tons of steel scrap have been recycled, especially in the production of casks for radioactive waste. While the ecological concept was convincing from the early beginning, it was the economic aspect which brought the melting process to the leading position in radioactive metal waste treatment.

PACKAGING CONCEPT IN A MONOLITH CONTAINER

As mentioned above, during the past few years, the production of waste packages re-using radioactive material became the preferable way - not only in Germany. It was the task of the companies GNS (Gesellschaft fr Nuklear Service mbH) and SIEMPELKAMP FOUNDRY to optimize the material quantity to be brought to the final repository, with special regard to the costs of this disposal.
Figure 1 shows the concept of the Monolith container, it illustrates the procedure in detail, as being performed in Germany. The package for final repository is produced in three steps:

Production of the Container for the Final Repository

Coming from the nuclear power plant, steel scrap with a specific activity below 200 Bq/g is melted and a package is produced suitable for transport and storage of radioactive material. The relevant demands will be discussed later. In our special project, the dimensions are as follows:

length	2,000	mm
width	1,600	mm
height	850	mm
wall thickness	150	mm
total weight	10.8	tons

Filling the Container

The monolith container produced by using low level radioactive material is transported to the nuclear power plant, where medium- or high-level radioactive material is loaded into the container. The lid is put on top of the container which is qualified as an IP2-container.

Producing the Monolith

The container is then brought back to the foundry, where radioactive steel is melted. Following the existing license for the melting shop, the specific activity may not exceed 200 Bq/g. Through special openings in the container lid, the residual space between the high-active waste components in the container is filled with liquid steel, which again is in the level of low radioactive material. The container is placed right in front of the furnace and liquid steel is poured through the openings in the lid, filling the entire container to form a solid iron block.

Going to the Final Repository

The cask is now an optimum size for final disposal; the outer geometry is suitable as demanded and no empty space wastes the volume of the final repository.

REGULATIONS TO BE OBSERVED IN GERMANY

Two types of regulations influence the design of this package:
- demands on the physical integrity of the package
- limits for activity content.

Figure 2 gives a survey of the different demands on the integrity of the package. Basically, for the transport, the IAEA Safety Series are the regulations which are in correspondence with the German GGVS (Ordinance on Transport of Dangerous Goods). For the three transports (from foundry to nuclear power plant, back to the foundry and then to the final repository or intermediate storage) industrial package Type II was chosen as package design.
For the production of the package itself, radioactive material was used. Thus the transport from the foundry to the nuclear power plants has to be done under observance of the regulations, which means that a drop test from a height of 30 cm has to be carried out as well as a water spray test and a drop of a bar of 1 kg in weight from a height of 1 m onto the package.
There are no additional demands on the integrity of the package coming from the intermediate storage Gorleben. In addition to the IAEA Safety Series, the conditions for final repository KONRAD are higher, even for Type IP II packages. KONRAD repository defines two categories (Classes I and II) with lower demands put on Class I containers. In this case, a drop test from a height of 80 cm without losing the integrity must be fulfilled. In addition, in a fire test of one hours' duration at a temperature of 800C, a loss of activity below 1 Mol of the contents of the package is allowed at maximum (see Fig. 3).
The second point of interest are the activity levels to be filled into the package following the different demands of the regulations.
According to IAEA Safety Series, the maximum allowed contamination of the outside surface of the package is 10 Bq/cm. This demand is easily fulfilled by cleaning the package before transport.
The second demand, from German GGVS conditions, is that the maximum allowed surface contamination of the material being packed into the package must be below 8 x 105 Bq/cm for b- and g-emitters and below 8 x 104 Bq/cm for a-emitters. The specific activity to be filled into the package must be below 2 x 10-3 A2/g. As to be shown later, this leads to a maximum allowed activity concentration of 8 x 108 Bq Co60 per package and approx. 106 Bq Fe55. Experience shows that these two nuclides are the most important and the only ones to be considered.
For the intermediate storage Gorleben, only the Co60-content limits the maximum allowed activity concentration in the package. In this particular case, the maximum allowed total activity is 9.25 x 1011 Bq Co60.
Finally, KONRAD conditions show the highest allowed activity level which is at 5.5 x 1012 Bq Co60. The limits for Fe55 are several orders of magnitude higher and lead to no restrictions.
In addition to that, the dose rate at the surface of the package must be below 2 mSv/h in average and 10 mSv/h as maximum value. The maximum allowed dose rate at a distance of 2 m must be below 0.1 mSv/h.
One final limiting condition for the packages is given by its maximum manageable weight, which in the final repository KONRAD is 20 tons. This means that the weight of the package plus the material to be inserted and the material to fill the space between inserted material may amount to a maximum weight of 20 tons.
The table below gives an idea of the different masses the final monolith container consists of. The weight of the package with a wall thickness of 150 mm is 10,850 kg. For the production of this cask, radioactive material in a range of 40% is used which leads to a sink of 4,340 kg radioactive material.

- Weight of package 10,850 kg
- incl. radioactive material (</= 200 Bq/g) used for production (40 %) 4,340 kg
- Material to be filled in (HAW) (20 % filling factor) 1,840 kg
- Material to be poured to fill residual space (</= 200 BQ/g) 7,310 kg

Total allowed mass per package 20,000 kg
The container with a volume of 1.18 m can take up 1,840 kg of high-active waste, representing a filling factor of 20%. This figure is the result of several tests carried out before.
80% of the volume of 1.18 m are residual space between the radioactive waste components and can be filled up with liquid steel in the foundry. This leads to 7,310 kg.
As you can see, the design of the package was chosen to come to a weight of 20 t in

total, which is acceptable for the final repository KONRAD mine.

COSTS FOR DIFFERENT WAYS OF DISPOSAL

Besides all ecological considerations, the economic factor actually defines the way to final storage. The costs of three different possibilities of final storage for radioactively contaminated steel scrap have been compared:

-Conventional package

In this case, material is packed into 200-l-drums, which are in turn loaded into a final storage container.This package is brought to the final repository.

-Supercompaction

The radioactive material is supercompacted; the pellets are put into final storage containers and then taken to the final repository.

-Monolith container

The radioactively contaminated metal scrap is treated as described above and the solid block is taken to final repository.

Figure 4 shows a cost comparison between the three different methods. Basis for the cost estimates are experiences gained at SIEMPELKAMP foundry, further cost factors were taken from the report of Teunckens et.al. "Economic Aspect of Melting and/or Recycling of Waste Metals from Decommissioning" (5, p. 309).

The basis for this cost comparison was the total weight of a monolith container, i.e. 20 tons. To come to comparable quantities of metal scrap to be treated, this quantity was reduced by the non-contaminated material used for the production of the final storage container.

Nevertheless, the volume of 2.5 cm to be stored in the final repository was maintained.

The density of unconditioned scrap being filled loosely into 200-l-drums is approx. 20%, a representative figure derived from the treatment of several thousands of tons in Germany.

For supercompaction, an average density of 3.5 tons/cm should be an optimistic value for this disposal method.

After packing the unconditioned metal scrap into 200-l-drums, these have to be packed into a final repository container, following the conditions of the German KONRAD mine. In this case, approx. 60% of the container volume are not being used.

It was assumed that - before being taken to the final repository - the waste material will be kept in an intermediate storage for 5 years. This leads to additional, considerable costs.

The costs for final repository in Germany today were calculated with a figure of 10,000 $/cm. This is the actual fee for final repository Morsleben, which is presently in operation, and it is made up of 8,500 $ repository costs and 1,500 $ documentation. The same figure is presently being discussed for the final repository KONRAD, but this price will be subject to development until KONRAD will open its doors.

The comparison shows that - including the costs for final repository - the Monolith container has an advantage of approx. 50 % versus unconditioned packaging and an advantage of approx. 30 % compared to supercompaction. When final repository costs are not considered, the advantage goes down to 15 % compared to untreated scrap and remains at approx. 25 % compared with supercompaction.

In any case, regarding the safety aspects, it is of advantage to have a solid block going to the final repository.

STATE OF THE PROJECT AND FURTHER PROCEDURE

This project is carried out by the project partners KRB Gundremmingen, EWN Greifswald, GNS Essen and SIEMPELKAMP Krefeld. A delay in the project was caused by the fact that this container has to pass a license procedure at the German authorities BfS (Bundesamt fr Strahlenschutz = Federal Agency for Radiation Protection) and BAM (Bundesamt fr Materialforschung und -prfung = Federal Agency for Materials Research and Testing).

The delay of one year is particularly due to the fact that the dimension of the Monolith container is not proposed in the preliminary final storage conditions as they exist right now. Approval has recently been given to continue with the development of this container, the license is expected to be granted in early summer this year.

In parallel to this licensing procedure, one prototype container was cast and tested with non-radioactive material.

A hot test container has been produced and is on its way from the foundry to KRB

Gundremmingen. In Gundremmingen, the container will be charged with high active waste coming from the reactor vessel of KRB plant. The container will be filled and tested. As soon as the results show that it works as demanded, four other containers will be produced and filled with the rest of the reactor pressure vessel - the highly active part in the central area.
It is envisaged to conclude the project by mid-1996.
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13-5
TRANSPORTATION RADIOLOGICAL RISK ASSESSMENT FOR THE FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL ENVIRONMENTAL IMPACT STATEMENT: AN OVERVIEW OF METHODOLOGIES, ASSUMPTIONS, AND MODELING APPROACHES
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ABSTRACT
The U.S. Department of Energy (DOE) is considering a policy to manage spent nuclear fuel (SNF) from certain foreign research reactors (FRRs). The policy, described fully in the Implementation Plan(1) would involve shipment of up to approximately 840 casks from locations overseas and in Canada, via truck or rail, to one or more DOE facilities, and could involve the shipment of SNF between DOE sites. This paper provides an overview of the transportation risk analysis performed for the Environmental Impact Statement (EIS) associated with this policy. Alternatives that involve the shipments of lesser amounts of fuel, and/or shipment of other materials (i.e., target material, vitrified waste) are also being analyzed.
The interim storage site or sites for the fuel will be determined under the Programmatic SNF Management and Idaho National Engineering Laboratory (INEL) Environmental Restoration and Waste Management Programs EIS (known as the SNF&INEL EIS), which will identify a preferred alternative from effectively 12 possible alternatives (1 of 5 DOE sites or 1 of 7 different site pairs) as the ultimate interim storage site. However, eight of the SNF&INEL EIS alternatives involve sites not immediately ready to receive fuel at the onset of this policy. Therefore, for each of these alternatives, a two-phased approach involving the use of sites currently capable of receiving SNF [the Savannah River Site (SRS) and INEL] is analyzed. Analysis has been performed for splitting the fuel between these two sites at least four different ways and for varying numbers of inter-site shipments. Additional complexity is added by the current approach of using about 10 different seaports.
The collective population risk, maximally exposed individual (MEI) risk, accident

risk, and nonradiological risk assessments were performed using established methodologies and codes. The risk associated with the policy varies by more than an order of magnitude, depending on which alternative is chosen under the SNF&INEL EIS, which site(s) are selected for short-term storage under this EIS, and whether truck or rail shipments are used.

INTRODUCTION

A draft EIS showing the potential environmental effects resulting from adoption of a proposed policy for management of SNF from FRRs is being prepared. The purpose of the proposed policy is to support nuclear nonproliferation objectives of the United States. Only spent fuel which contains uranium enriched in the United States would be accepted under this policy.

Environmental and human health effects of alternatives are analyzed. If the proposed policy is adopted and implemented, then spent fuel generated in FRRs could be imported to the United States and either: 1) be transported by ship to seaports in the United States, unloaded, and then transported by truck or rail to an interim storage site; or 2) arrive at the Canadian side of the United States-Canada border in an approved conveyance (truck or rail), and then be transported in that conveyance to an interim storage site.

The unique difficulty in preparing this analysis is that the scope of the policy involves the shipping of numerous types of SNF in at least 8 different casks from any of 10 or more ports to any of 5 DOE sites. This results in a potential for thousands of risk parameters for origin-destination pairs. For the purposes of analysis, the material was binned into representative groups, and the number of shipments of material corresponding to each group was estimated for each alternative.

This paper provides an overview of the approach that will be used in the FRR SNF EIS to assess the risks from the transportation of fuel. In additions, the most important assumptions and input parameters are discussed for analysis. A number of simplifying assumptions were necessary to make this analysis manageable and useful to the readers. Wherever possible, the methodology used is identical to that used in the SNF&INEL EIS(2) and described by Monette et al.(3).

APPROACH

The policy would involve transporting FRR SNF from the ports of entry (both marine ports and Canadian border crossings) to DOE sites, and could involve transporting FRR SNF between DOE sites. The proposed acceptance of FRR SNF is subject to the selection of a management site made in the SNF&INEL EIS(2). Regarding FRR SNF transportation, there are 12 distinct decisions that could be made from the SNF&INEL EIS(2). This decision is expected to identify the use of one or more of five candidate sites and seven distinct combinations of sites.

Certain assumptions are required in order to simply and consistently describe the manner in which FRR SNF is transported to the sites. First, the shipments were divided into East Coast and West Coast shipments, depending on the country of origin. SNF shipments from Europe, Africa, the Middle East, and parts of South and Central America were designated as East Coast shipments, and all others were designated as West Coast shipments. Shipments from Canada were assumed to enter the United States from either an eastern or western point of entry, depending on the Canadian point of origin. Under these assumptions, if all SNF identified is returned to the United States, the East Coast would receive approximately five-eighths of the cask shipments and the West Coast would receive approximately one-quarter of the cask shipments. Approximately one-eighth of the shipments would come from Canada into the eastern United States.

Eight of the alternatives involve sites that could not be ready to accept SNF at the onset of the FRR SNF program. Therefore, a two-phased approach is assumed using one or both of the sites that are ready to accept SNF (SRS and INEL) as a near-term storage location. Phase 1 is defined, for the purposes of analyzing transportation, as the period of time in which shipments of FRR SNF are transported to a near-term storage site. For analytical purposes, Phase 1 is assumed to last from the beginning of 1996 to the beginning of 2006.

The amount of fuel that would arrive in Phase 1 and Phase 2 cannot be precisely determined at this time. In order to proceed with the risk analysis, it is necessary to make assumptions based on the available information. The split between Phase 1 and Phase 2 depends on the rate at which casks are received and the time the Phase 2 site(s) is ready to receive fuel. For calculational purposes, the casks are assumed

to arrive at a uniform rate.
Since the SNF&INEL EIS(2) does not select a specific site or sites, the disposition of FRR SNF during Phase 1 is analyzed in this EIS. Logically, Phase 1 could entail anyone of four options: 1) splitting FRR SNF by fuel type (TRIGA, which stands for Training, Research, and Isotope reactors built by General Atomic, to INEL and aluminum-based to SRS), 2) splitting the SNF geographically by port of entry, 3) transporting all SNF to INEL, or 4) transporting all SNF to SRS. Not all Phase 1 strategies are consistent with all Phase 2 strategies.
Phase 2 would begin when the Oak Ridge Reservation (ORR), Hanford Site (HS), or Nevada Test Site (NTS) could be ready to receive fuel from ports and, when applicable, from a DOE site being used for near-term storage. In all cases, Phase 2 would be dependent on the decision of the SNF&INEL EIS(2) alternative. During Phase 2, all FRR SNF arriving at ports of entry would be transported to the appropriate site. Additionally, intersite shipments from the near-term storage site could also be arriving at the SNF&INEL EIS selected site(s).
In order to maximize the calculated impacts of accidents, a highly conservative approach was used to estimate the radionuclide content of the casks. The FRR SNFs were grouped into three classes and four fuel categories for the determination of bounding radionuclide inventories. This division was created to provide a conservative representation of radionuclide inventories and to provide a means for identifying the type of transportation casks to be used for estimating the number of shipments.
The selected fuel types for the determination of bounding radionuclide inventories are:
1. Special: aluminum-based fuels that do not fit the TRIGA or Materials Test Reactor (MTR) description.
    1a. Single-Element Reactors: SNF from research reactors that operate with one element (e.g., RHF of France).
    1b. NRU-Type SNF: SNF from Canadian Research Laboratories' research reactors (e.g., NRU and NRX) and from South Korean research reactors.
2. MTR SNF.
3. TRIGA SNF.
The ORIGEN2(4) code was used to calculate the radionuclide inventory in each SNF type, based on fuel burn-up, initial inventory, and minimum decay time. It was determined that, for calculational purposes, the use of IU-04 (Pegase) transportation cask maximizes the radioactive inventory and requires the shortest cooling period (a maximum of 1 year). All shipments were assumed to fit to one of the above selected fuel types at it's maximum radionuclide inventory for calculations of overland transportation accident risks.
It is important to note that the radionuclide inventories identified here are for calculation purposes only. In reality, the majority of the SNFs would have much lower radionuclide inventories than the bounding inventories listed here. The final assumption, and the assumption most important to risk, is that the casks will have an external dose rate that equals the regulatory limit for transportation vehicles. Calculations show that this dose rate limit (10 millirem per hour at any point 2 meters from the vehicle, per 49 CFR 173.441) could theoretically be achieved in a few casks; but in general, most casks will exhibit much lower radiation fields.
Incident-Free Risk Assessment Methodology
For each representative route, the incident-free risk was estimated for the population along that route. Incident-free analysis included both the affects of the radiation field associated with the cask and the emissions associated with the truck or rail vehicle used. Additionally, the hypothetical MEI was determined for each implementation approach.
Collective Population Risk: The RADTRAN 4(4) computer code was used to consider all major groups of potentially exposed persons. The RADTRAN 4 risk calculations for incident-free highway and rail transportation include exposures of the following population groups:
  Persons along the route (off-link population): Collective doses are calculated for all persons living or working within 800 m (0.5 mi) on each side of a transportation route. The total number of persons within the 1.6 km (1 mi) corridor is calculated separately for each route considered in the assessment.
  Persons sharing the route (on-link population): Collective doses are calculated for persons in all vehicles sharing the transportation route. This group would

include persons traveling in the same or opposite direction as the shipment, as well as persons in vehicles passing the shipment.

Persons at stops: Collective doses are calculated for people who may be exposed while a shipment is stopped en route. For truck transportation, this would include refueling stops, food stops, and rest stops. For rail transportation, stops are assumed to occur for classification purposes.

Crew Members: Collective doses are calculated for truck and rail transportation crew members.

The doses calculated for the first three population groups are added together to yield the collective dose to the general public. The dose calculated for the fourth group represents the collective dose to workers. The RADTRAN 4 incident-free dose models are not intended to be used for estimating specific risks to individuals. The RADTRAN 4 incident-free dose calculations are based on expressing the dose rate as a function of distance from a point source(5). Associated with the calculation of incident-free doses for each exposed population group are parameters such as the radiation field strength, source-receptor distance, exposure time, vehicle speed, stop time, traffic density, and route characteristics such as population density. The RADTRAN 4 code user's manual contains derivations of the equations and descriptions of these parameters(5).

The collective incident-free risks are calculated for each specific alternative as follows. Each alternative is first defined as a set of origin and destination pairs. Representative highway and rail routes are determined for each unique pair of DOE sites and ports. For each pair, RADTRAN 4 is used to calculate the collective risks to workers and the public for a single shipment based on representative radiological and physical properties of the SNF. These estimates for a single shipment are referred to as per-shipment risk factors. The number of shipments transported across each linkage is then determined for both truck and rail modes. The collective risks for an alternative are calculated by multiplying the number of shipments by the appropriate per-shipment risk factor.

MEI Risk: In addition to the incident-free collective population risk assessment, the risk to MEIs has been estimated for a number of hypothetical exposure events using RISKIND(6). The receptors include transportation crew members, inspectors, and members of the public exposed during traffic delays, while working at a service station, or living near a port of entry or DOE site.

The dose to each MEI considered is calculated with RISKIND for a given distance, duration, and frequency of exposure specific to that receptor. The exposure scenarios are not meant to be exhaustive, but were selected to provide a realistic range of potential exposure situations.

The RISKIND external dose model considers direct external exposure and exposure from radiation scattered from the ground and air. RISKIND is used to calculate the dose as a function of distance (mrem/hr for stationary exposures and mrem/event for moving shipments) from an SNF shipment based on the dimensions of the shipment. The code models the shipment as a cylindrical volume source, and the calculated dose includes contributions from buildup, cloudshine, and groundshine. As a conservative measure, potential shielding between the cask and the receptor is not considered.

Nonradiological Risk (Vehicle-Related): Vehicle-related health risks resulting from incident-free transport may be associated with the generation of air pollutants by transport vehicles during SNF shipment, and are independent of the radioactive nature of the shipment. The health end point assessed under incident-free transport conditions is the excess latent mortality due to inhalation of vehicle exhaust emissions. Risk factors for pollutant inhalation in terms of latent mortality have been generated(7). These risks are 1x10-7 mortality/km (1.6x10-7/mi) and 1.3x10-7 mortality/km (2.1x10-7/mi) of truck and rail travel in urban areas, respectively. The risk factors are based on regression analyses of the effects of sulfur dioxide and particulate releases from diesel exhaust on mortality rates.

Accident Assessment Methodology

To provide an assessment of SNF transportation accident impacts, two types of analyses were performed. First, an accident risk assessment was performed that takes into account the probabilities and consequences of a spectrum of accident severities using methodology developed by the NRC(8). The accident risk assessment used route-specific information for accident rates and population densities. For the spectrum of accidents considered in the analysis, accident consequences in terms of collective dose to the population within 80 km (50 mi) were multiplied by the

accident probabilities to yield dose risk. Second, to represent the maximum reasonably foreseeable impacts to individuals and populations should an accident occur, radiological consequences were calculated for an accident of maximum credible severity in each population zone.

Accident Risk Assessment: The accident risk assessment is treated probabilistically in RADTRAN 4. Accident risk is defined as the product of the accident consequence (dose) and the probability of the accident occurring. In this respect, the RADTRAN 4 code estimates the collective accident risk to populations by considering a spectrum of transportation accidents.

The RADTRAN 4 calculation of collective accident risk employs models that quantify the range of potential accident severities and the responses of transport packages (i.e., casks) to accident environments. The accident severity spectrum is divided into the same accident severity categories as were used in the SNF&INEL EIS(2). The models take into account the transportation mode and the type of packaging being considered. The accident rates, definition of accident severity categories, and release fractions used in this analysis are discussed further in a previous section.

For accidents involving the release of radioactive material, RADTRAN 4 assumes the material is dispersed in the environment according to standard Gaussian diffusion models. For the risk assessment, default atmospheric dispersion data were used representing an instantaneous ground-level release and a small diameter source cloud(5). The calculation of collective population dose following the release and dispersal of radioactive material includes the following exposure pathways:

- external exposure to the passing radioactive cloud,
- external exposure to contaminated ground,
- internal exposure from inhalation of airborne contaminants, and
- internal exposure from the ingestion of contaminated food.

For the ingestion pathway, state-specific food transfer factors, which relate the amount of radioactive material ingested by people to the amount deposited on the ground, were derived in accordance with the methods described by NRC Guide 1.109(9). Radiation doses are calculated using standard dose conversion factors in DOE EH-0070(10) and DOE EH-0071(11).

The collective accident risk for each alternative is determined in a manner similar to that described for incident-free collective risks. Accident risks are first calculated for each unique origin and destination pair ("per-shipment" risk factors) and then summed over all pairs to estimate the total risk for the alternative. The accident risk assessment uses site- and SNF-type-specific radiological and physical characteristics. In addition, the assessment uses route-specific population density information and accident rates derived for individual States.

Accident Consequence Assessment: The RISKIND code is used to provide a detailed assessment of the consequences of the most severe transportation accidents. The accident consequence assessment is intended to provide an estimate of the maximum potential impact posed by a severe transportation accident involving SNF.

RISKIND was used for the accident consequence assessment for two reasons. First, the code has the ability to model the complex atmospheric dispersion present in severe accident environments. The atmospheric dispersion is modeled as an instantaneous release using standard Gaussian puff methods. In addition, because severe accidents routinely involve fires, modeling of the potential radiological consequences takes into account physical phenomena resulting from the fire, such as buoyant plume rise. Second, RISKIND can be used to estimate the dose to MEIs in the vicinity of an accident. The location of the MEI is determined by RISKIND based on the atmospheric conditions assumed at the time of the accident and thermal characteristics of the release.

The consequences of the most severe accidents are calculated for both local populations and MEIs. The population dose includes the population within 80 km (50 mi) of the accident site. The exposure pathways considered are similar to those discussed above for the accident risk assessment. Although post-accident remedial activities (e.g., immediate evacuation of the public or cleanup of dispersed radioactive material) would reduce the consequences of an accident, these activities were not given credit in the dose calculations.

Because it is impossible to predict the exact location of a severe transportation accident, separate accident consequences are calculated for accidents occurring in rural, suburban, and urban population density zones. Moreover, to address the effects of the atmospheric conditions existing at the time of an accident, two

different atmospheric conditions are considered. The first case assumes neutral atmospheric conditions, and the second, stable conditions.
Release Fractions Used in Accident Assessment: Radiological consequences are calculated by assigning cask release fractions to each accident severity category. The release fraction is defined as the fraction of the radioactive material in a cask that could be released from the package in a given severity of accident. Release fractions take into account all mechanisms necessary to create a release of radioactive material from a damaged cask to the environment. Release fractions vary according to the SNF type and the physical and chemical characteristics of specific radionuclides within the SNF. For instance, most solid radionuclides are difficult to release in particulate form and are therefore relatively non-dispersible. Conversely, gaseous radionuclides are relatively easy to release in the likely event that the cask and SNF elements are compromised in an accident.
Two sets of release fractions were used in the assessment depending on the SNF type, consistent with the SNF&INEL EIS(2). Release fractions developed for MTR SNF were used for aluminum-clad fuels including BR-2, RHF, and NRU SNF; release fractions for TRIGA were used for the PRR-1 SNF.
For high-level waste (HLW) shipments, the modal study results are not applicable because of differences between SNF and HLW. Therefore, more conservative release fractions from NUREG 0170 are used for HLW accident analysis. Similarly, the NUREG 0170 scheme also is applicable for shipments of target material.
The values indicate that in the most severe accidents, 100 percent of the material is released from the cask -- a highly conservative assumption for most solid waste forms and somewhat conservative for a powder or cake-like material. The accident assessment also utilizes the fraction of the release that is aerosolized and the fraction of the aerosol that is respirable. The values for HLW and target material (assumed to behave as a loose powdered material) were taken from the recommendations provided in RADTRAN 4. These values are shown in Table I.
Therefore, the maximum total respirable release fraction for the most severe accidents is 5x10-8 for HLW shipments, and 0.005 for shipments of target material. The values shown above have been used in the accident calculations for shipments of target material and vitrified material for the FRR SNF EIS.
Nonradiological Accident Risk Assessment: The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities not related to the shipment cargo. This risk represents fatalities from mechanical causes. Nonradiological accident risks are calculated for each alternative by multiplying the total distance traveled in each State by the appropriate State fatality rate. In all cases, the nonradiological accident risks are calculated using round-trip shipment distances.
Modeling Approach
Several of the SNF&INEL EIS(2) alternatives involve consolidation of all SNF to INEL and/or the SRS and, therefore, are single-phase programs that would require no additional shipments. However, many of the possible options require the use of HS, NTS, and/or ORR, and thus would require intersite shipments. The number of intersite shipments is calculated based on the assumption that the equivalent of 10 seagoing FRR casks will fit into a single rail cask that would travel between DOE sites. Similarly, it is assumed that the contents of four FRR casks would fit into a single truck cask for intersite shipment. This assumption is based on a review of the capacities of candidate casks and of alternatives for managing the fuel during Phase 1. These assumptions, which are neither definitely conservative nor non-conservative, are considered to be reasonable and realistic. The number of intersite shipments for SNF&INEL EIS(2) alternatives that would require two-phased approaches varies between none and almost 200. The variation is caused by the large number of unique combinations of Phase 1 and Phase 2 approaches depending on the specific management sites selected. Additionally, the variation is affected by the assumption that larger truck and rail casks would be used for intersite shipments. Since the fuel may actually arrive at a variety of ports, average shipment risk factors were calculated for East Coast ports to each DOE site, and an average shipment risk factor for West Coast ports to each DOE site. An advantage of this calculation is that it does not require that a specific port be selected for analysis purposes. This approach provides a reasonable model of the average or expected affect of the FRR SNF acceptance policy might on the public. This approach is conservative since the dose rates and curie content of the fuel used for the

analyses were selected to be conservative, but, as realistic as possible since it is impossible to predict the distribution of shipments among the capable ports.
The upper and lower bound risk estimates for the FRR SNF policy were also calculated. The upper bound assumes that an acceptable port with the highest per-shipment risk factors is used for all shipments, and the lower bound risk estimates assumes an acceptable port with the lowest per shipment risk factors is used. In general, the highest risk factors result from the longest shipments, and the smallest risk factors from the shortest shipments.

RESULTS

Displaying the calculational results requires far more pages than could reasonably be included in this paper and even challenges a reasonable sized EIS Appendix. Figure 1 shows the format used in the EIS to report the impacts. The figure shows, for an SNF&INEL EIS single alternative, the ranges of risk estimates for carrying out the activities. The sizes and relative positions of the ranges are similar for each SNF&INEL EIS alternative. Since it is not within the scope of this EIS to compare programmatic alternatives, no attempt was made to graphically show more than one programmatic alternative on a single figure.
Figure 1 shows that the risk associated with crew radiation exposure, public radiation exposure and traffic accidents are within the same order of magnitude for truck shipments. The risk associated with vehicle emissions is about an order of magnitude lower, and the risk associated with radiological accidents is about three orders of magnitude lower. This is driven by the conservative assumption that all shipments are at the maximum regulatory limit. If dose rate based on past practice were to be used, the radiation exposures for the crew and public would be an order of magnitude lower and the traffic accident risk would be the highest risk associated with the policy.
Fig. 1. Typical Range of Estimated Fatalities for a SNF&INEL EIS Alternative
For truck shipments, the largest contributors to the collective population dose were found to be, in decreasing order of importance: 1) incident-free dose to members of the public at stops, 2) incident-free dose to transportation crew members, 3) incident-free dose to members of the public sharing the route (on-link dose), 4) incident-free dose to members of the public residing along the route (off-link dose), and 5) accident dose risk to members of the public. Approximately 80 percent of the estimated public dose was incurred at stops, 15 percent by the on-link population, and 5 percent by the off-link population. In general, the accident contribution to the total risk was negligible compared with the incident-free risk.
For rail shipments, the largest contributors to the collective population dose were found to be, in decreasing order of importance: 1) incident-free dose to transportation crew members, 2) incident-free dose to members of the public residing along the route (off-link dose), 3) incident-free dose to members of the public at stops, 4) incident-free dose to members of the public sharing the route (on-link dose), and 5) accident dose risk to members of the public. Approximately 70 percent of the estimated public dose was incurred by the off-link population, 25 percent by the population at stops, and 5 percent by the on-link population. As with truck shipments, the accident contribution to the total risk in general was negligible compared with the incident-free risk, even when the SNF type is selected to maximize the accident risk results.
Comparing truck risks to rail risks, Fig. 1 shows that crew radiation risks are lower when SNF is shipped by rail. This is reasonable since the train operating crew is further from the cask than the truck drivers. Vehicle emission risk is somewhat higher when SNF is shipped by rail because of the conservative assumption of one dedicated train for each cask. A railroad diesel engine emits more pollutants per trip than a large truck engine. Traffic accident and radiological accident risks are estimated to be lower for rail transport than for truck because of the lower accident rates, on a per-mile basis, for trains.
The single largest contributor to the collective population doses calculated with RADTRAN was found to be the dose to members of the public at truck stops. Currently, RADTRAN uses a simple point-source approximation for truck-stop exposures and assumes that the total stop time for a shipment is proportional to the shipment distance. It uses a similar model for rail stops, and includes a distance-independent stop time for inspections. The parameters used in the stop model were based on a survey of a very limited number of radioactive material shipments that examined a variety of shipment types in different areas of the

country(12). It was assumed that stops occur as a function of distance, with a truck stop rate of 0.011 h/km and a rail stop rate of 0.033 h/km. It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m. The population dose is directly proportional to the external shipment dose rate and the number of people exposed, and inversely proportional to the square of the distance. The stop rate assumed results in an hour of stop time per 100 km (62 miles) of travel. The identical assumptions were used in the SNF&INEL EIS(2).

CONCLUSIONS

The transportation risk assessment conducted for the FRR SNF EIS provides a comprehensive and flexible approach that can be applied SNF shipments of all types. The approach is consistent with the SNF&INEL EIS(2) and previous transportation risk assessments. The flexibility associated with the tabulation of a large number of per-shipment risk factors gives the public and decision makers the information needed to evaluate the alternatives presented in the EIS. The results of the transportation risk assessment can be used to evaluate the relative impacts of the various alternatives in the Programmatic EIS.

The conservatism of the approach is necessary for the National Environmental Policy Act and decision-making requirements. The two major areas of conservatism are caused by the use of the regulatory limit as the dose estimate and the use of RADTRAN's default stop model. The affect of the use of the regulatory limit is uniform throughout the analysis, and can only result in the overestimation of incident-free effects. Since the use of a more central estimate of radiation dose would not change the conclusion that the incident-free risk is higher than the accident risk, this assumption is considered acceptable.

The conservative estimate of the stop times, especially for truck transportation, may cause inaccurate representation of the relative risks of truck and rail transportation. Informal discussions with shippers of SNF indicate that the stop models are very conservative, however, no published data analysis exists to support this premise. There is no evidence of a possibility that either stop model is nonconservative. Therefore, until a more exhaustive data analysis of shipping practices is carried out, the analysis was performed with the best available data.

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13-6
INTERACTIVE DEVELOPMENT OF RADTRAN

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ABSTRACT
The RADTRAN computer code for transportation risk analysis, which has been under continuous development at Sandia National Laboratories since 1977, has evolved from a purely research tool into a publicly available system with a variety of applications. This expansion of the user community has substantially increased the need to make the system easier to use without decreasing its capabilities or the quality of output. A large set of modifiable RADTRAN input files has been available via TRANSNET for several years. One approach to assisting the user involves adding annotations/information to each of these files. A second approach is providing additional help in building new/modifying old input files.
Keeping the proposed information/annotation files separate from but closely coupled to the modifiable input files within the TRANSNET shell system allows the modifiable input files to remain as regular input files while providing rapid, automatic access to useful information about the analysis. In this way, the sample input files remain intact as regular RADTRAN input files and any files generated using associated on-line menus or editors may be readily converted into new input files. A single sample file is selected and used as an example to illustrate the prototype help features.

INTRODUCTION
The RADTRAN computer code for transportation risk analysis (1,2,3,4), which has been under continuous development and application at Sandia National Laboratories (SNL) since 1977 and is now available in the fourth release (RADTRAN 4), has evolved from a tool for research and internal applications into a publicly available system used by many with less than professional expertise in risk analysis. RADTRAN 4 and related codes comprise a computational system that is publicly available on TRANSNET (5), the SNL node on the Internet that provides access to the RADTRAN code system. The expansion of the RADTRAN user community has been paralleled by increased usage of the code system on TRANSNET. The system includes a user-friendly, menu-driven, "front-end" that guides the user in the creation and editing of RADTRAN 4 input files. Since the inception of the TRANSNET system (5), SNL has maintained a telephone help capability for RADTRAN users. Users who have problems, questions, or need project-specific data are able to speak directly to a RADTRAN code developer. From this direct user interaction, two of the authors of this paper noted certain types of problems were being repeatedly encountered. Some problems could be traced to ambiguous wording in the menus or documentation or unrealistic expectations regarding capabilities and/or flexibilities of the code system, but a few problems were intrinsic to the code. For example, a user discovered that a logical error occurred when zero was entered as the population density for a route segment. The code was modified because this was a circumstance other users were likely to encounter (large areas of the United States have zero population density). At first, such difficulties were corrected as they arose on
an ad hoc basis. Any alterations made in the code were posted on the RADTRAN revision-history bulletin board on TRANSNET. As the computational system became more complex and the user community became larger and less experienced, this approach no longer sufficed.
SNL has employed several strategies to address user needs. Workshops have been held,

most recently at the International High-Level Radioactive Waste Management Conference in Las Vegas. Recognizing the value of user feedback to SNL's efforts to continue improving both the user friendliness and the technical value of the RADTRAN system, in 1994 SNL initiated a new strategy of convening formal RADTRAN user meetings. Two SNL-sponsored meetings of RADTRAN users were held in 1994. At the first of these, the scope was limited to the RADTRAN 4 code system; notices were mailed to all individuals in the United States who either had current user passwords or who had requested executable tapes of RADTRAN 4 for use with their own mainframe computers. The second meeting had a broader scope, covering both the RADTRAN system and other codes and databases available on TRANSNET. At these meetings, users were presented with an outline of planned modifications/improvements and asked for feedback and suggestions for improvement. Reports of these meetings will be available this year.

The results of these meetings were positive and constructive. The suggestions for improvements addressed everything from highly specific single-parameter comments to such broad concepts as addition of a graphical user interface (GUI) and incorporation of a geographical information system (GIS). There were several criticisms of the existing input/file/generating menus on TRANSNET. These ranged from the general to the particular. One general criticism was that several common, inadvertent key-press combinations cause the user to exit without having saved any part of the file being developed. This has been corrected on the current operating system. An example of a particular comment deals with a default dispersability category (DISP=1). It cannot be modified by the user, but the menu screens do not explain this.

While GUI development may be appropriate for the long term, the simpler matter of making it easier for a user to construct a RADTRAN input file can be implemented with available technology within the existing computational environment. Because RADTRAN requires large amounts of input data (like all risk codes) and many new users are relatively inexperienced, it seemed more reasonable to help develop the users' understanding of what is needed for suitable input than to devote resources to development of a GUI that might look like a home-computer game but not really assist the user. As long as it continues to be one of the world's most powerful tools for serious risk analysis, little purpose is served by slick packaging before basic user needs are addressed.

The current RADTRAN computational system includes a collection of approximately 15 Transportation Technology Center (TTC) file sets derived from actual risk analyses performed by SNL for the DOE. These files were placed on the system primarily to make them accessible to critics and activists opposing U.S. Department of Energy (DOE) environmental assessments (EAs) and environmental impact statements (EISs) that the analyses supported. This step was taken in response to activists' criticisms that DOE/SNL risk analyses contained "hidden assumptions" that could not be reviewed by anyone who did not have access to a mainframe computer. Placing the input files on the system thus performs a public quality assurance and review function in that it allows anyone with Internet access or a personal computer (PC) and a modem to obtain the raw input files, to run RADTRAN, and to satisfy themselves that the results reported in the documents they support are indeed the actual RADTRAN outputs.

The current menu system also allows users to modify (edit) these files. Thus, critics are able to satisfy themselves as to the effect on the result of changing "assumptions." The latter capability was soon seen to be useful to serious analysts as well, because it can be used to modify a few parameters and thereby quickly analyze similar but not identical problems. However, as currently structured, the TTC files are "pure" input files and contain little information to help a beginning user identify the sources of the data. Little need was seen for such a user interface at first, because the source information was contained in the EAs and EISs that the input files were developed to support. However, the files soon were being used as "starting points" by persons who had no interest in the original problem but who did have a similar problem to analyze. Both the lack of explanatory information and the fact that the input files cannot be easily modified without consulting the RADTRAN 4 User Guide (4) were cited as non-user friendly features of the current system during the user feedback meetings. Therefore, a first step toward increased user friendliness is the association of background information with each TTC file.

BACKGROUND INFORMATION FOR TTC FILES

Much potentially helpful background information could be associated with each input file. For example, a user might be interested in knowing why a particular input file was generated. Other types of information the user might find helpful include general conclusions resulting from the RADTRAN run and the types of related analyses that are possible using simple modifications (with the modifications described in detail) of a particular file.

ON-LINE USER GUIDE

An on-line user guide also was recommended by some users. This would consist of making the text of the published User Guide (4) available on the TRANSNET system. This option has the drawback of not being accessible while the user is building or modifying an input file. For that reason, enhancing the file-generating menus with additional information is believed to be a more beneficial option. This option is discussed in the following section.

AVAILABLE RADTRAN INPUT FILES

Figure 1 shows the list of RADTRAN TTC file sets currently available to the TRANSNET user. One set of files (FOREIGN RESEARCH REACTOR SPENT FUEL) has been selected to illustrate the proposed user assistance. These files were developed to analyze transport of foreign research reactor spent fuel by several modes (6). Figure 2 shows the second step in file selection with the present current RADTRAN menus. The first line on this sample screen is "Enter a Category Number -> 15." This means that the user selected number 15 from the list in Fig. 1. The screen shows the full title of the citation and gives all the files available under Category 15. The last line in the table is "Enter subcategory number you wish to use (1-6) -> 1," which means that the Highway Routes subcategory was selected. Figure 3 shows the last line of Fig. 2 and then shows the titles of all the input files available for research reactor spent fuel analysis by highway (truck) mode in this file set. In the example, File Number 1 has been selected. Table I a,b shows the sample input file, ASTRAXT. As the figure shows, it is "pure" computer input and does not contain much information helpful to an inexperienced user.

NEW HELP FILES

Figures 4 and 5 are examples of background information associated with a particular TTC file; in this example, the ASTRAXT file. It tells something about the nature of the problem that was to be solved, why it was important, and gives detailed information about specific input parameters used in the analysis. Figure 6 summarizes the results of running this particular TTC file for both the first route-segment (3.2 km) and the total route and shows the types of outputs that may be obtained from RADTRAN. Note that probability and consequences as well as risk are given.

SPREADSHEET FOR BUILDING AN INPUT FILE

Several users have commented that in its current form, the RADTRAN menu system does not contain sufficient help for a beginner who isn't familiar with the User Guide (4). The Appendix, p. 1, p. 4. p. 6, p.7, p.10, p. 12, shows some of the help statements for a prototype enhanced menu system; information is given for each line in a RADTRAN input file. This experimental system now exists as a PC-based spreadsheet. The few default values remaining in RADTRAN, such as those for atmospheric dispersion, are listed under the appropriate keywords. Emphasis has been placed on identifying which input values are required for route-specific analysis and which are not. In addition, the user is frequently reminded about the necessity of making the sizes of the various data arrays consistent. Failure to enter the proper number of values in certain arrays is a common beginner's error. Isotope data may be automatically entered from RADTRAN's internal isotope database by use of an alphanumeric identifier. The spreadsheet lists all available isotopes and their identifiers but does not display the contents of the database itself. A complete data list for each isotope used can be obtained by running a full output of RADTRAN.

SUMMARY

This paper has discussed approaches for responding to user comments about ease of use of the RADTRAN system. The first approach to assist users involves adding annotations/information to the modifiable RADTRAN input files on TRANSNET. The second approach provides additional help in building new/modifying old input files.

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DEVELOPMENT OF A DOE ROUTE SELECTION METHODOLOGY FOR THE SHIPMENT OF HIGHWAY ROUTE CONTROLLED QUANTITIES OF RADIOACTIVE MATERIALS

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ABSTRACT

The U.S. Department of Energy (DOE) is developing route selection guidance for its unclassified Highway Route Controlled Quantity (HRCQ) shipments of radioactive material (primarily spent nuclear fuel, high-level radioactive waste, and certain large source radioisotopes). This guidance will provide DOE with a consistent means for routing HRCQ by highway and rail shipments and will allow shipment support planning activities on the part of DOE Traffic Managers, state and local officials, emergency response agencies, and other stakeholders. The major concerns that will be addressed in this guidance will be risk reduction, both radiological and nonradiological, and operational efficiency.
10 Currently, only highway shipments have specific US Department of Transportation (DOT) regulations that apply to selection of routes. Rail movements have no US DOT routing regulations. A consistent method of routing HRCQ shipments will provide all of DOE with a dependable, reproducible, and defensible way to ship. It will also provide the affected parties with the ability to effectively prepare for the shipments. This methodology will be developed with the assistance of the Transportation External Coordination Working Group and other stakeholder groups; transportation and shipping experts; and DOE traffic management personnel.
This paper outlines the regulatory framework within which DOE currently makes HRCQ shipments, the rationale for the route selection guidance, the steps that are being taken to develop the route selection guidance, and an initial set of guidance criteria and methodology that are being used as a basis for discussion.

INTRODUCTION

The U.S. Department of Energy (DOE) is developing route selection guidance for its unclassified Highway Route Controlled Quantity (HRCQ) shipments of radioactive material (primarily spent nuclear fuel, high-level radioactive waste, and certain large source radioisotopes). This paper outlines the regulatory framework within which DOE currently makes HRCQ shipments, the rationale for the route selection guidance, the steps that are being taken to develop the route selection guidance, and an initial set of guidance criteria and methodology that are being used as a basis for discussion.

REGULATORY BACKGROUND

Regulatory Background - Highway

Department of Transportation

In 1978, the Department of Transportation (DOT) initiated a rulemaking to establish a set of routing regulations for HRCQ shipments. The resulting regulations, often referred to by its docket number, HM-164, were developed based on extensive analysis and public input to provide a framework for minimizing risk. These regulations provide a framework for selecting routes for such shipments. The regulations require, among other things, that shipments be made along "preferred routes" (generally, these are interstate highways or state-designated alternatives) chosen to "to reduce time in transit" over the preferred route segment of the trip. The requirement to reduce time in transit does not always lead to an unequivocal choice of one route. For short trips between sites that are serviced by a limited number of interstate highways, the regulations will generally lead to a clear choice of one route. As the distance of the shipment increases, and the complexity of the interstate system between the origin and the destination sites increases, the choice of a single route based upon reduced time in transit may become more problematic. Through the years, Docket HM-164 has been re-opened several times to further refine and clarify the route selection regulations. HM-164C established some very specific requirements for choosing the pickup and/or delivery route to and from the preferred highway to be used. These requirements cause the analyst to evaluate the route selection in two steps. Paragraph 397.101(c)(2) in 49 CFR specifies that for pickup and delivery not over a preferred route, "the route selected must be the shortest distance route from the pickup location to the nearest preferred route entry location, and the shortest distance route to the delivery location from the nearest preferred route exit location (italics added)." Deviation from the shortest pickup and delivery route is authorized if such deviation:

1. Is chosen based on minimizing radiological risk, and
2. Does not exceed the shortest distance route by more than 25 miles, and does not exceed 5 times the length of the shortest distance route.

This means that if the shortest route to an interstate highway is three miles, then an alternate route could be used, if it does not exceed 15 miles (5 times the shortest route) and it can be shown to reduce radiological risk. If it is ten miles to the nearest interstate highway, an alternate route can be used if it does not exceed 35 miles (shortest distance + 25 miles) and can be shown to reduce radiological risk. The purpose of HM-164C is to minimize the amount of travel on secondary roads, which are considered higher risk roads.

The effect of these additional regulations is that route selection must be addressed for two different parts of the trip - (1) pickup and delivery routes and (2) preferred routes.

Nuclear Regulatory Commission

Transportation of spent nuclear fuel by highway does require advanced route approval by the Nuclear Regulatory Commission (NRC) and advanced notification to NRC of each shipment. NRC also requires advance written notification to States in which the shipment will travel, and requires that the written notification must include: (1) a listing of the highway routes to be used within the State, (2) the estimated date and time of departure from the point of origin of the shipment, (3) the estimated date and time of entry into each State, and (4) an appropriate statement that the schedule information must be protected as safeguards information. A change in arrival time greater than six hours requires notification to the State.

Department of Energy

DOE Order 5632.11, Physical Protection of Unclassified Irradiated Reactor Fuel In Transit, 9-15-92, provides direction on procedures for transporting a specific type of HRCQ shipments, irradiated reactor fuel. This Order specifies that such material shall be packaged and transported in accordance with DOE Orders 1540.1A, 1540.2, and 5480.3. On the subject of highway route selection, it says,

3. Motor carriers shall ensure that the transport vehicle operates over preferred routes selected to reduce time in transit except that an interstate bypass or beltway around a city shall be used when available. A preferred route consists of either or both:

a) An interstate highway system.

b) A State-designated route selected by a State routing agency according to Department of Transportation (DOT) guidelines and filed with that Department (49 CFR 177.835). [sic]

Regulatory Background - Railroad
Department of Transportation
There are currently no DOT rail route selection regulations for shipment of radioactive material.
Nuclear Regulatory Commission
Transportation of spent nuclear fuel by rail does require advanced route approval by the Nuclear Regulatory Commission (NRC) and advanced notification to NRC of each shipment. NRC also requires advance written notification to States in which the shipment will travel, and requires that the written notification must include: 1) a listing of the rail routes to be used within the State, 2) the estimated date and time of departure from the point of origin of the shipment, 3) the estimated date and time of entry into each State, and 4) an appropriate statement that the schedule information must be protected as safeguards information. A change in arrival time greater than six hours requires notification to the State.
Department of Energy
A memorandum outlining recommended practices for routing rail shipments of unclassified spent nuclear fuel (SNF) and high-level waste (HLW) made by DOE was issued in a memo by Lawrence H. Harmon, Transportation Management Division, Office of Defense Waste and Transportation Management, Defense Programs on 29 August 1988. The title of the memo was Current Recommended Practice For DOE Rail Routing Of Unclassified Spent Nuclear Fuel (SNF) and High Level Waste (HLW).
"Rail shipments of unclassified SNF and HLW are the responsibility of the traffic manager of the field office having program authority for its transport. Actual routes selected will be a joint effort among the DOE, its contractors, and the origin and destination carriers.
Because of the sensitivity of routing SNF and HLW through population centers, it is necessary for DOE, its contractors, and its carriers to consider additional specific route selection criteria prior to route finalization:
1. Minimize time, distance, number of carriers, and interchange points in transit.
2. Maximize the use of best track class considering that maximum safe speeds for each track class is regulated by the Federal Railroad Administration.
3. Apply lowest through rates and accessorial charges consistent with service requirements.
4. Obtain computer run from ORNL's "INTERLINE" (rail routing model) of the final alternate rail routes being considered--retain as part of the permanent route selection record.
5. Coordinate final route selection with Transportation Management Division (DP-121).
STEPS TO DEVELOP A ROUTE SELECTION GUIDANCE DOCUMENT
DOE is proposing a nine-step strategy for developing and implementing route selection guidance for HRCQ shipments. The first step in the process was to develop a strategy and have it reviewed by DOE's stakeholders. A parallel effort involved working with various transportation elements within the department, to develop draft discussion papers addressing issues associated with, and potential approaches to, route selection for each mode of transport. These draft documents served to stimulate discussion, comment, and institutional interaction. Based on feedback on the strategy and discussion papers, a draft guidance will be developed and distributed for DOE internal review. After review and revision at DOE, the document will be available to stakeholders and the general public for comment. Notice of this availability will be through the Federal Register. DOE will address all comments received in the review process. After final internal review and revisions at DOE, the guidance will be published and implemented.
METHODOLOGIES
A major part of the routing discussion papers included preliminary proposed methodologies for highway and rail. These preliminary methodologies were developed, not as a DOE position, but rather as a "springboard" for stimulating public input and discussion.
The proposed route selection methodologies use criteria that are widely accepted as being effective at reducing the risk associated with transportation of radioactive materials. The criteria were chosen based on 1) effectiveness at discriminating among routes, 2) relative contribution to various types of risk, and 3) availability of the required data.
The route selection methodologies synthesize input from a number of different

sources. The methodologies represent a simple approach to route selection, based on the premise that the best way to estimate and compare the total risk associated with shipments of radioactive materials over specific routes should be simple and straightforward. The methodologies do not measure risk in absolute terms. A number of simplifying assumptions are made in analyzing routes that preclude the process from developing specific risk measures, but still allow the comparison of relative risk associated with competing routes.

Work is currently underway to define specific values for time and population variables in the route selection methodologies. Values for these variables may change with each campaign based of differing requirements.

Highway Route Selection Methodology

The two major parameters used in the highway methodology are time in transit and population density. Other parameters that have been proposed by different groups include truck accident rates, potential property exposure, transit of environmentally sensitive areas, transit of culturally sensitive areas, emergency response time, proximity of difficult to evacuate populations (e.g. schools, prisons, hospitals, stadiums, shopping centers, etc.), transit of bridges and/or tunnels, inclement weather, and time of day considerations.

The proposed highway route selection methodology includes two independent, but related, efforts -selection of pick-up and delivery routes and selection of the preferred route to be used. Pick-up and delivery route selection is based on straightforward compliance with the provisions of 49 CFR 397.101(c)(2). This also determines the preferred route (i.e. interstate highway or state designated route) entry and/or exit points that are to be used. The selection of the actual preferred route to be traversed is based upon a process by which multiple routes are run through successive filters to eliminate routes that could be considered less desirable. The HM-164 regulations provide the first filter. The designated entry and exit points for the preferred route segment are chosen and the shortest distance pickup and delivery routes to those points are determined. The process then proceeds along two separate paths.

Pickup and Delivery Route Segment

1. Select proper origin and destination points for shipments.

2. From the origin point, choose the shortest route to the nearest preferred route entrance (in most cases this will be the nearest interstate highway). From the destination point, choose the shortest route from the nearest preferred route exit. A highway routing model can be used to determine the shortest route.*

3. Coordinate with State Departments of Transportation (or equivalent agencies) to verify that the shortest distance pickup and delivery routes are capable of handling the transport vehicles. This evaluation would consist of a review of clearances (height and width), weight limits (bridges), and local ordinances. If the selected pickup or delivery route is not acceptable, then select another route that meets the regulatory constraints.

4. Coordinate with local communities through which the pickup and delivery routes pass and provide an opportunity for local community comment and interaction on the final local route selection.

Preferred Route Segment

1. Generate a "base" HM-164 route between the chosen preferred route entry and exit points. This base route should be compliant with Federal regulations at 49 CFR 397.101 and .103 by using only preferred routes (either interstate highways and/or routes designated by States), using city by-passes when available, and should require the minimum transit time between the entry and exit points.

2. Generate optional HM-164 compliant routes to the base route until the cumulative travel time is XX percent greater than the base route.* The routes that are generated make up the feasible routes from which to select the shipment route. If no optional routes are within XX percent of the base route, choose the base route for shipment.

3. Calculate the total population along each of the optional and the base routes (a band of XX miles can be used). Eliminate all routes not within XX percent of the total population of the route with the minimum population. If only one route remains, choose that route for the shipment.

4. If more than one route still remains, determine the number of miles traversed through urbanized areas and the total population of the urbanized areas within a band XX miles wide. Divide this urban population by the number of miles traversing

urban areas to determine the potential population exposed along the transport corridor in urbanized areas. Select the route with the lowest urban population density value.
5. Write a report to document the process by which the final route was selected.
Railroad Route Selection Methodology
The two major parameters used in the rail methodology are impedance values and population density. Impedance values represent a combination of time, distance, and cost to transfer from one system to another. Other parameters that have been proposed by different groups include time in transit, cost of shipment, track class, grade crossings, accident rates measured by type of track, location, or carrier, potential property exposure, transit through environmentally sensitive areas, transit through culturally sensitive areas, proximity of difficult to evacuate populations (e.g. schools, prisons, hospitals, stadiums, shopping centers, etc.), and class of railroad company.
Rail shipments can move as general commerce, dedicated trains, or special trains. Rail shipments that move as general commerce, normally move over a maximum distance on the rail company that initiates the shipment. In this way, the originating rail company maximizes its profit. Dedicated trains carry only one commodity from origin to destination, stopping to refuel, to change crews, and if more than one carrier is required, to change locomotives. Dedicated trains tend to be used to transport large volumes of raw materials, such as coal. Layover time at railyards is kept at a minimum, reducing overall travel time considerably compared with general commerce train service. Special trains are dedicated trains with their own locomotive and crews. Certain restrictions can also be put on special trains including speed limits, stopping when meeting other trains, crew numbers, etc. The specific routes and other conditions that will be met by dedicated or special trains are negotiated through a contract between the shipper and the railroad.
The first step in route selection relates to determination of the type of rail service to be used - general commerce or dedicated (or special) train. Criteria for this determination will be based upon level of service required, institutional factors, cost, and other shipment- or campaign-specific considerations. Once the type of service has been determined, the route selection process can begin.
General Commerce
General commerce routes are selected by the rail company to coincide with normal operations. The DOE or contractor transportation manager should use available tools to confirm the reasonableness of the route chosen. Special programmatic or institutional issues may, at times, require specific route selection considerations. These should be discussed and negotiated on a case-by-case basis with the involved rail company. Issues that significantly impact route selection, may require the use of a dedicated or special train.
1. Select the origin and destination points for the rail shipment.
2. Use a rail routing model to determine a base route and feasible alternatives.* Feasible alternative routes should have an impedance value within XX percent of the impedance value of the base route.
3 Determine the population within XX miles of the track along each of the routes. Eliminate all routes not within XX population percent of the route with the minimum population.
4. Contact originating rail company to determine if the route they plan to use in general commerce is consistent with alternative routes remaining after Step 3.
5. If rail company proposes a route not remaining after Step 3, conduct a run to determine impedance and population along the proposed rail company route to determine if it is within acceptable limits of the base route.
6. Review the results with the rail company(s) to determine if a route acceptable to DOE can be negotiated.
7. If agreement cannot be reached between DOE and the rail company(s), consider use of a dedicated or special train.
8. If route selection issues can be adequately addressed within the context of general commerce operation, notify the affected states of the route that will be taken and coordinate with the states to address concerns that they have. The Federal Railroad Administration can be brought in to inspect portions of the track that are of concern.
Dedicated or Special Train
1. Select the origin and destination points for the rail shipment.

2. Use a rail route selection model, with parameters set for dedicated or special trains, to determine a base route and feasible alternatives. Select those routes with a calculated impedance within XX percent of the base route.
3. Determine the population within XX miles of the track along each of the routes. Eliminate all routes not within XX population percent of the route with the minimum population.
4. If more than one route still remains, determine the number of miles traversed through urbanized areas and the population of the urbanized areas within a band XX miles wide. Divide the population by the number of miles to determine the potential population exposed along the transport corridor in urbanized areas. Select the route with the lowest urban population density value.
5. Write a report to document the process by which the final route was selected.
6. Notify the affected States on the route that will be taken and coordinate with the States to address concerns that they may have about the route, i.e. track quality. The Federal Railroad Administration can be brought in to inspect portions of the track that are of concern.

CONCLUSION

DOE is developing rail and highway route selection guidance to be used for its unclassified HRCQ shipments. The process of developing this routing guidance involves interaction with DOE transportation management staff and affected stakeholders. DOE is developing this routing guidance to ensure a consistent set of criteria and methodology will be used in the selection of these shipping routes. A consistent route selection policy also allows DOE to plan and administer risk mitigation and public outreach programs in an effective and efficient manner.

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Co-chairs: Albert Castagnacci, Dusquesne Light;
Steven C. Slate, PNL

14-1

CONCEPTUAL DESIGNS FOR EVALUATING ALTERNATIVE LOW-LEVEL WASTE DISPOSAL METHODS FOR NEW YORK

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ABSTRACT

A paper presented at Waste Management '94 described the process for selecting a preferred method (or methods) for disposal of low-level radioactive waste in New York State. Six disposal methods will be compared: above-grade concrete vaults without an additional cover, covered above-grade concrete vaults, below-grade concrete vaults, vertical shaft mines, drift mines, and augered holes. In order to make this comparison, conceptual designs were developed for each of the candidate disposal methods. While they are called conceptual designs, these designs in many cases include much greater detail than is typical of a conceptual design, to enhance public understanding of how the disposal facility will function.

This paper describes the six conceptual designs. It indicates the features that are common to all six disposal methods and describes some of the major differences among the methods, including land areas required for the disposal facility, sizes of staffs, costs, volumes of materials being imported, and volumes of materials being excavated. It also lists some of the alternative design features that were considered but not used in the nominal conceptual designs.

INTRODUCTION

In response to the federal laws on low-level radioactive waste disposal, New York enacted the Low-Level Radioactive Waste Management Act in July 1986 and amended it in 1990. The Act created a Low-Level Radioactive Waste Siting Commission with responsibility for selecting both a preferred method and site for disposal of New York's low-level radioactive wastes. Under the amended Act, the Siting Commission must evaluate alternative disposal methods and select a preferred disposal method (or methods) before proceeding to select a site. The process for method evaluation and other steps in the method selection process are described in some detail in

Refs. 1 through 3. It is summarized in the following paragraphs.
Based on the input received during a public outreach program and the review of previous work by the Commission and others, the following six candidate disposal methods will be evaluated in the disposal method selection process:
- Above-grade concrete vaults without an additional cover.
- Covered above-grade concrete vaults.
- Below-grade concrete vaults.
- Vertical shaft mines.
- Drift mines.
- Augered holes.

This group of candidate disposal methods covers the full range of major conditions affecting permanent low-level radioactive waste disposal. It also includes the two alternative methods specifically mandated for consideration in the 1990 amendments to the 1986 Act: deep vertical shaft mined disposal and above-ground monitored retrievable disposal (represented by above-grade concrete vaults without an additional cover).
The Commission will evaluate the six candidate disposal methods with respect to the seven evaluation considerations listed in Table I. The relative abilities of each of the disposal methods to satisfy criteria that were formed from the evaluation considerations will be assessed. Factors for use in developing criteria and in evaluating the disposal methods are also shown in Table I. The evaluations will compare the six methods as the methods are represented by the conceptual designs briefly described here. Complete descriptions of the conceptual designs are contained in Ref. 4.

THE DESIGNS

To perform comparisons of alternative disposal methods, the Siting Commission prepared conceptual designs for the six candidate methods. Those conceptual designs were intended to meet design requirements specified in New York regulations and to be of sufficient detail to enable objective comparisons of the six methods with the aim of selecting a preferred method (or methods). In fact, it was often necessary to add more detail to the designs than is typical for a conceptual design, in order to enhance public understanding of how a disposal facility would function.

Common Features

Some features are common to all six disposal facility designs. These include features such as buildings for the receipt of waste, the placement of waste in concrete overpacks called modular disposal units (MDUs), and conduct of administrative functions. The designs were based on the receipt of a total of 5.5 million cubic feet of low-level radioactive waste over 60 years. This waste volume corresponds to the most likely or "expected" case described in Ref. 5.
For design purposes, all of the waste is expected to arrive at the disposal facility in either 55- and 85-gallon drums, 120- and 200-cubic-foot steel liners, 90-cubic-foot steel boxes, or as steel ingots that are the result of melting steam generators. It is estimated that an average of between 15 and 20 trucks a week will arrive at the disposal facility carrying waste. All of the waste except the steel ingots will be placed in rectangular MDUs of one of two sizes -- 6 feet by 6 feet by 9 feet and 8 feet by 8 feet by 9 feet. Once in place inside the MDUs, the waste packages will be grouted in place. The use of the MDUs is motivated, at least in part, by the requirement in New York regulations and Siting Commission policy that the waste be retrievable throughout the institutional control period, which is assumed to be 100 years. About 47,000 MDUs will be constructed on site: about 25,000 of the smaller size and 22,000 of the larger size.

Site Characteristics

Conceptual designs were developed for generic disposal facility sites described in Ref. 6, which provides information about the geology and hydrology of sites that are typical of the non-excluded areas in the State of New York. For the four near-surface methods (not mines), the generic site characteristics do not have a major effect on the designs, except that generally shallow groundwater conditions limit the depths to which waste can be placed below grade. Designs were prepared, to the extent necessary, for the mine disposal facilities in four geologic media: shale, salt, limestone, and igneous/metamorphic rock.

Specific Designs

Illustrations of the six conceptual designs are shown in Figs. 1 and Fig. 2. Three of the designs are based on massive concrete vaults that are subdivided into cells

by interior concrete walls. The roofs on the vaults and augered holes are tied structurally to the walls of the vaults. All of the vaults dedicated to Class B and C waste hold two layers of MDUs. The two above-grade vault designs that are dedicated to Class A waste hold three layers of MDUs, while the below-grade vaults and augered holes dedicated to Class A waste hold only two layers of MDUs. Each layer of MDUs in a vault cell contains either nine large MDUs or sixteen small MDUs. Each layer of MDUs in an augered hole contains either four large MDUs or seven small MDUs. All of the MDUs will be placed in the vaults or holes from above, using cranes.

Fig. 2. Illustration of a mined LLRW disposal facility.

The illustrations for the three vault methods are of the designs for containing Class A waste. The vaults for Class B and C waste differ slightly from those for Class A waste. The primary difference is that the Class B and C vaults are narrower, being only one disposal cell wide. All of the augered holes are the same, except that the Class B and C wastes are placed together in separate holes from the Class A waste. The vaults for disposal of Class A waste using above-grade vaults are about 475 feet long and 60 feet wide. For below-grade disposal they are about 700 feet long and 70 feet wide. For disposal of Class B and C waste in all three vault-based designs, the vaults are about 300 feet long and 30 feet wide. The augered holes are all about 26 feet in outside diameter. Holes of this size are larger than have typically been used in the past, but the large size helps keep down land area requirements for this disposal method. Use of the larger holes also reduces the amount of excavation needed and the amount of materials needed to construct the holes.

The covered above-grade vaults, below-grade vaults, and augered holes have earthen covers placed over the concrete vaults or holes. These earthen covers are 7 feet thick and consist of various layers of natural and man-made materials that serve functions such as drainage, prevention of water infiltration, prevention of animal and plant intrusion, and support of short-rooted plants at the surface.

New York regulations require that protective enclosures be placed over the disposal units (vaults and holes) while they are being filled and until they are covered. These protective enclosures are envisioned as insulated steel buildings with access for trucks carrying MDUs. The enclosures are part of the conceptual designs and are intended to protect the disposal units from weather and freezing temperatures. With the two above-grade disposal methods, the covers will not be emplaced until disposal facility closure, so some the protective enclosures will remain in place until facility closure, which could extend to as long as 60 years.

The two mine designs contain 650-foot-long Class A disposal rooms that will hold either 308 of the large MDUs stacked two high and two wide, or 612 of the small MDUs stacked two high and three wide. The disposal rooms for Class B and C waste are 600 feet long and contain 260 of the large MDUs, stacked two high and two wide. All of the MDUs for the mine methods will be placed using forklifts. The underground layout is identical for both drift and vertical shaft mine disposal methods and contains eight separate waste disposal units (panels) for Class A waste, each containing up to fourteen disposal rooms, and one waste panel for Class B and C waste containing four disposal rooms. One Class A panel and the Class B/C panel will be constructed before any waste is disposed of in the facility. As individual panels are filled, they will be temporarily closed. At the end of operations, the facility will be backfilled and seals will be constructed at strategic locations in the underground works and in the drifts or shafts.

The conceptual designs were developed to allow comparison of the inherent features of the six candidate disposal methods. The designs were not necessarily optimized and, since potential sites were not known, they will probably be changed considerably when they are developed further.

COMPARISON OF THE CONCEPTUAL DESIGNS

Table II provides information about some characteristic parameters of the conceptual designs. The parameters shown are some of those that will be used in the detailed comparison. Table II shows that the land area required for uncovered above-grade vaults and the two mine designs is approximately the same. The below-grade vaults require a little more land because, in the conceptual designs, the earthen covers over the vaults rise above the surrounding terrain and thus require a very gradual slope to prevent erosion. Covered above-grade vaults require even more land because they rise higher above the surrounding terrain. The largest land area is required by

the augered holes which, to prevent collapse of adjacent holes while new holes are being constructed, require significant spaces between holes and therefore result in larger land requirements.
The average facility staffs during operations cover a very narrow range (66 full-time staff persons to 70 full-time staff persons) for five of the six methods. The staff for the augered holes is larger because of the larger excavation, construction, and concrete work required, and because the larger land area creates greater maintenance and environmental monitoring requirements.
The estimated total lifetime costs for the below-grade vaults and the two mine methods are approximately the same. The costs for the uncovered above-grade vaults are slightly higher, primarily due to the requirement in New York regulations that protective enclosures be constructed over these vaults and kept in place until the end of the disposal facility closure. Also, in the conceptual designs it was assumed that a more expensive mix of concrete will be used for the uncovered above-grade vaults and that the walls and roofs will be thicker than for the other two vault designs. The thicker walls and roof were chosen to better withstand the effects of frequent freeze-thaw cycles. The covered above-grade vaults cost slightly more than the uncovered above-grade vaults because, while the same set of protective enclosures is needed, the cost of providing the earthen covers represents an additional expenditure for this disposal method. The largest lifetime cost is attributed to the augered holes. This is a result of much larger volumes of materials to be excavated, larger amounts of concrete to be poured, and the additional monitoring and maintenance cost associated with the larger land area required.
The volume of materials excavated on site is lowest for the uncovered above-grade vaults, representing primarily excavation to place the foundation of these vaults well below the frost line. The volume of materials excavated on site is higher for covered above-grade vaults than uncovered above-grade vaults primarily due to an assumption that some of the materials for the cover construction could be found on site. The two mine methods represent the next highest volume of materials excavated on site. This material consists primarily of the rock removed to form the accesses to the mines, rock from the excavation of the rooms where the waste will be placed, and rock from the construction of horizontal accesses to those rooms. The largest estimates of volumes of materials excavated on site are for the below-grade vaults and the augered holes. These volumes represent the earthen materials that must be removed to make room for the vaults and the holes.
In addition to materials excavated on site, large volumes of materials are expected to be imported from off the disposal facility site. Estimates of these volumes are shown in the last row of Table II. It is estimated that no material will have to be imported from off site for the mines and that only a relatively small volume of such material will be needed for the uncovered above-grade vaults, primarily to provide under-vault drainage. Below-grade vaults will require about 1 million cubic yards of material to be imported from off site, primarily to construct the earthen covers over these vaults. Covered above-grade vaults will require much more material to be imported because the volume of the covers will be much greater due to the fact that the vaults rise above the natural terrain. The augered holes will require the most material to be imported because the waste is much more spread out on the site, necessitating much larger covers.

ALTERNATIVE DESIGN FEATURES CONSIDERED

A number of alternative design features were considered for use in the conceptual designs. These features, and others, could become part of the final design for the New York low-level radioactive waste disposal facility, but were not included in the conceptual designs that will be used for the comparative evaluation. These alternatives are described very briefly below:

Disposal of some waste without using MDUs -- Some of the waste could be placed in the disposal facilities without first putting it in MDUs. This would most likely include waste that has short-lived radionuclides and lower concentrations of radionuclides. Except for the steel ingots, this alternative was not selected because of the strong emphasis in New York regulations on the ability to retrieve waste.

Use of more than one of the candidate disposal methods at a single disposal facility -- While the preferred method could involve combining more than one candidate disposal method at a single facility, the Siting Commission chose to

compare the six candidate methods first, before deciding whether combinations of methods should be considered.

Use of sand or gravel to fill spaces among the waste packages inside the MDUs -- Grout was chosen for filling around the waste packages in the MDUs because it provides structural stability, additional shielding, and chemical buffering beyond what sand or gravel would provide. Use of sand or gravel may also require more careful placement, raising the potential radiation dose to the workers who backfill around the waste packages inside the MDUs.

Loading the concrete vaults from the sides instead of from the tops -- Roofs do not have to be placed on the concrete vaults to provide a weather shield because New York regulations require the use of a separate protective enclosure over all of the disposal units when they are being filled. Therefore, there is no need for a roof on a vault while it is being filled. Side-loading makes it more difficult to reduce worker radiation doses since it is harder to keep the workers separate from the waste during emplacement.

Maintaining underground access to mines during the institutional control period for inspection and testing -- While this alternative would increase confidence in short-term postclosure performance, it would increase the cost of the mine disposal facilities, particularly for the vertical shaft mine facility.

Use of augered holes with much smaller diameters than those in the conceptual designs -- Augered holes are usually employed for disposal of waste that exhibits a high radiation dose rate external to the waste containers. It is usually a specialized method of disposal and not used for disposal of large volumes of waste. However, the comparison is being made at this time on the basis of disposing of all New York's waste for 60 years in a disposal facility using a single candidate method. It was estimated that if small-diameter holes were used there would be large increases in the land area required, excavation needed, and concrete needed compared to those required for the present conceptual design for augered holes.

Delaying backfilling and sealing of the mine underground facilities until the end of the institutional control period -- Constructing temporary seals to prevent access to the underground facility during the institutional control period and deferring backfilling and sealing until the end of that period would greatly improve the ability to retrieve the waste at a reasonably cost. However, it is inconsistent with New York regulations, which require closure to occur at the end of disposal operations.

Use of precast vault roofs -- Some proposed vault designs use precast roof slabs that would probably allow easier retrieval. However, it was felt that the monolithic vault roofs tied directly to the walls of the vault would be less likely to admit water into the vault interiors and would provide greater structurally rigidity to the vaults.

Increasing the MDU stack height from two to three (or more) MDUs in mine Class A waste disposal rooms -- This alternative would significantly decrease the size of each of the mine facilities and the associated construction costs. However, it would require a larger forklift and possibly create other logistical problems. The heaviest MDUs weigh over 40 tons.

SUMMARY

This paper briefly summarizes the designs that will be used in comparison of six candidate disposal methods for use in New York State. For further details the reader is invited to contact the New York State Low-Level Radioactive Waste Siting Commission for copies of any of the references.

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14-2

DOE'S PLANNING PROCESS FOR MIXED LOW-LEVEL WASTE DISPOSAL*

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ABSTRACT

A disposal planning process was established by the Department of Energy (DOE) Mixed Low-Level Waste (MLLW) Disposal Workgroup. The process, jointly developed with the States, includes three steps: site-screening, site-evaluation, and configuration study. As a result of the screening process, 28 sites have been eliminated from further consideration for MLLW disposal and 4 sites have been assigned a lower priority for evaluation. Currently 16 sites are being evaluated by the DOE for their potential strengths and weaknesses as MLLW disposal sites. The results of the evaluation will provide a general idea of the technical capability of the 16 disposal sites; the results can also be used to identify which treated MLLW streams can be disposed on-site and which should be disposed of off-site. The information will then serve as the basis for a disposal configuration study, which includes analysis of both technical as well as non-technical issues, that will lead to the ultimate decision on MLLW disposal site locations.

BACKGROUND

The Federal Facility Compliance Act (FFCAct) of 1992 requires the Secretary of Energy to develop and submit site treatment plans (STPs) for the development of both treatment capacity and technologies for treating mixed waste for each facility at which DOE stores or generates these wastes. These plans will identify how DOE will provide the necessary mixed waste capacity, including schedules for bringing new treatment facilities into operation. In collaboration with representatives from the States and the National Governors' Association (NGA), DOE has been evaluating candidate treatment options and developing these treatment plans.

Although the FFCAct does not specifically require DOE to address disposal of treated mixed waste, both DOE and the States realize that disposal issues are an integral component of treatment discussions and have representatives working on and discussing disposal issues. DOE established the FFCAct Disposal Working Group (DWG) in June 1993 to work with the States to define and develop a process for evaluating disposal options. The focus of the DWG process and discussions on disposal with the States has been to identify, from among the sites currently storing or expected to generate MLLW, sites that are suitable for further evaluation regarding their disposal capability. Sites considered to have marginal or no potential for disposal activities have been removed or postponed from further evaluation under this process. Remaining sites are being evaluated more extensively to define the technical capabilities of the sites. Ultimately, a number of sites are expected to be technically acceptable for disposal activities.

This paper describes how the disposal planning process was established, major activities from June 1993 to the present, progress to date, and the expected results from this process.

STEP 1: SCREENING PROCESS FOR POTENTIAL DISPOSAL SITES

The process adopted by the DWG for evaluating disposal options is summarized in Fig. 1. In October, 1993, DOE prepared a draft report "Framework for DOE Low-Level and Mixed Low-Level Waste Disposal: Current Overview" (DOE, 1993) describing the history and status of DOE's low-level and mixed -low-level waste disposal and also outlining a disposal planning process. The sites originally evaluated in this process were the 49 sites reported to Congress by DOE in the Mixed Waste Inventory Report (MWIR) (April 1993) as currently storing or expected to generate MLLW. In the initial step of this process, sites that were in geographic proximity were combined into a single site for consideration in subsequent steps. The following sites were combined:

Lawrence Livermore National Laboratory and Sandia National Laboratories, California;

Idaho National Engineering Laboratory and Argonne National Laboratory (West);

Sandia National Laboratories, New Mexico, and Inhalation Toxicology Research Institute; and
Oak Ridge National Laboratory, Oak Ridge K-25 Site, and Oak Ridge Y-12 Site

This grouping reduced the number of sites to 44.

The next step was the establishment of a screening process using exclusionary criteria for the 44 remaining sites. These criteria were developed by reviewing Federal and State laws regarding the siting of waste treatment, storage, and disposal facilities to determine whether any criteria existed which could be considered exclusionary minimum requirements for hosting disposal activities and which could be applied uniformly across sites. It was agreed at a joint DOE/States meeting in Tucson, Arizona, on March 3-4, 1994, that in order to be further evaluated for potential disposal activities, a site:

must not be located within a 100-year floodplain,
must not be located within 61 meters of an active fault, and
must have sufficient area to accommodate a 100-meter buffer zone.

The first criterion is derived from regulatory requirements under the Resource Conservation and Recovery Act (RCRA) and the U.S. Nuclear Regulatory Commission (NRC) which restrict the location of waste disposal facilities. The second criterion is derived from the requirements under RCRA which restrict the location of waste storage, treatment, and disposal facilities. The third criterion is derived from guidance from the U.S. Environmental Protection Agency, U.S. NRC and U.S. DOE concerning the minimum area required to properly operate such facilities.

Application of the three exclusionary criteria identified 18 sites that did not meet the criteria. The results were presented at a March 30-31, 1994, joint DOE/States meeting in Dallas, Texas. At the meeting, it was agreed that the 18 sites would be removed from further evaluation and that DOE would prepare "fact sheets" on the remaining 26 sites to provide additional site-specific information for identifying the strengths and weaknesses of the remaining sites for the purpose of disposal activities.

A predecisional draft of the report "Framework for DOE Mixed Low-Level Waste Disposal: Site Fact Sheets" (Gruebel et al., 1994) was prepared by the DOE and forwarded for comment and review on July 13, 1994, to the State representatives who had been discussing the disposal issues with DOE for further consideration as potential disposal sites. The DOE and the State representatives met on July 26-27, 1994, in Denver, Colorado, to discuss the Fact Sheet report and to consider proposals for elimination of sites from further evaluation. Prior to the meeting, the DWG had reviewed the information in the Site Fact Sheets and evaluated the 26 sites according to the following methodology:

A. The factors contained in the fact sheets were grouped into three categories:

1. Technical Considerationsfactors that represent the technical ability of a site to accommodate the waste disposal facility and minimize the risk of releases of waste constituents from the disposal facility (e.g., precipitation and evapotranspiration, tectonic and volcanic hazard potential, soil stability and topography, flooding potential, groundwater hydrology [depth to groundwater]).
2. Potential Receptor Considerationsfactors that gauge the potential magnitude of consequences in the event a disposal site ceases to function properly (e.g., population, significant groundwater resources, sensitive environment).
3. Practical Considerationsfactors that can potentially affect the development and long-term management of a disposal site (ownership, mission, MLLW storage and generation, regulatory considerations).

B. Each of the 26 sites was evaluated by the DWG according to the three categories. One of three results was assigned for each grouping: the site posed 1) a major problem, 2) a moderate problem, or 3) a minor problem. Major problems were defined as having features or attributes that make developing and operating a disposal facility extraordinarily difficult. Moderate problems were defined as significant problems that could likely be solved with additional efforts and resources. Sites designated as having minor problems were those with neither major nor moderate problems.

Based on the DWG evaluation, during the July 1994 meeting the DOE proposed to the States that 9 additional sites be eliminated from further consideration. The DOE and the States were able to agree that the following sites would be eliminated from further evaluation under this process regarding the sites' disposal capabilities:

Site State

Energy Technology Engineering Center California
General Atomics California
General Electric Vallecitos Nuclear Center California
Pinellas Plant Florida
Site A/Plot M Illinois
Additionally, DOE and the States agreed that due to its geographic proximity, the Knolls Atomic Power Laboratory at Niskayuna, New York, would be merged with the Knolls Atomic Power Laboratory at Kesselring, New York, for purposes of further analysis. The DOE and the States also agreed that the following sites, while not eliminated from further evaluation, would be given a lower priority for further evaluation:

Site	State
Weldon Spring Remedial Action Project	Missouri
Brookhaven National Laboratory	New York
Mound Plant	Ohio
Bettis Atomic Power Laboratory	Pennsylvania

Sites assigned a lower priority for further evaluation had issues that required further consideration, including whether the technical abilities of the site were adequately known, the volume of MLLW that may be generated by the site had been determined, and other arrangements for disposal of the sites' MLLW were adequate. The DOE and the States agreed to further evaluate these sites in terms of their ability to dispose of their own MLLW on-site only if no other options for disposal of their wastes could be identified through the disposal evaluation process. In no case would these sites be considered as a disposal option for wastes from other sites, and could be eliminated from further analysis should sufficient information suggest that their potential for disposal activities is too limited.

STEP 2: EVALUATION PROCESS FOR POTENTIAL DISPOSAL SITES

For the sites not eliminated from further evaluation or assigned a lower priority for evaluation, a more technically detailed analysis (performance evaluation) is being conducted to increase understanding of the strengths and weaknesses of a site's potential for disposal and to better identify what types of disposal activities could or could not occur at a site. The performance evaluation to be conducted for each of the remaining sites entails the collection of site-specific data related to the natural surroundings, geotechnical setting, groundwater and surface water characteristics, and other factors related to the disposal capabilities of each site. The sites being carried forward in this analysis are the following (Fig. 2):

Site	State
Lawrence Livermore National Laboratory, Site 300	California
Rocky Flats Environmental Technology Site	Colorado
Idaho National Engineering Laboratory	Idaho
Argonne National Laboratory	Illinois
Paducah Gaseous Diffusion Plant	Kentucky
Nevada Test Site	Nevada
Los Alamos National Laboratory	New Mexico
Sandia National Laboratories	New Mexico
Knolls Atomic Power LaboratoryKesselring	New York
West Valley Demonstration Project	New York
Fernald Environmental Management Project	Ohio
Portsmouth Gaseous Diffusion Plant	Ohio
Savannah River Site	South Carolina
Oak Ridge Reservation	Tennessee
Pantex Plant	Texas
Hanford Site	Washington

The goal of the performance evaluation analysis is to quantify and compare the limitations of 16 DOE sites for the disposal of MLLW. The objective is, therefore, to use a set of modeling assumptions of sufficient detail to capture major site-specific characteristics and yet be general enough for consistent application at all sites. The results of the performance evaluation will be represented in terms of concentrations of radionuclides in MLLW streams that a site can accept without violating the performance objectives prescribed in DOE Order 5820.2A. This information will be used to evaluate the sites and estimate the types of waste that may be disposed of at a given site. (It is important to note that although a

performance evaluation is planned for the West Valley Demonstration Project (WVDP) site, only on-site wastes will be considered because the WVDP Act prohibits the site from accepting off-site wastes). The performance evaluations of the 16 sites were initiated in August 1994, and are scheduled to be completed in June 1995. A progress report on the performance evaluation task will be issued during the PSTP process and a final report documenting the performance evaluation results will be issued in October 1995.

An important component in the performance evaluation step is the involvement of a Senior Review Panel. The Panel is established to provide independent review of the PE analysis. The panel consists of six highly reputable technical experts in various aspects of waste disposal. Five members of the panel were selected by DOE , and one member was selected by the National Governors Association. Two technical reviews have been performed to date, and two more are planned before the completion of the PE analysis.

STEP 3: CONFIGURATION STUDY

As illustrated in Fig. 1, much progress has been made in the disposal planning process. However, much work remains to be completed. The following steps outline the future activities after the performance evaluation that need to be completed in order to make an informed decision about the disposal of DOE MLLW. Coordination with the States will continue in the next steps in order to gain stakeholder input and to resolve issues at the earliest possible stage.

Develop Estimates of Waste Volumes and Radionuclide Concentrations in treated MLLW residues

Once treated methods for the MLLW waste streams are finalized through the FFCAct process, estimates of treated residue volumes and radionuclide concentrations in the treated residues will be developed for all waste streams. These estimates are needed to compare to the radionuclide concentration guidelines derived in the performance evaluation.

Compare Radionuclide Concentration in Treated Residue Estimates to Radionuclide Concentration Guidelines Derived in the Performance Evaluation

Radionuclide concentrations for each treated waste streams will be compare to those disposal values derived in the performance evaluation. Analysis will identify sites with on-site disposal capabilities and the treated waste streams they will be able to dispose on-site. Also the analysis will evaluate off-site DOE and commercial disposal capacity for those treated waste streams which cannot be disposed on-site.

Develop Sample Configuration for Disposal of Treated Residuals

Sample complex-wide configurations will be developed for the disposal of treated MLLW residuals. These configurations will take into account such technical issues as compatibility of radionuclides and capacity to handle projected residual volumes. Other types of issues will be weighed during the configuration discussions such as transportation distances and costs. Information from the DOE EM Programmatic Environmental Impact Statement (PEIS) will be included here. In addition, input from the States will be an important component in this step.

Develop a Draft Disposal System Configuration

Using the sample configurations as a basis, in coordination with NEPA requirements and with State and stakeholder input, a draft disposal system configuration will be developed. This configuration will be the basis for determining future funding and schedules for proposed disposal facilities.

INTEGRATION WITH THE STP PROCESS

The FFCAct does not require disposal to be included in the STPs; however, given the complex issues involved, DOE recognizes the importance of State input to facilitate resolution of issues related to disposal. Information on the disposal planning process is provided in the PSTP to continue to involve the States and to make them aware of DOE's continued work on this issue. A progress report describing this planning process and, in detail, the performance evaluation methodology is being prepared as part of the PSTP. Results of the performance evaluation for the 16 sites will be provided to the States when Compliance Orders are issued in October, 1995.

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14-4
CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITSPROGRESS REPORT ON FIELD EXPERIMENTS AT A HUMID REGION SITE, BELTSVILLE, MARYLAND
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ABSTRACT
This study's objective is to assess means for controlling water infiltration through waste disposal unit covers in humid regions. Experimental work is being performed in large-scale lysimeters 21.34m 13.72m 3.05m (75 ft 45 ft 10 ft) at Beltsville, Maryland. Results of the assessment are applicable to disposal of low-level radioactive waste (LLW), uranium mill tailings, hazardous waste, and sanitary landfills.
Three kinds of waste disposal unit covers or barriers to water infiltration are being investigated: 1) resistive layer barrier, 2) conductive layer barrier, and 3) bioengineering management. The resistive layer barrier consists of compacted earthen material (e.g., clay). The conductive layer barrier consists of a conductive layer in conjunction with a capillary break. As long as unsaturated flow conditions are maintained, the conductive layer will wick water around the capillary break. Below-grade layered covers such as 1) and 2) will fail if there is appreciable subsidence of the cover, and remedial action for this kind of failure will be difficult. A surface cover, called bioengineering management, is meant to overcome this problem. The bioengineering management surface barrier is easily repairable if damaged by subsidence; therefore, it could be the system of choice under active subsidence conditions. The bioengineering management procedure also has been shown to be effective in dewatering saturated trenches and could be used for remedial action efforts. After cessation of subsidence, that procedure could be replaced by a resistive layer barrier or, perhaps even better, by a resistive layer barrier/conductive layer barrier system. The latter system would then give long-term effective protection against water entry into waste without institutional care.
As mentioned in the preceding paragraph, a bioengineering management cover might well be the cover of choice during the active subsidence phase of a waste disposal unit. Some maintenance is required during that period. Final closure, using geological materials, could follow cessation of subsidence. No further significant maintenance would then be required. If the geological material used is merely a clay barrier to water infiltration, the cover will be "sensitive" to imperfect construction or degradation by penetrating roots. The roots will die and decay, causing markedly increased permeability of the clay with the passage of time. A system using a conductive layer under the clay layer as a water-scavenging system will, in comparison, be "robust." Roots will still degrade the clay layer but will not degrade the scavenging layer. A root hole through the conductive layer will be analogous to a hole through a wick. It will do no significant damage. The combination of a resistive layer with a conductive (scavenging) layer underneath is thus less dependent on perfect construction techniques and will be resistant to damage by root invasion. In the absence of subsidence such a system should function effectively for millennia.
Another very useful application of the resistive layer barrier/conductive layer barrier system would be to protect an earth-mounded concrete bunker disposal unit. In that case, the barrier system would shield the concrete from exposure to flowing water. The resulting stagnant alkaline film of water would tend to protect the concrete from degradation over a long time period. Similarly, a resistive layer barrier/conductive layer barrier system could be used to protect high-level waste. If high-level waste were disposed of in fractured rock, this system could be used to divert possible fracture flow water around the waste.

INTRODUCTION

Infiltration of water into the waste is the foremost problem associated with near surface disposal of low-level radioactive waste (LLW). Up to this time, disposal unit covers have generally been constructed from soil materials. In humid areas, these soil or clay covers have generally proved less than satisfactory; often, the cover itself has served as the principal pathway for water entry into the waste (1). Water infiltrating to buried wastes, contacting the wastes, and then exiting the area can reasonably be expected to be the most important of radionuclide transport agents. Some radionuclides, such as tritium (present as tritium oxide) and those present in anionic form, will essentially move with the flow of water; others, present as multivalent cations, will move much more slowly, but all will move to a greater or lesser degree. Clearly then, it is advantageous to reduce water infiltration to buried waste to as low a level as reasonably achievable. It is the purpose of our work to examine and demonstrate various approaches for achieving that goal.

Three kinds of waste disposal unit covers or barriers are being investigated in this work:

1. Resistive Layer Barrier
2. Conductive Layer Barrier
3. Bioengineering Management

The resistive layer barrier is the well-known compacted clay layer and depends on compaction of permeable porous materials to obtain low flow rates. A simplified model is shown in Fig. 1. Flow through porous media is described by Darcy's law (2). Investigations on flow through such layers have gone on for over 100 years, so further progress in this area can be expected to be slow.

The conductive layer barrier (1) is a special case of the capillary barrier (3). Use is made of the capillary barrier phenomenon not only to increase the moisture content above an interface, but to divert water away from and around the waste. During such diversion, water is at all times at negative capillary potential or under tension. A simplified model is shown in Fig. 2.

This system consists of a porous medium underlain by a capillary break (rock layer). Infiltration barriers such as a conductive layer barrier or a clay layer barrier (or a combination thereof) must fail if subjected to substantial shearing caused by waste subsidence. Reestablishment of a layered system after subsidence failure is a difficult undertaking and is exacerbated by the increasing complexity of the layered system. The failure potential (see CAUTION) of in-ground layered systems during the subsidence period argues for development of an easily repairable surface barrier for use during that period. To that end a procedure called "bioengineering management" was developed (4). The bioengineering management technique utilizes a combination of engineered, enhanced run-off and moisture-stressed vegetation growing in an overdraft condition to control deep water percolation through disposal unit covers. An artist's conceptual drawing is shown in Fig. 3.

EXPERIMENTAL AND DEMONSTRATION

In this section we will discuss experiments being conducted in large-scale lysimeters at a humid region site in Beltsville, Maryland (see Fig. 4).

Bioengineering Management

In bioengineering management the necessary run-off is provided by features installed at or above the soil surface rather than within the profile. The procedure, described by Schulz et al. (4), was designated bioengineering management. Its principal advantage is that subsidence can easily be managed by relatively simple, inexpensive maintenance of the above-ground features rather than by difficult reconstruction of below-ground layers. It should be noted that, after a length of time sufficient so that the organics have decayed and the waste containers have completely failed, subsidence will cease and a layered system could be then installed which could last over geological time periods.

In essence, the bioengineering management technique utilizes a combination of engineered, enhanced run-off and stressed vegetation in an overdraft condition to control deep water percolation through disposal unit covers. To describe it further: if a waste burial site is selected so that incoming subsurface flow is negligible, then precipitation is the sole source of input water. In a simplified model, that water has three possible fates: 1) evapotranspiration, 2) run-off, and 3) deep percolation. Evapotranspiration has a definite limit, governed by energy input. Ideally, deep percolation should be zero, leaving only the run-off component

available for unlimited manipulation. Positive control of run-off becomes difficult with the use of compacted porous media trench caps as the sole barrier to water infiltration. The compacted material tends to become more permeable with the passage of time, due to fractures caused by waste subsidence and from the inexorable process of root growth, followed by death and decay of the roots, thus creating water channels. Evapotranspiration cannot then use all of the infiltrating water, and water percolates downward to the waste. As stated before, evapotranspiration has a theoretical maximum dictated by solar energy input to the system; only run-off remains available for nearly unlimited management. This run-off can be surface or subsurface, as long as it occurs before water reaches the waste.

Surface run-off can be managed to as high as 100 percent by means of a perfect, leak-proof roof, which is expensive and hard to guarantee. Alternatively, adequate but not total run-off can be engineered rather inexpensively by using an impermeable ground cover over part of the surface to achieve high and controlled levels of run-off. Vegetation planted between areas of impermeable cover will extend over the cover to intercept incoming solar energy to evaporate water. Roots will extend under the cover in all directions to obtain water.

Such a system can be visualized by imagining a supermarket parking lot, where trees are planted in islands, surrounded by concrete curbs, within an extensive paved area. In this case, the trees are maintained in a drought environment due to the small soil surface available for infiltration of precipitation. The paving, along with the curbing around the trees, causes run-off of most of the precipitation. Aboveground, the tree's branches and leaves extend over the parking lot and intercept incident solar energy. Beneath the surface, the roots, in a drought state, explore outward under the paving for any available water. Utilizing this concept, it should be possible, by combining engineered run-off with vegetation, to maintain the soil profile in a potential overdraft condition on a yearly basis.

Initial investigations of the bioengineering management technique were carried out in lysimeters at Maxey Flats, Kentucky. Results obtained in seasonal 1984-1985 and 1985-1986 were reported by O'Donnell et al. (5). In that work, a fescue grass crop was used with an engineered cover of stainless steel. Following seasonal 1985-1986 the grass cover was removed, a new stainless steel engineered cover was constructed and Pfitzer junipers were planted in the lysimeters. After the junipers were established, percolation data were again collected in 1988 and reported by Schulz et al. (6). The woody junipers were excellent in preventing deep percolation of water in the lysimeter.

The encouraging initial results obtained in the Maxey Flats lysimeter experiment led to the establishment of a large-scale field demonstration at Beltsville, Maryland (Fig. 4). Figure 5 is a photograph of lysimeter 1, bioengineering management, taken in December, 1994, eight years after planting of the Pfitzer junipers. Alternating panels of aluminum and fiber glass were used as the hard cover. These plots, or lysimeters, are 21.3m (70 ft) long by 12.7m (45 ft) wide, and the bottoms are 3.05m (10 ft) below grade. Figure 6 shows a side view of construction details of lysimeters 1 and 2 (bioengineering management). The only difference between the two was the initial water level in the lysimeters. The water level was 90 cm above the bottom of lysimeter 1 and 190 cm above the bottom of lysimeter 2. The water level in the lysimeter simulates the water table in a flooded disposal cell. In addition to the two bioengineered lysimeters, two reference lysimeters (3 and 4) were initially constructed. They were similar to the former, except that they were merely planted with fescue grass. No hard cover was present, but surface slopes were similar to the two bioengineered lysimeters (i.e., a slope of 1:5). Performance data for the reference lysimeters are given in Fig. 7.

The water level in the two reference plots or trenches (lysimeters 3 and 4) rose until it was near the surface. At that time, water was pumped from the lysimeters to keep them from running over. The graphs of the water tables (i.e., water levels) in the bioengineered plots (lysimeters 1 and 2) show an entirely different story, as evidenced in Fig. 8. In both cases, the water table was eliminated. It appears that the bioengineering approach could prevent water infiltration to a disposal unit. It also could be used for a remedial action in dewatering existing problem sites such as Maxey Flats.

On February 4, 1988, lysimeter 4 was pumped out to prevent overflow. It was then discontinued as a reference lysimeter and converted to a rock-surfaced, resistive-layer barrier plot. Lysimeters 1 and 2 (bioengineered) and lysimeter 3

have been continued. A summary of run-off, evapotranspiration, and pumping from those three lysimeters is given in Fig. 9.
Figure 9 shows that there was very little run-off from the grass covered plot. Most of the precipitation was disposed of, via evapotranspiration, by the fescue crop, but this was not adequate to prevent the rise of the water table. Table I gives the run-off, evapotranspiration, and deep percolation in the bioengineered plots during the past seven years. There was no deep percolation during this period. Until seasonal 1993-1994 the evapotranspiration had been rising annually, probably as a result of the greater vegetative canopy intercepting a greater percentage of the precipitation. In 1988, 1989, 1990, 1991, and 1992 the runoff percentages were 80, 74, 70, 67 and 63, respectively (7). In 1993, the run-off decreased to 61% of the precipitation. In 1994 the run-off remained the same as in 1993. During 1989, the water table was completely eliminated in both plots (Fig. 8).
In addition to rainfall, run-off, and evapotranspiration measurements discussed above, neutron probe soil moisture measurements have been made continuously to monitor soil moisture changes in all six lysimeters depicted in Fig. 4. The neutron probe measurements will indicate whether there is a gain or loss of moisture from the soil profile, or perhaps a steady-state situation where there is little or no net gain or loss of soil moisture during a year.
A steady-state situation with relatively constant moisture "dry" soil above the waste would be highly desirable with a bioengineered cover. There would then be a large safety margin to protect the waste from infiltrating water.
Neutron probe apparatus, as supplied by the manufacturer, is calibrated against moisture measurements in sand. Such calibration is of unknown accuracy when applied to soil measurements. For this reason, the probe was calibrated using the same soil as in the lysimeters. Six hundred and twenty-eight kilograms (1,400 lbs) of soil were placed in a weighing lysimeter, and measurements were made over a seven year period. Calibration data obtained using the weighing lysimeter are given in Table II. The resulting curves, depicting the factory calibration and the weighing lysimeter calibration, are given in Fig. 10. It is evident that use of the factory calibration on sand would result in a very large error in soil moisture determination.
Results of some neutron probe measurements are shown in Fig. 11 for bioengineered lysimeters 1 and 2. The data are plotted as volumetric moisture content, as a function of soil depth, on specific dates. Only nine widely spaced measurement dates are shown, for clarity. From inspection of the figure it is seen that, at the start of the experiment in July, 1987, the moisture content of the soil increased with depth until the water table was reached, then became constant. By July, 1989, the water table had been eliminated from both lysimeters, and the soil profiles were drying out. However, the soil moisture content, although much lower in the soil profile than in July, 1987, still increased with depth. This same relationship was still evident November, 1994, although the soil profile had become still drier.
Figure 12 shows the moisture content of the soil profiles in lysimeters 1 and 2 at the end of each seasonal year. Following the complete removal of the water tables during the 1987-1989 period, the soil profiles were dried out further during the ensuing years. However, an unanticipated result turned up in lysimeter 1 at the end of seasonal 1993-1994. The moisture content of the soil profile increased slightly. To shed light on that result, the moisture content in the soil profiles at four depths were plotted monthly along with monthly rainfall data (Fig. 13a, b, c, d, e, f, g, h). Here we see seasonal cyclical variations in moisture content in the soil profiles, with peak moisture concentrations occurring in the early spring, following periods of significant rainfall and minimal evapotranspiration. That cycling is both obvious and expected. What was totally unanticipated was the increase in the moisture peaks in each of the last three years in lysimeter 1. The increasing amplitudes of the moisture curves do not appear to be a result of rainfall variations, nor are they present in lysimeter 2. During the coming year we shall see if the aforementioned trends continue, and if so, we will try to determine why. Increased stemflow resulting from water funneling downward with increased plant canopy size is conceivable, but if that is the case, why do we not have increasing amplitude of the seasonal moisture curves in lysimeter 2? Although the increasing amplitude of the moisture curves for the soil profile in the spring of 1992, 1993, and 1994 in lysimeter 1 are a concern, the results to date indicate that bioengineered closure, as described in this experiment, would maintain the cover

over buried waste in a "dry" steady-state condition. This would not only prevent water from percolating down to the waste, but would do so with a large safety factor.

Fig. 13. Plot of moisture content at the 61, 122, 183 and 244 cm levels as a function of time in lysimeters 1 and 2. Bar graphs of monthly rainfall superimposed. Moisture peaks occur in springtime, following periods of low evapotranspiration and the soil profile is again dried out in following periods of high evapotranspiration. This process prevents water from accumulating in the soil profile and percolating to the waste.

Resistive Layer Barrier

As previously mentioned, on February 4, 1988, lysimeter 4 was pumped out, discontinued as a reference lysimeter, and converted to a rock-surfaced resistive-layer barrier plot. The primary reason for constructing that particular cover is the likelihood of such covers being used for uranium mill tailings. An end view of that plot or lysimeter is shown in Fig. 14. This lysimeter was completed in the fall of 1988 and data collection (measuring performance) has begun. The most important information to be gained here will be the relative weighing of the advantages and disadvantages of rock surface vs. a vegetated surface.

In addition to the UMTRA or rock-surfaced resistive-layer barrier plot, a vegetated resistive layer barrier plot was constructed. The primary purpose of this plot is for comparative measurements. Essentially, this plot is similar to the rock-surfaced plot except that topsoil replaces the rock layer, and the plot is planted with fescue grass. A diagram of this plot is given in Fig. 15.

In Fig. 9, the fate of precipitation in the UMTRA and grass-covered, clay-layer lysimeters is given. There was more than twice as much run-off from the rock-covered plot as from the grass-covered plot. Although the data show no deep percolation through the clay layers to date in either lysimeter, there is little indication as to how much safety margin has been offered. Nor is it known how consistently such near-perfect clay barriers would be installed in a routine operation. That remains a problem for future consideration.

Another concern is the possible drying out of clay barriers. If this were to happen, the clay layer would not be as efficient a barrier for preventing radon escape as planned in the UMTRA application. In addition, drying out of the clay layer could lead to cracking, leading to subsequent leakage prior to resealing by wetting. Figure 16 gives the volumetric moisture content of clay in the rock-covered (lysimeter 4) and the grass-covered (lysimeter 6) plots. In no case did the clay layer dry out significantly. On the contrary, in the UMTRA or rock-covered plot, which was devoid of vegetation, there was a slight increase in moisture content with time, suggesting that some leakage of water through the clay layer may occur in the future. Lysimeter 6 has a clay layer and a grass cover. In this case, no increase in moisture content has been observed. On the contrary, to date the moisture content of the clay layer seems to be in a rather steady state, taken over the 6 year period of measurement.

Conductive Layer Barrier

If we consider the case of water flowing downhill in an unsaturated porous medium, we have the case shown in Fig. 17. The "holes" shown in the diagram could be a rock layer, affording a capillary break or capillary discontinuity (Fig. 18). Under appropriate conditions, water everywhere in these cross-sections will be under tension, and there will be no leakage. This might then serve as an excellent means of protecting waste by conducting water around the waste. Figure 17 simulates a conducting porous medium, such as a fine sandy loam soil, lying smoothly on top of a rock layer. Problems with water flow under saturated conditions could certainly arise where a less than smooth surface ends up being constructed as depicted in Fig. 19. That is, what happens if imperfections are constructed so that "pockets" of soil extend down into the rock layer? Figure 19 represents that case. Again, there will be no leakage, provided conditions are such that the water in all parts of the conductive layer remains under tension.

The big question is, can conditions required to maintain the necessary soil water tension be practically maintained while using this procedure to effectively protect waste disposal units? To answer this question the apparatus schematically depicted in Fig. 20 was constructed, i.e., "a soil beam." Several miniature soil beams (Fig. 21) were constructed for use in the laboratory so that a variety of candidate conductive-layer materials could be quickly evaluated.

A number of materials were evaluated using the miniature soil beams. It was quickly established that it would be necessary to construct a resistive layer barrier above the conductive layer barrier to have a practical system. The standard was set that the resistive layer barrier have an easily achievable conductivity of not greater than 10-6 cm/sec. On this basis it was found that material such as fine sandy loam could provide an effective conductive layer barrier, that is, conduct around the waste 100% of water percolating through the resistive layer. However, the measurements showed that such materials would not provide the desired (factor of 10) safety margin.

Further investigations turned up a material, diatomaceous earth, that would fit these requirements. Measurements of tension vs. distance of flow are shown in Fig. 22.

The results of this experiment in the 137 cm (4.5 ft) long beam suggest that, as long as the flow rate is no greater than 4.2 10-4 cm/sec, the soil water will remain under tension regardless of the soil beam length. These results show that with the use of diatomaceous earth for the conductive layer and following the easily achievable standard set above for the resistive layer, it should be possible to construct a barrier that would allow no water leakage to a waste disposal unit. However, before final selection of the diatomaceous earth as the conductive layer material, we believed it to be prudent to conduct tests in a large-scale soil beam. The large beam, shown in Fig. 23, has a soil beam length of 6.4 m (21 ft). As shown in Fig. 24, a matric potential of about -15 to -20 cm of water is maintained over the entire 6.4 m length of the beam when the flow rate does not exceed 3.1 10-4 cm/sec.

The studies carried out in the large soil beam closely confirmed the data obtained in the miniature beam. Accordingly, diatomaceous earth was used as the conductive layer material in the demonstration lysimeter (lysimeter 5). It has been estimated that purchasing and shipping the diatomaceous earth to a job site any place in the United States will add about $0.50 per ft3 of disposed waste. This is over the cost of using locally obtained soil, and based on waste being 3.05 m (10 ft) deep.

After the time-consuming task of selecting the conductive layer material was accomplished, a resistive layer barrier over a conductive layer barrier was constructed in lysimeter 5. It was completed in January, 1990. A local clay from Beltsville, Maryland, the Christiana Clay, was selected as the resistive layer barrier. Testing has shown this material more than meets specifications. A cross-section of the cover system is shown in Fig. 25.

Performance of this cover is shown in Figs. 9 and 16 (lysimeter 5). Until seasonal 1993-1994 the cover system was 100% effective in preventing water movement downward through the cover. In seasonal 1993-1994, 0.13 cm (0.05 in) of water passed through the cover to the pan shown in Fig. 14. Although that amount is an extremely small percentage of the total rainfall, in theory no water should have percolated through the cover to pan. It is possible that the cover system was compromised by the instrumentation installed to measure performance. During the coming year we will try to find out if this is merely an experimental anomaly.

Further Studies of Conductive-Layer Materials

For The Beltsville study, diatomaceous earth was selected for the conductive-layer material, based both on performance and cost considerations. Based on these two considerations only, diatomaceous earth would still be the material of choice, particularly since it has a much lower bulk density than sand and is therefore less expensive to ship. However, the engineering properties of sand are better known, thus sand may be more attractive to some installers. Therefore, we have been conducting further studies with various sands. Results of studies of the unsaturated flow characteristics of four different sands are given in Fig. 26. All these sands exhibit unsaturated flow rates that are about twice that of the diatomaceous earth at any given negative matric potential. The particle size distribution of the four sands is given in Table III. The mortar sand, for example, had the narrowest particle size range, and the foundry sand had the widest particle size distribution, although the particle size distribution did not have an important effect on the flow rates reported in Fig. 26. The Nevada dune sand and the Kelso dune sands are from large eolian deposits in the Nevada and California deserts, respectively. The Kelso deposit has been mined commercially. This work on the unsaturated flow characteristics of various materials is ongoing. Hydraulic properties will be studied over a larger range of matric potentials, further deposits will be located

and investigated, and these investigations will be described in a future report.

APPLICATION

The three procedures described in the Introduction may be used singularly or in combination to protect disposal units from percolating water. The principles apply equally to above-ground or below-ground disposal. For example, a combination of covers 1) and 2), described in the Table III. Particle size distribution of the four sands used in unsaturated flow studies reported in Fig. 26.
Introduction, could be ideal for a stabilized, shallow land burial facility, whether it is above or below ground; e.g., the subsurface disposal could be in below-ground vaults, and the above-ground disposal units could be earth-mounded concrete bunkers. A combination of a resistive layer over a conductive layer in a concrete bunker or above-ground application is shown in Fig. 27. The resistive (clay) layer is the primary barrier. The small amount of water passing through the clay layer will be diverted around the concrete bunker by the conductive layer. This cover over the concrete bunker can, in theory, be 100% effective, shielding the bunker from exposure to flowing water. This would result in a film of stagnant alkaline water at the gravel/concrete interface. The presence of this high pH, stagnant water would tend to protect the concrete from degradation over a long period. A resistive layer above must leak somewhat due to the imperfections in construction when using compacted porous material (clay).
The bioengineering concept could be advantageous for either a tumulus or shallow land burial unit that would be likely to exhibit subsidence. If desired, and after subsidence has ceased, a combination of covers 1) and 2) could be constructed with geological materials to give extremely long-term isolation without further maintenance (8). Another possible application of a combination of covers 1) and 2) described in the Introduction is shown in Fig. 28. Here, high-level waste is emplaced in a tunnel excavated in rock. If a fracture were present in the rock, and fracture flow occurred, the combination of a resistive layer and a conductive layer could provide excellent isolation of the waste from flowing water. Figure 29 depicts an application where only very low flow rates need be protected against (essentially, dropwise fracture flow). Here, the system could be simplified so that only a conductive layer with a capillary break is necessary.
Fig 26.Unsaturated flow characteristics of four sands. Soil water tension at various flow rates, measured in mini-soil beam shown in Fig. 21. Tension vs. horizontal distance from discharge point. Results suggest that at rates of about 10-3 cm/sec or less, water would remain under tension at any beam length. Slope of beam is 1:5.

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14-5

WATER INFILTRATION CONTROLS AT THE WEST VALLEY LOW-LEVEL WASTE DISPOSAL AREA

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ABSTRACT

The New York State Energy Research and Development Authority (NYSERDA) is implementing technologies to eliminate accumulation of leachate in low-level radioactive waste (LLRW) disposal trenches at the shut-down New York State Licensed Disposal Area near West Valley. Results to date show that installation of a slurry wall and a geomembrane cover on two trenches has stopped leachate accumulation in these trenches. Based upon this success, NYSERDA is preparing to cover the remaining trenches to reduce water infiltration into the trenches and to prevent additional leachate accumulation.

In addition, NYSERDA has initiated a pilot project to demonstrate the effectiveness of bioengineering management as a potential technique for controlling leachate accumulation in the trenches. The pilot project is being conducted on an existing LLRW shallow land burial trench.

INTRODUCTION

The New York State Energy Research and Development Authority (NYSERDA) recently installed water infiltration control technologies to remedy water management problems caused by a wet climate and a silty-clay soil, at the shut-down, commercial, Low-Level Radioactive Waste State Licensed Disposal Area (SDA) in Western New York. NYSERDA installed a slurry wall and geomembrane cover to address the immediate problem of water accumulation in two shallow burial trenches, and initiated a pilot project to assess the feasibility of a long-term bioengineering management program. This paper discusses our experiences leading to selection and implementation of these technologies, presents the results, and provides an outlook on our future applications.

The SDA occupies approximately 6.07x104 square meters of the Western New York Nuclear Service Center (Center). The Center, site of the world's first commercial nuclear fuel reprocessing plant, is located 50 kilometers south of Buffalo, New York, near the hamlet of West Valley. Nuclear Fuel Services Company, Inc. (NFS) operated and maintained the Center under a lease agreement with New York State. Currently, NYSERDA holds title to and maintains the 1.35x107 square meter Center on behalf of the people of New York State with the exception of approximately 8.09x105 square meters under exclusive use and possession of the United States Department of Energy (DOE) to perform the West Valley Demonstration Project (WVDP).

Description of the SDA

From 1963 to 1975 NFS placed approximately 6.8x104 cubic meters of packaged commercial low-level radioactive waste (LLRW) in two sets of parallel trenches referred to as the north and south areas; each containing seven trenches, numbered 1 through 7 and 8 through 14, respectively (Fig. 1). Typically the trenches are approximately 170 meters long, 6 meters deep, and vary in width from 6 meters at the bottom to 11 meters at the top, except for two specially constructed trenches. Trench 6 consists of a series of holes used for the disposal of high-specific activity wastes that required immediate shielding; and Trench 7 is a narrow, shallow trench in which the waste was encased in concrete. Wastes placed in the trenches were covered with the soil excavated during trench construction.

The trenches are constructed in the Lavery till, described as a silty-clay or clayey-silt, pebble-cobble till. It has minor amounts of small, random, discontinuous lenses of stratified sand, gravel, silt and rhythmic clay-silt laminations.(1) This area was used for disposal trenches because the high clay content and high degree of over-consolidation of the Lavery till make it virtually impermeable at depth. In-situ tests in bore holes indicate that the material has a hydraulic conductivity of about 1x10-8 centimeters per second (cm/sec). However,

samples taken from bore holes and excavations show that the upper three meters of the till is weathered as a result of desiccation. The weathered till is highly fractured and is generally more permeable than the unweathered till.

Early Water Problems and Management Activities

Because of the humid environment in the region and the very low permeability of the unweathered till, water enters the trenches by permeating through the upper weathered till or the trench covers and tends to accumulate there. Water that enters the trenches and contacts the disposed wastes is called leachate, and mobilizes the radionuclides and other waste contaminants contained in the wastes. In 1975, leachate levels rose above the original ground surface and seeped laterally through two trench caps into an adjacent brook at the north end of the SDA. NFS immediately ceased disposal operations and no waste has since been disposed of in the SDA. Following the seepage incident, NFS pumped and treated leachate from the disposal trenches. To minimize infiltration of water, NFS rehabilitated the north disposal area trench caps, incorporating a number of changes based upon experience with the southern trench cover design, which appeared to be more successful in limiting infiltration of water into the trenches.(2) However, abrupt increases in the leachate levels of Trenches 11 through 14 in the south area occurred starting in 1978. These increases were also attributed to seepage through desiccation cracks in the cover. Consequently, the covers of Trenches 11 through 14 were reworked to reduce this problem.

SITE MANAGEMENT BY NYSERDA

Monitoring and Maintenance Activities

NYSERDA assumed possession of the SDA in 1983 along with responsibility for surveillance and maintenance of the Center. During the summer of 1986, leachate levels in Trench 14 rose at a significantly faster rate than previously observed. An investigation confirmed the presence of a relatively large, permeable, stratified sand and gravel body that was projected to intersect Trench 14 near the southern end and came very close to the ground surface near the western boundary of the SDA. Historical information suggested that the sand and gravel body extended eastward beyond Trench 13 to Trench 12. A subsurface concrete barrier wall was installed immediately west of Trench 14 to block groundwater from entering the trench and to create a barrier to allow safe excavation of the sand and gravel unit, which was replaced with compacted silty-clay till from a nearby borrow area. Following this work, the Trench 14 water levels rose at a much lower, but steady rate until sudden increases in the Trench 14 leachate level occurred in late 1990 and early 1991, followed by a dramatic rise in the Trench 13 water level (Fig. 2). This occurrence repeated providing evidence for a suspected hydraulic interconnection through a sand and gravel body between Trenches 13 and 14 at the leachate elevation reached in Trench 14.

Hydrologic Investigation

In response to the initial increases in Trenches 13 and 14, NYSERDA initiated a hydrologic study to find the source and pathway of this water.(3) The study evaluated the potential for horizontal migration of groundwater into Trenches 13 and 14, as well as vertical infiltration through the cap of these trenches. A total of 28 shallow- and deep-screened piezometers were installed in logged soil bore holes south and west of these trenches. Precipitation, groundwater levels and trench water levels were monitored concurrently.

The two-year hydrologic study produced the following conclusions:

Heavy rainfalls contributed to abrupt leachate level increases in Trenches 13 and 14.

Sinkhole formation in the caps suggest that cracks in the cap serve as pathways for vertical water infiltration.

Groundwater, generally moving horizontally from west to east through the weathered till, contributed to leachate accumulation in Trench 14, while the unweathered till does not appear to contribute.

Consistently high water levels observed in the piezometers near a previously identified kettle hole deposit around the northwest end of Trench 14 indicate a possible pathway for groundwater to flow into the trench.

INFILTRATION CONTROLS

Selection of Engineering Solutions

An evaluation of alternatives to reduce or eliminate the influx of water into Trenches 13 and 14 was prepared while the hydrologic investigation was in

progress.(4) Alternative methods evaluated to minimize lateral flow included a slurry wall, soil freezing, and sheet piling; while those evaluated to limit vertical permeation included soil cover improvement, a geomembrane cover, and bioengineering management system.

Because the hydrologic study did not identify a single source for water infiltration, a combined solution consisting of a slurry wall to divert horizontal groundwater flow in combination with a geomembrane cover to obstruct the vertical seepage of precipitation through the cover was developed. The slurry wall was selected because of its natural properties, relative ease of implementation, construction cost, and absence of maintenance or operating costs. A geomembrane cover was selected because past soil cover improvement work only had limited success and bioengineering management (which has not gone beyond the research in test lysimeters) may not provide the immediate results that were needed. A very low-density polyethylene (VLDPE) geomembrane material was selected over other geomembrane materials for its combined flexibility, resistance to cold stress, resistance to tearing or stress cracking, and relative cost.

Construction

The slurry wall was aligned to run along the west (upgradient) side of Trench 14 and partially wrap around both the north and south ends (Fig. 1). The objective for the project was to cut off all desiccation cracks, fissures, or lenses of stratified coarse deposits that could transmit water to the trench. The 276-meter long slurry wall is approximately 0.9-meters wide and 9-meters deep, extending below the bottom of the lowest trench.

Slurry wall construction began in early September 1992, and was completed in three weeks. The soil excavated from the slurry trench was mixed with bentonite clay and water, then bulldozed into the trench to form a highly impermeable barrier to horizontal water flow. Mix design testing prior to construction demonstrated that adding one percent of dry bentonite to the native till would result in a barrier permeability in the area of 1 x 10-8 cm/sec. Due to the very soft consistency of the slurry wall, it was covered with soil to provide a suitable base for and to facilitate safe installation of the geomembrane cover over the top of the wall.

Surface preparation prior to installation of the geomembrane cover included stripping the grass cover from the trench caps, regrading areas to accommodate cover installation, and excavating the perimeter anchor trench for the membrane cover. The VLDPE cover extends from the center line of Trench 12 across Trenches 13 and 14 and also covers the slurry wall. It was extended to the center of Trench 12 to terminate at the crest of a slope for drainage considerations. The perimeter anchor trench is filled with stone, to also serve as a collection channel conveying precipitation runoff away from the trenches. Due to heavy rain and snow in October and November, installation of the geomembrane cover was completed in June 1993. The VLDPE cover was left exposed to limit the impact on the closure options, to allow closer monitoring for possible sinkhole formation, and to allow visual observation of the membrane for areas of possible leakage.

Results

Triaxial permeability test results of the completed slurry wall ranged from 1.1 to 2.9x10-8 cm/sec. After installation of the slurry wall, the Trench 14 water level immediately stabilized (Fig. 2). Since installation of the geomembrane cover, the water levels in Trenches 12, 13, and 14 have remained stable. Special slit trench monitoring wells, installed to monitor for groundwater mounding outside the slurry wall, have not exhibited a discernible build up. If ground water were to mound behind the wall, it could overtop the slurry wall and again provide a potential for infiltration into the trenches.

BIOENGINEERING MANAGEMENT PILOT PROJECT

Project Initiation

The purpose of the bioengineering management system is to change the humid climate conditions (where precipitation exceeds the evapotranspiration (ET) rate) to dry climate conditions (where potential ET exceeds the precipitation entering the cover soils), thus eliminating deep permeation of water through the trench cover soils into the waste. It has been demonstrated to be a successful means of controlling deep water percolation in test facility lysimeters at the U.S. Department of Agriculture's Experiment Station in Beltsville, Maryland.(6) Because it has not been previously applied to an existing disposal site, NYSERDA decided to proceed with a pilot project demonstration at the SDA. The objective of the pilot project is to

evaluate the effectiveness of this system in preventing vertical water infiltration through the trench cover soils and in dewatering the silty-clay trench cover soil, in the cool, humid, Western New York climate.

In Autumn 1993, NYSERDA installed a bioengineering management system on Trench 9 at the SDA. This trench was selected because its construction is typical of trenches at the SDA with a 2.4 meter-thick, compacted clay cover, and it has a consistent rate of water accumulation, that is believed to result from infiltration through the clay caps. Trench 9 does not appear to be influenced by groundwater or an interconnection with other trenches, making it suitable for this project.

Fiberglass panels mounted on wooden frames were placed perpendicular to the length of the trench to provide a hard cover for positive and predetermined runoff (approximately 80-90 percent of precipitation). Shallow rooted Pfitzer junipers, which are hardy in this region, were planted between the panels to provide for ET of precipitation not removed by the panels (Fig. 3).

Environmental monitoring is used to assess existing SDA environmental conditions and identify environmental changes or trends which occur in response to this pilot project. Project performance will be assessed over a five-year period by evaluating the following areas:

Ability of the system to reduce trench cover soil moisture.

The effect of the bioengineering cover on trench water levels.

Maintenance and repair needs of the junipers, fiberglass panel cover system and drainage system.

Radionuclide uptake in vegetation compared with that in grass cover systems.

Other factors affecting trench cap performance (e.g., nuisance animals, desiccation cracking, etc.).

Initial Observations

Since installation of the bioengineering management system, the junipers have done relatively well in terms of growth and appearance (i.e., color, vigor, etc.), following a set back the first winter. Rodents (e.g., mice, shrews) girdling bark and removing branches from the junipers damaged approximately 50-80 percent of the 3,000 junipers, killing approximately three percent during the first winter. Nursery professionals advised NYSERDA that the damage to similar vegetation was above normal throughout the area, likely due to the consistent heavy snow cover and the unusually cold weather experienced in the area during the winter of 1993-94. The loss of some of junipers was anticipated and, due to the initial overplanting, is not expected to affect the performance of the cover. The recovery and growth of the junipers in the first growing season indicate that the plant roots are becoming established and that the Western New York climatic conditions will not present an obstacle to plant survival.

The hard cover system required no maintenance the first year. As the junipers continue to grow and overshadow the panels, the panels will be less susceptible to weather and solar degradation.

Hypalon-lined drainage swales, along both sides of Trench 9, carry the runoff from the fiberglass panels away from the disposal trenches. The hypalon is anchored by extending stone fill for soil erosion control onto the ends of the hypalon. In Spring 1994, field inspections identified small pockets of water collecting under the hypalon at the end of the swales. Improvements to the anchorage system prevented further ponding of water under the liner.

Reducing vertical water infiltration is a fundamental objective of the bioengineering management system. From March through November, soil moisture was measured at one-foot intervals for depths from one to five feet at three locations distributed along the center line of Trench 9, in grass covered trenches, and in a membrane covered trench. Initial measurements show that moisture in the bioengineering management system trench cover soils are similar to those at the other locations. This is expected because the junipers are small and have not developed the extensive root system and foliage, which will provide the ET rate needed to reduce the soil moisture. As the junipers grow, it is expected that the Trench 9 cover soils will become drier.

Vegetation samples are collected to evaluate the potential uptake of radionuclides (carbon-14 and tritium) in plants on conventional grass covered caps, in the junipers on Trench 9 and at a control location outside the SDA. The levels of carbon-14 and tritium vary between locations and over time, not allowing any initial conclusions to be drawn.

Concentrations of radionuclides measured at the Trench 9 ambient air sampling station are consistent with levels measured at background locations one-half mile to 50 miles away.

IMPENDING ACTIVITIES

NYSERDA and DOE are jointly preparing the Environmental Impact Statement for Completion of the West Valley Demonstration Project and Closure or Long-Term Management of the Facilities at the Western New York Nuclear Service Center (EIS). The EIS will establish the basis for the approach to and schedule for long-term management of the leachate in the trenches.

However, based upon the initial success from implementation of the infiltration controls on Trenches 13 and 14, NYSERDA is preparing to install a geomembrane cover over the remaining grass covered trenches (Trenches 1 through 8, 10, 11 and the other half of 12). Even though current water infiltration rates through the grass covered, silty-clay caps are quite low, water that enters the trenches will accumulate and increase the potential for an uncontrolled release. It is likely that the accumulated leachate will eventually have to be removed and treated or solidified as part of any site closure strategy being considered in the EIS. Leachate removed from trenches will have to be managed and treated as low-level radioactive waste and possibly as RCRA mixed waste due to the potential applicability of a RCRA hazardous waste designation.

By reducing the potential for a release of leachate and reducing the generation of waste, this project is also consistent with the following regulatory and best management philosophies:

- Waste minimization
- Stabilization and source control
- Reduction of contaminant discharge to the environment

The project is intended as an interim measure until the assessment of long-term management and closure options for facilities for the SDA is complete. With an intended life of 10 years, the exposed cover is appropriate only as an interim measure. Because all closure alternatives for the SDA trenches will likely require treatment or solidification of the leachate, this interim measure is consistent with long-term management goals and closure options for the facility.

Design work has started, and installation of the geomembrane cover on the remainder of the grass covered trenches is scheduled for completion in 1995.

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FINAL DISPOSAL OF LILW IN SLOVENIA - PRESENT AND FUTURE ACTIVITIES

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ABSTRACT

The siting project for low and intermediate level radioactive waste repository in Slovenia started in late eighties. In this project only surface and near surface repository type was considered. The site selection was planned to be concluded at the end of 1994. The project, which was divided into four steps, successfully

followed the plan until the last fourth step. After the third step was concluded, the public presentation of the results, i.e. five most suitable locations, provoked extremely negative public response. The fourth step of siting, that includes also field investigations, was stopped. Since the continuation of the project was not feasible, the Agency for Radwaste Management, responsible for the project, decided to investigate some other possibilities: the site selection for underground repository of low and intermediate level radioactive wastes and volunteer siting. In this paper recent siting activities as well as previous siting project are presented..

INTRODUCTION

In Republic of Slovenia, the small country with the population of about 2 millions on approximately 20.000 km2, one nuclear power plant and one small TRIGA research reactor are in operation. The nuclear power plant (Westinghouse technology, total capacity 632 MWe) was constructed as a joint venture between Republic of Slovenia and Republic of Croatia and is in commercial use since 1982.

For safe and reliable operation of NPP and research reactor the appropriate radioactive waste management is of great importance. Being aware of that problem, the first preliminary studies concerning the low and intermediate level radioactive waste repository started before the nuclear power plant went into operation but the siting project actually started in late eighties, several years after the NPP went into operation. Due to the natural characteristics of our country as well as economical and technical reasons, the project was oriented more towards surface and near surface type, than towards underground disposal.

In 1990 the time schedule for the siting project was prepared and discussed in the parliament. According to this plan the siting project was divided in four steps and planned to be concluded in 1994.

In 1990 the Guidelines with the instructions for siting of low and intermediate level waste disposal facility in Republic of Slovenia were published by the Slovenian Nuclear Safety Administration (1). Only the criteria for surface and near surface options were considered. These Guidelines, including 43 obligatory site selection criteria, represented the basis for siting project. Subsequently the Guidelines were revised by international experts (2) and in 1991 Revision 1 was published in Official gazette (3).

The siting project was initially under the responsibility of the Division for radwaste management which was founded in 1986 and organized within the NPP Krsko. In 1991 the Agency for Radwaste Management, independent on radwaste producers, was established by the Slovenian government (4) and authorized to carry out the radwaste management projects, including siting of low and intermediate level radioactive waste repository.

LILW REPOSITORY SITE SELECTION - PROJECT HISTORY

The objectives and main results of the siting project are presented in Table I. Following the Guidelines and the siting schedule, the first step was started and concluded in 1990 (5). As the result the unacceptable areas were excluded by application of the exclusion criteria. Since some of the criteria, applied in this step, were extremely severe and rigorous, only few % of whole Slovenian territory remain available for further examination. For example, only the criterion of active faults presence excluded 97 % of our total area. The siting project continued by second step which was successfully concluded in 1991. Following the comparison criteria 36 suitable locations were selected (6). After conclusion of first and second step the results were publicly announced, but except some disagreements no significant reactions were noticed.

At that time the ecological movements in Europe have influenced the formation of similar groups in Slovenia. The changes in our political, economical and social system that occurred in late eighties and early nineties encouraged groups and individuals to become publicly active. New political parties were founded, among them the green party which became politically active. On the other hand the number of opponents to peaceful use of nuclear energy was rapidly increasing. Consequently the members of the green party and ecologically oriented groups started to show increased interest for the radioactive waste management and the site selection project.

The siting project still continued according to the schedule. At that time the third step of the repository project already started. This was the step at which the Agency for Radwaste Management took over the project. The technical part of the

project, including all criteria, specified for the third step except public acceptance, was concluded in 1993 (7). As a result of this step five most suitable locations were identified. In accordance with the practice from previous steps the results were presented to the public. The presentation was unsuccessful and has provoked a strong disapproval within the local communities where the locations were identified. The Agency for Radwaste Management was blamed not to provide regular and sufficient information on the project to the public and not to present and discuss the problem with the representatives of these communities prior the public presentation of the results. The opposition in these local communities was very strong and it soon became evident, that the public acceptance criterion at these locations can not be fulfilled. The expert group, that revised the project, decided to shift the public acceptance criterion into the fourth step so that the third step was considered as concluded (8). Nevertheless, it was impossible to proceed with the siting procedure at these five locations, since the field works at locations were planned to be performed within the fourth step. The project was temporarily stopped.

SITING PROJECT - PRESENT SITUATION

More than one year after the third step of siting project was concluded, the situation in our country, regarding the public acceptance of radioactive waste repository, has not improved. Due to the negative public opinion the continuation of the siting project on five selected locations was recognized as not feasible. The construction of the repository for radioactive wastes is shifted into the next century.

As long as there is no final repository for radioactive wastes in Republic of Slovenia they are stored on locations where they are produced. There are two such locations in Slovenia: the storage for the operational wastes from nuclear power plant on Krsko site and the storage of low and intermediate level radioactive wastes from all other producers in our republic (research, industry and medicine) at Research reactor center Podgorica. On the third location near village Zavratec temporary storage for the decontamination wastes after an accident at the Oncological Institute in Ljubljana in 1961, is placed. On first two locations besides low and intermediate level wastes the spent fuel elements are stored in spent fuel pools, as well.

In Table II. the present quantities of low and intermediate level radioactive wastes as well as storing capacities for these three locations are given. From the table it can be seen that the most critical situation is in NPP Krsko, where storing capacities will soon be occupied.

In nuclear power plant by optimization of working procedures a considerable reduction in waste production was achieved over the last several years. Besides this, another effort in minimization of radioactive waste quantities by the supercompaction of wastes, is currently taking place. It is expected, that by this campaign the present volume of low and intermediate wastes in Krsko will be reduced for approximately 360 m3. Regarding the limited storing capacities at NPP site, all these activities are necessary to provide normal power plant operation until the final disposal facility will be available.

Since the present siting project is stopped and it is not expected that the continuation of this project on selected locations would be feasible in near future the Agency for Radwaste Management is looking for other possibilities to proceed with the project. According to the experience in some other countries the Agency decided to start a new project for underground disposal of low and intermediate level radioactive wastes, which is hoped to be more acceptable to the public.

As the basis for siting procedure the Guidelines with the siting criteria for underground disposal should be prepared and issued by the authorities. For this purpose the Agency for Radwaste Management started the preparation of the siting criteria. Recently, the preliminary criteria have been concluded (9, 10). When the Guidelines will be issued the siting project will start.

The new criteria for the underground repository of low and intermediate level wastes take into account most of the recommendations of international experts (2), given for the siting of surface and near surface repository, some useful experience from other countries and also our own experience. The criteria are basically divided in two main categories: general criteria and geological criteria. Most of the general criteria are the same or very similar as in the Guidelines for surface or near surface repository, so it is expected that some results of the previous siting project can be used. However, significant differences are expected in application of

seismic and active fault criteria, since these criteria appear to be less rigorous in case of underground repository. Taking into account that in first project only active fault criterion excluded 97% of the territory, even small changes in this criterion can considerably increase the area of potential locations.
From lessons given in previous project we learned that public acceptance is the crucial moment in whole siting procedure. Therefore it is planned to support this site selection project for underground repository by intensive public informing and educational campaign from the very beginning. Public acceptance will be evaluated after each step. Since it is not likely to continue with the next step of site selection unless the public acceptance is achieved, it is difficult to predict the time schedule of this project, especially because it depends also on financial sources that will be available to the project.

SITING PROJECT - FUTURE ACTIVITIES

Besides the siting project for underground repository the Agency for Radwaste Management plans also some other activities connected to the site selection. Recently the survey of the abandoned mines and some other underground facilities was concluded (11). The project continues with the study on possible use of these abandoned mines and other underground objects to store or to dispose off the radioactive wastes (12).
Other possibilities for site selection are being studied, as well. Following the foreign experience the analysis of volunteer siting has just started (13). The first study in this field should identify the most suitable approach to invite the communities to volunteer their locality to host the repository.
The volunteer siting became more feasible since the Environmental Protection Act (14) was issued in 1993. By this act the financial compensations to local communities, which would host the project of general importance, are foreseen. This possibility is now being studied to be applied in the volunteer siting project for radioactive waste repository.
Special program is being prepared also for the activities concerning public relations and educational programs. Regular opinion polling and different presentation materials like brochures and booklets are part of this program.
At the same time some other studies concerning the low and intermediate level radioactive waste management like Preparation of preliminary acceptance criteria for radioactive wastes (15), Overview of the materials suitable for engineered barriers in LILW disposal (16) as well as Transportation of low and intermediate level radioactive wastes (17), are performed. However, final reports can not be prepared before the location of the repository is known.

CONCLUSIONS

Since the siting project for surface and near surface repository in Slovenia was stopped and it is not expected to be continued in near future, the Agency for Radwaste Management, which is responsible for the site selection, decided to start a new project for underground repository. The criteria for underground repository site selection have already been prepared. It is expected that the Guidelines will soon be issued by the Slovenian Nuclear Safety Administration and the project will start.
As seismic and active fault presence criteria are different in case of site selection for underground and surface repository , it is believed that by new siting project more potential locations will be available than in first case. However, the possibility to conclude the siting project successfully , depends significantly on public relations. Therefore, the Agency decided to pay strong attention to these activities from the very beginning of new project.
At the same time the possibilities for volunteer siting for radioactive waste repository are studied . Since such approach was already successful in some countries, the Agency intends to apply it in Slovenia as well, if the results of preliminary study will be positive.
The staff of the Agency for Radwaste Management put strong efforts in realization of siting project. By new approach and better public relations it is believed that the project will be successfully realized.

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HEAT GENERATION AND HEATING LIMITS FOR THE IRUS LLRW DISPOSAL FACILITY

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ABSTRACT

Heat generation from radioactive decay and chemical degradation must be considered when implementing low-level radioactive waste (LLRW) disposal. This is particularly important when considering the management of spent radioisotope sources. Heating considerations and temperature calculations for the proposed IRUS (Intrusion Resistant Underground Structure) near-surface disposal facility are presented. Heat transfer calculations were performed using a finite element code with realistic but somewhat conservative heat transfer parameters and environmental boundary conditions. The softening-temperature of the bitumen waste-form (38oC) was found to be the factor that limits the heat generation rate in the facility. This limits the IRUS heat rate, assuming a uniform source term, to 0.34 W/m3. If a reduced general heat-limit is considered, then some higher-heat packages can be accepted with restrictions placed on their location within the facility. For most LLRW, heat generation from radioactive decay and degradation are a small fraction of the IRUS heating limits. However, heating restrictions will impact on the disposal of higher-activity radioactive sources. High activity 60Co sources will require decay-storage periods of about 70 years, and some 137Cs will need to be disposed of in facilities designed for higher-heat waste.

INTRODUCTION

Heat generation from radioactive decay and chemical degradation must be considered when implementing low-level radioactive waste (LLRW) disposal [IAEA 1985]. This is particularly important when considering the disposal of radioisotope sources. This paper summarizes heating calculations that have been performed for the proposed Intrusion Resistant Underground Structure (IRUS) disposal facility. Heat limits are

proposed for the IRUS facility, and these are compared with radioactive decay and chemical degradation heating rates expected in IRUS waste. The implications of heating considerations on managing Canada's LLRW are also discussed.

The IRUS disposal facility (Fig. 1) is based on an open-bottom concrete vault, and is designed for radioactive waste with a hazardous lifetime of less than 500 years [Hardy, et al., 1988]. The facility is expected to be filled with higher-activity LLRW from Canadian nuclear research programs and radioisotope applications. The basic design principles for IRUS are to contain the waste within the vault (i.e., restrict the release of contaminants), and isolate the waste from the environment (i.e., prevent inadvertent intrusion).

The IRUS facility is 32 m long, 22 m wide and 9 m deep, and is divided into 6 cells that provide a total useable volume of about 3900 m3. The reinforced concrete walls are arched so that when the cells are empty, the external soil pressure creates compressive stresses within the walls, reducing the need for reinforcing bars. The walls are 0.61 m thick, and the reinforced concrete roof is 1 m thick. The floor is permeable to avoid the "bathtub effect", and is composed of two buffer layers: a 0.3 m thick mixture of sand (90%) and clinoptilolite (10%), and a 0.3 m thick mixture of sand (90%) and Dochart clay (10%). The clinoptilolite and clay have the capacity to sorb many critical radionuclides from aqueous solution, and thus reduce radionuclide escape from the vault. To avoid flooding, IRUS is to be located in a free draining sand deposit with its foundations at least one meter above the highest recorded water table. Final closure of the facility will include a multilayer earthen cover system about 2 meters thick.

During the operating phase, the IRUS facility will be covered with a temporary weathershield building. The building will contain a gantry crane for transferring materials to the cells using remote attach/detach capabilities so that entry into the cells is not required. The principal waste packages will be 0.4 m3 bales of compacted waste (90% volume) and 200 litre metal drums. The bales will include fibrous materials, plastics, and small quantities of metals. Most drums will contain a bitumen waste-form produced from liquid-solidification processing or ash immobilization. Waste will occupy about 50 percent of the IRUS volume. Voids between packages and layers of waste will be backfilled with sand (90%) and clinoptilolite (10%).

FACTORS LIMITING HEATING RATE IN THE IRUS FACILITY

Factors that could limit the heat generation rate within the facility were determined, as summarized in Table I. For each factor a maximum allowable temperature or temperature gradient was determined (column 3). The melting temperature of bitumen (38C) was identified as the factor that would limit the allowable temperature, and thus the heat load, in the IRUS facility.

Concrete strength is generally not affected by temperatures less than 38C [Mittelacher, 1992]. At temperatures above 75C, decreases in strength have been measured by Carette and Malhotra [1985]. Thermal stresses due to temperature gradients in the concrete walls during the operational and post-closure phases have been considered for the temperature design load of the IRUS structure [Mok, 1989]; the largest temperature gradient of the study was assumed to be a maximum allowable temperature gradient in the concrete for this analysis (i.e., 26.7C/m). The waste-form most sensitive to increased temperatures is the bitumen used for liquid solidification. This bitumen is similar to type 80/100-150, which has a melting point in the range of 38-53C [IAEA, 1993]. Bitumen's flash point temperature is well above its melting temperature, and is not a limiting factor. Microbiological degradation of hydrocarbons in bitumen occurs at temperatures from 0 to 70C. Under the most favourable conditions of temperature, pH, and O2 availability for degradation, the maximum microbial penetration of bitumen would be of the order of 0.7 mm per 100 years [Allison, et al., 1991]; therefore, it is not expected to limit the temperature in the IRUS facility. Temperature gradients in the waste will increase tritium migration. The allowable temperature gradient for tritium migration have not been determined, but this consideration is not expected to be critical. Finally, the steel containers of the waste are not given any credit as a barrier to waste migration, so corrosion rate temperature dependence will not limit the allowable temperatures.

TEMPERATURE CALCULATIONS FOR THE IRUS FACILITY

The finite element computer code, ANSYS 5.0A, was used to perform calculations of temperature in and around the IRUS facility. Calculations were performed for a base

case with uniform heating throughout the facility, and for a case where a limited number of high-activity packages were included. Analyses of sensitivity to the various thermal conductivities, boundary conditions and location of facility with-respect-to surface and aquifer were also performed. A 2-D cross-section of the two middle cells of the repository, where the temperature would be the highest, was used as a basis for the simulation. The cross-section was simplified, as shown in Fig. 2, using the following assumptions:
Fig. 2. Sketch of the ANSYS model of the IRUS facility
All cells have the same heat load, giving plane symmetry in the centre of the inner concrete wall.
The waste was assigned a uniform thermal conductivity based on thermal conductivity of building rubble materials, fibrous materials, plastics, sand, and bitumen.
The overburden layers (gravel, cobblestone, fine sand, and top soil) and buffer layer (90% sand, 10% clinoptilolite) were assigned the thermal conductivity of sand. This is justified since the thermal conductivities of these layers are similar, and minor differences in their thermal conductivities will not significantly impact the calculated temperatures.

Thermal Conductivities

Typical values and bounding ranges were identified for the thermal conductivities required for the temperature simulations (Table II). The bounding ranges were used for sensitivity studies, and to help establish a conservative reference scenario. The walls and the roof of the IRUS facility will be constructed of high-performance reinforced concrete with limestone aggregate. The typical thermal conductivity of concrete with limestone aggregate is 1.56 W/m.K; the selected range of 1.2 to 1.8 W/m.K is slightly more conservative than the typical range. The thermal conductivities of the sand, overburden and buffer is typical of Ottawa area sand with a moisture content of 2-4%. The effect of moisture migration on thermal conductivity of sand was also considered; it was concluded that the selected range of the thermal conductivity of sand covers the effect of moisture migration satisfactorily.
The thermal conductivity of the heterogeneous waste/backfill mixture expected within IRUS has not been explicitly determined. The dry environment within the vault, coupled with the slightly elevated temperatures expected, suggest that the thermal conductivity of dry sand (0.35 W/m.K) would be a reasonable starting point. Fibrous materials and air gaps in the compacted bales may decrease this value somewhat. The thermal conductivities selected for this study cover the range of realistic values, although the "typical value" is likely to be somewhat low.

Boundary Conditions

The boundary conditions were established based on annual averages of environmental parameters. Annual averages of environmental parameters were used throughout this study, since seasonal temperature variations are insignificant two or three metres below the ground surface [Wildsmith, 1976]. Parameters such as air temperature, ground temperature, aquifer temperature, solar radiation, wind speed and ground-to-sky radiation for the Chalk River/Ottawa Valley region were considered. Since the radionuclides in the IRUS facility will be active for several decades, slightly conservative ranges of annual averages of all parameters were used, based on measured data.
Adiabatic boundary conditions were assigned at the vertical boundaries, as shown in Fig. 2. The location of the right boundary was sufficiently far away to not affect the temperatures in or around IRUS facility. At the horizontal boundaries, constant temperature boundary conditions were assigned. The ground surface temperature range was based on the sensitivity analysis of the environmental parameters; therefore, it covers the effects of all those parameters on the ground surface temperature. The effect of heating in the IRUS facility on the ground surface temperature was also examined. It was found that heat load of 2 W/m3 (which is about 5.5 times higher than the established limit) affects the ground surface temperature only by 0.2 to 0.3C.

Effect of Voids Within the Vault

Voids may form within the vault, either through incomplete backfilling around waste or waste slumping. The effects of voids on the heat transfer characteristics were investigated. Horizontal air gaps up to 0.5 m were considered. In the ANSYS model, the closed air gap is represented as a solid material having an effective thermal

conductivity of the air gap [Holman, 1981], which includes the effects of conduction, convection and radiation. The effective thermal conductivities, found for various air gap heights for a heat load corresponding to the temperature limit of 38C in the waste, were used in the finite element simulation.
For vertical air gaps, it was found that the effective thermal conductivities were within the bounding values of the thermal conductivity of waste, and in some instances slightly higher than, therefore, their thermal effect is effectively covered by the waste thermal conductivity.
Calculations
The problem of the IRUS heating was simplified to five parameters, as listed in Table II. Temperature calculations were performed to determine the heat generation rate which produces the limiting temperature of 38C. Calculations were made for a typical case with uniform heat generation throughout the waste, and then single parameter sensitivity analyses were carried out to lower and upper bounding values. Various schemes were considered that allowed some higher-activity packages to be placed in IRUS. Such schemes must be simple to implement operationally, and they must be amenable to simulation. A suitable scheme reduces the general heat limit for waste, but allows for higher-activity packages to be placed anywhere within 1.2 m of the top or bottom of the facility. To simulate higher activity packages, some finite elements were assigned higher heating rate than others according to selected perturbation schemes (i.e., higher heating in top and bottom layers).
RESULTS
The calculated heat rate limits for IRUS that give a maximum temperature of 38C are presented below. In all cases the temperature gradients in the IRUS structure are less than 50% of the imposed 26.7C/m limit.
Heating Rate Limit with Uniform Heat Source
Calculations using the typical values for heat transfer parameters given in Table II, and assuming uniform heat generation throughout the waste, give an IRUS heat rate limit of 0.63 W/m3. Single parameter sensitivity analysis showed that the heat limit is not very sensitive to the parameters in Table II except for the kwaste. The heat load is, in most cases, in the range of 0.5 to 0.7 W/m3 except for the lower bounding value of kwaste when it is 0.35 W/m3. Analysis also showed that a 1 m increase in the distance between the IRUS facility and the aquifer or ground surface decreased the heat load by about 0.02 W/m3. Due to uncertainty in some of the parameters, and recognizing that the IRUS facility will need to maintain its performance for several centuries, the reference case was defined as having a typical thermal conductivity of waste form and conservative bounding values of all the other parameters. This conservative reference case gives an IRUS heat rate limit of 0.34 W/m3. If the waste form thermal conductivity can be better defined, then the heat limit can be adjusted using the data in Table III.
Heating Rate Limit with Higher Activity Package Perturbation
The higher activity package perturbation calculations were performed using the same reference case boundary conditions and thermal conductivities as described earlier for the uniform heating source. The calculations show that IRUS could handle a small number 0.4 m3 packages with heating rates of about 10 W/m3. If a general heat limit of 0.1 W/m3 is applied to the facility, then high-activity packages with heating rates up to 1.6 W/m3 could be placed anywhere within 1.2 m of the top or bottom of the facility.
HEATING WITHIN IRUS
Heating in LLRW occurs from radioactive decay and chemical reactions during the waste degradation process. In general, heating in LLRW is expected to be small, although some radioactive sources do produce significant power.
Radioactive Heating
In solid-LLRW, all radioactive decay energy, except neutrino energy, is deposited locally (neutrino interaction cross sections are negligible). The heat energy and corresponding Bq/Watt for some representative radionuclides in LLRW are listed in Table IV, columns 2 and 3, respectively. These values were derived from average energy released per decay provided in ICRP Publication 38 [1983].
As presented earlier in Section 4, the heat rate limit in IRUS could be implemented as either a uniform limit of 0.34 W/m3, or alternately, as a base value of 0.1 W/m3 with some higher-activity packages permitted with heating rates up to 1.6 W/m3. Radionuclide limits for these different options are presented in columns 4, 5, and 6 of Table IV. These are single nuclide limits, and a sum-of-fraction rule must be

applied to mixtures of radionuclides.
To assess the impact of heating limits on IRUS waste acceptance, the expected LLRW inventory must be considered. Preliminary inventory estimates suggest that the total activity in the IRUS vault will be about 4.0 x 1014 Bq, with 3H (1.6 x 1014 Bq), 137Cs (1.1 x 1014 Bq), and 60Co (2.1 x 1013 Bq) being the highest- activity longer-lived nuclides (Hardy et al., 1991). The total activity results in about 40 watts of radioactive heating [Chan et al., 1994], which translates to an average heating value of 0.01 W/m3. This low value suggests that a general heat limit of 0.1 W/m3 would be accep for IRUS, thus permitting a limited number of packages with heating rates up to 1.6 W/m3.
The IRUS facility could be used for the disposal of sealed radioactive sources, although heating rate and risk-based limits will exclude acceptance of some sources. Table V shows the distribution of activities for the higher-activity sealed sources currently licensed in Canada [Beriault, 1994]. Only sources with a half-life greater than three years are included in the , since shorter half-life sources can be easily stored until their heat generation rates are low. The final column in the shows the IRUS activity limit in a 0.25 m3 package, based on a heat limit of 1.6 W/m3 and assuming that the waste occupies 50% of the vault volume. Sealed sources that exceed this limit are indicated with shading in the .
The data in Table V show that only 60Co, 137Cs, and 241Am sources are above the IRUS heat limit in significant quantities. The very-high-activity 60Co sources, which represent several 100 kW of power, would need to be stored for a period of about 70 years before they qualify for disposal in IRUS. Although the 137Cs sources represent a much smaller heat source than the cobalt, their longer half-life makes decay-storage impractical. High-activity cesium sources will either have to be dismantled to distribute the heat source, or disposed in a facility that can accommodate higher-heat waste. The higher-activity americium sources are not likely to be sui for near-surface disposal (excluded by risk-based limit), and these sources will likely be disposed in a geologic facility.
Heat Generated from Biogeochemical Processes Associated with Waste Decomposition
Leaching experiments have been performed on compacted bales of LLRW to monitor their degradation. The degradation process can be related with simple redox reactions to eventually give heat generation rates. To extrapolate the leaching measurements to give an estimate of the IRUS heat generation rate, a correction factor must be applied to reflect the drier environment expected in IRUS.
Microbial degradation of organic matter in landfill sites is one of the major sources of heat. This reaction gives the theoretical yield of anaerobic degradation of cellulose [Pirt, 1978; Rees, 1980]:
1 kg $C_6H_{12}O_6$ → 0.25 kg CH_4 + 0.69 kg CO_2 + 0.056 kg dry biomass
+ 632 kJ heat (1)
The water content and the flora dictate the regime of the reaction (aerobic or anaerobic). In a closed system such as a landfill or a confined aquifer, the most energy-yielding substrates will be assimilated preferentially either until depletion, or until the concentrations of these substrates become too low to be favourable thermodynamically. The substrates are generally reduced in this approximate order: O_2, NO_3^-, Mn(IV), Fe(III), SO_4^{2-}, CO_2, N_2 [Scott and Morgan, 1990]. Several redox couples may coexist, due, in part, to the relative closeness in redox couples and heterogeneities in the reactor.
Four bales of compacted low-level wastes were sealed in separate metal boxes to monitor the decomposition process. Each box was connected to a pump in a closed loop and water was recirculated daily over the bales to keep them wet, but not flooded. Major ions, gases, dissolved organic matter (DOM), etc., were monitored periodically [Caron, 1994] on all the bales. Two complete sets of redox-sensitive parameters were used on two dates (spanning over approximately one year) for mass balance calculations. The parameters showing the most important changes are shown in Table VI (O_2, SO_4^{2-}, NO_3^- and Mn were measured but their contribution was small). They were related together with a series of redox reactions (Table VII) to determine the contribution of each reaction in that time period. The energy released is given by the enthalpy DH (negative values denote exothermic reactions).
Heat Production Calculations
The time "zero" of the experiment was set when water was added to each bale. The results of 6 are shown for only one bale as an example. The most important assumptions related to these calculations are:

the gas generation rates were linear over the duration of the experiment, based on the long bale-degradation half-life (30-2400 years)[Caron et al., 1995];

all the DOM was acetic acid, which is reasonable based on elevated levels of acetate measured in these bales, [Caron, 1994];

nitrogen was assimilated to produce ammonia, and not displaced. Ammonia was not measured, but the calculated concentration (~0.08 mole/L) is not unrealistic of landfills [Rees, 1980; Lisk, 1991; Ehrig, 1983].

Only the spontaneous reactions (noted with a negative DGr value) were used because microorganisms are unlikely to go against an energy expense for fulfilling their energy needs. Hydrogen gas was not detected, but its assimilation (reaction 4 in 7) and production are more than likely.

The total energy released to the surroundings is the sum of all the enthalpies of each of the reactions (7):

Eq. (2)

where i corresponds to reactions 1 to n, and Ci is the relative contribution of reaction i. This amounts to 1734 kJ for bale #6, which is higher than the energy released (840 kJ) from the stoichiometric reaction (1), corrected for the same amount of cellulose. The major difference between our calculation and equation (1) is the contribution of metal corrosion (reaction 1 in Table VII) and CO2 reduction to methane (reaction 4 in Table VII). Credit is not given to energy losses due to conversion to biomass.

Degradation Rates Applicable to the IRUS Environment

To estimate the heating rate expected in IRUS, the heating rates derived from the leaching tests must be corrected to reflect the dry environment expected in IRUS. Gas generation rate measurements by Torok and Haas [1992], using the same bales degrading in dry conditions, allow for a reasonable estimate of the required correction factor. Their reference gas generation rate was 0.021 l/kga. Gas generation rates in the leaching tests (Table VIII) were 5 to 60 times higher than this reference value, so the heat generation was normalized to the same rate to give corrected values.

CONCLUSIONS AND RECOMMENDATIONS

Factors that could limit the temperatures and temperature gradients in IRUS were investigated, and the melting temperature of the bitumen waste form (38C) was found to be the most restrictive. Heat transfer calculations, based on realistic, although somewhat conservative heat transfer parameters and boundary conditions, suggest that a heat rate limit of 0.34 W/m3 is appropriate for IRUS, assuming a uniform heat source throughout the facility. Alternatively, a general heat limit of 0.1 W/m3 could be applied, with heating rates up to 1.6 W/m3 within 1.2 m of the top or bottom of the facility. Assuming a 50% waste packing efficiency, the heat limits for the waste packages will be double the facility limits. For example, a general package heat limit of 0.2 w/m3 can be applied, with some high-activity packages accepted with heating rates up to 3.2 w/m3. Heat limits are sensitive to the waste form thermal conductivity. The proposed heat limits are based on a somewhat conservative waste-form thermal conductivity, and this should be reviewed once the actual IRUS inventory is known.

Heating in LLRW occurs from radioactive decay and chemical reactions during the waste degradation process. For most of the LLRW expected to be disposed of in IRUS, radioactive decay heating is expected to be about 0.01 W/m3. The IRUS facility may also be used for the disposal of sealed sources. A review of sealed sources licensed in Canada shows that most sealed sources fall within the 1.6 W/m3 heat limit noted earlier. The exceptions to this are 60Co, 137Cs, and 241Am. 60Co sources could be decay-stored for about 70 years at which time they could be economically disposed of in IRUS. On the other hand, the longer-lived 137Cs and 241Am sources will likely need to be disposed of in facilities that can accommodate higher-heat loads (e.g., fuel-disposal facility).

Heating generation from waste degradation due to biogeochemical processes has been evaluated based on leaching experiments with compacted waste bales. The results, corrected to account for the dry environment expected in IRUS, suggest that chemical heating will contribute about 0.004 W/m3, which is a small fraction of the proposed IRUS heat limits.

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14-8

THE NATIONAL ACADEMY OF SCIENCES AS "SCIENCE COURT": REVIEW OF THE WARD VALLEY SITE BY THE BOARD ON RADIOACTIVE WASTE MANAGEMENT

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ABSTRACT

A panel of scientists, appointed by the Board on Radioactive Waste Management of the National Academy of Sciences/National Research Council (NAS), is completing a review of the so-called "Wilshire Report" and the Ward Valley site which has been licensed by the State of California for the Southwestern Compact's regional low-level radioactive waste disposal facility. The review was requested in March 1994 by Interior Secretary Bruce Babbitt who asked the Board to evaluate certain "earth science concerns" raised in the "Wilshire Report." Completion of the Academy's evaluation has been delayed and its report is now expected in early March 1995. Academy review of a proposed project that has already been licensed by the authorized governmental agency is unique and raises a number of institutional issues. NAS review may or may not be consistent with statutory and regulatory standards adopted and implemented by federal and state governments for low-level radioactive waste disposal facilities. The delay of the project due to the review and the ultimate role of the report in the federal decision process in response to the State of California's request to purchase the Ward Valley property from the U.S. Bureau of Land Management (BLM) are also matters of public policy concern. Only when the Academy completes its review and makes its report public will we know how it has dealt with these institutional issues and with what degree of success.

INTRODUCTION

In June, and again in December 1993, three geologists employed by the U.S. Geological Survey (USGS) at its office in Menlo Park, California raised concerns about the suitability of the Ward Valley site for California's and the Southwestern Compact's proposed low-level radioactive waste disposal facility. Their document, prepared for release at a press conference held in Los Angeles on December 8, 1993, has come to be known as the "Wilshire Report" after the lead author Howard Wilshire.

The Ward Valley site is federal land under the jurisdiction of the U.S Bureau of Land Management (BLM), an agency within the Department of the Interior. The California Department of Health Services (DHS) application to the BLM to purchase 1,000 acres at Ward Valley has been pending since July, 1992. (The BLM has been fully involved on the California LLRW project with DHS since 1988 when it was clear that the California site would be on BLM land.) On March 14, 1994, Secretary of the Interior Bruce Babbitt requested the Board on Radioactive Waste Management (BRWM) of the National Academy of Sciences to evaluate earth sciences concerns in the Wilshire report and to deliver a report to him by December 1, 1994. The Secretary's request followed several actions and counter-actions taken by the Department of the Interior during the preceding fourteen months. (See section on Political Antecedents on page 3.)

BACKGROUND

Environmental documentation for the Ward Valley disposal project was issued in April 1991 by DHS and BLM as a Final Joint Environmental Impact Report and Environmental Impact Statement, and, following public hearings on the joint EIR/EIS and the draft license, the public comment period was closed in August, 1991. In September, 1993, the BLM released a Final Supplemental Environmental Impact Statement (FSEIS) which found that the "Ward Valley facility will meet or exceed all environmental health standards with no contamination of ground or surface water," confirming the findings of the earlier documentation. On September 16, 1993, the DHS issued a license to US Ecology, Inc., the company that had been designated "license designee' in late 1985. In August 1993, Secretary Babbitt requested the Governor of California to hold one more hearing, and the Governor agreed. In that request, the Secretary indicated his objective of reaching a decision on the land sale by the end of the year. On November 24, 1993, Secretary Babbitt canceled the hearing he had requested that California conduct, and put the land sale on hold. On March 14th, 1994, the Secretary requested the review by the BRWM which is the subject of this paper.

COMMITTEE COMPOSITION, TASK STATEMENT AND SCOPE OF TECHNICAL ISSUES

The "Committee to Review Specific Scientific and Technical Safety Issues Related to the Ward Valley, California, Low Level Radioactive Waste Site" appointed by the BRWM is comprised primarily of earth scientists in the disciplines of geophysics, geochemistry, tectonics, hydrology of the unsaturated zone, hydrology of fractured media, and sedimentary geology. The committee also includes a civil engineer and desert ecologist.

The Committee's Statement of Task states, in part:

"The Committee will undertake an examination of the relevant data, reports, license

application material, and other documents that address the following specific issues related to the Ward Valley, California site:
1. Potential infiltration of the repository trenches by shallow subsurface water flow, including the possible presence of tritium in the deeper soils, and interpretation of C-14 ages.
2. Potential transfer of contaminants through the unsaturated zone to the ground water.
3. Potential for hydrologic connection between the site and the Colorado River.
4. The absence of plans to monitor ground water or the unsaturated zone down-gradient from the site.
5. The potential for failure of proposed engineered flood-control devices.
6. Potential damaging effects on the desert tortoise habitat.
7. Potential interference with revegetation and reestablishment of the native vegetation.
"The objectives of the study are 1) to assess the adequacy of the site studies relative to the above enumerated issues and the validity of the conclusions concerning site performance that are the subject of debate, and 2) to determine if the enumerated concerns are valid, significant, and unresolved and, if so, to assess the potential impacts on site performance.
"The committee will comment only on the scientific and technical issues. It will not evaluate the site nor be a party to any approval process."
The committee held two public meetings in the City of Needles, California (the community nearest the Ward Valley site) on July 7-9 and August 30-September 1, 1994. The panel visited the site and listened to presentations by the DHS, US Ecology, and the Wilshire group and their respective contractors and consultants. The panel also heard public comment during "open microphone" sessions at the end of each day, with presentations generally limited to 5 minutes. The committee also invited written submissions. These public portions of the committee's work raise several policy issues related to procedure and the standard for evaluation.

POLITICAL ANTECEDENTS TO SECRETARY BABBITT'S REQUEST TO THE NAS

In January 1993, Secretary Babbitt rescinded the Ward Valley land sale to California which had been approved by former Secretary Lujan. He also rescinded Mr. Lujan's record of decision. In August, in a letter to Governor Wilson, Mr. Babbitt requested that California hold one more hearing on the adequacy of the Ward Valley site. The Secretary outlined a schedule that anticipated a decision by Interior on the land sale by the end of the year. In September, the BLM issued its FSEIS as described above. On September 16, 1993, the DHS issued its license to US Ecology, and the Governor announced that he would agree to hold the hearing requested by Mr. Babbitt. On November 24, 1993, Secretary Babbitt canceled the hearing he and Governor Wilson had agreed to hold. The Secretary's stated reasons were that California had issued a license and that the license was being challenged by a lawsuit in state court in which petitioners claimed that state law requires an "adjudicatory hearing" before a license may be granted. But Governor Wilson had personally notified Mr. Babbitt in advance that a license decision would be made, pursuant to the order of the appellate court, prior to the hearing on federal issues related to the land transfer. Secondly, the adjudicatory hearing issue had already been resolved by the state Court of Appeal. The Secretary's basis for delay was removed in February 1994 when the state trial court in Los Angeles granted a motion for summary judgment and ruled that state law requires no such hearing. This ruling was based on the Sacramento Appellate Court's ruling in May 1993. Whatever Mr. Babbitt's reasons for requesting the NAS/BRWM review, it has provided the Administration with more than a year's delay past the Secretary's original decision schedule of December 1993.

ISSUES RAISED BY THE ACADEMY'S REVIEW AND PROCEDURES

Fairness/Openness

Ward Valley opponents have charged that the appointed Ward Valley panel is not "balanced." In a July 1, 1994 letter to Dr. Bruce Alberts, President of the National Academy of Sciences, U.S. Senator Barbara Boxer (D-California), an outspoken opponent of Ward Valley since 1991, stressed the importance of the case under study, "where the issue is whether the proposed nuclear dump could radioactively contaminate the Colorado River for generations to come." Senator Boxer complained of "an apparent heavy bias in favor of those with contracts and other ties with the nuclear industry and associated institutions involved in the promotion of nuclear activities in this country." She also questioned the objectivity of panel members

who "have published on the very issues the panel is to resolve, having taking (sic) positions contrary to those of the U.S. Geological Survey (USGS) scientists whose work they are to review." Neglecting the large role assigned to Dr. Wilshire and his allies in the committee's public sessions in Needles, she complained about the agenda's "...precluding any participation by any representative of or scientist with any of the entities opposing Ward Valley." (Emphasis in the original.) Senator Boxer urged Dr. Alberts to "clean up this mess."

In his response, Dr. Alberts described the process by which the National Research Council (NRC) selects "an expert committee with the appropriate scientific and technical qualifications to investigate the issue at hand. The selection of the committee is made by the NRC Chairman after reviewing candidates suggested by internal and external sources. The NRC committee selection process also strives to balance potential biases with regard to the issues in dispute and to avoid conflicts of interest that would compromise the study." With respect to the Ward Valley panel, Dr. Alberts wrote, "Based on the completed forms and the discussion in executive session, my staff and I have confirmed that the committee as a whole is technically highly qualified, is reasonably balanced, and does not have a bias or conflict of interest that would prevent it from reaching highly-credible scientific conclusions regarding the seven technical issues in dispute." Dr. Alberts noted, "As an aside, you might be interested to know that three members of the committee, including the chairman, were recommended by environmental organizations; none of the committee members was suggested by industry or by proponents of the use of Ward Valley for low-level radioactive waste disposal."

Dr. Alberts went on to describe the limited scope of the NRC study: "...seven very specific technical issues in dispute between the USGS geologists, the State of California and Wilshire et al. As with any NRC committee, it will not address issues extraneous to its charge --however important and broad those issues may be. In planning its recent meeting, both parties to the controversy (Wilshire and associates and the State of California) were given the opportunity to choose their agenda items, and were given the amount of time they requested for each presentation. At the end of the meeting on July 9 the committee chairman asked Dr. Wilshire if he had enough time to present his case. Dr. Wilshire responded that he had."

A Ward Valley opposition group that addressed the NAS committee's meetings in Needles has complained that they were not treated fairly. The facts are otherwise and show much greater deference paid to opponents than to organizations that use radioactive materials, generate LLRW, and will be the customers for the Ward Valley disposal facility.

At a meeting of the NAS Board on Radioactive Waste Management at the Academy's Beckman Center in Irvine, California on December 15, 1994, a spokesperson for the Committee to Bridge the Gap (CBG) stated:

"We became very concerned that the panel, even as unbalanced as it was, needed to hear both sides. The site owner and the proposed site contractor were given days of time to make formal presentations. Experts from the site opponents were precluded from making any formal presentation. Not permitted at all. My group, the Federation of Scientists, others have sued the Department of Health Services and US Ecology over this issue and were, in fact, victorious. They were permitted to make formal presentations; our group was not. The public was permitted to make five-minute comments at the end of each session from a peanut gallery, but the project's opponents were not permitted formal presentations." a, b

The CBG spokesperson then described a thirty minute presentation he was allowed to make prior to the last day's meeting in Needles with members of the committee attending voluntarily an informal session at the rear of the meeting room. He characterized this as speaking from the "back of the bus."

In fact, during the two three-day meetings in Needles, the Wilshire group put on numerous witnesses including the three geologists themselves, as well as geochemists, hydrologists (the Wilshire Report authors have stated on the record that they are not hydrologists), and a biologist. In addition, a member of the Committee to Bridge the Gap Technical Review Panel, a hydrologist, made a formal presentation as a consultant to the Wilshire group. At the NAS meeting in early July, CBG was allowed thirty minutes to make a presentation on groundwater protection issues.

Despite requests to the NAS Committee by Cal Rad Forum, no privileges were extended

to waste generators comparable to those extended to opponents. A request for thirty minutes for an informal presentation was denied.
Representatives of Cal Rad Forum and other supporters of the Ward Valley project used the open microphone sessions to make short presentations refuting claims by opponents concerning the composition of the waste stream and the need for the disposal facility in the Southwestern Compact region, to rebut statements by the Wilshire group regarding the significance of assertions of connections between ground water at Ward Valley and the Colorado River, and to outline provisions of the U.S. Nuclear Regulatory Commission's regulations which have been ignored by the Wilshire group, CBG, and others who are in opposition to the Ward Valley project and to the Low-Level Radioactive Waste Policy Act. Subsequent to the public meetings in Needles, Cal Rad Forum supplemented its oral comments made during the open microphone sessions with written submissions. The NAS Ward Valley committee repeatedly invited written submissions stating that their work would be based primarily on written material. The Cal Rad spokesperson (Pasternak) indicated his group's willingness to play by whatever rules of procedure the committee established. The CBG spokesperson urged the committee to overrule its chairman regarding access to the microphone.
It should be anticipated that Ward Valley opponents will continue to attempt to discredit the composition and fairness of the committee and its procedures (witness the "peanut gallery" comment about the committee's open microphone sessions). The apparent purpose is to discredit the outcome in advance so that the issues put forth as the basis of their opposition may be kept alive. Their complaints should be taken with a pillar of salt.
STANDARD OF REVIEW
A license has been issued by the California DHS to US Ecology to construct and operate a low-level radioactive waste disposal facility at Ward Valley. The license has no practical effect until acquisition of the site by the State of California. The license was issued by the agency authorized by federal and state law to exercise regulatory oversight and was pursuant to adopted state and federal regulations. Environmental documentation published by BLM and DHS includes both a Draft Joint Environmental Impact Report/Statement and a Final Joint EIR/S by DHS and BLM to comply with both state and federal environmental laws, as well as a Final Supplemental Environmental Impact Statement issued by the BLM. The DHS has also prepared a Safety Analysis report. The license and EIR certification are now under appellate review in state court; therefore, the Academy's review is concurrent with ongoing legal processes in the state of California for which the standard of review will be the applicable laws and regulations.
The decision by the Secretary of the Interior (in which the White House apparently participated) to request a review by the BRWM is seen by many as part of an effort by the federal Administration to "second guess" California's regulatory licensing decision. Intervention by the federal government in California's low-level waste disposal program has been criticized by many including the former chairmen of the oversight committees in the Congress: Senator Bennett Johnston and Congressmen Philip Sharp, John Dingell, and Richard Lehman. Given the regulations which have guided the applicant (US Ecology) in its site characterization work and the regulatory standard of review which has guided the state of California, it is fair to ask the NAS Ward Valley panel, what standard of review will be applied? What will be the framework for the evaluation of the concerns in the Wilshire Report? Will the work of the Committee be guided by or compatible with the governing regulations which constrain and direct the State's administrative agency and courts?
View of the State of California on the Standard of Review
At the start of the Ward Valley committee's first meeting in the City of Needles on July 7, 1994, Elisabeth Brandt, Chief Counsel and Deputy Director for the Department of Health Services addressed the committee from the point of view of an attorney who is a regulator. She described how the Department, for ten years, has "worked to elevate science over emotionalism, publicrelation, and bureaucratic convenience." She stated that "Science drove our effort for two reasons." These are that "California wanted to approve a site and a facility design that the experts could honestly say would be excellent from the technical point of view" and "...the state has a profound need for a low-level radioactive waste site to support its users of radioactive materials - medical, academic, biotechnology, industrial, and power-generating." She said "The approach which California has taken to the Ward

Valley facility should be very compatible with the interest of this committee."
Deputy Director Brandt stressed the importance of making decisions and the necessity to bring regulatory review to closure. She went on to "raise one point that is of great importance to those of us who deal not in abstract questions, but in the practical realities. Misters Wilshire, Howard, and Miller have raised seven different issues. We have answers to each of these general issues which give us sufficient confidence to go forward. Still, I would be the last person to claim that definitive answers to all of the detailed issues they have raised are known today. As an academic matter, claims can always be made that additional research and new physical measurements are necessary to increase some element of scientific certainty."
"However, I believe that the issues which they claim require further physical and analytical work do not pertain directly to the safety of the proposed Ward Valley facility. They may be issues of valid research or academic interest, and addressing them may improve our understanding of this part of California. But, given our current knowledge, they are not issues that need to be answered to determine whether radionuclides from the disposal trenches will or will not endanger public health or safety or the environment." (Emphasis added.)
She urged the members of the committee to "maintain a clear mental separation between those issues that affect health and safety related to the proposed Ward Valley facility and those issues that might tempt you as scientists to want to learn more, but that either do not relate to the safety of waste disposal, or did not need to be answered to make a licensing decision."
And, describing the Department's regulatory role: "Our concern as the regulatory agency was to thoroughly answer those questions that relate directly and measurably to health, safety and protection of the environment. We requested US Ecology and its contractors to study the site to the point where they and we were satisfied that we knew everything that was necessary to a full safety assessment for the proposed facility. Then we stopped."
And finally, "In making our determination that harm would not occur, we used two guides. The requirements of the state and federal statutory and regulatory framework, and the principle, derived from this guidance, that any exposures and all environmental effects, even when they are well within specific regulatory limits, must be as low as reasonable achievable, or ALARA."
"... The NRC has established reasonable health and safety standards which we must prove we can meet. And we have proven that the Ward Valley facility can meet those standards. But it will also operate under the ALARA principle to exceed the expectation of those standards."
View of Users of Radioactive Materials (Cal Rad Forum) on the Standard of Review. The Nuclear Regulatory Commission's "Reasonable Assurance" Standard at Part 61.23
During the open microphone session at the Committee's public meeting in Needles on August 30, 1994, Dr. Pasternak presented a statement by Dr. Budnitz and himself on the standard of review from the perspective of scientists who are also former regulators.c A written statement was also submitted for the record. Drs. Budnitz and Pasternak supported the remarks of DHS Chief Counsel Brandt: "As scientists who are former regulators, we endorse the Department of Health Services" approach and urge you to relate your scientific inquiry to the regulatory standards which must be met before a license can be issued." Drs. Budnitz and Pasternak cited the "special place" which NRC's regulations occupy in the California licensing framework. "In 1983, the California Legislature directed the Department of Health Services to adopt regulations for low-level waste disposal consistent with 10 CFR 61. In 1984, the Department adopted the federal regulations by reference. These regulations establish a "reasonable assurance" standard for license review." The statement went on to provide examples of how "reasonable assurance" applies: protection of the general public from releases of radioactivity, protection of inadvertent intruders, that the radiation protection standards of Part 20 will be met, long-term stability of the disposed waste and the site to eliminate to the extent practicable the need for site maintenance following closure, meeting the technical requirements for land disposal facilities, etc.
The statement by Drs. Budnitz and Pasternak cited errors in the Wilshire analysis that display a lack of familiarity with the NRC regulations. The criticism that plans are lacking for monitoring groundwater or the unsaturated zone down-gradient from the site assumes, in effect, that regulatory monitoring requirements will not

be followed. The Wilshire Report's assertions regarding interconnections with the Colorado River also ignore regulatory requirements. The Budnitz-Pasternak statement notes:

"Another example of how regulatory guidance can inform evaluation of the proposed disposal site relates to the assertions by Wilshire, Howard, and Miller that there is potential for hydrologic connection between the site and the Colorado River. Having postulated such hydrologic connection, Wilshire, et al stop with no further evaluation of what such connection might or might not imply with respect to reasonable assurance of protection of the public health and safety. None of the geochemical and radioactive decay processes which can reduce the concentrations of radionuclides are mentioned...Radioactive decay is, of course, time dependent, and familiarity with NRC guidance could have prevented this error. NRC's comments on its regulations contain the following (Federal Register, vol. 47, page 57450, December 27, 1982.):

The Commission believes that reliance should be placed on siting requirements which will keep water away from wastes, result in low volumes of contaminated water being released, and provide a long travel time for decay. (Emphasis added.)

"Of course, no connection to the Colorado River has been established, and both the project applicant and the Department of Health Services are on record as believing that none exists. However, even if such a connection were to exist, travel times to the Colorado River by any postulated route are in the thousands of years. This is far longer that the 500-year hazardous life of Class C waste. Without an evaluation of the physical and geochemical processes such as radioactive decay and sorption that would control radionuclide migration in groundwater (in the hypothetical case that radionuclides reach groundwater), not to mention a health physics evaluation of the impact should any radionuclides ever reach the Colorado River or any other source of drinking water, a critique based on postulated interconnections is irrelevant."

OTHER UNCERTAINTIES IN THE NAS REVIEW AND ITS ULTIMATE USE

The NAS Review

Safety of disposal of low-level radioactive waste is enhanced by factors in addition to those provided by a superior site. The NRC's regulations include a waste classification system and waste form and packaging requirements, financial requirements, requirements for post-closure institutional controls, and operator qualifications. All of these enhance the ability of the disposal project to isolate the wastes safely. Given the focus of the committee's inquiry on earth science issues, how will the committee factor in these portions of the regulations? What role, if any, will health physics and risk analysis considerations have in the NAS review?

Standard of Review by the Secretary of the Interior

To date, the Secretary of the Interior has not said how he and the Administration will utilize the NAS report in their decisionmaking on the Ward Valley land sale. In our view, it is very unlikely that the NAS report will conclude that the DHS made an error in issuing a license to US Ecology to develop and operate a LLRW disposal facility at Ward Valley. Unless the report identifies significant, new safety or environmental impacts issues not addressed by the DHS, the 1,000 acre Ward Valley site should be sold without further unnecessary delay to the State of California.

BENEFITS TO DATE OF THE NAS REVIEW

The Academy review encouraged the USGS to complete work on a report on the characteristics of the unsaturated zones at the Beatty, Nevada and proposed Ward Valley, California sites sooner than might otherwise have been the case. The USGS report, "Estimates of Percolation Rates and Ages of Water in Unsaturated Sediments at Two Mojave Desert Sites, California-Nevada, " by David E. Prudic, U.S. Geological Survey, Water-Resources Investigations Report 94-4160 was completed in late August 1994. This official, peer-reviewed report on the unsaturated zones at the Beatty and Ward Valley sites concludes, on the basis of chloride profiles, that most precipitation does not penetrate the soil to a depth of more than a few meters, and that the estimated movement of water below a depth of 10 meters is of the order of only 3 to 5 centimeters per 1,000 years. These characteristics lead to the conclusion that regulated disposal of LLRW at these sites poses no threat to ground water.

As a result of the NAS study, a number of agencies have reviewed estimates of the amount of plutonium likely to be disposed at the Ward Valley site during its

thirty-year operational life. Data compiled by the Idaho National Engineering Laboratory from waste shipment manifests show that the amounts of plutonium shipped for disposal declined markedly beginning in the early 1980s. The DHS, the USNRC, and a report by the Congressional Research Service reflect a consensus that the amount of plutonium likely to be disposed at Ward Valley is several orders of magnitude less than the amount assumed for the Ward Valley performance assessment in the license application.

CONCLUSIONS

An evaluation of the "Wilshire Report" and the Ward Valley site is underway by a committee convened by the National Academy of Sciences/National Research Council Board on Radioactive Waste Management. Completion of the study report, originally expected in December 1994 is now expected in early March 1995.

Opponents of the Ward Valley project have criticized the composition of the committee and the fairness of the committee's procedures. The competence and balance of the committee has been capably defended by Dr. Bruce Alberts, President of the NAS, and a review of the committee's proceedings show far greater deference was paid to opponents of the project than to supporters in terms of time allowed for presentations and accepting witnesses for formal presentations.

The NAS review probably prompted USGS to complete work on a study of the vadose zones at Ward Valley, California and Beatty, Nevada sooner than might have been the case otherwise. The findings of this official, peer-reviewed USGS report confirm the existence of characteristics that make these arid sites suitable for safe disposal of LLRW. As a result of the NAS review, reevaluation of the amounts of plutonium likely to be disposed of at Ward Valley have been carried out by the Department of Health Services, the Congressional Research Service, and the USNRC. These reviews indicate a much smaller amount of plutonium is likely to be disposed at Ward Valley during its thirty-year operating life than assumed for the purpose of the Ward Valley performance assessment in the license application.

A license, dependent upon the land transfer for practical effect, has been issued for the proposed Ward Valley LLRW disposal project, and judicial review of the license and accompanying environmental documentation is ongoing. The standard for both administrative action and judicial review are clear; they are governed by statutes and regulations of the federal and state governments. The NAS review is concurrent with judicial review, but the standard for the NAS review is not clear. Both the California Department of Health Services and Cal Rad Forum have urged the committee and the BRWM to look to the legally adopted regulations as a framework for their review.

Upon release of the NAS report, the Secretary of the Interior should sell the land for the Ward Valley site to the State of California without further delay unless the report identifies significant new issues not addressed by the DHS in its licensing record.

Session 15 -- Hydrogeologic Considerations in the Disposal of Radioactive and Mixed Waste

Co-chairs: Michelle Rehmann, Energy Fuels Nuclear;
Nancy Rothermich, Martin Marietta Energy Systems

15-1

AN ALTERNATIVE CONCEPTUAL MODEL FOR THE SATURATED ZONE AT YUCCA MOUNTAIN NEVADA

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ABSTRACT

State of Nevada funded research on water level fluctuations in the vicinity of Yucca Mountain has led to the formulation of a different and more complex conceptual model of the saturated zone hydrology than has previously been envisioned or analyzed for the site. The alternative saturated zone model posed herein requires a close coupling of thermal, tectonic and hydrologic processes in a structurally controlled domain.

INTRODUCTION

This paper summarizes separate studies funded over the past few years by the State of Nevada Nuclear Waste Project Office. These studies include the analyses of water level data from the saturated zone and computer modeling exercises of unsaturated zone hydrologic data. The results of these individual studies, together with

observations of the potentiometric surface and temperature data, led to the development of a different, more complex model of the saturated zone than presently analyzed in performance assessments.

These studies suggest that saturated zone flow around Yucca Mountain is complex and not fully understood. Evidence exists for locally focused recharge. Recharge can also be focused along fault and fracture zones in both the saturated and unsaturated zones. The data also indicate that there exist hydrologic domains bounded by faults. These domains are only loosely connected and respond to recharge and seismic disturbances differently. In addition, the seismic responses of the water table indicate significant coupling between the hydraulic system and local stress conditions.

ANALYSES

Water Level Oscillations

Analysis of water table oscillations was performed on eight water table wells in the vicinity of Yucca Mountain. The period, phase shift, amplitude of water level measurements was analyzed over a several year period. The results of this study indicated that while amplitudes were small, there were marked similarities in periods and phase shifts between wells on the west side of the mountain while wells located on the east side of the mountain show different similarities.(1) Figure 1 shows the locations of the wells studied indicating similar grouping.

The similar groups of wells line up along a direction parallel to the major north to south trending fault zones. This was the first indication that, 1) the flow field at Yucca Mountain was structurally controlled and, 2) that there were two separate, or weakly coupled flow systems. The finding of separate systems was also supported by chemistry data produced by the Desert Research Institute which indicated chemistry on the east side was quite different than water chemistry on the west. (2) The frequency of oscillations were shorter on the east side than the west, exhibiting an average of 2.4 years on the east and 2.6 years on the west. These frequencies were compared with a plot of rainfall distribution from 7 rain gauging stations in a broad region surrounding Yucca Mountain. It showed an approximate 2.5 year cycle in rainfall periodicity. This could indicate that even these deep wells at Yucca Mountain are responding to the rainfall frequency.

Effects of Earthquakes on Water Levels

As part of the DOE environmental monitoring program, the water levels in environmentally sensitive springs and nearby wells are routinely monitored. Of these springs Devil's Hole is a critical habitat for the desert pupfish, an endangered species. Devil's Hole is located about 25 miles south of Yucca Mountain near Ash Meadows adjacent to the Stewart Valley Fault which trends in a northwest to southeast direction. As a result of a series of earthquakes experienced during the summer of 1992, Devil's Hole water levels underwent changes. These earthquakes were the June 28th 7.5 magnitude earthquake centered at Landers, California, the 6.6 magnitude earthquake which occurred 3 hours later centered near Big Bear, California, and the June 29th, 5.6 magnitude earthquake centered at Little Skull Mountain, Nevada, just 23 kilometers southeast of Yucca Mountain.

The response to these earthquakes differed in the various wells located near Devil's Hole. The level in Devil's Hole showed a temporary drop, while a nearby well indicated a significant rise. This difference was puzzling, but it was noted that the location of these two monitoring points was separated by the Stewart Valley Fault and that perhaps the fault was influencing the responses. This led to examination of all the DOE water level data available in the Yucca Mountain region during the time period of the earthquakes. The water level data showed four distinct types of response to the earthquakes as follows:

- an upward temporary spike
- a rapid upward change with an apparent long-term stabilization at a higher level
- a downward temporary spike
- a rapid downward change with an apparent long-term stabilization at a lower level.

Table I

Because of potential fault control of these responses, the wells were plotted and compared with fault traces in the area. Figure 2 shows the locations of wells which experienced water level changes greater than 15 cm, and the locations of major fault zones. Increasing water levels are indicated by upward arrows and downward trends by downward pointing arrows. It can be seen that most of the wells which experienced increased water levels are closely associated with the northwest

trending shear zones. The downward trending water levels do not correlate to the shear zones but align more closely to normal faults.
The most obvious explanation for these responses might suggest that some areas experienced compressive strain, reducing effective pore volume and raising water levels, while other areas experienced tensile strain, increasing effective pore volume and lowering water levels. Alternatively, the result of nearby earthquakes and sudden changes in local stress may be to cause changes in fracture aperture where fractures are held at relatively low closure pressure. This would lead to nonlinear permeability changes in the host rock causing large scale adjustments in the hydrologic system including rapid water table elevation change, altered flow patterns, and adjustments in flow velocities. This process may be coupled with thermal flow as well so that adjustments in fracture and fault properties may affect the locations and magnitudes of local convection features. In this case the explanation for the observed water table responses might be a unique combination of effects from strain on hydraulics and thermal flow for each monitoring location. These observations have several implications. First, that the level of the water table is structurally controlled and is closely linked to the ongoing tectonics in the region. These observations may also indicate that the hydraulic conductivity field in these fault zones is transient, not steady as is currently assumed in performance assessment models. Other impacts to the flow field are local changes in hydraulic gradient and direction or diversions of the flow field. One well experienced a water level rise of 16 feet, which in flat gradient areas of the flow field could cause temporary, if not permanent changes in flow direction and velocity.
Unsaturated Zone Analyses
Analyses done as part of the INTRAVAL project included predictions of water content within deep boreholes at Yucca Mountain and calculations of potential recharge using traditional runoff techniques. These analyses have been documented in a previous paper and are not repeated here. (3) The results of these studies indicated that significant infiltration through the unsaturated zone was possible and that this infiltration may be controlled by faults and fracture zones. Models applied at INTRAVAL which simulated flow through discrete fractures (simulated as high conductivity zones), did a better job of matching water content and were more consistent with water chemistry data. These fracture controlled infiltration models also allowed for more rapid infiltration through the unsaturated zone than could be accounted for in the more traditional matrix flow models. Areas where significant infiltration occurs are limited and most likely occur along fractures and fault zones and areas of fault intersections. This focused infiltration concept also allows for very high flux rates to occur locally, rather than as a uniform distribution of rainfall. Uniform distributions were a commonly used assumption in performance assessments completed at that time.
Other Observations
The US Geological Survey has also published information relevant to this conceptual model of the saturated zone flow field. These data include the newly revised potentiometric surface data and temperature data at the water table. The potentiometric data indicate a series of embayments or potentiometric lows coincident with northwest trending shear zones running along the eastern flank of the mountain. This would indicate that water is draining into these fault zones. The temperature distribution at the water table is shown as Fig. 3. This temperature data shows a cold water plume coincident with the Ghost Dance Fault zone, which runs through the center of the repository block. An interpretation of this temperature data is that cold water is moving from the high gradient area to the north of Yucca Mountain via the Drill Hole Wash Fault or other nearby faults, into the Ghost Dance Fault. This interpretation is considerably different than the published flow models of the US Geological Survey which indicate a more uniform and generally eastward flow field across the Yucca Mountain block. (4) The water level measurements alone may not be accurate enough to show movement of a narrow fault controlled plume of cold water. It is also possible that the anisotropic effects of the fault control or dominate the flow at this location.
Additionally, published observations of the US Geological Survey show other areas where similar anomalous gradients such as those encountered at Yucca Mountain occur. (5) One anomalous gradient appears just to the north of the Ash Meadows region and coincides with a fault intersection. It was also concluded that the flow field in

the Nevada Test Site region was highly compartmentalized rather than a continuous uniform field.
Finally, Czarnecki analyzed a breach of the large hydraulic gradient as a potential scenario to be considered. (6) In his model, the resultant flow field was quite similar to the temperature plume shape, shown in Fig. 3. This more southerly flow was different than the normally modeled base case, primarily due to rock property changes and higher head gradients to the north. This same type of flow response is expected for the model proposed herein.
Alternative Saturated Zone Model
Taken together the above information resulted in a different conceptualization of the flow field than that currently being considered. The model is structurally controlled by fault or fracture zones of a self-similar fractal nature . Fracture zone intersections play a key roll in the distribution of recharge, velocity fields and pathways. The model is also dynamic rather than static, and has the potential to change rapidly due to tectonic movements.
The proposed model of flow is shown in Fig. 4. This figure shows the potentiometric surface and the proposed flow paths. It allows some water movement to occur across the mountain block from east to west, primarily via discrete northwest trending fracture zones. The Solitario Canyon Fault zone creates a resistance to eastern flow but does not prevent it. A steep hydraulic gradient exists at the location of the Solitario Canyon Fault and is equal to 35 meters of head difference over a lateral distance of about 1000 meters. Water movement across this fault probably occurs as a result of intersections with northwest trending shear zones and creates flow to the next lower level of the water table. The water table directly under Yucca Mountain is very flat.
Colder flow also enters the Yucca Mountain block from the northwest across a very steep hydraulic gradient. This gradient is equal to over 300 meters of head change across 2500 meters distance. The faults in the Drill Hole Wash region no doubt play a role in the transport of water across this hydraulic barrier. Where the Drill Hole Wash Fault or those near it intersect the northern extension of the Solitario Canyon Fault a potential breach may occur and allow the colder water north of the steep gradient to move down this fault zone and subsequently into the Ghost Dance Fault. The proposed model and the potentiometric surface also suggest that another fault zone exists just to the south of the repository footprint. This zone may also be transporting water from the Solitario Canyon side of the block toward the east. This area of the mountain has not yet been mapped in detail.
The proposed alternative model leads to many challenges for site characterization because a number of questions must be answered. Some of them include:
1.What changes to the existing site characterization program are needed to fully examine a complex, compartmentalized, and dynamic flow model containing fractal-like properties rather than a simplistic model of the flow field with statistically described parameters?
2.What are the implications to repository performance assessments or risk assessments of the following?
The flow field is dynamic and water-level changes can be expected due to tectonic events, which could lead to changes in gradients and possible local reversals or diversions in flow directions.
Velocity fields may also change with time, perhaps compression along the northwest trending shear zones causes increases in velocities, while extension along north south trending faults may cause decreases in velocity.
3.What are the effects of the large temperature differences encountered across the mountain block? Are they affecting the velocity field in a significant way?
While we cannot answer all of these questions at present, we can ask that the DOE investigate the implications of this alternative model. Any future modeling and performance assessments, to be credible, must be consistent with the hydraulic and temperature data at the very least. The emerging isotope chemistry data should also help to better define the flow field and answer some of the above questions.
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15-2
FLUID FLOW THROUGH VERY LOW PERMEABILITY MATERIALS: A CONCERN IN THE GEOLOGICAL ISOLATION OF WASTE
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ABSTRACT
The geological isolation of waste usually involves the selection of sites where very low permeability materials exist, but there are few earth materials that are truly impermeable. Regulatory concerns for the containment of radioactive material extend for geologic periods of time (i.e., 10,000 years or more), and it becomes nearly impossible to "assure" the behavior of the site for such long periods of time. Experience at the Waste Isolation Pilot Plant (WIPP) shows that very slow movements of fluid can take place through materials that may, in fact, have no intrinsic permeability in their undisturbed condition. Conventional hydrologic models may not be appropriate to describe flow, may provide modeling results that could be in significant variance with reality, and may not be easy to defend during the regulatory compliance process. Additionally, the very small volumes of fluid and very slow flow rates involved are difficult to observe, measure, and quantify.
The WIPP disposal horizon is excavated 655 m below the surface in bedded salt of Permian age. Salt has some unique properties, but similar hydrologic problems can be expected in site investigations were other relatively impermeable beds occur, and especially in deep sites where significant overburden and confining pressures may be encountered. Innovative techniques developed during the investigations at the WIPP may find utility when investigating other disposal sites.
The details of flow in these very low permeability units is quite complex and difficult to quantify. Vertical drillholes yield inconsistent data, even when closely spaced, but horizontal drillholes provide consistent and comparable data sets. Flow may be constrained to a relatively few, fairly discrete, bedding planes and radial flow (as assumed in most modeling) toward an excavation or drillhole may not occur. Fluid preferentially occurs in the more argillaceous beds and is not uniformly distributed throughout the salt. The pore spaces in some units may be so small that surface tension forces become significant and Darcy's Law may have to be applied in a modified form or may not hold at all. Additionally, some previously unsuspected flow mechanism may be acting, such as compaction in the pillars driving brine out of poorly compacted clays.
Ongoing work at the WIPP is expected to continue to advance understanding of flow through very low permeability materials. The study of flow under these conditions will become increasingly important as additional waste disposal sites are designed that require assurance of their safety for geological periods of time.
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The geological isolation of waste usually involves the selection of sites where very low permeability materials exist, but there are few earth materials that are truly impermeable. Regulatory concerns for the containment of radioactive material extend for geologic periods of time (a mandated 10,000 years). The study of ground water and the engineering aspects of hydrogeology have traditionally focused on ground water as a resource (1, 2). As a result, the focus has been on the conditions of flow where the quantity and quality of the water available are adequate for human

use and the materials containing the water allow it to flow (or be induced to flow) rapidly enough so that useful quantities can be produced in reasonable amounts of time.

Disposal sites are usually chosen on (or in) materials with very low permeability for which conditions of flow are less well understood. In order to predict the long term behavior of any such disposal site, we must first have experience with and an understanding of the detailed behavior of the systems acting at the disposal site. Uncertainties in the prediction of repository behavior can be significantly reduced by continued site investigations, but can never completely be eliminated. For these and other reasons, the National Research Council (3) concluded that science cannot "prove" (in an absolute sense) that a disposal site will be "safe" as defined by existing Environmental Protection Administration standards and United States Nuclear Regulatory Commission regulations.

Numerous recent studies have demonstrated that describing flow under low permeability conditions stretch the limits of our knowledge for shales (4, 5, 6), clays (7, 8), unsaturated fractured rock (including shales, clays, tuff, and basalts)(9), and evaporites (10, 11). Recent experience at the Waste Isolation Pilot Plant (WIPP) shows that very slow movements of fluid can take place through materials that may, in fact, have no intrinsic permeability in their undisturbed condition. Conventional hydrologic models may not be appropriate to describe flow (12), may provide modeling results that could be in significant variance with reality, and may not be easy to defend during the compliance process. Additionally, the very small volumes of fluid and very slow flow rates involved are difficult to observe, measure, and quantify.

The WIPP disposal horizon is excavated 655 m below the surface in bedded salt of Permian age. Considerable stratigraphic variations occur within the salt beds. Although dominantly halite, individual units range from clear halite through argillaceous halite to polyhalitic halite. Interbeds include clay seams and anhydrite beds. Water is present in several ways within the Salado Formation (11): (1) within hydrous minerals such as gypsum and clays, (2) as fluid inclusions within halite and other crystals, (3) within intergranular pores and open fractures, and (4) as intergranular moisture within poorly consolidated clays which occur within salt crystals as well as between halite, anhydrite, and other crystals.

The undisturbed clear halite units may be effectively impermeable. State-of-the-art permeability testing (13, 14) was unable to measure any permeability, indicating that if it exists at all, intrinsic permeability of the clear halite units is less than 1 x 10 -23 m2 (0.01 nanodarcy). Those halite units that contain a few percent clay are more permeable, typically less than 1 x 10-20 m2 (10 nanodarcy). The permeability of the interbedded anhydrite units are several orders of magnitude greater, typically between 1 x 10-19 and 1 x 10-18 m2.

Salt has some unique properties, including the fact that at repository depths it deforms plastically. Studies undertaken during the Brine Sampling and Evaluation Program (BSEP) at the WIPP (15, 16, 17, 18, 19, 20, 21) were directed primarily toward the environment in and directly adjacent to the underground excavations. These studies and others (22, 23, 24, 25) show that the rock immediately surrounding the excavation is altered significantly from its original state due to the influence of deformation induced by the rock excavation and the movement of salt toward the excavation. The common theme running through the BSEP investigations relates to the presence and movement of brine in rocks that saw little to no fluid migration prior to the development of deviatoric stress accompanying excavation and the permeability enhancement caused by elastic expansion and brittle deformation of the salt and anhydrite units.

A halo of deformation forms around the excavations, whether they are rectangular or circular in cross section (Fig. 1). The development of this halo of deformation around an underground excavation at the WIPP, sometimes described as the Disturbed Rock Zone, is discussed by Deal and Roggenthen (11). They point out that there are generally two parts to the deformational envelope around underground excavations in salt: an outer zone where dilatency and microfracturing occur with pore pressures above atmospheric (zone C in Fig. 1), and an inner zone characterized by macrofracturing and pore spaces where the pressures are essentially at atmospheric (zone B in Fig. 1). Some authors tend to treat the inner zone, which includes the volume of rock that has separated (decoupled) from the host rock, as simply a growing part of the excavation comprising the "Actual Opening" (26). Brine moving

toward the excavation behaves differently in these two zones, and it is important to consider both of them when discussing brine seepage into the WIPP excavations.
FIG. 1
The salt at the WIPP originated as a stratified and bedded sedimentary rock and consists of alternating sequences of halite, argillaceous halite, polyhalitic halite, clay layers, and thin anhydrite beds. As a result, there are numerous horizontal discontinuities. There are clay partings and thin (1-3 cm) clay beds, as well as beds of anhydrite ranging from a few millimeters to a meter or so in thickness. The anhydrite beds are brittle and do not deform plastically at repository depths. Typical storage rooms are 4 m (13 ft) high, and 10 m (33 ft) wide. Therefore, the deformational sequence is complicated by the effects of geometry and the stratigraphy as the disturbed envelope is driven toward a circular geometry (Fig. 2).
FIG. 2
Evidence is abundant that the excavation geometry around the openings at the WIPP is modified by these discontinuities and inhomogeneties. It includes failure of roof and floors due to heaving, separation along clay seams, and the development of macrofractures in ribs (11, 22, 25). The patterns of fracturing and deformation observed at the WIPP is shown in Fig. 3.
FIG. 3
The details of flow in these very low permeability units is quite complex and difficult to quantify. Vertical drillholes yield inconsistent data, even when closely spaced, but horizontal drillholes provide consistent and comparable data sets (21). Flow may be constrained to a relatively few, fairly discrete, bedding planes and radial flow (as assumed in most modeling) toward an excavation or drillhole may not occur (21). Fluid preferentially occurs in the more argillaceous beds and is not uniformly distributed throughout the salt (18). The pore spaces in some units may be so small that surface tension forces become significant and Darcy's Law may have to be applied in a modified form, such as the piece-wise method suggested by Deal et al. (18), or may not hold at all. Additionally, some previously unsuspected flow mechanism may be acting, such as compaction in the pillars driving brine out of poorly compacted clays (18).
A number of modeling efforts have been made in an attempt to predict seepage into the WIPP excavations. Seepage into a horizontal drillhole 7.6 cm (3 in) diameter and 46 m (150 ft) long is predicted to be on the order of 0.01 liters per day if a permeability of 1 x 10-22 m2
(0.1 nanodarcy) is used for the undisturbed salt. Three drillholes of that dimension have been monitored for over 2 years and all three accumulate fluids at seepage rates on the order of 0.01 to 0.02 liters per day.
The way in which flow rate varies with time is important. If flow rate eventually reaches a steady rate, then there may be some far-field brine that flows through the body of the undisturbed rocks to reach the repository excavations. If flow rate continues to decrease and eventually ceases (21), then there is no significant amount of brine derived from the far-field and only brine released from the disturbed rock zone due to depressurization will enter the repository excavations. Observations are presently being made at the WIPP to determine which of these conditions exist.
If no far-field flow exists and radial flow occurs in all directions toward a waste storage room, then release of brine from the disturbed rock zone around the excavations due to depressurization is estimated to produce about 150,000 liters of brine (21). This volume is on the same order of magnitude as the volume of brine (220,000 liters) necessary to corrode all the metal in the waste and waste storage drums (21). Anoxic corrosion will consume brine and produce metal oxides and hydrogen. If the volume of brine entering the repository is less than that required to completely corrode the metal, then all the brine that comes in contact with metal will be consumed.
As pointed out above, there is good evidence that the assumption of radial flow may not hold for the WIPP. The undisturbed clear halite units have such low permeability (or none at all) that flow is probably constrained and only occurs horizontally, parallel to bedding. In that case, less than one tenth of the 150,000 liters estimated above may enter the repository to react with the metal stored there. If compaction of the clays is the source of the brine rather than release of brine from the disturbed rock zone due to depressurization, then even less brine may enter the

repository.
Continued site investigations at the WIPP are defining the bounds of the brine seepage phenomena. After extensive modeling, field investigations, and experiments, some uncertainties still exist. Although the details of flow in the geologic units at the WIPP are quite complex and difficult to quantify, it is clear that only very small quantities of brine moving very slowly are involved. Although field investigations show that very little brine actually enters the excavations (see BSEP references cited previously), it is impossible to completely eliminate all uncertainty. The fact that some uncertainties remain should not be sufficient to prove that the disposal site is "unsafe." Some other criteria should be used to determine whether or not the relative risk of transporting this waste to the WIPP and disposing of it there is less (and more acceptable to society) than the risk of leaving it where it is or disposing of it in some other way.
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15-3

DOE'S PERFORMANCE EVALUATION PROJECT FOR MIXED LOW-LEVEL WASTE DISPOSAL*

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ABSTRACT

A performance evaluation (PE) is an analysis that estimates radionuclide concentration limits for 16 potential Department of Energy (DOE) mixed low-level waste (MLLW) disposal sites based on the analysis of two environmental exposure pathways (air and water) to an off-site individual and an inadvertent-intruder exposure pathway. Sites are analyzed for their ability to attenuate concentrations of specific radionuclides that could be released from wastes in a hypothetical MLLW disposal facility. Site-specific data and knowledge are used within a generic framework that is consistent across all sites being evaluated. After estimates of waste concentrations for the three pathways are calculated, the minimum of the waste concentration values is selected as the permissible waste concentration for each radionuclide. The PE results will be used as input to the process for DOE's MLLW disposal configuration.

Preliminary comparisons of results from the PE and site-specific performance assessments indicate that the simple PE results generally agree with results of the performance assessments, even when site conditions are complex. This agreement with performance-assessment results increases confidence that similar results can be obtained at other sites that have good characterization data. In addition, the PE demonstrates a method to provide DOE decision makers with a simple, conservative, defensible, and easily understandable analysis that provides results similar to those of more complex analyses, and which can be used to aid MLLW disposal configuration plan development.

INTRODUCTION

The Federal Facility Compliance Act (FFCAct) of 1992 (1) requires DOE to work with its regulators and with members of the public to establish plans for the treatment of DOE's mixed wastes. Although the FFCAct does not specifically address disposal of treated MLLW, both DOE and the States recognize that disposal issues are an integral part of treatment discussions. The DOE established its MLLW Disposal Workgroup in June 1993 to work with the States to define and develop an evaluation process for disposal options. This joint DOE-State process has narrowed the sites for further evaluation from 49 to 16.

A PE has been developed to quantify and compare the limitations of 16 DOE sites for the disposal of MLLW. The principal goal in developing the PE is to estimate, for residues resulting from the treatment of MLLW, permissible concentrations of radionuclides that could be disposed at each site. The PE consists of simple analyses consistent with the approach used in many low-level waste (LLW) performance assessments. The objective is to use a set of modeling assumptions of sufficient detail to capture major site-specific characteristics and yet be general enough for consistent application at all sites. Additionally, the analyses must ensure that no systematic biases are introduced, the sites are analyzed consistently, and all major assumptions are clearly stated.

Although the approach is simple, every effort is made to ensure that the PE is technically adequate for the intended purpose and that the PE results reflect the strengths and weaknesses of each of the 16 sites for the disposal of MLLW. To ensure technical adequacy, the following strategy was adopted:

Use the existing knowledge, analyses, and data at each site to the extent practical;

Use well-established policies and recommendations on disposal-related issues; and

Use extensive and continuous reviews from both internal and external experts.

The PE analysts rely extensively on interactions with site personnel to utilize important research, site characterization, modeling, and other analyses that have been performed. These interactions include visits by the PE analysts to each of the sites. Based on discussions with personnel who have spent years studying their sites, the PE analyses incorporate the best documented understanding of the sites into the PE framework.

FRAMEWORK OF THE PERFORMANCE EVALUATION

The PE is solely a radiological assessment even though the waste is mixed with hazardous components that are subject to Resource Conservation and Recovery Act (RCRA) requirements (2). This approach is taken by assuming that the hazardous component of the waste is treated and the disposal facility designed per RCRA specifications. Further analysis of the performance of hazardous components currently is not required by the Environmental Protection Agency (EPA) or any state with RCRA primacy.

The PE estimates permissible concentrations of radionuclides in disposed waste using the performance objectives specified in DOE Order 5820.2A (3) as "performance measures" for the radiological assessment. Performance measures are used in the PE to relate radionuclide concentrations in disposed waste to permissible doses to individuals for three exposure pathways: ingestion of drinking water and all-pathways exposure to atmospheric releases to an individual at the performance boundary (i.e., point of compliance), and all-pathways exposure to an inadvertent intruder. All of the performance objectives in DOE Order 5820.2A could be used as performance measures in the PE. However, experience with performance assessments (4, 5) has shown that some of the performance objectives are generally more restrictive than others. Thus, the performance measures used in the PE are:

4 mrem (0.04 mSv) per year from the drinking-water pathway for releases to water;

10 mrem (0.1 mSv) per year from all pathways for atmospheric releases; and

100 mrem (1 mSv) per year from all exposure pathways for chronic exposure of inadvertent intruders.

Based on guidance for conducting performance assessments (6), calculations for the PE use a performance boundary that is located 100 m from the edge of the disposal facility, and the period for consideration is 10,000 years from the time the disposal facility is closed.

Two generic disposal facilities are considered in the PE: a RCRA-compliant, below-ground trench and a RCRA-compliant, above-ground tumulus. The size and shape of each generic facility is assumed to be the same for all sites. The generic

disposal facilities were selected to provide consistency in evaluation of the 16 sites.
The same list of radionuclides is used in the PE to enable comparisons of the 16 sites. Because the actual radionuclide inventory that might be disposed in a facility is unknown, the duration of contaminant release is also unknown. Therefore, a continuous release of radionuclides from the disposal facility is assumed. Assuming a continuous source is considered to be reasonable and conservative.
The waste form in the PE is assumed to be grouted treatment residuals. Grout, which consists primarily of hydrated Portland cement and fly ash, is often used to stabilize both wastes containing hazardous metals and residues resulting from thermal treatment. Grout is the primary waste form considered in three LLW performance assessments: Oak Ridge Solid Waste Storage Area 6 (4); Savannah River Z-Area (5); and Hanford grout (7).
For the water and atmospheric pathways, the sequential attenuation of contaminants that occurs between the waste in the disposal facility and the performance boundary is represented by "concentration reduction factors" (CRFs). The CRF approach is used so that intermediate results can be displayed to allow for comparisons of effects of the disposal facility and site on overall performance. For the water pathway, the CRFs represent concentration attenuation related to the contaminant release rate from the facility and dilution of leachate with uncontaminated groundwater. For the atmospheric pathway, the CRF represents the reduction in contaminant concentration provided by diffusion through the overlying soil and by atmospheric dispersion to the performance boundary.

METHODOLOGY

Three sets of calculationsfor the drinking-water, atmospheric, and inadvertent-intruder pathwaysform the foundation of the PE. In general, for each pathway the maximum permissible contaminant concentration at the performance boundary is determined for each radionuclide by using the performance measures and the appropriate pathway or scenario dose conversion factors (annual effective dose equivalent per unit concentration) based on EPA dose conversion factors (8). Conceptual models for environmental flow and transport are developed based on interactions with site personnel. For the water and atmospheric pathways, the concentration reduction provided by the environment is estimated using results of site analyses and data evaluation. For the intruder analyses, concentration reduction is estimated for appropriate exposure pathways for several intrusion scenarios; interactions with site personnel focus on selecting appropriate "credible" intrusion scenarios based on site-specific conditions. Once estimates of waste concentrations are calculated for the three pathways, the minimum of these values is selected as the permissible waste concentration for each radionuclide.

Water Pathway

Different hydrogeologic environments cause the water flow to vary considerably from one site to another; therefore, an acceptable conceptual model and its associated assumptions must be supported by site-specific knowledge and data. Depending on site-specific conditions, pathways to be considered may be surface water, the vadose zone, and groundwater. Once the water-flow paths are established, similar radionuclide-transport assumptions are imposed on each site to provide consistency in the PE analyses. A generic conceptual model incorporates site-specific geometry and water-flow pathways into a simple transport analysis (Fig. 1). Results are compared with available site-specific analyses to identify areas where differences exist.

FIG. 1

Site-specific data required to perform the water-pathway analysis come from site personnel either as data from site characterizations and reports or from site-selected literature. The minimum data that are required are natural infiltration rates, distance between the disposal facility and groundwater, Darcy flow rates, porosity, ambient moisture content, mixing depth in the aquifer, dry bulk density of the porous media, and solid/liquid partition coefficients of the porous media.
Two CRFs are calculated for the drinking-water pathway: one for the concentration attenuation between the disposed waste and leachate exiting the bottom of the disposal facility, source CRF; and one for the concentration attenuation between the leachate exiting the disposal facility and the water at the performance boundary, the environmental transport CRF for drinking water. The source CRF is defined as the

dimensionless ratio of the waste concentration to the resulting leachate concentration. Desorption with advecting flow is the mechanism used to describe the leaching of radionuclides from the grout, consistent with analyses in LLW performance assessments that have evaluated grouted waste forms (4, 5, 7). Because the desorption model is based on a grouted waste form, the radionuclide-specific values for the source CRF for each type of generic facility are the same for all 16 sites.

The environmental transport CRF for drinking water is defined as the ratio of the concentration of the leachate exiting the disposal facility to the resulting concentration in water at the performance boundary. The concentration attenuation represented by the transport CRF for drinking water consists of dilution due to mixing with uncontaminated groundwater and/or surface water. Effects of radioactive decay and decay product ingrowth are also included in the analysis.

Atmospheric Pathway

The conceptual model for evaluating the atmospheric pathway is derived from performance assessments for LLW disposal facilities (5, 7, 9, 10). The model is generalized for the PE and uses site-specific data for many of the parameters. Only the volatile radionuclides 3H and 14C are considered in the PE for atmospheric transport. In the model, radionuclides are transported from the disposal facility to the soil surface by vapor diffusion. When the radionuclides reach the soil surface, they are entrained in the air and are transported to an assumed receptor located at the performance boundary. Atmospheric dispersion also causes minor attenuation. Site-specific data required to perform the atmospheric pathway analysis come from site personnel either as data from site characterizations and reports or from site-selected literature. The site-specific data required for the soil diffusion calculation are the same as those required for the water pathway. Atmospheric dispersion data include probabilities of wind speed and direction and stability class.

Three CRFs are used to account for the attenuation encompassing diffusion to the ground surface, mixing in air, and dispersion in the atmosphere to the performance boundary. The CRF for diffusion in soil is conservatively modeled (i.e., large diffusion constants) in an attempt to bound releases from alternative transport mechanisms (e.g., desiccation cracks, burrowing animals, and root uptake). The CRF for diffusion in soil is defined as the ratio of the radionuclide concentration in the waste to its resulting concentration in the upper one centimeter of soil. The CRF for mixing with air is defined as the ratio of the radionuclide concentration in the upper one centimeter of soil to its resulting concentration in air at the facility boundary. The CRF for atmospheric dispersion is defined as the ratio of the concentration at the facility boundary to its resulting concentration at the performance boundary 100 m from the edge of the disposal facility. Gaussian dispersion is the concentration-attenuating mechanism used in the dispersion CRF. The effect of radioactive decay is included in the analysis.

Inadvertent Human Exposure

Standard scenarios that were developed for performance assessments of LLW disposal facilities (4, 11) are used in the PE to analyze inadvertent intrusion. These scenarios were selected based on experience indicating that the chronic agriculture (homesteader) and post-drilling scenarios are generally the most restrictive scenarios for most sites and facility designs. In some cases, however, these scenarios are tailored for site-specific conditions. The agriculture scenario includes establishment by an intruder of a permanent homestead directly above a disposal facility with the foundation of the home extending into the waste; some of the waste exhumed from the disposal facility is mixed with native soil in the intruder's vegetable garden. The post-drilling scenario considers the construction of a well for a domestic water supply by an intruder who resides permanently nearby a disposal facility; the well is drilled through the disposal facility.

Total intruder doses consist of doses from several exposure pathways (e.g., external exposure to the waste, ingestion of food or soil, and inhalation of suspended soils). Factors that are used to calculate the doses and, therefore, scenario dose conversion factors include, for example, correction factors for the amount of natural soil mixed with exhumed waste, plant to soil concentration ratios, shielding factors, and fraction of time exposed through a certain exposure pathway. Some of the factors used to calculate the scenario dose conversion factors are constant, while others may vary depending on such factors as exposure pathway and disposal

technology. The effect of radioactive decay prior to intrusion is included in the analysis.

DISCUSSION

The PE project provides estimates of the maximum permissible concentrations that can be disposed of at 16 DOE sites for 58 radionuclides based on the most constraining of three performance measures: 4 mrem (0.04 mSv) per year dose from drinking water, 10 mrem (0.1 mSv) per year dose for all pathways from atmospheric releases, and 100 mrem (1 mSv) per year dose for all pathways from inadvertent intrusion. These estimates of concentrations are based on a relatively simple analysis considering existing generic and site-specific knowledge and data.

The primary use of the PE results will be to compare the estimates of permissible concentration limits for disposal with concentration estimates for stabilized MLLW residues resulting from treatment processes. The comparison will provide indicators both of the amounts and types of waste that are acceptable for disposal at each site and of the wastes that are not acceptable for disposal at each site.

Comparison of results from the 16 sites requires that the general analysis framework be as consistent as possible, while site-specific input is required to ensure that the analyses are representative of site conditions. A simple flow and transport analysis that allows incorporation of site-specific flow conditions meets both these objectives. The project framework and results are reviewed by internal and external panels to ensure consistency of application among sites and to ensure that results and conclusions are supported by the data and analyses. Interactions with site personnel familiar with site-wide characterization efforts, environmental restoration efforts, and performance-assessment and modeling efforts ensure that the best understanding of site characteristics and data relevant to waste disposal are incorporated into the analyses.

The PE is a deterministic analysis, and a quantitative uncertainty analysis is not performed. The degree of uncertainty associated with site-specific data and conceptual models varies from site to site, depending on the current level of understanding of radionuclide transport and availability of data to support the conceptual model. Less uncertainty may exist at sites where extensive site characterization work has been done. Uncertainty is addressed qualitatively in the PE by discussing confidence in data and results based on the quality of the data source (e.g., well-documented site-specific data, site-supported literature data, or unsupported assumed data) and level of understanding of flow and transport at the site.

Care has been taken in executing the PE to avoid bias in analyses that draw comparisons between sites with vastly differing attributes (e.g., small versus large sites, western versus eastern sites, or arid versus humid sites), and efforts have been made to ensure that the results of the PE do not indicate unwarranted advantages or disadvantages for any disposal site. Sensitivity analyses are used to identify the magnitude of change in parameter values required to change the controlling pathway or scenario. For example, one sensitivity analysis will determine the increase in infiltration rate required to change the controlling pathway for a permissible radionuclide concentration from the intrusion pathway to the drinking-water pathway. Analysis results indicating that a large change in infiltration rate would be required indicates the robustness of results to the assumed value of infiltration. Similar analyses will be performed for other input values and modeling assumptions.

Preliminary comparisons of PE with performance-assessment results have shown that the results of the simple PE model generally agree with results of the performance assessments, even when site conditions are complex. No direct comparisons can be made at sites with no existing performance analyses, but, based on the comparisons with performance-assessment results, the PE results are expected to provide a reasonable representation of site conditions when good characterization data are available. Sites with little characterization data will have the greatest uncertainty in results.

CONCLUSION

While the PE cannot be used as a substitute for performance assessments, it demonstrates a method to provide DOE decision makers with a simple, conservative, defensible, and easily understandable analysis that provides results similar to those of more complex analyses. Simple representations of complex phenomena may be the only way to gain public acceptance of MLLW disposal facilities that evoke deep

concerns in a cautious, non-technical public.

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GROUNDWATER FLOW MODELING FOR THE CENTRAL INTERSTATE COMPACT, BUTTE, NEBRASKA, LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITY

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ABSTRACT

A regional, three-dimensional groundwater flow model was developed to support the characterization and licensing of the Central Interstate Compact's Low-Level Radioactive Waste Disposal Facility located near the town of Butte in Boyd County, Nebraska. This model was used to simulate saturated groundwater flow in the unconsolidated sediments overlying the Pierre Shale. The model was calibrated for steady-state conditions representative of long-term average hydrologic stresses, which included recharge from precipitation, evapotranspiration from off-site wetlands, and pumping from production wells. Model results were used to confirm the recharge, groundwater levels, flow velocities and directions on-site, all of which are key to performance assessment. The most likely groundwater pathways off-site were also identified.

INTRODUCTION

The Central Interstate Compact is seeking a license from the State of Nebraska to construct and operate a low-level radioactive waste disposal facility at a site located near the town of Butte in Boyd County, Nebraska. The groundwater modeling presented here was conducted to support the characterization of the site and

demonstrate compliance with the requirements of Title 194 - Nebraska Department of Environmental Control, Chapter 5 -Technical Requirements for Facilities, Section 001.01B, which specifies that the disposal site shall be capable of being characterized, modeled, analyzed, and monitored. The capability to model the site consisted of developing a conceptual model from hydrogeologic data acquired as part of site characterization. The significant hydrogeologic features of the conceptual model were then quantified and incorporated into a numerical model. The site was deemed "modelable" when consistency between the conceptual and numerical models was achieved. Parameters that cannot be directly measured at the site, including the long-term average recharge to the saturated zone, were also determined by model calibration. This paper describes the results of this site characterization effort.

SITE DESCRIPTION AND CONCEPTUAL MODEL

The Butte site lies on a 320-acre tract of land located between Ponca Creek and the Niobrara River in central Boyd County, Nebraska. The site itself occupies 110-acres of the southwest portion of this property. Geologic boreholes, 117 in number, were used to characterize the stratigraphy and geology of the 320-acre area, which range in age from late Cretaceous to Quaternary. Overlying the late Cretaceous Pierre Shale and the Niobrara Formation is a thin (10 to 40 ft) sequence of unconsolidated sediments. These late Tertiary- to Quaternary-age deposits are divided into four stratigraphic units: surficial fines, upper sand, fine-grained sediments, and contact zone. This sequence appears to exist over most of the modeled region, except in the area adjacent to Ponca Creek where terrace sand and gravel deposits are present. Observation wells were completed in 51 boreholes at 39 locations. Hydraulic tests were conducted in open boreholes and screened observations wells to estimate the hydraulic conductivities of the various stratigraphic units. Groundwater levels have been measured on a weekly to monthly basis from late 1989 through 1994.

These site investigations have determined that the shallow groundwater flow system present in the unconsolidated deposits at the site is the principal system for subsurface migration of radionuclides. Because of the very low hydraulic conductivity of the underlying Pierre Shale, groundwater flow is restricted primarily to the shallow system. Recharge rates and the locations of recharge/discharge zones are influenced by the high potential evapotranspiration (40 to 46 in./yr) relative to the average annual precipitation (23.67 in./yr). Because the potential evapotranspiration exceeds precipitation, only a small fraction of precipitation or snowmelt that infiltrates the soil leaves the root zone and reaches the water table as recharge. In areas of topographic lows where the water table is shallow, such as off-site wetlands, evapotranspiration exceeds infiltration due to the proximity of the water table to the ground surface. Such off-site areas discharge groundwater by means of evapotranspiration.

On a regional scale, the shallow groundwater system is bounded to the south by the regional groundwater divide, to the north by Ponca Creek, to the east by Dizzy Creek, and to the west by an unnamed Ponca Creek tributary (Fig. 1). Given these boundary conditions, the regional groundwater flow is generally north-northeast from the regional divide towards the Pierre Shale outcrops exposed along the bluffs of Ponca Creek and its tributaries and the lower reaches of Dizzy Creek. Groundwater discharge occurs along these outcrops in the form of springs and seeps. Off-site wetlands, where the water table is shallow and evapotranspiration exceeds infiltration, function as groundwater sinks from which groundwater is lost to the atmosphere by evapotranspiration. These wetlands locally reverse the general north-northeast groundwater gradient. Locally, piezometric surfaces developed from observation well measurements show that groundwater flow is relatively uniform across the 110-acre site in the north-northeast direction. Once off-site, groundwater flow continues toward the wetland group located to the northeast of the 110-acre site, and to Wetland 01 in particular, which is located in the northeast portion of the 320-acre area.

MODEL DESIGN AND CALIBRATION

Modeling Strategy

A regional modeling approach was adopted that allows the areal and vertical extent of the model to be defined by natural hydrogeologic boundaries. Because the boundary conditions largely determine the flow pattern in steady-state simulations, boundary conditions were specified at defined physical and hydraulic features (e.g., outcrops, streams, regional groundwater divides). The presence of distinct saturated hydrostratigraphic units (upper sand, fine-grained sediments, contact zone) required

a three-dimensional representation. Because nuclide transport to potential receptor locations requires hundreds to thousands of years, average groundwater flow velocities and directions are of primary interest. A steady-state model was therefore used to simulate the regional groundwater flow under long-term average conditions. The computer code MODFLOW (1) was chosen to construct the numerical model. Because MODFLOW enjoys wide use, the code has had the benefit of extensive testing and verification, and the code is recommended for groundwater flow analysis by the NRC.

Finite-Difference Grid

The block-centered, finite-difference grid developed for simulation of the regional groundwater flow is shown in Fig. 1. Areally, the grid extends from the regional groundwater divide to Ponca Creek in the south-north direction and from an unnamed Ponca Creek tributary to Dizzy Creek in the west-east direction. The long axis of the grid is rotated 30 degrees from the east-west direction to minimize the total number of nodes in the rectangular, finite-difference grid and to allow alignment of the grid with the northwest-southeast trending topographic features that define off-site wetlands. The horizontal finite-difference grid resolution was chosen to be commensurate with data resolution. Within the characterized 320-acre area, a grid size of 250 by 250 ft was adopted, which allowed reasonably accurate representation of the characterized spatial variation in hydrostratigraphic unit thickness and elevation. Outside the characterized area, a coarser grid ranging in size from 250 by 500 ft to 500 by 500 ft was selected. Vertically, the finite-difference grid was resolved into four layers corresponding to represent the three hydrostratigraphic units. Layers 1 and 2 were used to represent the upper sand unit and fine-grained sediment unit, respectively. Layers 3 and 4 were used to represent the contact zone unit. Two model layers (versus a single layer) were used to represent the contact zone to allow for the possibility of groundwater flow in the contact zone under water table conditions.

Boundary Conditions

Boundary conditions were established at natural hydrogeologic boundaries defining the extent of the regional model. The Pierre Shale outcrops north and northeast of the Butte site and functions as a physical boundary along which flow from the shallow groundwater system may discharge in the form of springs or seeps. The Drain Package in MODFLOW was used to simulate this condition. If the piezometric head is above the outcrop elevation at a given location, discharge occurs in proportion to the difference between the piezometric head and outcrop elevation; otherwise, there is no discharge across this boundary. Drains were represented in the finite-difference cells falling on the Pierre Shale outcrop. Drain conditions were also imposed along the ephemeral streams bounding the eastern and western model extremities. This condition allows groundwater to discharge if the piezometric head elevation of the underlying groundwater system is greater than the streambed elevation, the discharge rate being proportional to the difference between the piezometric head and streambed elevations. No discharge occurs if piezometric head remain below the streambed elevation. The regional groundwater divide lying to the south of the Butte site forms a hydraulic boundary across which the flow is zero. This boundary was taken to be coincidental with the surface topographic divide, which itself is coincidental with the top-of-bedrock divide. Vertically, the top of the Pierre Shale forms a no-flow boundary, given its very low hydraulic conductivity. A no-flow condition was therefore specified over the base of the contact zone.

Hydrologic Stresses

Hydrologic stresses imposed in the model include groundwater recharge from precipitation, groundwater discharge by evapotranspiration, and groundwater discharge from pumping wells. The basis for assigning recharge and discharge areas and rates, and pumping rates follows.

Recharge was imposed over the entire model domain, which itself was divided into zones of normal and high recharge on the basis of surficial geology. The zone of normal recharge extended from the regional groundwater divide to the southern extent of the terrace sand and gravels lying south of Ponca Creek, while the zone of high recharge was defined by the extent of the terrace deposits. This division was made to account for the higher infiltration rates and higher saturated groundwater flow rates likely to occur in these deposits, which are much coarser than the sediments on the characterized 320-acre area. Recharge was estimated by calibration, as there

is no universally applicable method for estimation other than calibration (2). Due to the shallow depth of groundwater and high potential evapotranspiration, groundwater discharge by evapotranspiration in low lying areas is a characteristic of this region. Areas of groundwater discharge by evapotranspiration were therefore defined in the topographically low areas of wetlands. The locations and areal extents of wetlands were determined from National Wetland Inventory maps (Butte NW, Nebr.-S. Dak; Butte, Nebr.-S.Dak.). Small wetlands (i.e., smaller in area than a single finite-difference cell) were not considered as discharge areas because their contribution to the overall water balance is negligible. Evapotranspiration from wetlands was modeled using MODFLOW's Evapotranspiration Package. This routine assumes that evapotranspiration is a function of depth to the water table. When the water table is at the ground surface, the evapotranspiration rate is taken to be its maximum value, the potential evapotranspiration rate for the region (45.6 in./yr). Conversely, when the water table is at or below the extinction depth (an estimated 5 ft), evapotranspiration is zero. Between these limits, the evapotranspiration rate decreases linearly with increasing depth.

Data on the 20 pumping wells in the modeled area were obtained from a well canvass. Since only one well was metered, pumping rates for the remaining wells were estimated from the number of people, number and type of livestock, and lawn and garden irrigation demands placed on each well. These data were furnished by each well owner or user. Pumping rates were then determined using unit consumption rates appropriate for the region. To assess the accuracy of this procedure, the estimated pumping rate (0.76 gal/min) and measured pumping rate (0.49 gal/min) for Well 4, located just east of the characterized 320-acre area, may be compared. All pumping wells were assumed to extract their water from the contact zone.

Hydrostratigraphic Representation

Construction of a regional groundwater flow model required that the hydrostratigraphy defined for the characterized 320-acre area be extended to the natural hydrogeologic boundaries defined above. The hydrostratigraphy for the model domain was developed by extrapolating information developed for the characterized area, supplemented by additional geologic and hydrogeologic data available for the region. Supplemental data sources included driller's logs acquired during the well canvass, stratigraphic test hole logs and geologic maps given by Souders (3), and test pit logs and geophysical profiles for the adjacent borrow source area located on the quarter section due west of the site. These data were used to develop a geologic map depicting the top of Pierre Shale and a number of geologic profiles showing the elevations and thicknesses of the surficial fines, upper sand, fine-grained sediment, and contact zone in the model domain.

Details of the geologic interpretation are not within the scope of this paper and are reported elsewhere (4). The thicknesses of the layers shown in the profiles generally reflect the thickness and range of variability observed at the 320-acre area, modified as required by the data. The surficial fines unit, which is part of the vadose zone, was assumed to persist across the entire area as a layer about 5 ft thick on the average. The upper sand is known to account for most of the local small-scale topographic mounding on the 320-acre characterized area and was allowed to do so on the profiles; characterization data indicates this unit may be absent beneath wetlands. The fine-grained sediments unit is indicated by the data to be absent or strongly reduced in thickness across the Pierre Shale high south of the site, becoming much thicker farther south toward the Niobrara River. The contact zone was considered to range generally between 3 and 5 ft in thickness across the study region. Only the upper sand, fine-grained sediment, and contact zone were represented in the model; the surficial fines unit was excluded because this unit is not saturated by average groundwater levels.

Hydraulic Conductivity

The horizontal hydraulic conductivities of the upper sand unit and fine-grained sediment unit were assigned the geometric mean values determined by field testing on the characterized 320-acre area, i.e., 1.4x10-3 cm/sec for the upper sand and 1.5x10-5 cm/sec for the fine-grained sediment. The horizontal hydraulic conductivity for the contact zone was determined by calibration (see below) but constrained to fall within the 3.5x10-5 to 6.8x10-2 cm/sec range determined by field testing on the 320-acre area. An additional constraint required that the calibrated contact zone hydraulic conductivity be within an order of magnitude of the observed value in a finite-difference cell for which an observation exists. The vertical hydraulic

conductivity of each hydrostratigraphic unit was taken to be 1/10 the horizontal value, which is within the range commonly observed for regional model applications (2).

Model Calibration

Model calibration was achieved by adjusting the recharge and contact zone hydraulic conductivity until the modeled piezometric heads and water budget (spring flows) matched the observed record. The observed piezometric heads used for calibration were chosen to reflect long-term average groundwater levels. Within the characterized property, the observed heads were determined by time-averaging hydrographs for the contact zone wells for the 3-year period extending from 1990 through 1992. This period was selected as representative of long-term average conditions because the mean annual precipitation for these three years (26.67 in.) nearly equals the mean annual precipitation (23.67 in.). Outside the characterized property, observed heads were estimated from the static depth to groundwater obtained through the well canvass and well location and ground surface elevation estimated from USGS topographic maps (10-ft contour interval). The observed heads estimated in this manner must be regarded as approximate (5 ft) because this procedure provides a one-time versus time-averaged value and because of uncertainty in assigning elevations to the groundwater. These data were therefore given less weight in the model calibration. The spring flows referenced by US Ecology (4), which were measured in a year of below average annual precipitation, were used for calibration. Again, these data represent one-time measurements that likely underestimate the spring flows for the time-averaged period and were treated accordingly during calibration. The observed piezometric heads used for model calibration are shown in Fig. 2.

Model calibration was expedited by first constructing an equivalent, two-dimensional, areal model. Calibration parameters were then adjusted in this areal model to (1) minimize the residual errors (observed head minus modeled head) so that the mean of the residual errors was approximately zero, and (2) match approximately the total spring flow to that observed. Recharge was obtained first by matching the observed heads and head gradients across the characterized area and the observed total spring flows, using the geometric mean transmissivity of the contact zone. The model was further calibrated to better match the observed head distribution by spatially distributing the total transmissivity into zones, and then adjusting the transmissivity of each zone to more accurately reproduce the observed heads. A three-dimensional model was then constructed by distributing the calibrated total transmissivity for a given zone into transmissivities for each of the three hydrostratigraphic units. Given that the hydraulic conductivity and saturated thickness of the upper sand and fine-grained sediment are known, adjustments in transmissivity/hydraulic conductivity were made in the contact zone only. Predictions from the three-dimensional model were essentially the same as those from the calibrated two-dimensional model. Further calibration using the full three-dimensional model was unnecessary.

MODEL RESULTS

The modeled regional head distribution for the contact zone is shown in Fig. 2. Results for the contact zone showed this unit to be saturated over most of the modeled region, although the unit is likely unconfined and only partially saturated in the area between the site and the regional groundwater divide and in other areas where the top of Pierre Shale is high in elevation. Regional flow patterns inferred from the head distributions in the contact zone show groundwater generally flowing from the regional groundwater divide to the north-northeast towards the Pierre Shale outcrop and east towards Dizzy Creek. The groundwater sink created by the group of wetlands near the northeast corner of the characterized 320-acres modifies the general flow pattern, locally reversing the predominant regional gradients to the north-northeast. The head distributions in the overlying model layers were similar when saturated, with small head differences across hydrostratigraphic units. This result is consistent with well pair observations on the characterized 320-acres. The head distribution for the upper sand indicated that the upper sand is, for the most part, unsaturated under long-term average conditions. Regional groundwater flow is therefore restricted primarily to the contact zone under these conditions.

Locally, the head distribution for the contact zone are shown in Fig. 3. These results show that groundwater flow across the 110-acre site is nearly uniform in the north-northeast direction. After groundwater leaves the site, flow patterns are

controlled by the group of wetlands (around the northeast corner of the 320-acre area) from which groundwater is lost by evapotranspiration. Most of the groundwater leaving the site flows through the contact zone to Wetland 01, while the remaining fraction is captured by a pumping well just east of the characterized 320-acres (Well 4). Results of a particle tracking analysis using MODPATH (5) are illustrated in Fig. 3, assuming releases from the areal extremities of the Class A and Class B/C disposal units. The plotted pathlines show that most radionuclides originating from the cells would collect in Wetland 01. Nuclides originating from the southeast corner of the Class A unit would be captured in Well 4.

A water budget for the regional model was completed to assess the magnitudes of the inputs to and outputs from the system. On a long-term average basis, about 22 gal/min are supplied to the modeled region by recharge as determined by calibration, i.e., 0.053 in./yr normal recharge over the site and most of the modeled region, and 0.5 in./yr high recharge over the terrace sand and gravel deposits. Pumping wells discharge about 4 gal/min, drains (springs) along Dizzy Creek, the Ponca Creek tributary and the Pierre Shale outcrop discharge about 14 gal/min, and off-site wetlands discharge about 4 gal/min by evapotranspiration. The modeled spring discharge compares favorably with the observed discharges from springs 10 and 71 that total 11.2 gal/min, i.e., the modeled value is of the same magnitude but in excess of the observed value.

DISCUSSION

Based on the consistency between site characterization data, conceptual model, and numerical model, the Butte site was found to be "modelable" and in compliance with this particular site suitability requirement. Additional information obtained from the regional model included an estimate of the long-term average recharge, a quantity otherwise difficult to estimate for a shallow groundwater system in a subhumid climate and a key parameter for performance assessment. The regional model confirmed the location of the compliance point on the northern site boundary for the performance assessment of the groundwater pathway. The analysis further showed that the off-site pathways terminate in a wetland just northeast of the 110-acre site and in an adjacent pumping well.

Results from the regional model were used to support the development of a local performance assessment model that describes the leaching of radionuclides from the waste cells and their transport to a compliance point on the site boundary. These processes were represented in a two-dimensional vertical slice model located along a groundwater pathline identified by the regional model. The performance assessment modeling was conducted for a hypothetical release of radionuclides under a very conservative set of assumptions, e.g., high recharge and high water table conditions persisting indefinitely. The performance assessment model demonstrated that the dose resulting from a radionuclide release would be below the criteria established by the State of Nebraska.

A sensitivity analysis of the regional groundwater flow model described in this paper is currently underway. US Ecology and its principal subcontractor, Bechtel National, Inc., continue to work with the State of Nebraska to achieve a timely review and licensing decision on the Butte site.

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15-5

HANFORD'S RADIOACTIVE MIXED WASTE DISPOSAL FACILITY

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ABSTRACT

The Radioactive Mixed Waste Disposal Facility, is located in the Hanford Site Low-Level Burial Grounds and is designated as Trench 31 in the 218-W-5 Burial Ground. Trench 31 is a Resource Conservation and Recovery Act compliant landfill and will receive wastes generated from both remediation and waste management activities.

On December 30, 1994, Westinghouse Hanford Company declared readiness to operate Trench 31, which is the Hanford Site's (and the Department of Energy complex's) first facility for disposal of low-level radioactive mixed wastes. As part of their startup approval process, the Department of Energy is currently conducting a line management review of the facility and it's operating basis.

INTRODUCTION

The Department of Energy and it's contractors have taken an important step forward in their program for management of mixed wastes at the Hanford Site. The completion of construction of Trench 31 in the low-level waste burial grounds provides the Site with it's first disposal facility for mixed waste.

An aerial view of Trench 31 is shown in Fig. 1.

This facility not only provides a disposal pathway for qualifying waste currently in storage at the Site, but also provides a disposal pathway for residues from planned mixed waste treatment facilities.

The facility also provides near-term disposal capacity for Site remediation activities. Long-term disposal capacity for remediation wastes may be provided by a larger facility (currently in regulatory review and development). Trench 31 operations will provide valuable operating experience and data (bulk remediation waste handling, leachate generation and disposal) that will aid development and planning for this larger facility.

FACILITY DESCRIPTION

Trench 31 is Resource Conservation and Recovery Act (RCRA) compliant, with a double composite liner and leachate collection and removal systems. The landfill is rectangular, with approximate base dimensions of 76 meters by 30 meters. The side slope ratio for the landfill excavation is 3 horizontal to 1 vertical. The bottom of the landfill excavation slopes slightly to facilitate leachate collection, giving a variable depth of approximately 7.6 meters to 9.1 meters. There is a recessed section at the eastern end of the landfill excavation that houses the sumps for leachate collection and removal. Access into the landfill is provided by a vehicle ramp (8% slope) along the southern perimeter.

From top to bottom, the liner system incorporates the following layers (see Fig. 2):

Operations layer - The bottom and sides of the landfill are covered with a 0.9 meter deep layer of soil to protect the liner system during waste placement operations.

Primary Leachate Collection System - This layer consists of geotextile, drainage gravel, geonet/geotextile geocomposite, primary high density polyethylene (HDPE) geomembrane, and a 0.5 meter soil and bentonite clay (eight to ten percent) admix layer.

Secondary Leachate Collection System - This layer consists of geotextile, drainage gravel, geonet/geotextile geocomposite, secondary HDPE geomembrane, and a 0.9 meter soil and bentonite clay admix layer.

The primary leachate collection system is composed of drainage gravel and perforated drainage pipes that lie along the centerline of the trench bottom, at the base of the side slopes, and down the "upslope" side if the access ramp. A secondary leachate collection system is installed above the secondary liner system. The leachate collection systems are designed to direct leachate to the sump area located at the east end of the landfill. Pumps are located in the sump area, and provide for removal and storage of leachate in a 37,850 liter storage tank sited at the eastern crest of the landfill.

The landfill was designed with consideration for the 24 hour, 25 year peak precipitation event (>4 centimeters precipitation). This 24 hour event is conservatively estimated to result in a maximum of 400,000 liters of liquid collected. The layout of the facility as described is shown in Fig. 3.

Construction of a second landfill lobe has been completed and is currently going through the operations acceptance process. The design of this lobe is nearly identical to Trench 31. It is currently planned that this second lobe will not be used until Trench 31 is filled and capped with an interim cover, in order to minimize volumes of leachate requiring management.

WASTE SOURCES

The predominance of the wastes to be disposed of in the facility will be remediation wastes. Remediation alternatives for waste sites along the Columbia River are being developed through the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) process and/or Resource Conservation and Recovery Act (RCRA) past-practice corrective measures actions. When developed and selected, these alternatives may involve relocation of wastes to a location more central to the Hanford Site (further separated from the river and groundwater). Trench 31 is intended to provide disposal capacity to support such a relocation alternative for remediation waste. Trench 31 will also provide sufficient disposal capacity for remediation waste to "span the gap" until a larger centralized disposal facility, the Environmental Restoration Disposal Facility, comes on line as proposed in September, 1996.

It is anticipated, based on historical knowledge and characterization data from the remediation sites, that the remediation wastes to be disposed of in the landfill will be predominantly soils or debris with very low concentrations of radionuclide and hazardous constituents. The facility operating basis has been developed to support packaged or unpackaged (bulk) waste receipts.

A small relative volume of the landfill capacity (less than 2000 cubic meters) will be used for disposal of wastes generated through waste management activities. Some of the containerized waste currently stored within the Hanford Central Waste Complex may qualify for disposal in the landfill without treatment. Evaluations are underway to determine what portion of the stored wastes may be disposed of directly without further treatment. Trench 31 will provide disposal capacity for some of these stored wastes, as well as for other newly generated wastes, which qualify for disposal.

WASTE ACCEPTANCE

Primary waste acceptance criteria have been developed for mixed low-level radioactive disposal in this landfill. These waste acceptance criteria for the landfill are included in the "Hanford Site Solid Waste Acceptance Criteria" document. Procedural requirements for waste sampling and analysis, waste receipt, and certification prior to acceptance into the landfill will be based on these waste acceptance criteria.

Waste acceptance criteria currently include prohibitions on acceptance of certain wastes. Prohibited wastes include, but are not necessarily limited to, such items as free liquids, reactive wastes, greater than Class C wastes as defined in 10 CFR 61.55, RCRA land disposal restricted wastes, transuranic wastes (there are quantifiable limits applied to many of these).

The waste acceptance criteria also address the compatibility of the accepted wastes with the landfill liner materials, with the RCRA permits for all affected facilities (including those that will handle the leachate), and with other applicable regulatory documentation (e.g. the delisting petition for the facility that will eventually treat and dispose of the leachate). Testing for liner compatibility (9090 testing) has been completed for a limited number of constituents, additional testing is underway and will likely expand the range of compounds found to be compatible with the liner system.

These waste acceptance criteria will likely continue to evolve as the regulatory processes are completed for the remediation sites (ie. Record of Decision outcomes may influence criteria), as additional waste characteristic information is obtained as remediation progresses and field observation information is obtained, and as ongoing tests to supplement liner compatibility data are completed.

LEACHATE GENERATION AND DISPOSAL

Leachate handling and disposal is an important consideration once landfill operation begins. The facility will be operated to insure that leachate generated is consistent with treatment/disposal facility requirements and that leachate generation is avoided or minimized to the maximum extent possible.

The characteristics of solid wastes accepted in the landfill for disposal will be have to be carefully managed to remain consistent with permits, regulatory requirements and process parameters for facilities which will be involved with

handling, storage, treatment and disposal of the leachate. Of particular concern, waste codes assigned to the solid wastes accepted for disposal will have to be managed in order to comply with current permitting documentation for leachate handling facilities. If multiple listed waste codes are accepted into the facility, the F039 waste code ("multisource leachate") will apply to the leachate per RCRA requirements. Modifications to the applicable permits and other regulatory documentation (i.e. delisting petitions) are being pursued to support as much flexibility in waste receipt as possible. The list of acceptable waste codes will likely continue to evolve as needed and as regulatory requirements allow.

The scheduling of waste receipts at the facility is important in limiting the volumes of leachate requiring management. It is desirable to minimize the length of time between start of operations and closure of the facility (placement of interim cap or final closure) in order to minimize the volume of leachate generated. Requests for forecast waste volumes and schedule information have been sent to the generators. These will be used to develop operating scenarios for the facility which will minimize leachate volumes.

Modeling of leachate production was conducted using the Hydrologic Evaluation of Landfill Performance (HELP) model. Using conservative assumptions (such as the facility is open for 10 years, minimal operating cover layers, etc.), the average annual leachate production of leachate was estimated to be about 590,000 liters. Experience at nearby commercial hazardous waste disposal facilities and at Hanford indicate that this is a conservative estimate and that actual volumes may be somewhat lower.

It is planned that leachate collected from the operating landfill will initially be transferred to the Hanford double-shell tank system for storage. Treatment and disposal of the leachate once received in tank farms will be as for any other dilute tank farm waste stream. Disposal will be facilitated through the use of existing tank farm and liquid effluent facilities, such as the 242-A Evaporator, the Liquid Effluent Retention Facility (LERF) and the 200 Area Effluent Treatment Facility (ETF).

The transfer of the leachate to the tank farms supports operation of the landfill in the near-term, but may not represent the best long-term alternative. Tank farm space is a limited and valuable commodity, and is best reserved for higher priority, less dilute, liquid waste streams. In the interest of limiting the volumes of waste sent to the tank farms, direct transfer of the leachate to the 200 Area Effluent Treatment Facility (ETF) is being pursued.

CONCLUSIONS

The Department of Energy has been in a mode of interim storage of mixed wastes at Hanford, this facility represents the first step towards a final solution (disposal). Qualified wastes currently in storage at Hanford can now be disposed of, and development and implementation of waste treatment processes can continue to proceed with the knowledge that a facility exists which can receive mixed waste residues. Trench 31 will also allow disposal of mixed wastes resulting from remediation activities, and thus supports the Hanford Restoration mission. This facility can also serve as an important pilot test of disposal operations to be used to support full scale remediation activities.

15-6

APPLICATION OF BINARY MIXING MODELS TO THE INTERPRETATION OF NITRATE AND RADIONUCLIDE FLUCTUATIONS IN THE VADOSE AND SATURATED ZONES, SOLAR EVAPORATION PONDS, ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE, GOLDEN, COLORADO

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ABSTRACT

The Rocky Flats Environmental Technology Site (RFETS) has utilized surface impoundments (Solar Evaporation Ponds or SEPs) for the storage of high nitrate, liquid plutonium- and uranium-process wastes since 1953. Emptying of the ponds is now nearly complete and decommissioning is underway. The Interim Measures/Interim Remedial Action (IM/IRA) proposes to place all SEP-contaminated wastes beneath an engineered cover constructed of natural materials and to demonstrate long-term

protection of the environment. A critical aspect of the IM/IRA design is an assessment of the environmental threat posed by sub-SEP vadose zone soils if left in place. Excavation of the soils significantly increases the volume of waste required to be covered and thus the extent of the engineered cover. Simple binary mixing models are derived and used to interpret existing groundwater data collected since 1989, and to evaluate the extent of on-going contaminant leaching from the vadose zone which serves as the basis of a predictive model for future performance. Data analysis indicates that nitrate and radionuclides are mobilized from both vadose zone pore waters related to old (ca. 1960) liquid pond wastes, and from vadose zone solids during seasonal high groundwater stands. The mixing model is also used to implicate sorption control of uranium and colloid-facilitated transport of americium and plutonium. The study supports the decision to excavate vadose zone soils down to the mean seasonal high groundwater table and install a drainage layer (to limit groundwater rise) as a base for soils and other SEP wastes scheduled for inclusion beneath the engineered cover.

INTRODUCTION

Liquid process wastes from plutonium and uranium operations at the Department of Energy's, Rocky Flats Plant near Golden, Colorado (now known as the Rocky Flats Environmental Technology Site or RFETS) were discharged into various surface impoundments beginning in 1953. Collectively these ponds are known as the Solar Evaporation Ponds (SEPs). Details of the site are contained in numerous documents including DOE (1) and illustrated in Fig. 1. The nature of the plutonium refining process involved large quantities of nitric acid for dissolution of the fissionable metal. Prior to the incorporation of adequate refluxing equipment, the highly acidic spent nitric acid wastes with traces of unrecoverable plutonium and uranium were placed in the SEPs along with other industrial wastes. Documented SEP leakage is known to have been responsible for unquantified releases of nitrate to groundwaters and surface waters. Direct discharges of process wastes to the SEPs largely ceased by the mid 1980s but various contaminated waters have remained in the impoundments until recently.

As of this writing, all SEPs have now been deactivated and all but one are completely emptied of liquids and sludges. A proposed action under the negotiated Interim Measures/Integrated Remedial Action (IM/IRA) is to construct a "dirty" closure whereby all contaminated materials from SEP activities (sludges, liquids, liners etc.) would be enclosed on-site beneath an engineered cover designed to be protective of the environment and human health for 1000 years. Additional contaminated materials considered for inclusion beneath the engineered cover are those vadose zone soils previously exposed to SEP leakage. Inclusion of these soils with the other wastes beneath the engineered cover represents a significant increase in volume. As part of the IM/IRA design, the
extent of ongoing groundwater contamination due to leaching of contaminants from vadose zone soils was investigated as a predictor of future behavior and as a basis for justifying their disposition relative to inclusion with the other wastes beneath the engineered cover.

Absolute contaminant concentrations in groundwater proximal to the SEPs reveal long (4 to 6 years) and short (quarterly) term variability. All or part of such variability may be attributed to 1) a changing contaminant sources (e.g., cessation of leakage, decreased contaminant concentration in leakage, or exposure to contaminated soils due to water table rises) and/or 2) changing dilution factors with uncontaminated groundwater. In order to sort out the relative contributions of these factors to the observed contaminant concentration variability, and to determine if vadose zone soils are actively supplying contaminants to the groundwater (and thus might be a liability in the future if allowed to remain in place), it was first necessary to evaluate the effect of dilution. Following the method described by Whittemore (2) and others, groundwater concentrations of two components, both assumed to be conservative (chloride and nitrate), were ratioed and compared to the predicted ratios for various mixtures of background groundwater and SEP waters. Correspondence of observed data with predicted values offers support for a simple mixing hypothesis.

Averaged over several years, groundwater in the most nitrate-contaminated OU4 monitoring wells appears to be a product of mixing of background groundwater with SEP water compositions intermediate between ca. 1963 and 1991 pond waters. Groundwater in monitoring wells as close as 30 meters to the nearest SEP have

experienced very little dilution to date, apparently due to low groundwater flow rates. More proximal wells have become more dilute with time and indicate a shrinking contaminant plume. On a more detail time scale (seasonal), however, the degree of dilution appears to be affected by groundwater table fluctuations and suggests that during normal seasonal rises, groundwater may encounter labile nitrate sources in the vadose zone, presumably in the form of trapped SEP leakage in soil pores. Uranium/chloride ratios also vary with water table elevation and suggest that a net increase in groundwater uranium mass occurs during seasonal high water stands. These observations indicate that the vadose zone soils above the mean seasonal high water table pose a risk if allowed to remain in place and if groundwater rises further.

HISTORY OF SOLAR EVAPORATION PONDS (SEPs)

A brief history of the SEPs is useful for interpreting mixing curves presented later in this paper. Unlined (briefly) and lined impoundments have occupied an eight acre site in the northeast portion of RFETS since December, 1953. Current capacity is 10.7 million gallons. Their purpose was initially to provide temporary storage for low-level radioactive liquid process wastes and ultimately long-term waste reduction by evaporation. Existing ponds are collectively known as the SEPs and the entire area, including remnants of previous ponds, is being treated as Operable Unit (OU4). The history of SEP utilization is complex based on construction, maintenance and sparse discharge records. Process knowledge also suggests complexity because discharge compositions would have changed over time as plutonium and uranium refining procedures improved, but unfortunately. very little data exists on pond water compositions, especially during the period from 1963 to 1991.

Primary process waste discharges to the 207-B Ponds ceased in 1974 and to 207-A and 207-C in 1986. Treated sanitary waste water, recovered downgradient contaminated groundwater and minor industrial wastes were added to various ponds in later years. Removal and treatment of water and sludge has been on-going over the last five to ten years and currently only 207-C has not been emptied. A comprehensive pond water sampling and analysis program conducted in 1991 revealed that the water present at that time was dramatically less contaminated relative to water in the same ponds in 1963.

Evidence of leakage of pond liquids to the underlying vadose zone is abundant, both in the form of SEP operational records and compositions of downgradient groundwaters and surface waters. Although improvements in pond construction techniques were implemented and frequent repair measures taken documented leakage events were common through 1970 and sporadic thereafter. No estimate of volumetric leakage from the SEPs is available. A large Interceptor Trench System (ITS) was installed and upgraded to collect contaminated groundwater and return it to the ponds. High nitrate (2500+ mg/L NO_3^- + NO_2^- as N) concentrations have been observed in downgradient monitoring wells since their installation in 1989.

At present, contaminated materials resulting from SEP activities in OU4 include liquids stored in tanks, consolidated (cemented) sludges, unconsolidated sludges, pond liners and vadose zone soils. A proposed action under the terms of the IM/IRA negotiated for the site, is to excavate contaminated soils, further consolidate liquids and sludges, and to place all solid materials on an engineered drainage layer positioned above the water table and to cover the entire volume with an engineered cover constructed of natural materials. The design criteria include no unacceptable impact to groundwater for 1000 years.

HYDROGEOLOGIC SETTING AND GROUNDWATER CONTAMINATION

The site is located on a low interfluve pediment between North and South Walnut Creeks (Fig. 1). Surface geologic units are composed of thin accumulations (2-5 meters) of complexly interbedded silts and silty clays of the Quaternary Rocky Flats Alluvium (RFA) and derived colluvium. The RFA unconformably overlies bedrock which consists of nearly flat-lying claystones, shales and local fine sandstones of the Cretaceous Arapahoe/Laramie formations. Operationally, the entire thickness of the RFA, colluvium, the weathered top of bedrock and bedrock sandstones present at the unconformity are considered to constitute the Upper Hydrostratigraphic Unit (Upper HSU). Laboratory-determined hydraulic conductivities in the RFA exhibit a large range (7×10^{-2} to 1×10^{-9} cm s^{-1}) but are generally taken to be low (1×10^{-7} cm s^{-1}), as are overall hydraulic gradients (0.03), indicating very low Darcy velocities (1×10^{-3} m/yr). Effective hydraulic conductivity, based on well recovery data, appear to be significantly higher (1.4×10^{-4} cm s^{-1}) perhaps due to macropores, and

gradients in the vicinity of plumes (see Fig. 1) are higher (0.07) and may have been higher still in the past due to pond leakage. Conservative Darcy velocities therefore may be as high as 3 m yr-1. Overall groundwater flow directions are indicated on Fig. 1 but in detail are highly complex due to natural lithologic variability, earth moving disturbances, and extensive buried pipelines, drains and foundations.

The mean seasonal high groundwater table is generally within a few meters of the base of the SEP structures and locally may contact the liners. In general, leakage from SEPs is thought to have locally saturated the vadose zone, mixed with the groundwater and migrated to the north and northeast toward North Walnut Creek. The ITS was installed in such a way as to intercept and capture this flow path although current data suggest some component of the flow may pass beneath the ITS trenches locally.

Figure 1 also illustrates the location of monitoring wells proximal to the SEPs that have been completed in the Upper HSU and are sufficiently saturated for normal quarterly groundwater sampling. Coverage is sporadic for a variety of reasons but is sufficient to reveal downgradient groundwater contamination. Distributions of average 1992 groundwater nitrate concentrations (values represent NO3- + NO2- as N averaged over four quarters) are also shown on Fig. 1. A contaminated plume defined by monitoring wells H, I and J issues from the northern ends of 207-A and the 207-B series SEPs and is shown truncated by the ITS. A stylized plume is also shown downgradient of 207-C but data to support its existence is circumstantial. Minor nitrate contamination is present to the southeast of 207-B South and appears to be responding to a minor southeasterly component of groundwater flow.

Identifying uncontaminated, upgradient (background) monitoring wells is problematic at OU4 because the bulk of the RFETS industrial complex lies upgradient. A detailed examination of all available data (not presented here) using trilinear diagram analyses indicates that only groundwater in monitoring well A is similar to site-wide background groundwater and does not appear to have been affected by SEP leakage or other industrial contamination. This well is taken to represent background for all subsequent analyses but in reality, groundwater upgradient of OU4 and therefore groundwater which received SEP leakage may already have been contaminated to some extent by other industrial activities.

DERIVATION OF CONSERVATIVE COMPONENT MIXING EQUATIONS AND PLOTS

Simple 2-component conservative mixing equations have been presented by Whittemore (2) and others and applied to the identification of brine sources in contaminated surface waters and groundwaters. In these cases, dissolved chloride and bromide ions are assumed to be unaffected by heterogenous (aqueous-solid) processes and are thus considered "conservative". If multiple potential sources of brines exist (e.g., separated oil field brines and road salt runoff) and each possesses a different [Br/Cl] ratio, then it can be shown that each source will produce a unique mixing curve relative to a common diluent in [Br/Cl] vs. Cl space. By comparing an observed contaminated water composition with the mixing curves, it is possible to identify the source of the brine. For the purposes of this study Cl- and (NO3- + NO2-) reported as N, hereafter denoted simply as NO3-(N) were selected as the two conservative components as no Br- data exists and nitrate is of obvious interest. The extreme solubility of nitrate salts (>100,000 mg/L NO3-(N)) and abundance of dissolved nitrate (up to 2800 mg/L NO3-(N)) indicates that nitrate may be safely considered conservative. A mixing curve for two solutions (A and B), each containing two conservative components (1 and 2) is derived as follows:

Solution A (e.g., a concentrated brine)
C1A = concentration of 1 in solution A (e.g., Br- or NO3-(N))
C2A = concentration of 2 in solution A (e.g., Cl-)
VA = volume of solution A in mixture
Solution B (e.g., a background water)
C1B = concentration of 1 in solution B (e.g., Br- or NO3-(N))
C2B = concentration of 2 in solution B (e.g., Cl-)
VB = volume of solution B in mixture
Mixture (e.g., contaminated groundwater)
C1 Mix = concentration of 1 in mixture
C2 Mix = concentration of 2 in mixture
VA + VB = VMix
RV = VA / VB

See Eq. 1
Figure 2 illustrates a generic mixing curve calculated as described above. Note the relative sensitivity to mixing ratios at various points along the curve.
NITRATE IN GROUNDWATER
Figure 1 illustrates the distribution of nitrate in groundwaters only in the immediate vicinity of the SEPs. The sparse data reveal a significant nitrate plume extending northwest from SEP 207-BN to the point where it should be truncated by the ITS. This plume is consistent with the principal groundwater flow direction toward North Walnut Creek. Historically, data suggest the existence of another major plume emanating from SEP 207-C but no monitoring wells exist in that area. Nitrate groundwater contamination is also present outside the area of Fig. 1 and downgradient of the ITS at a distance of more than 300 m from the SEPs. Assuming the estimated effective Darcy velocities (3 m yr-1) in the RFA it is clearly impossible to explain the overall distribution of groundwater nitrate. It is apparent that overland flow of contaminated water, sourced in springs and seeps and possibly hidden drains, is partially responsible for the observed overall distribution. To interpret the source(s) and pathways contributing to the nitrate observed in the most contaminated monitoring wells (H, I and J on Fig. 1), a binary mixing model as described in the previous section was used.
OU4 Binary Mixing Model
Figure 3 illustrates various mixing curves developed for groundwater beneath the SEPs. All curves were constructed as described previously. Theoretical mixing curves are shown for a single upgradient groundwater (A) mixed with waters from four SEPs as they existed in 1963 and 1991. The dilute end-member groundwater represents a mean of water quality data collected from monitoring well A, the most defensible candidate for "local" background groundwater at OU4. Site-wide background groundwater contains, on average, less Cl- than well A but similar [NO3-(N)/Cl] ratios. Meteoric precipitation would also be expected to contain less Cl- than A and potentially lower [NO3-(N)/Cl]. Reddy and Caine (3) give 0.152 mg/L and 0.507 for Cl- and [NO3(N)/Cl], respectively, for precipitation at a site 32 km northwest of RFETS. Given this information, the dilute end-member of the mixing curves might be best considered to be an area on the plot rather than a point.
Six concentrated end-members are identified according to source SEP and year. Although substantial differences existed between pond compositions in 1963, as a group, the 1963 waters are clearly distinctive from the 1991 pond waters and consequently result in distinctive theoretical mixing curves. The curves describe where groundwater compositions would plot if they originated from mixing various proportions of background groundwater and a specific pond water. Given the period of time spanned by the two sets of curves and the complex evolutionary history of the ponds as described above, it is entirely possible that some undocumented intermediate pond water compositions might better represent the concentrated end-members for the majority of groundwater/pond water mixes.
All available quarterly groundwater nitrate and chloride data collected during the period 1987 to 1993 from monitoring wells H, I, and J are also shown on Fig. 3. Note that over this period groundwater from wells I and J each exhibit limited variability in [NO3(N)/Cl] - Cl space, indicating a stable groundwater chemical regime at those locations. The plotting positions of I and J data between the group of 1963 mixing curves and the group of 1991 curves suggest that some intermediate pond water composition was the source of the nitrate contamination in those wells. Additionally, the similar [NO3-(N)/Cl] ratio of I and J groundwaters suggests a common contaminant source and the lower Cl- concentrations in I well groundwaters indicates greater dilution at that location. The fact that groundwater at neither well I nor well J (located 20 and 60 m downgradient of the nearest pond margin, respectively) appear to represent dilutions of contemporaneous pond water (i.e, 1991 pond water) may be explained by the slow groundwater flow rates and their distances from the ponds. Greater dilution at the more source-proximal well I location may be evidence that the source has abated and the maximum contaminant concentrations have passed that location - analogous to a decaying breakthrough curve.
Monitoring well H is located immediately adjacent to SEP 207-BN but appears to be lateral to the main body of the groundwater nitrate plume. Groundwater from well H shares a similar [NO3-(N)/Cl] ratio with I and J, suggesting a common contaminant source, but exhibits a much wider range of Cl- concentrations and, by inference, a much wider range of dilutions of that source. A consistent decrease in Cl-

concentrations in well H groundwater (from >300 to <50 mg/L) with time is also seen, indicating a relatively rapid increase in source dilution. Such temporal behavior would be consistent with position of well H relative to the main axis of the plume where the affects of a waning source would occur most rapidly.

A mixing curve passing through the plots of groundwater compositions from wells H, I and J may be constructed as shown in Fig. 3 by calculating the compositions between a hypothetical intermediate pond water X and a hypothetical background groundwater Z representing the mean of all site-wide Upper HSU groundwaters (see previous discussion of background variability). This hypothetical mixing curve may then be used to estimate the dilution factors necessary to produce the observed groundwater compositions. (Note that the dilution factor is very sensitive to the chloride concentration selected for the brine endmember.) For example, well J groundwaters appear to be a 1:1 mixture (background groundwater:pond water), well I groundwaters about 2:1 and well H groundwaters ranging from 2:1 to 80:1.

Temporal Variations in Groundwater Nitrate

Figure 4a illustrates the nitrate and chloride concentration history of monitoring well H. With the exception of the final sampling period (which may reflect damage to the well casing which resulted in abandonment of the well), nitrate concentrations have decreased steadily and dramatically since mid-1987. By comparison, [NO3-(N)/Cl] has remained nearly constant over the same period, ranging from 3.3 to 5.2. The contrast between these two quantities supports the important conclusion that the source of nitrate contamination is remaining essentially constant in composition and only the flux of that source to the groundwater is changing (i.e., a changing dilution factor). Recall that pond water compositions are known to have changed over time and that 1991 SEP sampling reveals that the pond water present during the period shown in Fig. 4a was more dilute. In the absence of [NO3-(N)/Cl], one might conclude that pond leakage rates have remained constant and improving groundwater quality is due to improved pond water quality. Rather, it would appear that the "old" pond water source is diminishing but continues to affect present-day groundwater quality.

Figure 4a also shows that the groundwater elevation in monitoring well H. High [NO3-(N)/Cl] ratios appear to occur during periods of high groundwater stands. Although simple correlation analysis does not support this apparent relationship, close examination reveals that quarterly samples for water quality measurements were not collected consistently with respect to peak groundwater elevations. Other monitoring wells in the vicinity of the SEPs exhibit similar trends but in all cases the groundwater table elevation fluctuates at a higher frequency than that at which water quality samples were collected. It is apparent that monthly sampling would be required to directly quantify this relationship. If high groundwater stands are indeed associated with elevated [NO3-(N)/Cl], such a relationship would support the existence of a concentrated "old" pond water reservoir in the form of residual vadose zone pore water. In this scenario, groundwater rising into the vadose zone would mix with vadose zone pore water, acquire some of the dissolved contaminants and transfer those contaminants to the saturated zone as groundwater recedes. A simple mixing model, based on assumptions of effectively infinite nitrate solubility, saturated vadose zone porosity and field capacity predicts the decay of nitrate concentrations with successive seasonal flushing events.

RADIONUCLIDES IN GROUNDWATER

Behavior of radionuclides in the OU4 subsurface is a point of obvious concern but data is limited spatially and temporally, in part due to the large volumetric requirements for water analyses and low-yield nature of most monitoring wells. In the case of monitoring well H, damage to the well casing resulted in its removal from service in early 1992. Figure 4b illustrates the fluctuation of U-238 and U-238/Cl in monitoring well H. (As expected, U-235 and U-233/234 and the respective chloride ratios behave similarly.) Dissolved uranium concentrations appear to be depressed in Fall 1991, roughly coincident with a period of relatively high groundwater surface elevation. This decrease in dissolved uranium is initially associated with a sharp increase in U-238/Cl, followed by a decrease.

Applying the same mixing equations described earlier (substituting dissolved uranium for NO3(N)), and the uranium concentrations in the same hypothetical background and pond water necessary to explain the behavior of nitrate, the data presented in Fig. 4b can be tested against the predictions of a simple mixing model. A mixing plot containing the measured groundwater U-238 and U-238/Cl data (analogous to Fig. 3 but

not presented here) reveals that observed groundwaters do not fall on the mixing curve and actually describe an inverse curve relative to the hypothetical mixing line. The different behavior of uranium as compared to nitrate and the failure of the simple mixing hypothesis to explain the observed uranium concentrations is expected since dissolved uranium is unlikely to behave conservatively due to sorption and/or precipitation processes. Figure 4b indicates that during the initial rise of the groundwater into the vadose zone, dissolved U-238 decreases. Chloride also decreases during this period (as can be determined by the [U-238/Cl] ratios) but to a greater extent than does U-238. An equivalent decrease in both quantities would be consistent with simple dilution and result in a constant [U-238/Cl] ratio. It is evident, however, that the [U-238/Cl] ratio increases. This observation indicates that U-238 was added to the groundwater during the high water stand, probably as a result of transferral (dissolution or desorption) from the vadose zone stationary solid phase to the aqueous phase. Subsequent depression of the U-238/Cl ratio may represent the reverse process in the saturated zone. In support of a solid/liquid partitioning mechanism, a range of Kd values (approx. 15 to 30 L kg-1) was calculated using U-238 groundwater activities (as shown in Fig. 4b) and activities for vadose zone soils collected from the H borehole (1.5 - 2.0 pCi/g). Such Kd values are close to the preferred value of 43 L kg-1 proposed by Looney et al. (3), indicating the reasonableness of sorption control of U-238 groundwater activities.

A very limited amount of Am-241 data exists for groundwater from monitoring well H. Am-241 is present in the OU4 environment as a daughter product of Pu-241 (half life = 14.4 yr), a component of weapons-grade plutonium (Shefelbine (5)). Such a short half life would allow significant in-growth of Am-241 from plutonium contamination within the time frame of SEP operations and could remain proximal to the parent plutonium in the vadose zone. The fluctuation trends of Am-241 and [Am-241/Cl] in well H closely resembles that of NO3(N) and [NO3(N)/Cl] and thus suggesting extensive americium solubility. This is significant in view of the fact that americium is generally thought to have Kd values greater than those for uranium (Looney (4) recommends 100 L kg-1) and may be solubility limited in carbonate-rich environments (Triay et al. (6)). Mobility or apparent solubility of a species with a recognized affinity for solid phases may imply an association with suspended matter and/or colloidal matter. Colloid-facilitated transport of plutonium is generally considered to be potentially important and has been tentatively demonstrated for plutonium and americium elsewhere at RFETS by Harnish et al. (7). Consequently, the behavior of americium in the OU4 vadose zone may be a function of plutonium behavior which is influenced by colloidal matter.

CONCLUSIONS

Simple binary mixing models for conservative groundwater constituents (Cl- and NO3-) have been usefully applied to the interpretation of groundwater-vadose zone interactions in the vicinity of the SEPs at RFETS. Data demonstrate that recent groundwater compositions were mixtures of background groundwater and older, more concentrated, pond water rather than dilute contemporaneous pond water. This implies the retention of older pond leakage as vadose zone pore water. Temporal correlations of [NO3-(N)/Cl] ratios with groundwater elevation indicate that trapped pore water is contacted by and mixes with groundwater during seasonal high water stands resulting in periodic pulses of nitrate to the saturated zone. Limited data also suggests that radionuclides are mobilized from the vadose zone during high water stands. Uranium isotopes respond to groundwater fluctuations but do not obey the mixing model derived from nitrate and chloride concentrations. Uranium appears to be influenced by sorption processes characterized by a Kd in the range of 16 to 26. Americium, a daughter of weapons-grade plutonium, resembles the behavior of nitrate implying high (apparent) solubility but is likely associated with colloidal plutonium and is therefore more mobile than might be expected from literature Kd values. In the case of this site, the vadose zone appears to have the potential to act as a long-term contaminant source after elimination of leakage from the surface ponds. Excavation of contaminated vadose zone soils is recommended followed by isolation from rising groundwater.

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Session 16 -- Unique Site Remediation Solution from the WERC University Design Contest
Co-chairs: Ron Bhada, NMSU;
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16-1

INTERNATIONAL ENVIRONMENTAL DESIGN CONTEST FOR UNIVERSITIES - A REALISTIC UNIQUE APPROACH TO ENVIRONMENTAL EDUCATION

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ABSTRACT

The Waste-management Education and Research Consortium (WERC) is a consortium of three universities, a community collge, and two national laboratories with the mission of generating resources to address issues associated with all aspects of environmental management.

For the past four years, WERC has implemented an environmental design contest for competition by institutes of higher education. This is the only environmental design contest of its kind in the world. The contest is structured to give university student groups an opportunity to exchange information via a national contest for design, development, and testing of an environmental control process. Each year a practical environmental problem is presented to the competing teams approximately nine months prior to the actual competition. Each team prepares a total plant design for the solution of the environmental problem. The design stresses not only the technical solutions but also includes such factors as economics, risk analysis, health, regulations, public policy and communications. Some participating teams have used the contest problem as a part of a capstone design course. The judging is performed by experts from academia, industry, and government agencies.

In April 1994, twenty-five teams from throughout North America, including one team from Mexico, competed in the contest which focused on remediation of a contaminated site followed by reclamation of the remediated site using water harvesting as one of the techniques. The 1995 contest focuses on a waste handling and removal, as well as, a waste remediation/stabilization process. The program has continued to accomplish its objectives of providing a design challenge and a medium of exchanging information in the environmental area between various participating teams, industry and government.

This paper presents the 1994 Design Contest problem statement; what is expected from the participating universities; deliverables; as well as the list of participating teams and the list of 1994 award winners.

INTRODUCTION

The Waste-management Education and Research Consortium (WERC) was created in 1990 by the U.S. Department of Energy as a partnership between New Mexico State University, the University of New Mexico, and the New Mexico Institute of Mining and Technology in collaboration with Los Alamos National Laboratory and Sandia National Laboratories; the Navajo Community College joined as an affiliate in 1991. More than 2000 students (pre-college, college, and professional) receive these educational benefits, and more than 40 technology development projects are progressing on all aspects of waste management and environmental restoration. Currently, there are undergraduate and graduate educational programs throughout the world on environmental management and engineering. However, there has been no vehicle to bring students from various universities together for discussions and solutions of major environmental issues. Consequently, for the past four years, a unique and innovative environmental design contest for universities throughout the Americas (including Canada and Mexico) has been conducted by WERC. Practical environmental problems are presented to the competing teams approximately 9 months prior to the actual competition. The design contest is co-sponsored by industrial organizations, the Department of energy and other governmental agencies. The contest is structured to give university student groups from all over America an opportunity to exchange information and participate in an international contest for design, development, and testing of a realistic environmental control process. The judging, is performed by experts from academia, government, and industry and is based on technical, as well as other criteria such as economics, risk analysis, health regulations, public policy and communication.

1994 DESIGN CONTEST

In April 1994, twenty-five teams from throughout North America, including one team from Mexico, competed in the contest which focused on remediation of a contaminated site followed by reclamation of the remediated site using water harvesting as one of the techniques.

The issue was described in realistic terms in a simulated letter from the company management to an engineer who had recently joined the company. The young engineer is asked to form a group to design and construct a bench-scale working model and demonstrate the pollution control process. The engineer is given the following four tasks:

1) A written two-part report, one for remediation and the second a reclamation process. The report was to include:

- a process design
- a detailed total plant design
- the engineering basis for the design
- the economics of the process
- a discussion of the legal and health implications\
- a plan for presentation to the community for public acceptance so that problems are minimized after construction

2) A practical bench-scale working model of the process to demonstrate functionality of both the remediation and reclamation processes.

3) A brief oral presentation of the design including economics, health, business development, regulatory, and other related issues.

4) A poster presentation containing highlights of the design, economics and other issues.

In addition to trophies, the participating teams received cash awards in several categories which exceeded $ 30,000.

1994 DESIGN CONTEST STATEMENT

Introduction: A large area (hundreds of square miles) in an arid region of the Pacific-Northwest has been contaminated with fallout from a neighboring manufacturing region. The site is to be both remediated and reclaimed. Remediation will be defined as reducing the concentration of identified contaminants below the threshold values listed below. Reclamation will include the use of a water harvesting system to enhance the growth or natural vegetation or agricultural crops on the site.

Water harvesting is an ancient concept that has been applied to increase biomass production in arid and semi-arid lands. Water harvesting concepts currently applied in arid lands continue to be somewhat primitive technically and small in scale. The desire to improve the technical state of water harvesting by finding, selecting, designing and/or testing sealants for catchment areas and by developing equipment

that can apply the sealants on very large areas (hundreds of miles) in relatively short times (a few years) in a cost effective manner. These water harvesting catchments (sealed areas) may be tied into no drainage growing strips.
The design team is given the following three tasks:
1). Develop and demonstrate a bench scale process to remove the identified contaminants from a 5kg sample of the soil. Develop a conceptual design of the process applicable to the field scale project.
2) Develop and demonstrate a non-geomembrane, UV resistant, water repellant, erosion resistant, sealant that can be applied to the soil surface as an aid in water harvesting.
3) Develop a conceptual design for a machine capable of applying the sealant developed in task 2, to the surface of the remediated area.
Regional and Soils Description: The area to be remediated is located adjacent to a large river in an arid climate. Natural vegetation is desert shrub and bunch grasses. Soil material is the result of catastrophic flooding followed by deposition of river alluvium. The resulting material is a mixture of cobbles and sandy loam soil material. The following particle size description is typical of the material at the remediation site.
Soil (see insert a):
To approximate this material with a soil from New Mexico, we have chosen the soil series Casito (Petrocalcic Ustollic Paleargrid). This soil is formed in alluvium at the base of mountain watersheds. It is found on alluvial fans and terraces. It contains a similar mixture of cobbles and fines as the remediation site; however, the source of the alluvium is storm runoff rather than catastrophic flooding and river deposition. Provided below is some additional information on the contaminants in this soil.
Contaminants:

Category	Concentration per kg of soil
1) Pesticides, herbicides, insecticides	
Lindane (C6H6Cl6)	150 mg
Methoxychlor (Cl3CCH(C6H4OCH3)2)	150 mg
Endrin (C12H8OCL6)	150 mg
2) Metals	
Cadmium (Cd)	350 mg
Silver (Ag)	100 mg
Copper (Cu)	100 mg
3) Organic Compounds	
Ethyl Benzene (C6H5C2H5)	75 mg
Methyl ISO-Butyl Ketone (CH3CO-CH2-CH(CH3)2	100 mg
4) Halogenated Compounds	
Chloroethene (CH2CHCl)	75 mg
Tetra Chloroethylene (C2Cl4)	100 mg

The contamination is a surface type contamination (<1' deep) and must be removed to the following levels:

1) Pesticide, herbicides, insecticides	
Lindane (C6H6Cl6)	10 mg
Methoxychlor Cl3CCH (C6H4OCH3)2	10 mg
Endren (C12H8OCl6)	10 mg
2) Metals	
Cadmium (Cd)	15 mg
Silver (Ag)	15 mg
Copper (Cu)	15 mg
3) Organic Compounds	
Ethyl Benzene (C6H5C2H5)	10 mg
Methyl ISO-Butyl Ketone CH3-CO-CH2-CH-(CH3)2	10 mg
4) Halogenated Compounds	
Chloroethene (CH2CHCl)	14 mg
Tetra Chloroethylene (C2Cl4)	14 mg

Detailed Discussion of Tasks and Evaluation Criteria:
Task 1: The remediation demonstration will only deal with the fraction of soil material with particle diameter <2mm. This is because of the logistical trouble in transporting samples containing large cobbles. Therefore, each team will be supplied with a 5kg soil sample containing particles <2mm diameter. Contest rules will stipulate that the presence of the cobbles will not inhibit any process developed from removing contaminants from the fine fraction of the soil. However, teams must consider the logistical problems posed by the presence of large cobbles. For example, insitu processes must account for problems of surface roughness, wear and tear on equipment, etc, that the cobbles would create. Batch processes must handle the presence of the cobbles or account for their removal.
Task 2: Utilizing current knowledge of water harvesting sealants, select and test a nontoxic sealant for catchment areas. The test should be conducted at the bench scale per ASTM D5093-90. The sealant should have the maximum possible UV resistance. (The tests are subject to verification at an independent lab by the judges.) The sealant should be the lowest possible cost per unit gallon.
The sealant developed must not be a geomembrane or other "liner" material. It must be a non-toxic, UV resistant chemical formulation that is applied to the soil surface in a mechanized fashion. Literature and/or experimental data must be presented to document the UV resistance. The sealant must have water-repulsive properties to facilitate water harvesting, and be resistant to erosion on slopes up to 30. Demonstration of sealant properties and application must include soil with cobbles. Samples of the complete soil mixture with cobbles will be provided at the contest location. Practice by the design team must be done on samples created locally from the soil information provided here.
Sealant testing must conform to ASTM D5093-90 standards. Demonstration of sealant properties at the contest site will include water shedding and erosion resistant properties. Each design team will apply their sealant to a sample 2' wide x 3' long x 1' deep posed at 30. Each sample will have a standard amount of water applied at a standard intensity and total runoff of soil, sealant, and water will be measured, and used in the evaluation of the design. Amounts of water applied and application intensities will be provided at a later date. Water infiltrating into the soil and seeping from the bottom of the sample container will be measured separately and count against the design.
Task 3: Prepare a conceptual design of machine to apply the sealant on catchment areas. The catchment areas can be assumed to be nonvegetated, compacted beds of sand, gravel and cobble, or mixtures of the same that have been pregraded to remove gullies and rills.
The machine developed to apply the sealant chosen must be able to work on slopes up to 30. Evaluation criteria include: 1) cost, 2) speed of application (e.g. acres/day) and 3) reliability.
The design problem does not involve design of the water harvesting system. The team is not responsible for estimating the topography of the harvesting area, size of the area to receive sealant, etc. The only job for the team is to design a machine capable of applying the sealant over large areas of up to 30% slope.
The conceptual design should include sketches or mockups, and conceptual drawings. The conceptual design should also include an estimate of the capital cost to design, develop and test a full-scale machine along with estimates of amount of area that can be sealed per day (8 hours), number of operating days per month and number of downdays per month for maintenance and repair.
RESULTS AND CONCLUSIONS
The participating teams competed in a three-day event containing the paper design and presentation, as well as the bench-scale demonstration. 1994 Design Contest participating teams are listed in Table I. The winners of awards are shown in Table II. The top four solutions are presented following this paper. The winning designs are being examined by DOE for possible application to actual site remediation. If selected, the DOE will work with the individual schools to work out the contractual arrangements.
The 1995 contest focuses on a waste handling and removal, as well, as waste remediation/stabilization process. The program has continued to accomplish its objectives of providing a design challenge and a medium for exchanging information in the environmental area between various participating teams, industry and government.

16-2

SPARTAN ENVIRONMENTAL TECHNOLOGIES PROPOSAL FOR SITE REMEDIATION AND RECLAMATION PRESENTED TO WERC CORPORATION

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ABSTRACT

The WERC Corporation requested proposals for harvesting water from 100 square miles of arid land and for remediation of five acres of contaminated soil. The Michigan State University (a.k.a. Spartan Environmental Technologies or SET) design was selected as the best in the international competition held in Las Cruces, NM in May of 1994. This paper outlines Spartan Environmental Technologies' proposal. SET evaluated nine alternatives for water harvesting and selected one based on the use of fly ash as technically feasible and most economical. Ten site remediation alternatives were investigated. Solvent extraction with water rinsing was selected as the best alternative. SET considered legal and regulatory issues, health standards, and community concerns in selecting the design alternatives for proposal to WERC.

INTRODUCTION

Each year, the Waste-management Education & Research Consortium sponsors an international design contest under the auspices of the Department of Energy. The contest invitation is in the form of a request for proposals from a hypothetical company called the WERC Corporation. In 1994, the contest required the contestants to prepare two designs: (1) water harvesting and (2) site remediation of a contaminated soil. Michigan State University students formed a design team composed of 27 students from five departments: civil, environmental and, chemical engineering, crop and soil science and resource development. They formed a "company" called Spartan Environmental Technologies (SET) to investigate the problem and alternative solutions. The following discussion is a summary of their investigations and proposals.

WATER HARVESTING

The theory of water harvesting is to collect water over a large area and to utilize this water for irrigation. Sealing of the surface to minimize infiltration increases the runoff and the potential volume of water that can be collected. As requested by WERC, SET's proposal addresses only the process for sealing the terrain and does not cover the methods for water collection. The alternatives shown in Table I were considered for possible use.

Based on initial cost estimates, expected effectiveness and toxicity, three alternatives were selected for further study: water softening sludge, water based siliconates and fly ash. The results of infiltration studies are shown in Tables II, III and IV. Six tests were conducted using fly ash mixed with water. No significant amount of water infiltrated the fly ash treated soils in a 60 minute period. These data indicated that the fly ash incorporated into the soil with water yielded the least infiltration. Optimization studies on the proportions of fly ash, soil and water were conducted by measuring the surface runoff compared to a control surface of geomembrane (Table V.)

Several processes were investigated for the application of the fly ash: surface application of a fly ash/water slurry, incorporation of a fly ash/water slurry into the soil, and incorporation of the fly ash into the soil followed by the addition of water. Incorporation of the fly ash followed by the application of water yielded the most satisfactory results. A thick sealant layer is created by mixing the fly ash into the soil. The application of water will "set" the surface of the soil/fly ash mixture to form a strong, impermeable layer. If cracking occurs, subsequent precipitation events will provide water that will react with lower layers of the soil/fly ash mix and a new impermeable layer will form. Thus, the sealant is self-repairing. Should gross erosion occur, this system is easily repaired by incorporation of another fly ash/water layer.

The proposed full scale design involves a four step process. The site initially will

be prepared by removing any obstacles and vegetation that would inhibit vehicle travel. Then dry fly ash will be applied to a depth of approximately 1/2 inch using a commercially available dry material applicator. Next, a cultivator will be utilized to incorporate the fly ash into the top two inches of soil. In the final step, approximately 1/8 inch of water will be sprayed onto the soil/fly ash mixture to activate or harden the mixture.

REMEDIATION

WERC Corporation identified five acres of their southwest operation that required remediation. The soil contained unacceptable levels of lindane, endrin, methoxychlor, cadmium, silver, copper, ethylbenzene, methyl iso-butyl ketone (MIBK), chloroethene (VC), and tetrachloroethylene (PCE). These contaminants were in the top foot of soil. The initial concentrations provided by WERC Corporation and the remediation criteria are listed in Table VI.

In preliminary screening SET investigated the alternatives shown in Table VII. Using the method of Jury et. al., it was found that the volatile compounds would be removed from the soil within 15 days. (1) Because of the large safety factor in actually applying the model (i.e., the implementation of the remediation project would take more than 20 times the time for volatilization), SET determined that no treatment would be required for thee compounds. Based on the screening study, solvent extraction followed by water rinsing was selected as the treatment alternative to be investigated in the laboratory.

The extraction process designed by SET is a five stage batch extraction employing two different solvents. The first four stages utilize a one molar solution of di(2ethylhexyl) phosphoric acid (DEHPA) in hexane. The DEHPA, acting as a liquid ion exchange agent, extracts the metals while the organic compounds four stages are followed by a water rinse in the fifth extraction.

The process flow diagram is shown in Fig. 1. The contaminated soil enters the process through a trommel screen to remove oversized material. The DEHPA in hexane is added to the contaminated soil at a solvent:soil ratio of 1:1, volume/weight (V/W). The mixture is blended rapidly for thirty minutes and then allowed to settle for thirty minutes. The liquid fraction is decanted off and sent to the solvent recovery unit. This process is repeated three additional times. After decanting the fourth time, water is added as a final rinse at a water:soil ratio of 1:1 (V/W). The process generates the following waste streams: 1) spent DEHPA, 2) spent hexane, 3) decanted water, and 4) soil raffinate. These are further treated prior to discharge to the environment as shown in the process flow diagram.

The spent DEHPA solution will have a relatively high concentration of metals. This flow is pumped to a gravity separator. The DEHPA fraction from the separator is pumped to an acid stripper/settler. In the stripper, the liquid ion exchange reaction will be reversed. Metals will be dissolved in the acid and the DEHPA will be regenerated for reuse. Thus, a majority of the DEHPA will be recovered and reused. The silver may be recovered from the stripping solution through electrolysis using a parallel plate alternating electrode system. The remaining metals are removed from the solution by precipitation, solidified and disposed of in a RCRA Subtitle C landfill.

The spent hexane solution containing the pesticides is regenerated in a batch distillation unit. The recovered hexane is reused. The residual organics from the distillation process will be disposed of in a RCRA Subtitle C incinerator. The off-gases from distiller will be treated by chemical oxidation.

The decanted water will also be treated in the silver recovery and precipitation units. Once this stream is treated it will be discharged to a public owned treatment works. The residual solids are disposed of in a RCRA Subtitle C landfill.

The soil raffinate will be processed in a thermal desorption unit. The thermal desorption unit, operating between 120 -130 C, will volatilize the hexane. The hexane gas is recovered in a condenser. The off-gases are treated by chemical oxidation. The soil from the desorption unit will be placed back on the site.

IMPLEMENTAION ISSUES

Federal legislative and regulatory requirements, as well as New Mexico Environmental Department rules , were reviewed to determine WERC's responsibilities in implementing these projects. For the reclamation project, the requirements of the National Environmental Policy Act (NEPA), Safe Drinking Water Act, Clean Water Act (CWA), and Occupational Safety and Health Act (OSHA) must be met. For the remediation project, the requirements of NEPA, Comprehensive Environmental Response,

Compensation and Liability Act, Resource Conservation and Recovery Act, CWA, Clean Air Act, and OSHA must be met.
Health issues are a major concern to anyone working at the job sites and to the surrounding community. SET proposed that all employees be trained in legal, materials safety, emergency response and equipment operation aspects of each project. Protective equipment will be readily available on site. For the remediation site, access will be limited by security measures and engineering measures will be taken to control and monitor contaminant migration. Risk management techniques will be used where contaminants and treatment chemicals are used.
Community relations committees and public hearings are recommended to keep the public informed.
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16-3
THE MESA STATE COLLEGE APPROACH TO THE 1994 ENVIRONMENTAL DESIGN CONTEST: AN EXTRACTION PROCESS FOR SOIL REMEDIATION AND BENTONITE FOR SOIL SEALING
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Mesa State College
ABSTRACT
The first part of the 1994 Environmental Design Contest sponsored by the Waste-management Education and Research Consortium required teams to remediate contaminated soil to specified cleanup levels. MSC selected a process in which contaminants are removed from soil by heating with dilute nitric acid and air sparging. Aqueous effluent is treated by carbon adsorption and cation exchange. The basic idea for the process came from a participant's experience with plutonium processing at a DOE facility. MSC's application of this approach to the contest problem represents a first step in successful technology transfer from DOE to the private sector. The second part of the contest required teams to develop a sealant for application to soil to promote water harvesting in an arid area. MSC selected bentonite as a sealant because of its low cost, availability, and its acceptance as a natural material. The team then developed a simple method of applying the material to the land surface using conventional farm equipment.
PREFACE
Twelve students from the Environmental Restoration and Waste Management program at Mesa State College (MSC) in Grand Junction, Colorado participated in the 1994 Environmental Design Contest. Since 1991, MSC has offered an Associate of Applied Science degree in Environmental Restoration Engineering Technology. A Bachelor of Science degree in Environmental Restoration and Waste Management has been offered since Fall 1993. Participating for only the first time as a four-year program, the MSC team finished in fourth place overall.
An abridged version of the team's report is provided below.
INTRODUCTION
The Students for Environmental Awareness, Research, Compliance, and Health (SEARCH) team members from Mesa State College located in Grand Junction, Colorado, are pleased to have joined the Environmental Assurance Group of WERC Corporation. We, the SEARCH members, fully support the ethics of social responsibility expressed by management and place great importance on conduct of operations in a manner that ensures protection of human health and the environment as well as full compliance

with all legal and ethical requirements.
Our proposal responds to your request for support to deal with one of your WERC site operations located in the Dona Ana County area just outside of Las Cruces, New Mexico. This extensive site has received surface contamination from fallout originating from a neighboring manufacturing region. Contamination includes specific pesticides, metals, organic compounds, and halogenated organic compounds. The contamination is of sufficient magnitude that the soil must be remediated and contained in order to prevent more serious problems in the future. In addition to remediation of contaminants to concentrations below threshold values, reclamation of the site must also be accomplished. Reclamation should include use of a water harvesting system to enhance growth of natural vegetation or agricultural crops on the site in the future.

SOIL SEALING

Technical approach

The SEARCH team investigated several options for addressing WERC Corporation's request for the use of a sealant to reclaim its 100 square mile site. Although we have selected what we believe to be the most viable sealant option, we urge WERC to reconsider its plans to use a sealant for site reclamation. We believe that use of institutional controls and/or other less drastic actions (e.g., treatment through a revegetation program or construction of small catchment basins) will be more effective and much more publicly acceptable. We make this recommendation because the intended purpose of WERC's reclamation efforts is to enhance the growth of natural vegetation or agricultural crops on the site in the future. Any engineering solution (i.e., application of sealant) will do extensive environmental damage to the site that will likely be counter to this objective. The New Mexico site is located adjacent to a large river and ground water exists beneath the site at depths ranging from 8 to 10 feet. Given the availability of water and the aforementioned concerns, we hope that consideration is given to other reclamation options. Many such options can probably be applied at costs considerably less than any sealant option available.

SEARCH team members identified five criteria to be used in evaluating potential sealant materials. The sealant must: 1) exhibit water repellency to prevent both infiltration of water into the soil and erosion of the soil surface; 2) be resistant to decomposition from exposure to ultraviolet light; 3) be non-toxic to plants, animals, and humans; 4) be easily handled and capable of being applied to the land surface in an efficient manner; and 5) not be a geomembrane, as specified by the WERC corporation. Three materials were identified for detailed consideration: an asphalt-bentonite emulsion, a liquid polymer, and bentonite clay. The bentonite clay was selected as the sealant that best met the performance criteria. Bentonite-treated soil was demonstrated to have excellent water repellency and is known to have minimal toxicity and good UV durability. Bentonite is also available at low cost. The liquid polymer was rejected after extensive testing because it failed to meet the water repellency criterion. The polymer formed a hard crust on the soil surface that cracked following the initial tests. The asphalt-bentonite emulsion was rejected because of the presence of hydrocarbons and reports that water collecting in catchments lined with asphalt-bentonite will turn a black color. Difficulties with the emulsion binding up in a sprayer and being difficult to apply were also reason for rejection.

We propose to use standard agricultural equipment for application of the bentonite to the land surface at a rate of 1.5 pounds of bentonite per square foot. A widely-available seed and fertilizer applicator pulled by a tractor of at least 50 horsepower will be used to distribute the bentonite. The hopper on the applicator will be 10 feet wide with openings on the bottom spaced 3 inches apart. Flow rate can be adjusted by a lever on the applicator and by the speed of the tractor. The optimum speed for the applicator is 5 miles per hour, which will allow a single machine to cover one acre in 40 minutes. The hopper can hold 800 pounds of bentonite, which will be applied at a rate of 1.5 pounds per square foot. In order to reduce or eliminate refilling time, a truck and auger system will be driven alongside the applicator to provide continuous delivery of bentonite to the hopper. Simultaneous use of 5 applicators in this fashion will allow the 100 square mile area to be covered in 266 work days.

After application of bentonite to the land surface, a culti-packer will be used to incorporate the bentonite into the top 3 inches of soil. The culti-packer is a

common piece of farm equipment that uses a system of tines and rollers to break up, till, and smooth the soil. A 120-horsepower tractor is required to pull the culti-packer.

The equipment described here is widely available, both new and used. The known durability of the equipment should minimize depreciation and preserve value for resale at project completion.

Testing

The different sealants were tested according to the following procedure. Dried, sifted, and homogenized samples of local soil conforming with WERC's specifications were placed in one square foot pans. The liquid polymer was applied by spraying onto the soil in several thin coats; four different dilutions with water were tried. The bentonite (Swell Seal #10) was mixed with in different proportions with soil, then distributed as a three-inch layer on top of the soil. A pressurized sprayer was used to add water at a rate of 0.5 gallons over a 9 minute period to simulate a hard, heavy rainfall. The soil pan was inclined at a 30 degree angle for this operation. The mass of soil washed into a catchment was measured to determine the degree of erosion. The volume of water washed into the catchment was measured to determine water repellency. Test results are shown in Table I.

Soil without any sealant suffered a great deal of infiltration and erosion. Application of bentonite sealant decreased both infiltration and erosion. The best results were observed when part of the bentonite was incorporated into the top three inches of soil and the rest was applied in a thin layer on the soil surface. Bentonite can limit infiltration to as little as 17 percent of applied water. The polymer was as efficient as bentonite in the initial tests, but was rejected because of later cracking. Although bentonite also cracked upon drying, it softened and resealed upon becoming wet again.

Cost

The unit cost for sodium bentonite sealant is $23 per ton. At the selected application rate of 1.5 pounds per square foot, the cost for the sealant is $753 per acre or $48.2 million for the entire 100 square mile site. This cost includes a 5 percent waste allowance for over-application and spillage. Information from the "Means Site Work Cost Data" reference book indicate that operational costs will be approximately $36 per acre, including support operations. The equipment and labor costs sum to $2.3 million for the entire site. Including roughly 2 to 3 percent for overhead, total cost for soil sealing is $52 million. Assuming an average annual precipitation of 6.5 inches and 20 percent infiltration, the volume of water collected per acre per year is 132,286 gallons. The cost per gallon of harvested water (after one year) is $0.006 per gallon.

SOIL REMEDIATION

Technical Approach

SEARCH members investigated, researched, and compiled information on possible technical solutions. Several meetings were held to evaluate and select the most promising remediation processes. At the conclusion of our feasibility study, we conceptualized the Pollutant Extraction/Cation Exchange (PECE) process as our preferred alternative. We have determined that the PECE process is the most effective alternative in terms of processing rate and yield, equipment setup time and reliability, operating cost, and functionality. The PECE process complies with technical and regulatory requirements and is a safe and publicly acceptable process.

In the PECE process, contaminated soil is placed in a dissolution vessel containing nitric acid (0.1 M) at a mass ratio of 5:2 (acid solution to soil). Heat is applied to the vessel and a gentle air sparge is introduced. The temperature is held at 90 C for forty-five minutes, allowing soil contaminants to become dissolved. Vapors are collected and returned to the dissolution vessel through a condenser system. After cooling, the solution is drained through a primary filter. The soil is rinsed, dried, and sampled; clean soil is returned to the site.

The filtrate is then pumped bottom-to-top through a granular activated carbon (GAC) unit to remove dissolved organics. Sampling is conducted to verify removal of the organics. The pH is also measured and adjusted to a value greater than four to ensure effective cation exchange in the next step of the process. Technical grade Chelex-20 resin was selected for cation exchange. Chelex-20 is a coarse mesh resin useful for large-scale cleanup. The clean effluent produced by this process is sampled and collected in a basin for use in reclamation and water harvesting

activities.
Two secondary wastes are created by the PECE process: (1) contaminated activated carbon and (2) metals loaded onto the cation exchange column. The activated carbon will be transported to a treatment, storage, and disposal (TSD) facility. The metals loaded onto the cation exchange column will be sold through one of several regional "waste exchanges" to companies that has expressed interest in recovery of these metals.
PECE equipment will be truck-mounted in order to minimize setup time and expedite site remediation. System components will be constructed of stainless steel with a corrosion-resistant coating. Each component will be bolted to a flatbed trailer and supplied with an impact-resistant cover. Additionally, the entire trailer will be covered for transport.
Our research determined that the dissolution and cation exchange technologies have been successfully applied for many years in the defense nuclear industry to destroy contaminants and recover various metals for further production. By incorporating several modifications of these widely-accepted technologies, we have successfully demonstrated that this technology transfer can be effectively utilized to address WERC's current environmental restoration concern at its New Mexico operation; the PECE shows great promise for application to future environmental restoration needs as well.
Bench-scale testing
A one-liter three-port distilling flask was used as our dissolver vessel and three ground glass stoppers were fitted with rubber corks. The first contained a thermometer; the second contained the air sparge tube, constructed of bendable glass tubing; the third, the center port, contained the 13-inch condenser unit. Tygon tubing was attached to the fill and drain openings in the condenser and a cooling water source was added. Tygon tubing was also attached from the air sparge unit to the air compressor. The distilling flask for the dissolution unit was placed in an electric heating mantle.
Seventy-five grams of contaminated soil were weighed and placed in the dissolution unit with 400 milliliters of 0.1 M hydrochloric acid were added. The heating mantle and air compressor were each activated and the mixture was heated and circulated for 1 hour after reaching the optimum temperature of 90 C.
Following filtration, the dissolved phase solution was placed in a flask which held 10 grams of granular activated carbon. A magnetic stir rod was installed and the flask was placed on the magnetic stirring device, stirred for 10 minutes, and filtered to a third flask. The pH of this solution was approximately 6.20.
A 50 mL graduated buret was used as an ion exchange column. The buret was filled with 10 grams of Chelex-20 resin. The resin had been hydrated 3 times with deionized water (150 mL). The dissolved phase solution was processed through the cation exchange column at the rate of one drop per second.
Treated effluent samples were analyzed for pesticides by gas chromatography using laboratory standard operating procedure (SOP) U.S. EPA SW-846, Method 8080. The determination of cadmium, copper, and silver was done by inductively coupled plasma-atomic emission spectrometry. The determination of volatile organic compounds by gas chromatography-mass spectrometry was done using laboratory SOP U.S. EPA SW-846, Method 8240. All laboratory quality control requirements were met.
Under acidic conditions and high Eh, metal contaminants in this study's soil are soluble. If the soil is allowed to come to equilibrium with the amount of fluid used in the PECE process, no undissolved metal contaminants should remain. Agitating and heating the acidic solution enhances the rate at which equilibrium is attained. The conditions that the organics are exposed to, high Eh and low pH, are highly oxidizing. Any organics not oxidized to carbon dioxide and water are dissolved in the acid solution. The organic contaminants can then be concentrated onto activated carbon because of its strong affinity for organic contaminants. Volatile organics that are not condensed back into the reaction mixture are collected on a carbon filter placed at the outlet of the condenser.
Concentrations of metals and pesticides in the extract from the dissolution process ranged from hundreds to thousands of milligrams per liter. The efficiency of metal removal from the soil was low, ranging from only 1 percent for copper to 25 percent for cadmium. No residual pesticides were detected in the soil. Results for the volatile organics were ambiguous. Careful consideration of the chemistry of this process led us to believe that replacing hydrochloric acid with nitric acid would

improve performance for metals. Nitric acid was used during the contest, apparently with considerably better results.
Following GAC treatment, pesticide concentrations in the dissolution extract were reduced to 1.6 ug/L for lindane, 0.23 ug/L for methoxychlor, and 0.1 ug/L for endrin. After treatment by cation exchange, silver was reduced to non-detectable levels, cadmium to 5.9 ug/L, and copper to 16.3 ug/L.
Costs
In the full-scale implementation of PECE, we estimate that 21,600 pounds of soil can be treated per day. Approximately 925 work days would be required to remediate the estimated total weight of contaminated soil. We recommend that three treatment units be used simultaneously to reduce processing time.
For three treatment units, total capital costs were estimated to be $306,071. Operating costs are estimated at $2,601,074, which includes costs for incineration of spent activated carbon.
NON-TECHNICAL CONSIDERATIONS
Community Relations
Establishing an open, two-way relationship with the community is critical to the success of any remediation project. It is our ethical responsibility to ensure that such a program is in place prior to beginning any site activities and that the program establishes mechanisms for communication that will prevent misconceptions between WERC Corporation and local citizens. Because numerous recent environmental issues have received bad publicity and received increased public involvement, we propose appointment of a public relations official to oversee our community relations program. Our public relations representative will be knowledgeable about environmental remediation/reclamation and will exhibit strong communication skills. These capabilities will be essential in dealing with government officials, private citizens, and advocate groups to create win /win situations.
Our community relations plan will emphasize the successful technology transfer being applied to the Las Cruces site. Technologies similar to our PECE process were used in the defense nuclear industry during the cold war. We have demonstrated that this existing technology can be economically and effectively converted to new, positively-perceived applications in the areas of environmental restoration and waste management.
SEARCH members will ensure that the community relations plan emphasizes the following reclamation and reclamation activities:
Performance of environmental monitoring throughout remediation and reclamation activities to ensure that contaminants are controlled on site and that the surrounding environment is protected from spread of contamination;
Performance of post-closure monitoring;
Evaluation and controls to eliminate any potential public hazards; and
Full cooperation of WERC Corporation and the SEARCH team with all Federal, State, and local authorities.
We will establish several mechanisms to encourage public participation and to obtain public input; these activities will be designed to ensure that the community understands and does not oppose our remediation approach. Planned activities include public meetings with measures in place to facilitate attendance (e.g., selecting meeting locations particularly accessible to the general public; advance distribution of public notices; press releases provided well in advance to local radio, newspaper, and television stations). Special efforts will be made to supply visual aids and information to a representative of KRWG, University of new Mexico, since it serves as the television station exclusive to the Las Cruces area. Special efforts will also be made to work with the local newspapers, Las Cruces Sun News and the weekly Bulletin. Our public relations official will be our single point-of-contact and will be responsible for contacting the media. We will establish schedules and a special telephone line to ensure that our public relations official can be easily contacted.
We must make genuine efforts to identify and understand community perceptions and concerns and must demonstrate sincerity and compassion when addressing public concerns. The SEARCH team will seek to identify any potentially affected interests (i.e., individuals or groups who may, in any way, perceive that our reclamation or remediation activities may impact them). Potentially affected interests include local media; key community leaders; local government officials; local environmental groups including the Sierra Club, the Mesilla Valley Club of Audubon, the Native

Plant Society, and the Natures Park Group; and the Concerned Citizens of Sunland Park, who have recently been involved with hazardous waste issues in New Mexico.
Our community relations plan will address the history behind the site contamination and will express the mission of the WERC Corporation related to both the Las Cruces remedial action and overall corporate endeavors. The goal will be to establish and maintain public confidence in the company without over publicizing the issue.
We will ensure that our remediation and reclamation mission and its importance are clearly expressed at the public meeting and that the public understands the risks involved if no remedial action was performed on the site. We will openly address any aspects that may considered potential risks and precautions taken to address them. Activities that may be perceived as risks include potential for air pollution from dust raised during excavation; effluent produced in the process and effluent discharge; and use of hydrochloric acid in the PECE process.
Our community relations plan will directly address Emergency Plan and Community Right to Know Act (EPCRA). Key issues that will be addressed include procedures for responding to releases, procedures for notifying the pubic, process to evaluate releases and their impact on the area and population, MSDS information for all chemicals present, and conduct of periodic emergency drills.
Our public meetings will encourage involvement. Because approximately 40 percent of the Las Cruces community is Spanish-speaking we will ensure that our public relations official is bilingual and that bilingual facilitators are available at each meeting. Our remediation and reclamation processes will be explained in non-technical terms that will promote understanding and allow for questions and input. Sufficient time will be allowed for all to openly express suggestions, questions, and concerns.
SEARCH will promote public acceptance by emphasizing economic benefits and jobs for the community and future land use for the community once remedial action and restoration activities are complete. We will provide information regarding our employees' qualifications and expertise and explain the measures taken to ensure worker safety.
Health & Safety and Regulatory Considerations
A standard, site-specific health and safety plan was prepared for the proposed operations. The plan included provisions for training, medical surveillance, hazard communication, site operations, and emergency response.
Major environmental laws and regulations were analyzed for their effect on the proposed operations. Of particular concern is the proposed discharge of treated water from the soil remediation process to a basin for subsequent use in site reclamation. It is anticipated that a permit under the National Pollutant Discharge Elimination System (NPDES) may be required for this discharge. However, if the basin is less than 1 acre in area, an exemption from this requirement may be possible. New Mexico does not have NPDES authority, so application will have to be made to EPA Region VI in Dallas, Texas.
CONCLUSIONS
The SEARCH team urges the WERC Corporation to consider reclamation options other than application of a sealant. We believe that suitable options could be identified that would be less costly, less harmful to the environment, and more publicly acceptable.
The SEARCH team did, however, research several options and selected bentonite as the sealant of choice on the basis of extensive testing and the acceptability of results according to guidelines provided by WERC. We have demonstrated the ability of bentonite in limiting infiltration and erosion, and have found bentonite to be widely available at reasonable cost.
The SEARCH team believes that the Pollutant Extraction/Cation Exchange (PECE) process is an excellent process to treat WERC Corporation's contaminated soil and that test demonstrations to be held in Las Cruces in April will support our claims. We have developed and tested a bench-scale model of our process that represents successful transfer of technology developed in the nuclear weapons complex and applied it to this environmental restoration problem.
We carefully reviewed and verified each detail in calculating the effectiveness of our process. However, we recognize that several improvements to optimize performance can be made after additional testing. For example, tests with different molarities of acetic acid could be conducted since acetic acid is considered to be safer for use in environmental restoration. As always, field testing is recommended prior to

full-scale operations.
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REMEDIATION OF WERC SITE: PRELIMINARY REPORT
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INTRODUCTION
WERC Corporation requested that a remediation plan be developed for a site that is contaminated with hazardous materials. A five acre portion of the site is contaminated with heavy metals, halogenated organics, nonchlorinated organics, and pesticides to a depth of one foot. Target remediation goals range between 82% and 96% removal of the initial concentration depending on the compound.
REMEDIATION
ECoWVU conducted a thorough investigation of treatment options. Emphasis was placed on maximizing clean soil yield, minimizing costs and waste generated, and complying with all federal, state, and local regulations. After extensive literature review and laboratory testing, soil washing was determined to be the best type of remediation system available.
Soil washing of all soil size fractions is the only technology capable of removing both heavy metals and organic contaminants while satisfying the goal of maximizing clean soil yield. In addition, soil washing can remediate the soil within the one year deadline and is generally cost-effective. ECoWVU tested several soil washing agents to determine their effectiveness in removing the soil-bound contaminants. Based on these experiments, sodium dodecyl sulfate (SDS), a biodegradable, nontoxic, anionic surfactant, exhibited high removals of the organic contaminants, and 1N HNO3 was effective in removing the metal contaminants.
Therefore, ECoWVU recommends a treatment train consisting of excavation and dry separation, followed by soil washing with both an SDS and 1N HNO3 wash. In Fig. 1, a schematic of the chosen treatment process is presented for a soil processing rate of 100 tons/day. This alternative utilized a two phase soil washing process that is capable of treating soil to the desired contaminant levels at a cost of $575/ton. The system was designed to remediate 11,000 tons of contaminated soil in 42 weeks (including construction and startup). A clean soil yield near 100% is expected. Using an innovative system of SDS and 1N HNO3 reuse and regeneration, and treatment of ancillary wastewater, minimal residuals are produced.
Excavation and Storage
A 50,000lb track excavator will be used to excavate the 11,000 tons of dry soil at a rate of 500 yd3/day. Four hundred tons of soil will be excavated per week and placed in a 100 foot by 100 foot air-tight building, which also contains the dry size-separation unit. Earthwork can be completed in about 24 days.

Soil Washing Process
Each day, 100 tons of contaminated soil will be fed into an inclined trommel unit, enclosed in the airtight building, for dry separation. Volatilization of the volatile and semi-volatile contaminants should occur as the soil particles >2 inches (5cm) are separated. Because of the likelihood of the contamination of sands with heavy metals and pesticides, it was decided that all particles <2 inches would undergo soil washing.
About 50% of the soil will be greater than 5cm, and will be fed by conveyor belt to a high pressure spraying unit, which flushes any remaining fines from the cobbles. The clean cobbles are swabbed to ensure successful removal of the contaminants and returned to the site. The material <5cm will be made up of approximately 66% sands and 34% silt and clay. This material will be transported to two 40,000 gallon soil washing tanks (referred to as SWU-1) where a 2.5% SDS solution, with a 5:1 liquid:soil ratio (by weight), employing a 4-hour mixing and 4-hour settling schedule. Following settling, the supernatant is decanted and fed to the second soil washing unit
(SWU-2) to repeat the aforementioned process with a new batch of contaminated soil. Then, 1N HNO_3, at a 5:1 liquid:soil ratio, is added to the half-washed soil in SWU-1, utilizing a 2-hour mixing and 4-hour settling schedule. Following settling, the 1N HNO_3 supernatant from SWU-1 is decanted and transferred into two 30,000 gallon holding tanks and later used to acid wash the soil in SWU-2. The process is staggered so one batch of soil is being washed with surfactant while the other is washed with nitric acid, maximizing the amount of soil treated per week and adding operational flexibility. Before each wash, the appropriate amount of SDS, water, and concentrated HNO_3 will be added to the unit to restore the original liquid:soil ratio and maintain wash solution concentrations. Washing solutions will be regenerated after two complete cycles. The SDS solution is renegerated using a liquid exchange column containing methylene chloride. The HNO_3-metal waste water is treated using flocculation/coagulation followed by settling.
Following the SDS and nitric acid washes, the soil is transferred to a smaller tank where a 2:1 clean water rinse is performed to remove the wash solution residuals. The soil slurry settles briefly to remove the sand fraction. The supernatant, containing silt and clay, is mixed with a nontoxic polymer to promote settling and fed to a high speed centrifuge to ensure maximum separation of fines. Each batch of clean soil will be tested to ensure sufficient contaminant removal. The used water is pumped to a holding tank, where a portion of it is used to replenish the liquid streams. The remaining used rinse water is fed through two 10' by 2' diameter ion exchange columns (11) for metal removal and through a granular activated carbon column for organic contaminant removal. (12) The resultant metal and organic contaminant concentrations will be much less than 1 mg/L and thus, will be suitable for discharge to the local POTW. (13) A portion of this water will be recycled to the high pressure cobble wash step.
Further Research Requirements
Before final implementation of the remediation process, ECoWVU strongly recommends that additional research be conducted. Also, the soil washing process should undergo the pilot and field-scale testing according to the USEPA's Three-Tiered Approach for Site Remediation. (15) Results from pilot and field-scale tests will be used to optimize process performance, determine possible process modifications, and develop better cost data.
Materials and Methods
ECoWVU conducted extensive laboratory testing to select the most effective extracting solutions and to optimize their use in the treatment process. Based on literature reviews and preliminary lab tests, the following three washing fluids were identified as having the potential to remove the contaminants of concern: anionic surfactants, NaOH, and HNO_3. Pesticides, because of their hydrophobic nature, are tightly bound on the soil. Surfactants are able to transfer soil-bound hydrophobic, oily compounds into a water based emulsion by reducing the interfacial tension between the soil and water phases. (16) Sodium dodecyl sulfate (SDS), an anionic, biodegradable, nontoxic surfactant, removed over 90% of soil-bound pesticides and organics at ten times the critical micelle concentration (0.25%). (5) Because its effectiveness had been demostrated, SDS was selected for further testing. NaOH was selected because pesticides hydrolyze at high pH values. (17) Metal-removal using acids is well-documented. (17) HNO_3 was selected for further

study over other acids, such as HCl and H2SO4, because they have the potential to form precipitates with metal contaminants. Following the literature review and initial metal removal experiments, a series of soil washing experiments were conducted to determine: 1) the effectiveness of each solution in removing contaminants, 2) the optimal liquid:soil ratio and reaction time, and 3) the effect of the wash solution order. Table I provides a list of the soil washing experiments. Clean soil from the WERC site was contaminated to the prescribed levels using WERC's contamination recipe. Vinyl Chloride was excluded from the testing procedure because it was difficult to obtain and because it is improbable that it would remain on soil samples, due to an extremely high vapor pressure. For each soil washing experiment, a predetermined amount of contaminated soil was placed in a Nalgene container and a specified volume of washing fluid was added. The soil slurry was then placed on a shaking table for a predetermined period. After shaking, the soil slurry was centrifuged for 5 minutes at 7000 RPM to separate the liquid from the solids. The supernatant liquid was saved and analyzed for metal content on a Perkin Elmer Atomic Absorption (AA) unit. The concentration of orgainc contaminants on the washed and centrifugal soil were determined using EPA Method 8080 (Pesticides) and Method 8240 (VOC/SVOC). An unwashed, contaminated soil sample was analyzed to ensure the contamination method successfully placed the specified amounts of the compounds on the soil. Spiked matrices and several levels of recoveries were also carried out to ensure the required level of quality control/assurance.

Settling tests were performed with the wash solutions to determine the time required to separate the soil from the wash solution, allowing ECoWVU to evaluate the need for centrifuges in the process design.

Soil Washing Results

A summary of the experimental soil washing conditions, the final soil contaminant soil concentrations (mg/kg), the initial contaminant concentrations, the WERC proposed cleanup targets, and the analytical detection limit for each contaminant are presented in Table I.

Volatile and semi-volatile concentrations were below detectable limits for all soil solutions tested. Based on volatilization studies on the contaminated soil, researchers at ECoWVU believe that VOC/SVOC removal occurred predominantly through the volatilization of these compounds, and the dry separation unit was designed accordingly. SDS HNO3 met all of the target levels except Ag. Based on results in Table I, ECoWVU researchers concluded that an initial 2.5% SDS wash at a 5:1 ratio for 4 hours, followed by a 5:1 1N HNO3 for 2 hours would meet all cleanup goals for the contaminants. With optimization of the system, even the target level for the lowest scoring contaminant should be reachable.

BUSINESS PLAN AND ECONOMIC ANALYSIS

For the effective evaluation of alternatives, an economic analysis was conducted, and our remediation and reclamation solutions were chosen. Costs are presented in March 1994 dollars and are considered accurate to within +50/-30 percent. The total cost of the 1 year remediation plan is $6.3million ($575/ton).

ECoWVU has chosen an innovative technology called soil washing to solve the remediation problem. The total cost of the project will be $6.3 million ($575/ton), using 82 BTU/kg of contaminated soil. Fixed and variable costs are $2.2 million and $4.1 million, respectively. A further cost breakdown may be found in Table II.

Some benefits of the chosen process are the maximization of clean soil yield, the meeting of cleanup goals, and the minimization of secondary waste. Generally, soil washing costs range from $50 to $250 per ton. (17) However, most commercial vendors do not treat the fine fraction, and therefore have less yield and more waste. For example, if soil particles <2mm were disposed off-site, the cost would be $4.0 million, but the clean soil yield would be only 50 percent.

Also, because of the 1-year duration, several pieces of equipment may be sold for a depreciated value. A seven-year life (20) is assumed for the equipment, and a potentially profitable option is the reuse of the process for other remediation projects or the marketing of these services to other corporations in need of remediation technology. ECoWVU strongly recommends that WERC consider these options further.

LEGAL, HEALTH, AND REGULATORY CONSIDERATIONS

Operation of ECoWVU's proposed treatment train and sealant application for the remediation of the contaminated site requires compliance with specific Federal, State, and local regulatory standards. In 1992, New Mexico readopted the sections of

the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Liability, and Compensation Act of 1980 (CERCLA) as the New Mexico Hazardous Waste Management Regulations (NMHWMR). (7) New Mexico state regulations are consistent with Federal regulations unless otherwise noted.
Under RCRA, ECoWVU's TSD facility would be considered a generator of hazardous waste. A generator of hazardous waste is "any person, by site, whose act or process produces hazardous waste...or whose act first causes hazardous waste to become subject to regulation" (40 CFR 260.10(a)).(22) The soil washing process requires that the soil be transferred from the soil matrix to the liquid/gaseous matrices; therefor, WERC is considered a generator of hazardous waste.
The proposed TSD facility that ECoWVU will manage must comply with one of the following two air emission limits for volatile organic compounds. The facility must either limit total organic emissions with process vents to less than 1.4 kg/h (3 lb/h) and 2.8 Mg/yr (3.1 tons/yr), or reduce its total organic emissions by 95 percent (40 CFR 264.1032).(22)
Discharge limits for pH and the metal contaminants in the WERC soil are specified by the local Publicly Owned Treatment Works (POTW) in Table III.
The organic compounds that may be present in the liquid effluent are not directly regulated (with respect to specific discharge limits) by federal, state, or local statutes. Representatives from the Las Cruces POTW indicate that an Industrial Discharge Permit would be required and would likely be attained for the treatment project. The waste stream from the remediation process will be checked for compliance on these limits at least once a year by the Las Cruces POTW.
Under EPCRA, ECoWVU must report unplanned releases of any hazardous materials and the quantity released to each environmental medium (soil, air, and/or water). ECoWVU must notify a 911 operator if a release occurs on-site or during transportation to the licensed disposal facility. EPCRA requires that ECoWVU provide a material safety data sheets (MSDSs) to the local fire department and state and local emergency response commissions. If an employee is being treated at a health care facility for an on-site injury, the MSDS information must be provided to the health care professionals. Under the Occupational Safety and Health Administration (OSHA), ECoWVU must maintain an accurate record of on-site deaths, illnesses, injuries, and exposures to toxic substances that are incurred by its employees. (21)
The most basic duty of ECoWVU under OSHA with respect to worker safety is to furnish its employees with an environment that is free of hazards that are likely to cause serious physical harm or death. The hazard communication standards (HCS) under OSHA requires the following safety procedures of ECoWVU: 1) A MSDS for each hazardous substance that employees will be exposed to must be easily assessable to all facility workers, 2) A hazard communication program containing provisions for labeling of containers, MSDSs, and an employee training program must be written and implemented, and 3) OSHA certified employee training must be provided to employees by ECoWVU. Under OSHA regulations employees must be educated as to the engineering safety controls, work practices, and the proper usage of the personal protective equipment (PPE). The PPE that is required based upon OSHA's specifications in 29 CFR 1910.120 must meet the most stringent requirements of Level C protection. ECoWVU feels that with standardized training in the use of all Level C PPE, there should be minimal threat to employee health and safety. To mitigate the problem of heat stress that may result from working in a full body suit at the given site temperatures, cooling vests will be available for all employees.
To control air emissions of VOCs that may result from agitation of the contaminated soil during excavation, the soil should be excavated into a dump truck and then covered with a heavy canvas sheet while transporting the soil to the on-site storage facility. (7) ECoWVU will not have to comply with transporter standards as outlined in 40 CFR 263 when engaging in on-site transportation of the contaminated soil. (1) ECoWVU will hire a licensed transporter to remove the waste (in the form of solid and/or liquid residuals) from the site. The transporter will come on-site to get the waste after ECoWVU has properly labeled, marked, and contained the hazardous waste. The manifest reporting requirement must be signed by ECoWVU and given to the transporter at the time the waste leaves the site. The transporter must keep the manifest with the hazardous waste at all times and the document must show any transfer of the hazardous waste to another transporter or to the disposal facility. The final manifest signed by all parties should be returned to and kept by the generator for at least three years. (21)

ECoWVU's must obtain a final RCRA permit as stated in 40 CFR 270 as adopted in the NMHWMR part IX, sections 901 and 902. (7) Additionally, ECoWVU's TSD facility must obtain an EPA identification number for purposes of record keeping (40 CFR 264.11). Under RCRA, ECoWVU may only store hazardous waste on-site for a period of 90 days (40 CFR 262.34(a)). In order to store hazardous waste on-site for more than 90 days, ECoWVU must obtain a permit from the New Mexico Environment Department. (7) At the close of the TSD facility's operation, RCRA and CERCLA closure requirements must be met. ECoWVU must prepare a written closure plan to be submitted to the EPA with the initial permit application.

COMMUNITY RELATIONS

A Community Relations Plan (CRP) is extremely important for any project, particularly one that deals with hazardous wastes. The CRP may be used effectively to help establish a good working relationship with the community. A successful CRP allows the public to be involved in all steps of the process, from the initial screening of alternatives to the final selection of the treatment plan. WERC should designate an independent, experienced spokesperson to create information repositories and inform the public of all actions undertaken. The most important steps necessary to involve the public are putting articles in the local newspaper(s), posting signs at local establishments, hold interviews with community leaders and interested parties, and to place informational ads in the newspapers, radio, and television for at least 50 days.

CONCLUSIONS

ECoWVU's team has designed a dual unit, multi-phase soil washing system that meets contaminant removal goals and achieves a clean soil yield approaching 100 percent. After excavation and soil separation, the fines are washed in a 2.5% sodium dodecyl sulfate solution (5:1), followed by a 1N HNO_3 solution (5:1). The cleaning solutions are then recycled to minimize waste. The system is designed to process 100 tons of contaminated soil per day at an approximate cost of $575/ton and complete the remediation of the five acre site in 42 weeks (including construction and processing).

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EXECUTIVE SUMMARY
The WERC Corporation has requested that Environmental Consultants of West Virginia University (ECoWVU) develop a remedial plan for a site contaminated with several hazardous substances. A five acre portion of the site is contaminated with heavy metals, halogenated organics, nonchlorinated organics, and pesticides to a depth of one foot. Target remediation goals range between 82% and 96% removal of the initial concentration depending on the compound.
ECoWVU conducted a thorough investigation of treatment options in order to select the most effective treatment that would fulfill WERC Corporation's stated criteria. Emphasis was placed on maximizing clean soil yield, minimizing costs and wastes generated, and complying all federal, state, and local regulations. ECoWVU selected a dual unit, multi-phase soil washing system that met contaminant removal goals and achieved a clean soil yield of about 100 percent. The system is designed to process 100 tons of contaminated soil per day at an approximate cost of $575/ton and can remediate the five acre site in 42 weeks (including construction and processing).

16-5
REMEDIATION AND RECLAMATION OF A CARBONATE-RICH SOIL CONTAMINATED WITH ORGANIC AND METALLIC COMPOUNDS
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ABSTRACT
This work concerns the remediation and reclamation of soils from a hypothetical site contaminated with lindane, methoxychlor, endrin, cadmium, silver, copper, ethyl benzene, methyl iso-butyl ketone, chloroethylene and tetrachloroethylene. An ex-situ soil washing process involving batch addition of sodium dodecylsulfate, ethylenediamine tetraacetic acid (EDTA) and sodium thiosulfate was designed for treating the soil. The liquid effluent from the soil washing process was treated with granular activated carbon (GAC) to remove metal-EDTA complexes and organic pollutants, while a cementation bed was used to recover the silver. GAC also was used to treat gas-phase effluent. A bench scale study showed that the remediation process was capable of reducing all but one of the contaminants in the soil to target levels. The bench scale design was scaled up on a mass basis to design the full-scale remediation system. Estimated remediation time for the 12.95 km2 site was 90 days.
Site reclamation involved the application of a two-layer, impermeable sealant to the remediated and surrounding 259 km2 -area for the purpose of water harvesting. The first layer is composed of a cellulose xanthate slurry to be prepared on site. The purpose of this slurry is to enhance the structural integrity of the soil. The second layer is a spray mixture of unrefined paraffin wax and an anti-stripping agent to provide impermeability. The sealant was shown to be non-toxic, UV resistant, and erosion resistant. Bench-scale tests did not produce the predicted

runoff yield of 87%. However, only minor adjustments in the mixture are needed to achieve the desired yield. A modification of an agricultural tractor with additional insulation, heat tracing and exhaust re-routing will be used for the application of the sealant. The sealant application for the proposed 259 km2 area is projected to be completed in one year and five months.

INTRODUCTION

In August 1993, the Waste-Management Education & Research Consortium (WERC) invited a team of students at Michigan Technological University to participate in an international environmental design competition. The goal of the 14-member team was to develop processes for the remediation and reclamation of a hypothetical, contaminated site near Las Cruces, New Mexico. The site was contaminated by industrial fall-out from a neighboring manufacturing region which had deposited lindane, methoxychlor, endrin, cadmium, silver, copper, ethyl benzene, methyl iso-butyl ketone, chloroethene and tetrachloroethylene. The extent of the site to be remediated was 12.95 km2 in area and 0.3 m in depth, which translates into a total of 8.2 million kg of soil to be treated. Design of a remediation process for the soil was challenging because of the mixture of organic and metallic compounds, the wide range of contaminant chemical properties, and the chemical characteristics of the native soils. Design of the reclamation process also was difficult because of the physical characteristics of the site, the stringent efficiency criteria, and the requirement of using a non-geomembrane material. Two objectives were designated for solving these problems: 1) design a process for remediating the contaminated soils from the initial concentrations to concentrations specified for clean closure of the site, and 2) design a non-geomembrane sealant and a machine for sealant application to produce site suitable for water harvesting.

SOIL REMEDIATION

Background

The design approach was dictated by the major aspects of the problem: the large volume of soil to be remediated, the shallow depth of contamination, the composition of the Casito series soil, the 24-hour bench-scale testing time constraint, and the chemical properties of the contaminants. Ex-situ soil washing was the technology selected for remediating the contaminated soil.

common process in ex-situ soil remediation is the separation of oversized materials. Most of the contaminants are preferentially adsorbed onto the soil organic matter and are equally distributed on a surface area basis. This means that more contamination will be present on the smaller fraction due to the large surface area per unit mass. The larger soil fractions, which account for 50% of the total mass of contaminated soil, contains a very small potion of the contamination.

Chemical extraction is defined by the U.S. Environmental Protection Agency as a separation process that uses additives to remove insoluble pollutants from the soil. (1) The additives used in our remediation process were the surfactant sodium dodecylsulfate (DS) and the complexing agents ethylenediamine tetraacetic acid (EDTA) and sodium thiosulfate. The technique most frequently used for the extraction of metals is acid washing. However, the 1-8% percent carbonate content of the native Casito soils would make an acid wash difficult and expensive for low pH metal extraction. Removal of metals from soil by the addition of EDTA has been shown to be an effective alternative to acid washing for metals extraction. EDTA is a strong chelating agent which forms highly stable metal ion complexes with stability constants greater that 107 in stoichiometric 1:1 ratios. (2) The metal ions of concern in this case were Ag+, Ca2+, Cd2+, and Cu2+ with stability constants of 7.3, 10.7, 16.5, and 18.8, respectively. Although EDTA will form copper and cadmium chelates, only slight amounts of calcium will be complexed because calcium has a low solubility (0.14 g/ml) and low stability constant and the fact that the EDTA solution will be slightly basic. (3) Adaptation of a photo processing technique using sodium thiosulfate was chosen to remove the residual silver from the soil. A 2:1 molar ratio of sodium thiosulfate to silver was found to be optimal. (4) The addition of a surfactant effectively reduces the distribution coefficient between the soil and liquid (5) through the formation of micelles. This causes the equilibrium concentration of the contaminant in the micelle pseudophase to increase, and the time to reach equilibrium is shortened. Studies done by Jafvert (6) using DS as a surfactant and the pollutant pyrene were used to design a process for removing pesticides from the soil. The time required for equilibrium was calculated from a first-order rate equation. In this system, DS was chosen because its properties are

well known, it is easily obtained, and is non-toxic, non-recalcitrant, and anionic. In comparison to cationic or nonionic surfactants, DS should not adsorb appreciably to the soils (7).
GAC was chosen for removing the EDTA metal complexes (8) and the DS micelles (9) from the soil processing water and for treating the gas phase captured from the soil treatment unit. The Polanyi potential theory was used to determine the Freundlich parameters for liquid phase adsorption (10). Gas phase GAC parameters for the removal of volatiles were estimated from the Dubinin-Redushkevich equation (11). A software package was used to determine component break through times and reactor volumes. Carbon usage rates and GAC columns were sized accordingly.
A cementation process which incorporates scrap iron or iron wool was selected to remove silver from the liquid effluent. Gould et al. found that this process was effective for removal and recovery of the pure metals as an amalgam. (4)

Process Design Summary

Figure 1 shows a schematic summary of the soil remediation process. The following steps are performed in the application of the soil remediation design.
Step 1. Excavated soil is screened through a 4.76 mm sieve, and all over-sized material is returned to the site.
Step 2. Soil is placed into a constantly stirred batch reactor and washed four times. EDTA is utilized during the first wash to remove cadmium and copper. During the second wash, the silver is removed by sodium thiosulfate while DS is added to remove the more soluble organics. The third wash implements DS to remove the less soluble organics. Clean water is then flushed through the system as a final rinse. The reactor is covered and vented to gas phase GAC during mixing operations due to stripping of the volatile components MIBK, ethyl benzene, chloroethene, and tetrachloroethylene.
Step 3. Upon completion of the soil washing process, the tank is drained, allowing the effluent to flow through a rotary drum filter. The separated liquid is then pumped through storage tanks to allow for the optimal flow rate through the liquid phase GAC system. Liquid phase GAC is used to remove lindane, endrin, methoxychlor, EDTA-metal complexes, and tetrachloroethylene from the process water streams. Silver is separated from the thiosulfate-silver complex by passing the solution through a cementation tank containing scrap iron.
Step 4. The treated soil is dried using a decanting-type, paved, drying bed. Since the contaminants concentrations in the treated soil are not zero, an application for an innovative technologies permit will be submitted for in-situ biological remediation to zero concentrations if the soil is returned to the site.

Testing Results

Individual processes were evaluated in the laboratory prior to a bench-scale demonstration. Jar tests were performed with varying concentrations of surfactant to maximize the removal of radio labeled (C14) surrogate organic compounds, which were measured with a scintillation counter. Batch runs with EDTA and sodium thiosulfate established optimum concentrations. Metals concentrations were analyzed by atomic absorption spectrophotometry. GAC and cementation experiments were not run prior to bench-scale testing. A sieve analysis and a Walkley-Black analysis for total organic carbon content for each size fraction were performed. The results showed that the large size fractions could be left untreated.
Full-scale processes were approximated in the laboratory by bench-scale models in order to optimize remediation techniques. A bench-scale model was constructed and tested during the spring 1994 WERC Design Competition. Five kg of contaminated soil was placed in an 11.5 L bucket. The first 10.1 L soil wash consisted of 8 g of EDTA (10% excess) to produce a 2.36 mM solution. 233 g of DS solution (15 mM) and 1240 g of sodium thiosulfate solution (495 mM) were added for the second wash. DS was again added at the same concentration for the third wash. Clean water was used as a final rinse. The times for individual washes were 0.66, 1.11, and 1.66 hours, respectively. In place of the rotary drum filter, a centrifuge and buchner funnel were used to separate the clean soil from the wash water. The clean, dewatered soil was transferred to an aluminum pan and dried on a hot plate. The wash water was passed through glass tubing packed with GAC and the bucket was periodically vented to gas phase GAC to ensure VOC removal. A simulation of the cementation was performed by placing the final waste water in a beaker containing steel wool.
Table I. shows the concentrations of the soil contaminants used as remediation targets for the bench-scale remediation, as reported by an analytical laboratory.

The results show that only lindane was not removed to the soil contaminant criteria. During the bench-scale demonstration, only two surfactant washes were performed due to unforeseen filtering problems caused by frothing. This may explain why the lindane exceeded the final concentration criteria. Application of reduced surfactant concentrations may overcome the filtering problems by reducing froth formation. The silver cementation process appeared to produce positive results, although no laboratory analysis was performed.

Estimated Full-Scale Costs

A full-scale design was devised on a mass basis by scaling the bench-scale results to treat 12.95 km2 of soil in less than 90 days. Table II shows the breakdown of costs for full-scale remediation.

SOIL RECLAMATION

Background

Selection of the optimal sealant for water harvesting was based on a number of independent factors which may be loosely grouped into efficiency criteria and site-specific criteria. Efficiency criteria are the physical and chemical properties of a material which determine its suitability for use as a water harvesting sealant. The efficiency criteria used during sealant selection were:

1. impermeability
2. non-geomembrane or other "liner" type material
3. low toxicity
4. high resistance to failure by ultra-violet (UV) light exposure
5. color
6. ease of application
7. cost

Once a sealant meets the efficiency requirements, its usefulness was evaluated with respect to unique conditions at the site of application. These site-specific criteria include:

1. stability on slopes up to thirty degrees
2. durability in freeze/thaw cycling
3. high temperature resistance
4. compatibility with Casito soils, especially large cobbles and clays.

Based on these criteria, a feasibility study was conducted to determine the best sealant for the hypothetical site. Due to the large scale of the proposed water harvesting system, particular emphasis was placed on the cost of the sealant material. For this reason, the scope of research into potential sealants was limited to readily-available, commercial compounds which have been pre-engineered for cost-effectiveness and reliability. The sealants which were considered for the site include: aromatic and aliphatic urethanes, silicones, high molecular weight acrylate monomers, rubberized asphalt waterproofing compounds, and petroleum-based waxes.

The results of a feasibility study indicated that the optimal sealant was a petroleum-based wax. Waxes are highly impermeable when applied in a continuous layer due to their non-polar nature. They are also light-reflective; relatively easy to apply in liquid spray, liquid brush or solid chip form; and they exhibit low enough toxicity to be used in food products. (12) Waxes range in price from $0.46 per kg for highly refined paraffin to $0.35 per kg for unrefined wax, which is a byproduct of oil refining processes. A variety of waxes are available with differing melting points to accommodate the required resistance to high temperatures.

The most important factor in the success of a water harvesting sealant is sustained impermeability. The impermeability of the waxes results from their hydrophobic nature. Petroleum-based waxes are simply straight chains of hydrocarbons, meaning that they have single bonds between all carbon atoms in the molecule. (13) Therefore, if a uniform, continuous layer of petroleum wax is applied to soil, the theoretical water penetration should be near zero. However, real world applications of wax sealants suffer from loss of water repellency from UV light or ozone exposure, lack of structural stability due to erosion from freeze/thaw cycling, and lack of sealant continuity due to uneven application.

Published runoff efficiencies of early field experiments using paraffin alone at an application rate of 1.0 kg/m2 averaged better than 90% in the first year and 87% over the following seven years. (14) Unfortunately, due to weak Van der Waals type binding forces, wax alone does not exhibit stability from erosion and freeze/thaw cycling. However, these qualities can be improved by supplementing the wax with an antistripping agent and/or soil stabilizer. (15) Similar results have been published

using the residual wax/antistripping agent and soil stabilizer combination, at an application rate of 0.25 kg/m2, in an identical configuration to that proposed for the hypothetical site. In published studies, increasing the wax application rate from 0.25 to 1.0 kg/m2 had no noticeable effect on water repellency. (15) All of the cited studies tested sealant designs on soils which were relatively uniform in grain size and contained little pore space in comparison to the Casito series. The combined factors of a good rating against the selection criteria and the accessibility of previous wax-based water harvesting results made a petroleum-based wax the optimal sealant for the hypothetical site.

Reclamation Design Summary

The proposed sealant for the site is comprised of two application layers: a soil stabilizer and a heated mixture of residual wax/antistripping agent. The sealant will be applied after the remediation of contaminants. To enhance the structural integrity of the sealant, a cellulose xanthate soil stabilizing solution is first be applied to the site. Because cellulose xanthate has a very short storage life, acquisition of the stabilizer is accomplished through on-site production. Production is relatively simple; it is based on the following reaction, called xanthation, in which the cellulose chain is abbreviated as cell: (8)

cell-O- + CS2 = cell-O-CS2-

High purity cellulose, required for production, is obtained by repulping high quality, post-consumer, de-inked office paper. (16) The waste paper is shredded and chemically repulped using a batch process in which the paper and an 18% (by weight) NaOH solution will be mixed. Chemical repulping was chosen instead of mechanical repulping, because mechanically generated pulp would hinder the efficiency of the xanthate stabilization. (17)

After the reaction equilibrates, the repulped waste paper is treated with a commercially manufactured "dump chest" to remove the colloidal coatings on the paper. (16) Once the cellulose has been regenerated and purified, the slurry is rerouted to the batch reactor tank. In the tank, a 1:1 molar ratio of carbon disulfide is added and mechanically stirred to form cellulose xanthate. Subsequently, the xanthate is diluted with water to attain a 0.4% solution. Cellulose xanthate has a shelf life of only a few days, however the solution may be refrigerated to extend the possible storage time to about one year. (17) Following on-site production, the 0.4% xanthate solution is spray-applied directly to dry soil at a rate of 1.5 L/m2 of soil. (15) The soil stabilizer is allowed to air-dry prior to the next phase of sealant application.

The antistripping agent/unrefined wax mixture is applied after the soil stabilizer. The wax will be placed in the heated tank of the application machinery and allowed to melt. Asphalt antistripping agent will be added to form a 2% (by weight) solution. The melted mixture is spray-applied over the stabilized soil using a rubber-tired agricultural machine at a rate of 0.25 kg/m2. A representation of the soil/sealant interface is presented in Fig. 2.

The most efficient application of sealant occurs when the soil is dry. Therefore, it is recommended that both phases of the sealant application be conducted on warm, clear days when the moisture in the top 7 to 8 cm of soil is minimized. Accounting for weather-induced down time, both phases of the reclamation would be anticipated to be completed in approximately one year and five months.

As part of the reclamation design, a machine modification was designed to facilitate application of the sealant using ordinary agricultural equipment. A rubber-tired tractor with a tank and spray bar was modified. Additional heat tracing and exhaust rerouting was provided so that the sealant is applied at the appropriate temperature and rate.

Bench-Scale Testing

To test the sealant application, the proposed reclamation process was simulated using a bench-scale model. The bench-scale model included a soil testing box that was mounted at an angle of 30 degrees. The box interior was lined with polyethylene to ensure accurate measurement of infiltration during testing. The lowest seam of the box was left with a small gap to direct any infiltration water to an open-top 5.0-cm PVC collection pipe. The test box was filled with soil to a depth of 30 cm. De-inked, non-gloss office waste paper was hand-shredded and repulped by mixing with an 18% (by weight) NaOH solution for ten minutes in a glass beaker. The pulp was removed, drained and padded with paper towels. In a closed top container, the cellulose pulp was mixed with carbon disulfide in the following ratio: 2.5 g of

cellulose per 1.5 mL of carbon disulfide. (17) The solution was allowed to equilibrate for four hours and then diluted to a 0.4% solution with distilled water. The cellulose xanthate was applied uniformly to dry soil in the test box at a rate of 1.5 L/m2 using a hand-pump spray container. The xanthate solution was allowed to air dry for one hour.

The combination of 98% (by weight) unrefined wax and 2% (by weight) antistripping agent was melted together and mixed in a large beaker over a hot plate/magnetic agitator. The wax/antistripping agent mixture was brushed uniformly onto the soil surface at a rate of 0.25 kg/m2. Subsequent to application of the soil stabilizer and unrefined wax/antistripping agent, the soil test box was exposed to normal site weathering for a period of 24 hours.

After completion of the 24-hour weathering, the test box was exposed to a simulated precipitation event. The precipitation was simulated by applying a known volume of water using a standard shower mechanism for a period of one hour. During the simulated rainfall, run-off water was collected and compared to the known volume of precipitation applied, and a run-off efficiency value was established. The run-off efficiency was found to be less than 10%. At such a poor efficiency, it appears that the sealant had little effect on the water repellency of the soil. It is anticipated that this poor performance may be attributed to two causes: the soil conditions of the Casito series, and the effects of high temperature.

In addition to runoff testing, the sealant was submitted to toxicity testing using the Microtox testing method. Photobacterium Phosporium, a microbial bioassay, was exposed to run-off water and infiltration water from the sealant for periods of five and fifteen minutes. No toxicity effects were observed.

Full-Scale Implementation and Cost Estimates

From the bench-scale testing results, it is suspected that the residual wax/antistripping agent adhered only to the walls of large voids in the Casito soil matrix, leaving a portion of the void uncoated and open for water infiltration. With even a small number of voids open for infiltration, water was channelled into the open pores, thus causing infiltration into the soil. For the full scale application, a larger unrefined wax/antistripping agent application rate is recommended, perhaps 1.0 kg/m2 or more. The higher application rate will overcome the presence of the larger pore spaces.

High temperature tolerance is another important physical property of the sealant. The temperature tolerance of a wax is a function of its congealing/melting point. For the purpose of this problem, the wax must persist in solid form at air temperatures expected in a semi-arid climate. The manufacturer's congealing point value for the proposed sealant is 64 degrees C, therefore it was expected to remain functional for normal site conditions.

However, at temperatures near 27 degrees C, the sealant experienced unusual softness of texture. By nature of their high oil content, unrefined waxes are not brittle, as are refined waxes, at normal ambient temperatures. Instead, they are more semi-solid and very pliable at normal temperatures. This textural softness and pliability led to slow flow of the sealant into the upper layer of soil. With such flow, the sealant did not act as an impermeable film atop the soil. Rather, it melted to fill the interstitial space of the top portion of soil, and compromised the continuity of the sealant cover.

Prior to full-scale application, it is recommended that further testing of the unrefined wax/antistripping agent and soil stabilizer combination sealant be conducted in a semi-arid environment and with soils having large pore spaces. Recommended modifications to the proposed sealant design for future testing include increasing the application rate, and selecting a petroleum wax with a higher, laboratory-verified congealing point. Both of these proposed modifications for further testing will lead to a substantial cost increase for full scale application. In utilizing this sealant system, the proposed 259 km2 application can be completed in a one year and five month time frame. Estimated capital, operating and labor costs for the full site are summarized in Table III. The unit cost is estimated as $146,426/km2.

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16-6

DESIGN, ECONOMICS, AND IMPACT OF SOIL REMEDIATION AND RECLAMATION AT THE WERC CORPORATION SITES

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ABSTRACT

A large contaminated area was discovered on a site owned by the Waste-management Education and Research Consortium (WERC) Corporation in an arid region of southern New Mexico. Contaminants included: 1) Pesticides; 2) Metals; 3) Organic compounds; and 4) Halogenated Compounds. In response to a request by the WERC Corporation, a team of Montana Tech environmental engineers assembled to conduct research and to formulate an innovative, economical, and successful method of cleaning up this contamination. All the requested tasks, and an investigation of subsequent legal, health, and economic issues, were completed and outlined in a proposal.

A two-step approach was developed to effectively treat the combination of organic and metal contaminants that exist in the soil. In the first step, soil is heated to volatilize the organic contaminants. Vapors are captured for off-site treatment. In the second step of the remediation process, soil washing and chemical extraction remove the metals from the soil matrix, transferring them to a liquid solvent. The clean soil can then be reapplied to the site.

Cost is an important consideration as WERC only generates an annual revenue of $7-$8

million dollars. The remediation system processes 4.8 tons per hour at a cost of $222 per ton; cleanup will be completed in 130 days. The total remediation project cost is $2,659,537.

SITE DESCRIPTION/DESCRIPTION OF CONTAMINANTS

The site is located in an arid region of New Mexico; vegetation consists largely of desert shrub and bunch grasses. Catastrophic flooding of a nearby river deposited cobbles and sandy loam in the upper soil strata. Contaminated soil is contained on a five (5) acre site, is limited to the top one (1) foot of soil, and is the result of fallout from a neighboring manufacturing region. Contaminants include: 1) Pesticides, Herbicides, Insecticides; 2) Metals; 3) Organic compounds; and 4) Halogenated Compounds.

It is assumed that the neighboring industrial activity was shut down before November 19, 1980, and thus, contaminants at this site are governed by the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). Because these wastes are deemed to be a threat to human health and the environment, they must be addressed. However, removal and treatment must be conducted such that the safety of workers is assured.

REMEDIATION

Process Design

In order to remove the contamination from the soil, a multi-phase approach was taken. The organic constituents are removed in the first step, followed by metals removal and recovery in the next. The first step consists of an innovative approach to a proven technology: low temperature thermal desorption. The second step utilizes countercurrent soil washing to remove all metals followed by precipitation of those metals from the process water with lime kiln dust. The following flowchart describes the combined process chosen for this design. (See Fig. 1.)

Fig. 1. Process Flowchart.

Step One

The low temperature thermal desorption process has been used by some companies to remove small amounts of volatile or semivolatile organics. (1) This design uses the desorption process to remove the organics, halogenated organics, and also the non-volatile pesticides. The system consists of three main units; the thermal processor, the baghouse, and the condensing system; each of which are described below.

The thermal processor consists of four intermeshed hollow flight augers which are heated internally with hot oil. Each auger measures 30 inches in diameter and 28 feet long and sits in an enclosed heated auger housing. Both ends of the housing have close wall to auger tolerances to prevent air from entering the processor (less than 30 percent of the gas stream). The processor is fed by a conveyor which discharges through a vibratory screen into a surge hopper located above the thermal processor. The hopper continuously feeds the augers which will heat the soil to approximately 450 degrees Fahrenheit, removing the contaminants. The contamination is drawn through an insulated hood in the center portion of the auger housing and routed to the condensing system. The soil is then discharged from the augers into a conditioner where they slide down a ribbed heat exchanger to cool the soil and heat process water for the metals removal process. Some additional water is misted at the end of the conditioner to control dust emissions from the exiting soil. Figure 2 on the following page, shows this portion of the desorption system.

Fig. 2. Thermal Processor.

The coarse fragment which is screened off at the beginning of the process falls to the side where it is stockpiled and later reincorporated into the final clean soil (studies documented by the EPA have shown that in contaminated soils of varying particle size, the coarse particles--those over two millimeters in diameter--typically contain less than one tenth of one percent of the total contaminant). (2) The larger fragment will be tested occasionally to determine the amount of contamination it contains. If the levels are below those required on the WERC Cleanup, the material will be reincorporated into the final clean soil. If the coarse fraction does not meet these levels, it will be remediated in one of three methods:

Washing in a single cell batch reactor where the water can be captured and treated within the condenser or disposed of off-site;

Crushing and reincorporation into the feed material; or

Reintroduction to the hot fine soil stream inside the conditioner near the exit of

the thermal processor to strip the organics; it will then be screened off again and treated separately for metals.
This design will assume that under normal circumstances, treatment of the coarse fraction will not be necessary. Often, the highest concentration of metal and organic contaminants can be found in the fine fraction of any soil. This fact coupled along with the Environmental Protection Agency's (EPA) precedence of allowing minimally contaminated coarse fractions to go untreated is the basis for our assumption. Discussion with EPA project managers reaffirmed this assumption and the presented alternatives.
The contaminant vapors from the thermal processor are then routed via insulated duct through a pulse-jet baghouse to remove particulate material. Based on the expected gas flow rate, the baghouse will measure eight feet wide, nine feet long, and ten feet tall. It will house two compartments each capable of handling the gas flow rate in the event of a breakdown. The fabric filters selected were six feet long Teflon bags; they were chosen for their high temperature and corrosion resistance. The dust will be periodically blown off of the fabric filters using compressed air when the automatic pressure sensor reflects too large of a pressure drop across the baghouse. Depending upon the characteristics of the dust collected on the fabric filters, the material may be reprocessed, incorporated with the cleaned soil, or stored for off-site disposal. The performance aspects of the baghouse are presented in the Technical Assessment section.
The still hot vapors are then directed to the condensing system which consists of two condensers, a carbon adsorption bed, and the fan. A stainless steel air cooled condenser is first used to remove most of the pesticides and some organics. The gas then passes through a refrigerated condenser that will remove the remaining organic contaminants. The temperature within the second condenser is designed to be approximately 230F lower than the inlet temperature. This is a fairly high temperature to be condensing contaminants which will result in short term cost savings in equipment and long term cost savings in power consumption. The condensate is run through a simple oil-water separator and the organic contaminants are barrelled for off-site disposal. The water that is recycled, will be used for dust suppression of the raw untreated soil and for soil conditioning prior to treatment.

A small carbon adsorption bed is placed before the fan outlet to polish the exiting gas stream. Because of the relatively small flowrate, not much bed area would be necessary. An approximate three foot depth of 6 x 16 mesh activated carbon will be used; this should only contribute an additional three inches of pressure drop (see the Technical Assessment).
Step Two
The metals removal process utilizes soil washing and chemical extraction technologies to remove the metals from the soil matrix and dissolve them in the acidic solvent The process transfers the contamination to a liquid medium that is easily treated. The removal efficiency of the process is dependent on the soil: solvent ratio, the solvent characteristics, and contact time.
Metal contaminated soil exits the thermal processor and enters the five-chamber washing system. The soil is agitated in each tank and settled to the bottom where it is collected and sent to the next tank. At the same time, water is moving from tank to tank in the opposite direction. This will yield the cleanest soil at the end of the fifth tank. Metals are then removed from the water through precipitation using lime kiln dust. (See Fig. 3.)
Fig. 3. Counter Current Soil-Washing System.
The tanks will be constructed of stainless steel to avoid the corrosion effects of the acid. Likewise all pumps and tubing will be constructed to withstand the corrosive nature of the acid solvent. The soil will be mixed with .50 M nitric acid for a residence time of five minutes per tank to allow the acid solvent to dissolve the metal contaminants. The rationale behind the counter-current washing system design is to wash the cleanest soil with the cleanest solvent, thus ensuring the optimum contaminant removal from the soil. The soil will be removed from the bottom of the tanks and slurried to the top of each consecutive tank. Pneumatic mixing will allow the soil to mix with the acid as it slowly settles to the bottom of the tank. The clean soil will be removed from the bottom of the last tank of the counter-current soil washing system and slurried to the rotary drum vacuum filter. This drying system will consist of a rotary drum wrapped in filter fabric and

situated in a trough containing the soil slurry. As the drum rotates, it will pick up the slurry. A vacuum located inside the drum will draw the acid through the filter fabric. This recovered acid will be combined with the contaminated acid exiting the counter-current soil washing system and will be pumped to the acid treatment process. The dried soil will be removed from the drum by a stationary scraping knife. From this point, the clean soil will be conveyed to a height of eight feet and allowed to stockpile for several hours. A clean front end loader will be used to remove the clean soil to a storage location. When all of the soil is remediated, a small bulldozer will be utilized to spread the soil over the site.
The contaminated acid will be remediated by the precipitation of the metals as hydroxides. To accomplish this, the pH of the acid will be raised to 5.5 at which point the metals will form hydroxides and fall from the solvent in a solid phase. Kiln dust--a by-product of cement plants--has been chosen as the agent to adjust the solvent pH, allowing the utilization of one waste product (the kiln dust) to remove another waste product (the metals).
The kiln dust will be fed with a front end loader into covered hoppers located on top of the tanks. The precipitation step will take place in a double batch reactor system to allow for the necessary mixing time to sufficiently raise the pH. While one tank is filling with the appropriate amounts of solvent and kiln dust, the solution in the second tank will be mixed with a turbine mixing system and drained. The drained slurry will be transferred to a second rotary drum vacuum filter system which will recover the clean acid solvent and produce a waste sludge of kiln dust and metal hydroxides. This waste sludge will be removed from the site and disposed of at the nearest hazardous waste facility in West Moorland, CA.

TECHNICAL ASSESSMENT

Unit Operations

The thermal processor consists of a four-screw auger and trough with each auger measuring 30 inches in diameter and 28 feet long. Each auger is hollow and has high temperature oil continually flowing through it to heat the soil. Using the results from the bench scale runs in conjunction with HoloFlight heat transfer modeling, the feed rate of the soil was set at 4.8 tons per hour, or approximately 3.2 cubic yards per hour. Soil moisture will be maintained at approximately 20% by volume using recycle water separated in the condensing unit. The retention time of 20 minutes was necessary for the soil to reach the required remediation temperature of 440 degrees Fahrenheit. The bench scale testing indicated that a 20 minute retention time was more than adequate to reduce the contaminants to the required levels; therefore, with further testing of the full scale and better heat transfer, the retention time could possibly be decreased, and the feed rate could possibly be increased.
In the pulse-jet baghouse, air is filtered from the outside of the bags to the inside. A cage in each bag prevents them from collapsing. The bags are cleaned by short (30-100 millisecond) blasts of high pressure air from a compressed air manifold at the top of the baghouse. The shock wave from the compressed air blast shakes the dust off of the bags. Each bag is pulsed every few minutes. The advantage of this type of baghouse is that it does not have to be shut down for the bags to be cleaned; air-stream treatment can continue while the bags are being cleaned. The design utilizes high temperature Teflon bags and a maximum filtering velocity of ten feet per minute. With the expected dust loadings, the pressure drop is predicted to be less than ten inches of water using Leith and First's pressure drop model. (3) Particle removal efficiency for this type of pulse-jet baghouse is typically 99.9% or better.
The condenser consists of a stainless steel air cooled condenser followed by a low temperature refrigerated condenser. The use of the air condenser should reduce the amount of power consumed by the system while removing a large portion of the contaminants. The refrigerated condenser was designed with a U-value of 2.0 and should reach temperatures around 230F below the inlet temperature. Stainless steel refrigerated coils will be used to prevent corrosion. The condenser system (both condensers) will occupy approximately a ten feet by ten feet square area. A simple oil water separator will be used to reduce the volume of the condensate waste stream. The treated water is then reused on site for soil conditioning while the organic contaminants are barrelled for off-site disposal.

Prototype

A prototype thermal processor was developed to mimic the operating conditions that will be present in the full-scale unit. This was done by creating a small batch

reactor that heats and agitates the soil in a similar manner to that of the thermal processor. The contaminated air from the heated reactor is then drawn through a condensing unit and carbon adsorber before entering the atmosphere. (See Fig. 4.)
Fig. 4. Bench Scale Prototype.
The prototype is constructed of an eight inch diameter aluminum cylinder, 18 inches long, 1/8 inch thick. Aluminum was chosen because of its strength, low weight, and ability to with stand high temperatures. One end is sealed with a 1/2 inch thick aluminum plate that is connected through a coupling to a stationary electric motor by a 1/4 inch steel shaft.
The other end of the cylinder is sealed with a removable 1/2 inch thick aluminum plate. The air-tight seal for this cap consists of high temperature silicone o-rings. The center of this end plate is drilled and threaded to accept a 1/2 inch steel pipe that will draw off the contaminated vapors. On the inside of the cap, this pipe will have a small filter placed over it to prevent dust from entering the gas stream. A quick-couple slip fitting will be placed on the protruding steel pipe that will allow the cylinder to turn freely while still permitting air-tight gas flow. Following treatment, this end may be quickly removed, the motor can be uncoupled, and the cylinder may be lifted out and emptied. Heat is provided externally to the cylinder by a propane heating element fixed to the frame.
The contaminated vapors are drawn through the filter, the steel pipe, and into a short section of insulated copper tubing which leads to a condensation/adsorption unit. To simulate the condensers on the full scale design, three ice-bath condensers are placed in series to cool the vapors and condense the organics. Each condenser will contain cooled water to provide as much cooling surface area as possible. As a polishing step, the gas stream enters an activated carbon packed impinger. Finally, the gas passes through an adjustable vacuum pump and enters the atmosphere.
EXPERIMENTAL RESULTS
Gas chromatography/mass spectrometry (GC/MS) in conjunction with Soxhlet Extraction (EPA methods 3540 and 8240) was used to assess the technical feasibility of the prototype unit. The mass spectrometer was set up for ion selective detection to increase sensitivity. Three known concentrations of each contaminant were used to generate the calibration curves used for the quantitative and qualitative assessment. Unfortunately, due to improper extraction techniques, data for ethyl benzene was not obtained. However, it is the design team's consensus that the most difficult contaminants to remediate should be the non-volatile pesticides. It can also be noted that tetrachloro-ethylene and chloroethylene were not detected in any of the samples analyzed. It is assumed that the very small amount that was added to the soil had evaporated due to the volatility of these compounds.
Two different experiments were conducted in order to determine the optimum temperature and retention times for the thermal processor. Preliminary experiments were carried out using a temperature controlled oven with a gas exhaust port. Final experimentation was conducted on the bench scale prototype using a propane fuel source and revolving at approximately 10 to 15 revolutions per minute.
Three experimental runs were conducted to find the lowest (i.e. most cost efficient) temperature to run the thermal processor at while still getting an acceptable amount of degradation. The first half of Table I below shows the results of these runs. Even at the lowest temperature attempted, all of the contaminants are removed. To be safe, the temperature for the thermal processor will be set at 230C. This temperature may be adjusted based on further research; however, it is a likely possibility that the temperature can be reduced or the residence time decreased resulting in greater cost savings in the long run.
To determine the soil feed rate, the residence time for the soil would be necessary. A second experiment was conducted in which the residence time was varied while the temperature remained constant at 230C. The second half of Table I above shows that there is a notably smaller reduction of pesticide concentration in the 15 minute sample when compared to the 30 minute sample. When the 30 minute sample is compared to the 60 minute sample there is a small reduction in some concentrations, but not enough to warrant doubling the retention time. Even in the 15 minute sample, all contaminants met the required cleanup guidelines. Therefore, to be safe, the retention time can be set near 20 minutes.
Through assessment of the experimentation, the lowest--most cost effective--temperature was set at 230C and the optimum retention time was determined to be 20 minutes. When both of these conditions are met, the soil will be remediated

to the required levels. However, statistically valid experiments varying both retention time and temperature should be conducted in the future to account for any interaction between the two variables.
For the soil washing process, three parameters required investigation to determine the optimum system: the soil:solvent ratio, mixing time required, and solvent characteristics. A soil:solvent ratio was estimated based upon the amount of water needed to dissolve the amount of copper sulfate that would exist in one kilogram of soil. The ratio of 30 grams of soil to 150 ml of solvent was used for all subsequent experimentation.
The success of the metals extraction was studied by measuring the metal concentration of the washing solvent after washing with an atomic absorption (AA) analyzer for each of the three metal contaminants. A laboratory test was set up to determine the optimal mixing time and acid concentration using nitric acid as the washing solvent. Contaminated soil was washed with three different solvent concentrations: .01 M, .10 M, and 1.0 M HNO_3. The washing periods included: five, ten, and fifteen minutes.
The optimal mixing time in all cases was five minutes, and .01M acid was the optimal washing solvent for removing the silver and cadmium. Based on percent reduction, five washes would be necessary to remove the silver and cadmium. Also, .01M acid was not strong enough to remove copper in five washes; therefore, the optimum soil washing system would include five washing steps with .5 M nitric acid.
Table II depicts the removal efficiencies resulting from the experiment. The data show that copper would not be removed to the necessary 15 ppm with the .10 M HNO_3 solvent. The .50 M HNO_3 reduces the copper content to acceptable levels in five washings.
The metal hydroxide precipitation test was based on the following assumption: when adding kiln dust to the contaminated solvent, the pH of the solution would remain constant during the period in which the metals were precipitating. Upon precipitation of all metals, the pH would again begin to rise. For the experiment, three 450 ml samples of .10 M HNO_3 were directly contaminated with the amount of metals that would exist in 90 grams of soil. Kiln dust was added in increments of one gram. The pH was recorded when it stabilized and more kiln dust was added. It was determined that 5.25 grams of kiln dust to 450 milliliters of acid was the optimum ratio for metals removal from the solvent.

OPERATING PLAN

The plant must run 24 hours a day if all material is to be processed in one summer. Also, this will minimize the amount of heat and power lost due to shutdown cooling and startup reheating of the auger oil. Three shifts of four to six people will be necessary to run the operation. A safety officer will be on site at all times to enforce safe work techniques and conduct all sampling procedures. One or two bulldozers will push the soil from the five acre site to the centralized plant. From here, one man operating a small loader can load contaminated soil onto the feed conveyor. When the surge bin is full, he can move the accumulating coarse fraction with the same loader to the nearby stockpile location. On the exiting end of the soil washing process, another operator can use a different non-contaminated loader to move the accumulating clean soil into a stockpile for its future reapplication to the site. Upon full drying of the accumulated material, a bulldozer will be used to push the soil over the site. One of the employees will serve as shift supervisor. At a soil feed rate of 4.8 tons per hour, the full 12,000 tons of soil will be remediated in 2500 hours or about 105 days. Assuming approximately 80 percent utilization to account for mechanical failures and upkeep, the job time is estimated at 130 days or 3125 hours.

ECONOMIC ASSESSMENT

Table V illustrates the total cost for remediation equipment, labor, and operation and maintenance. Hours and rates along with operation and maintenance systems for this project have been estimated based on the time invested and current competitive wage estimates. Capital equipment is estimated to cost $400,500; Pre-operations labor is estimated at $170,025; and Operations and Maintenance over the project life is estimated at $2,089,012. The total Remediation Project Costs are $2,659,537 or $221.63 per ton (assuming approximately 12,000 tons of soil).

CONCLUSION

A unique remediation and reclamation plan was designed to remediate contaminated soil at the WERC site. Montana Tech environmental engineers have designed a

multistep system that includes a thermal desorption unit followed by a counter-current soil washing unit. This system can efficiently reduce contamination in the soil to acceptable levels while minimizing cost and waste streams. The remediation system first vaporizes the contaminants from the soil using heated augers and then condenses them out of the gas stream for off-site disposal. The metals are then washed out of the soil using an acidic solution. Contaminated soil is processed at 4.8 tons per hour and the WERC site can be completely treated within a four month period by a small work crew. The total cost for soil remediation is $2,659,537, or $222 per ton; however, it should be noted that the majority of this cost is due to the disposal of pure contaminant waste streams. There is, however, a high probability that the process would have a commercial value to the WERC Corporation. After the initial project costs and may even provide additional revenue in the future.

Health and safety issues are being addressed through the creation of a site safety and health plan which addresses worker safety, personal protective equipment, and emergency procedures. Legal issues regarding site remediation have been addressed including the regulations associated with any releases of wastes which will occur at the remediation plant.

Our design creatively address all of the WERC Corporation Management's concerns regarding the effectiveness, implementability, and cost of our Low Temperature Thermal Desorption/Soil Washing Project. Not only will the surrounding community benefit from a cleaner environment, but the WERC Corporation will also benefit from a safe, effective remediation and reclamation process that shall provide future economic benefits.

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Session 17 -- Poster - HLW/TRU/Transportation
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17-1

A DYNAMIC SIMULATION MODEL OF THE SAVANNAH RIVER SITE HIGH LEVEL WASTE COMPLEX

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ABSTRACT

A detailed, dynamic simulation model of the entire high level radioactive waste complex at the Savannah River Site has been developed using SPEEDUP(tm) software. The model represents mass transfer, evaporation, precipitation, sludge washing, effluent treatment, and vitrification unit operation processes through the solution of 7800 coupled differential and algebraic equations. Twenty-seven discrete chemical constituents are tracked through the unit operations. The simultaneous simulation of concurrent batch and continuous processes is achieved by several novel, customized SPEEDUP(tm) algorithms. Due to the model's computational burden, a high-end workstation is required: simulation of a year's operation of the complex requires approximately three CPU hours on an IBM RS/6000 Model 590 processor. The model will be used to develop optimal high level waste (HLW) processing strategies

over a thirty year time horizon. It will be employed to better understand the dynamic inter-relationships between different HLW unit operations, and to suggest strategies that will maximize available working tank space during the early years of operation and minimize overall waste processing cost over the long-term history of the complex. Model validation runs are currently underway with comparisons against actual plant operating data providing an excellent match.

INTRODUCTION

Process Overview

The mission of the Savannah River Site (SRS) High Level Waste (HLW) System is to receive and store radioactive high level wastes in a safe and environmentally sound manner, and to convert these wastes into forms suitable for final disposal. The planned forms are: borosilicate glass to be sent to a Federal Repository; saltstone grout to be disposed of on-site; treated wastewater to be released to the environment; and benzene for destruction by incineration. Also, the tanks used for storage of the wastes must be left in a state such that they can be decommissioned and closed in a cost-effective manner under the applicable Federal Facility Compliance Agreement.

The operations (Ref. 1) required to fulfill the above mission are illustrated in Fig. 1, with the processes indicated in rectangles and the coupling connections between the processes shown as numbered streams. Incoming high level wastes (stream 1), consisting primarily of insoluble metal oxides and hydroxides (sludge) suspended in aqueous solutions of sodium salts (supernate), are received into HLW Storage and Evaporation (Tank Farm). The majority of the waste is from nonradioactive process chemicals. Fission and activation products, and actinide components, are mainly present in the sludge phase, except for cesium which has a high solubility and remains in the supernate. The purpose of the Tank Farm is to safely store these wastes until downstream processes are available for further processing. Most of the supernate is evaporated to solid saltcake to reduce its volume and mobility. Incoming decontaminated overheads from the separations evaporators are sent to the Effluent Treatment Facility (ETF) (stream 13).

Insoluble sludges that settle to the bottom of the waste receipt tanks in the Tank Farm are transferred (stream 2) to Extended Sludge Processing (ESP) via hydraulic slurrying techniques. In ESP, sludges high in aluminum are processed to remove previously insoluble aluminum compounds. All sludges are washed with water to reduce their soluble salt content. The spent washwater from this process is sent back to the Tank Farm (stream 3), while the washed sludge is sent to Vitrification (stream 4).

Saltcake (primarily crystallized sodium, potassium, and cesium salts) in the Tank Farm is redissolved using hydraulic slurrying techniques and sent to In-Tank Precipitation (stream 5), where the salt solution is processed to remove radionuclides, which are concentrated into an organic precipitate. The precipitate is filtered, washed with water to remove soluble salts, and sent to Vitrification (stream 7). The decontaminated filtrate is sent to Solidification (stream 6).

In Vitrification (Defense Waste Processing Facility - DWPF), the precipitate (stream 7) is catalytically decomposed and separated into two streams: a mildly contaminated organic stream which is sent to storage and eventual destruction in the Consolidated Incineration Facility (stream 11); and an aqueous stream containing virtually all of the radionuclides which is combined with the washed sludge from ESP (stream 4) and sent to the glass melter. To prepare feed to the melter, the washed sludge from ESP has been chemically adjusted, stripping out a significant amount of mercury which is purified and sent to mercury receivers (stream 12). The precipitate and sludge are mixed with glass frit and sent to glass melting. The glass melter drives off the water and melts the wastes into a borosilicate glass matrix which is poured into a stainless steel canister. The canistered wasteform (stream 9) is sent to on-site interim storage, and will eventually be disposed of in a Federal Repository.

The water vapor driven off from the melter along with other aqueous streams generated throughout Vitrification are sent to the Tank Farm for evaporation, storage, and eventual further processing (stream 10).

Overheads (i.e., the fraction which is evaporated off) from the Tank Farm evaporators are combined with overheads from separations processes and other low-level streams and sent to the ETF (stream 13). In the ETF, these low-level wastes undergo further decontamination. The treated effluent becomes outfall which

flows to the site environment (stream 14), while the contaminants removed in the treatment process are concentrated and sent back to ITP (stream 15) where they are mixed with the decontaminated salt solution from ITP. This mixture of low-level waste is then sent to Saltstone (stream 6) where it is combined with cement formers and pumped as a wet grout to a vault (stream 16). In the vault, the grout hydrates and cures, forming a grout monolith that will eventually be closed as a landfill.

Model Overview

The Integrated Flowsheet Model (Ref. 2), the subject of this paper (hereafter referred to as the "IFM"), was designed to capture the processes described in the previous section via the use of the SPEEDUP dynamic simulation package (Ref. 3). The first phase of the IFM focuses on predicting the available Tank Farm space over the next five to ten years. Thus the current emphasis is on performing accurate dynamic volume balances for 30 of the 51 tanks, with simplified models of many of the facilities outside the Tank Farm. Succeeding phases will add detail to the models as the need for such is identified.

SPEEDUP is a comprehensive plant modeling package which offers steady-state and dynamic simulation capabilities in a flowsheeting environment. The user need only create sets of differential-algebraic equations to describe a particular process (e.g., conservation and rate equations) and then define how the various processes are coupled together (e.g., "output 2 of tank_48 is input 1 of tank_49"). The equation syntax is an intuitive one illustrated in Eqs. 1-3. The differential equations are primarily statements of the time-derivative of mass for various constituents. The algebraic equations are state equations (e.g., density as a function of composition), logical IF-THEN-ELSE constructs determining process control, and intermediate expressions which are used to compute elements of the differential equations (e.g., reaction rates).

SPEEDUP carries out a symbolic translation of the equations created by the user and automatically generates a FORTRAN solution algorithm for the dynamic problem. Automatic timestep selection is used during the integration of the equations. Discontinuities (e.g., a transfer flow turning on or off) are automatically detected, and at that precise time the solution re-initializes itself to continue the time integration on the other side of the discontinuity. Thus all discrete events are accurately captured. The user of the IFM need only define the applicable process equations and instruct SPEEDUP to solve the equation set. Process changes are almost trivially implemented in the IFM: the user need only modify the basic equations, and SPEEDUP automatically regenerates the necessary solution algorithms.

Each of the "boxes" in Figure 1 is broken down in the IFM into a series of unit operations, with many additional streams coupling those unit operations internal to the "box" . Generally, a single unit operation is associated with an identifiable part of the HLW complex (e.g., a particular tank, evaporator, etc.) or a process control operation (e.g., a diversion box that switches transfers from one tank to another). Currently, approximately 7800 differential-algebraic equations are used to represent 72 distinct unit operations coupled by 172 streams, each carrying 27 chemical compounds. The computation burden is variable, dependent on the number of discontinuities encountered during the simulation and on fundamental process time scales. Typically, a one year simulation of the HLW complex requires 2-3 CPU hours on a IBM RS/6000 Model 590 workstation. About 10000 problem variables are viewable from the analysis results, with an intuitive point-and-click plotter being available within the SPEEDUP environment.

We found that the SPEEDUP environment (Refs. 4,5) lent itself naturally to the rapid and efficient development of this rather complex dynamic model. The discipline imposed by the SPEEDUP modeling language allowed a multi-disciplinary team gathered from several SRS divisions to work effectively in a concurrent fashion. Once the fundamental model interfaces were agreed to by the team, the individual modelers were able to proceed independently in developing and verifying their assigned unit operation models. As individual models were completed, they were smoothly integrated into the growing overall model. The entire modeling process (design, implementation, integration, final checkout, and documentation) required less than one year with the equivalent of five full-time modelers.

The IFM will be used by process engineers to simulate in a dynamic fashion the response of various parts of the HLW complex to varying operational strategies and to study the integrated effects of one part of the complex upon another. From another perspective, the IFM will be used by production planners to study the

dynamic effects of various feed strategies, optimal batching sequences, the impact of varying the start-up of various parts of the complex, etc., all from the viewpoint of best utilizing the available resources over a projected thirty year future for the site's high level waste complex.

UNIT OPERATION MODELS

The major elements of the Integrated Flowsheet Model (IFM) will be described in this section, which is organized to be consistent with the logical and material flows shown in Figure 1. Currently, the IFM consists of 32 models of 72 unit operations (a unit operation being, for example, a single sludge washing tank). There are fewer models than unit operations due to the replication feature in SPEEDUP: more than one unit operation may be described by the same model as long as the form of the process equations are identical for the operations (while the parameters in the equations may differ in the individual unit operations). There is insufficient room to describe such a number of models in detail (see Ref. 2), so we restrict ourselves to the general types of processes modeled. In addition, we will discuss the techniques which allowed us to employ a fundamentally continuous process modeling tool to treat cyclic, batch processes. The chemical compounds tracked in the IFM are: H2O, NaNO3, NaNO2, NaOH, KNO3, CsNO3, NH4NO3, NH4OH, NaAlO2, Na2U2O7, Fe(OH)3, Al(OH)3, Sr(OH)2, C6H6, C6H6O, CsOH, KOH, CsB(C6H5)4, NaB(C6H5)4, Na2(C6H5)BO2, KB(C6H5)4, NH4B(C6H5)4, NaTi2O5H, NaTi2O5.Sr(OH)2, NaTi2O5.Na2U2O7, miscellaneous soluble salt, and miscellaneous insoluble sludge.

Waste Generators

SRS liquid waste, as received in the waste tanks, is made up of many waste streams generated in the separations process during the recovery and purification of transuranic products and unburned fissile material from spent reactor fuel elements. To prevent corrosion of the carbon steel waste tanks, these wastes are neutralized to excess alkalinity (pH > 12) before transfer to the Tank Farm underground storage tanks. The two incoming streams shown in Figure 1 are subdivided into five streams in the IFM. The separation process details are considered to be external to the IFM and are thus represented as external boundary conditions to be imposed upon the model: the user defines time-dependent flows and compositions for the five input streams into the IFM.

Storage and Evaporation

SRS operates a Tank Farm consisting of 51 waste tanks (ranging in volumetric capacity from 2700 m3 to 4900 m3; i.e., up to 1.3 million gallons) and three evaporators. Radioactive waste, as received in the Tank Farm, can be reduced to about 25% of its original volume and immobilized as crystallized salt by successive evaporation of the liquid supernate. Such dewatering operations have been carried on routinely at SRS since 1960 and have allowed the site to avoid the need to build an additional 70 storage tanks.

In the IFM, each tank is individually represented through basic equations representing a simple, lumped parameter, perfect mixing mass balance:

```
$tank_mass = Sin (stream_in_density * stream_in_flow) -
                                    tank_density * Sout (stream_out_flow)
                (1)

$tank_mass * tank_massfrac(1:nocomp) +  tank_mass * $tank_massfrac(1:nocomp) =
              Sin (stream_in_density * stream_in_flow * massfrac_in(1:nocomp)) -
              tank_density * tank_massfrac(1:nocomp) * Sout (stream_out_flow)   (2)

tank_density = function(tank_massfrac)  (3)
```

where the above is written as an example of the actual SPEEDUP syntax with "$" denoting the time derivative and "a:b" a vector operation over the elements "a" to "b", with "nocomp" being the number of chemical compounds being tracked. The summations are over the number of streams into and out of the tank, each of which would be written out explicitly and named uniquely in the SPEEDUP implementation. The variables are typically given names that make their significance obvious. The density function is based on experimentally measured data for the SRS Tank Farm compositions: it is written as a simple algebraic form in the equation set. The volumetric flow rates are determined by the batch control logic described in a later sub-section. Eqs. 1-3 represent nocomp+2 simultaneous differential-algebraic

equations. In SPEEDUP, the unknowns may be on either or both sides of the equation: the translator is able to parse the equation and identify which variables are to be solved simply from the structure of the equations and the given input.
For a salt tank, the above set of equations would be written once for the supernate region and once for the saltcake region, with additional coupling terms in the equations to reflect the dissolution or crystallization processes. In order to minimize the computational burden, the vector equations for each region only span the compounds normally present in that region. Currently, the simple salt tank process model compares the tank supernate density against a reference specific gravity. If the density exceeds the set point, salt precipitates at a given rate, thereby removing mass from the supernate balance and adding it to the saltcake balance via the new coupling terms. If the opposite is true, the mass exchange proceeds in the other direction until the reference specific gravity is attained in the supernate.
Similarly, a parametric pseudo-steady state model is employed for the evaporator. The evaporator overheads are given as a function of the specific gravity of the evaporator feed tank. The evaporator itself is defined to produce concentrate at a prescribed specific gravity, with the concentrate delivery rate equal to the evaporator feed rate minus the overheads. The remaining degrees of freedom in the formulation then uniquely specify the feed and delivery rates, and the composition of the concentrate delivered to the salt tank.
Extended Sludge Processing
Neutralizing the waste from the separations processes produces a mixture of insoluble sludges, primarily iron, manganese, aluminum, and uranium hydroxides in a salt solution. Although the sludge has settled and compacted in the various waste tanks for 5 to 40 years, that settled volume contains a significant fraction of interstitial liquid bearing dissolved sodium salts. If this salt were not removed, the vitrification process would require a larger frit-to-waste ratio (to control the rheology of the molten glass), which would greatly increase the number of glass canisters to be produced and stored, and hence the cost of the overall process. These soluble salts are removed from the interstitial liquid by repeated dilution in processing tanks in the Extended Sludge Processing (ESP) facility. In addition, some of the sludge inventory contains large amounts of aluminum (with a negative impact on the product glass rheology) which is dissolved by heating the sludge in a high caustic concentration. The supernate containing the dissolved aluminum is then decanted before the start of the salt washing process.
A variation of the standard IFM tank model described in the previous section is used to model the sludge washing process. If needed, the aluminum dissolution is treated via additional mass transfer terms in the supernate and sludge mass balance equations (much like the salt precipitation and dissolution is handled in the salt tank model of the previous section). The batch process control algorithms set the aluminum dissolution mass transfer terms nonzero during the appropriate time interval, and then back to zero when completion of the process is detected.
The volumetric dilution which reduces the salt content during ESP washing is already handled by the basic tank mass balance algorithms. But the process control becomes more complex. After the wash water is added to the tank, the sludge solids are suspended and agitated for a period of time using slurry pumps, after which the slurry is undisturbed for another interval of time to let the sludge settle. The subsequent decant is designed to remove supernate down to a level just above the settled sludge interface (whose position and settling rate change with time). A FORTRAN procedure which encodes an empirical, four-region sludge settling model is linked into the SPEEDUP tank model in order to compute the time dependent sludge settling level. That level is then passed to the batch process control algorithms to control the turning on and off of the tank flow which represents the decant. The dynamic model in IFM couples all these varying time-dependent processes into a consistent tank response.
During the dynamic simulation, the IFM monitors the sodium content in the sludge washing tank and will automatically terminate the washing process when the desired end-point is reached and go on to the next phase of operation. In addition, during the washing process, the IFM will add corrosion inhibitors at the appropriate time and in the correct amount to maintain the tank within the specified corrosion limits. Both operations mimic the actual events that would occur in the real facility and are controlled by the batch process algorithms.

The sludge washing process can move large amounts of wash water as decant throughout the HLW complex: a typical wash cycle can utilize 1300 m3 (one-third of a million gallons) of wash water, with three or more washes to process each batch of sludge. A co-washing strategy is usually adopted which utilizes two tanks in tandem, with the wash water from one cycle in one tank being used for the next wash cycle in the other tank. Any decant can be diverted to a number of different tanks. All this flexibility is also provided in the simulation with the IFM. With the tank models having high and low level set points built into them, the IFM analyst can make a number of different strategy choices in managing tank volumes and clearly see the outcome as success or grid-lock in the Tank Farm. The IFM captures all the significant system dynamics of the actual plant.

Salt Processing

The saltcake in the Storage & Evaporation portion of the HLW complex is redissolved for pumping to the In-Tank Precipitation (ITP) process where the solution is decontaminated in a batch process. The soluble radioactive metal ions (soluble cesium and the soluble fraction of strontium, uranium, and plutonium in part per million levels) are precipitated with sodium tetraphenylborate or adsorbed on sodium titanate to form insoluble solids. The resulting precipitate, which contains most of the radionuclides, is filtered to concentrate the solids and then sent to the vitrification process. The remaining decontaminated salt solution is transferred to Solidification following benzene stripping.

This part of the IFM represents yet another variation on the fundamental mass balance algorithms in which new terms appear in the appropriate tank model equations in order to represent the dynamics of precipitation chemistry, dissolution of solids, solubility of tetraphenylborate compounds, adsorption onto sodium titanate, benzene formation via radiolytic decomposition, and benzene evaporation. Benzene stripping is represented with a steady-state model: the user specifies the benzene concentration in the effluent flow of the stripping column, and the model performs overall and component mass balances to obtain the flows and compositions of the top and bottom streams leaving the column.

Detailed descriptions of this particular model may be found in the References (2,4) and will not be repeated here. The main impacts on the overall IFM are the stiff, non-linear behavior now imposed on the equation set due to the reaction rate stoichiometries, and the additional discontinuities introduced in the solution by having another batch process cycling through its discrete steps (i.e., in addition to the ESP batching). Both new elements have the effect of significantly increasing the computational burden, but are crucial to an accurate representation of the overall process.

Vitrification

The Defense Waste Processing Facility (DWPF) consists of several facilities: Late Wash, the Vitrification process, and Saltstone. Late Wash receives washed precipitate from ITP and reduces the nitrite concentration (used for corrosion inhibition) by filtration and dilution. During the process, the slurry is reprecipitated to capture cesium (which may have returned to solution in ITP) and then reconcentrated. The filtrate produced during the filtering process is stripped of benzene, chemically adjusted, and transferred back to ITP for reuse.

In the IFM, the Late Wash precipitation reactions are assumed to be instantaneous and complete. Just as in the ITP model, the Late Wash model adds 150% of the sodium tetraphenylborate required to stoichiometrically react the incoming nitrate salts. The precipitate is assumed to be continuously washed to reduce the nitrite concentrate to the prescribed value. Simple mass balance equations are used to model the processes.

In the Vitrification process, the liquid HLW which is processed in ITP/Late Wash and ESP is immobilized as a glass solid. The operations include chemically treating the two quite different waste streams, mixing them with ground borosilicate glass (frit), heating the mixture in a melter, and pouring the molten mixture into 3 m tall by 0.6 m diameter stainless steel canisters to harden.

In the IFM, the Vitrification process is approximated as a simplified, continuous (rather than a batch) process for the purpose of translating the outputs from Late Wash and ESP into waste product streams: glass, organic waste, and recycle to the Tank Farm. The model has been shown to give the same material distributions in the output flow streams as those in the reference steady-state design analysis when the model input matches the design input stream compositions.

Vitrification proceeds through three processing cells. In the Salt Processing Cell (SPC), the precipitate slurry is acid hydrolized and steam stripped in order to remove the aromatic organics from the melter feed. Formic acid used in the SPC is absent in the IFM, thus the reaction representation is necessarily simplified: the formulation is designed to reproduce the main output of interest at this stage, the organics. At this point the hydrolized precipitate is transferred to the Chemical Process Cell where washed sludge enters from ESP. Nitric acid is added to react components and provide the required reduction-oxidation balance. Frit is added and the final mixture is sent to the Melt Cell (which is not explicitly modeled in the IFM). In the IFM, ten simplified oxidation equations are used to apportion a fraction of the components to the recycle stream and the oxides that all go to the glass. All reactions are assumed to be instantaneous and complete. Distribution parameters (e.g., fraction of sludge components transferred to the recycle stream) are user-defined constants. While mass is preserved in the model, the densities of the oxides are not known, so the density of the glass is not calculated. Glass canister production in IFM is based on a volumetric flow rate that results from a volume balance. The mass of the glass is based on a nominal density taken from the reference design calculation. The IFM controls the vitrification process by adjusting the sludge flow to meet the sludge weight fraction criterion, by adjusting the precipitate flow to meet the prescribed precipitate mass fraction in the glass, and by adjusting the frit flow to make up the balance of the nominal glass production rate. Mercury production is not treated in the current version of the IFM. Neither is the Consolidated Incineration Facility currently represented in the IFM.

The Vitrification part of the IFM computes two of the four major outputs of the HLW complex: glass canisters and recovered organics.

Solidification

In the Saltstone Production Facility, aqueous salt solution is received from ITP and then combined with a blend of cement, fly ash, and blast furnace slag to generate nonhazardous, low level waste, saltstone grout.

The grout is pumped to a covered cell of an above-grade, concrete vault where it solidifies. In the IFM, the saltstone model computes the salt solution feed from ITP based on the nominal cement premix flow rate, and calculates the volume of dilution water required to bring the feed into compliance with limits set on strontium, sodium, and hydroxides. From the resulting mass balance and the given grout density, the model will compute the volumetric production of grout, which is then translated into the number of vaults filled.

The Solidification part of the IFM computes the third of the four major outputs of the HLW complex: wet grout.

Wastewater Treatment

The Effluent Treatment Facility (ETF) decontaminates the influent wastewater through a series of steps consisting of pH adjustment, sub-micron filtration, heavy metal and organic adsorption, reverse osmosis, and ion exchange. The treatment steps concentrate the contaminants into a smaller volume of secondary waste which is further concentrated by evaporation. The clean, treated effluent is discharged to the on-site environment, while the contaminated evaporator bottoms are pumped back to ITP for eventual disposal in Saltstone. In the IFM, the ETF model takes as inputs the time-dependent flows from the five evaporator overheads tracked in the IFM. Perfect mixing of the input streams is assumed. The user defines a split fraction for the volume going to the outfall, which is taken as pure water. The mass balance relation then computes the flow rate and composition of the concentrate and returns it to ITP.

This Wastewater Treatment part of the IFM computes the final of the four major HLW complex outputs: treated effluent.

Batch Process Control

SPEEDUP was initially developed as a continuous process simulation tool. In its language constructs, no integer arithmetic is allowed. There is no way to easily store "previous timestep" values of variables, and it is difficult to recognize when a variable has changed for the first time in order to initiate some new action. Thus it is somewhat more awkward to use SPEEDUP to simulate cyclic, batch processes than for continuous ones. Fortunately, SPEEDUP has a facility for linking customized FORTRAN into the continuous model via its External Data Interface (EDI), and we have done all our batch process control in the IFM via that facility (Ref.

6). Since process control is a key part of the IFM, it will be described in some detail.

EDI consists of a set of ordered, dummy subroutine calls that are always available in a simulation. All the user has to do is recognize the call order and fill in the dummy subroutines with customized FORTRAN which carries out the desired set of operations. This is where we have located the IFM batch control. Each batch process is set up as a "recipe" consisting of several steps. For example, the ESP washing process consists of 14 steps during which aluminum dissolution, decants, inhibitor additions, wash additions, agitation, settling, and time delays occur. We have used the EDI subroutines to encode these batch recipe steps, primarily via manipulations of integer and real arrays to identify when a process is on or off. Within the continuous SPEEDUP model, we have set up unit operations which represent junction boxes coupling M tank outputs to N tank inputs. These input/output streams in the couplers contain "logical valves" (i.e., scalers between 0.0 and 1.0) which are turned off and on by the EDI interface. These scalers are used to compute the current values of stream_out_flow and stream_in_flow in Eqs. 1-2. Thus SPEEDUP calls the EDI FORTRAN when a batch operation initiates; the FORTRAN changes the appropriate scaler to begin the batch operation and computes the time at which the batch operation will be complete (so that SPEEDUP knows when to call EDI again for the next step in the recipe); control returns to SPEEDUP so that the continuous model can respond to the batch operation changes via a change in a coupler state; and the simulation proceeds onward in the operation.

The EDI FORTRAN is broken down into six modules. One controls the ESP batch process and another the ITP batch process. A third controls the sludge feed from ESP to DWPF. Two more are used to dynamically alter model parameters such as tank operating limits, evaporator utility, or feed stream flow and composition from waste generation. This module is also used to simulate transfer line availability so that even though a batch recipe step may call for an operation, the user can define that transfer line to be down for a particular time interval and thus delay the operation. The sixth module is used to carry out manual transfers which the user may choose to define in addition to the normal automated processes: e.g., the user can choose to let the normal model control the evaporator system recycle operations, or alternatively use this sixth module to prescribe all the evaporator control via detailed input. The modular structure allows the coding to execute only those code blocks that are appropriate to the processes which are currently underway.

As a result of our implementing batch process control through EDI, the IFM has great flexibility in modeling both continuous processes (e.g., evaporator operations) and discrete batch processes (e.g., sludge washing) simultaneously. The entire waste complex model consists of many different sorts of processes with varying natural time constants and sequence patterns. In the IFM, each process can be operating simultaneously and independently, and still couple to other processes when the time comes for an interaction between the processes. The power of such generality is required in order to run the full spectrum of scenarios necessary to simulate thirty years of operation of the entire HLW complex.

Model Verification and Validation

The IFM is undergoing a formal verification and validation process. In the verification phase (which has been completed), each individual model was subjected to an independent review (by a modeler not involved in its creation) in order to verify that the correct equations were properly encoded, and that the model execution satisfied the formal definitions in the detailed Software Requirements Specification document (Ref. 7). The verified individual models were then tested en masse in an integrated form to verify that the Software Requirements Specification continued to be satisfied in the whole.

The second part in determining the pedigree of the IFM, validation, is currently underway. This consists of comparing the simulation results against actual plant operating data. The results of one such comparison are presented in Fig. 2, where a 130-day operating history of one of the evaporator systems is shown. The darkened points are the actual plant data, while the open points are the results calculated by the IFM. The lower curve shows the tank volume changes in the evaporator feed tank which is coupled to this particular evaporator, while the upper curve shows the tank volume changes in the concentrate receipt tank to which that evaporator delivers material.

The feed tank volume decreases as supernate is fed to the evaporator, and increases

as supernate is recycled from the concentrate tank or fed in from other tanks. The concentrate receipt tank volume increases as it receives output from the evaporator, and decreases as it feeds recycle to the evaporator feed tank. In addition, within the concentrate tank, mass transfer occurs between the supernate and saltcake regions as the salt precipitates out upon cooling of the concentrate. The saltcake volume effects the times at which recycle occurs since as the cake builds up, less supernate volume is available for recycle. Thus precipitation effects are implicitly shown in the total volume results in Figure 2 via the transfer times. Clearly, the agreement between the plant data and the simulation model is very good. The main discrepancies are in the slopes of some sections of the lower curves which indicates that in some modes of operation, the idealized volume transfer rate in the IFM does not precisely duplicate the actual transfer rate in the plant.

Similar validation is planned for all the other parts of the IFM where plant data is available. Where the model is to simulate those parts of the HLW complex which are not yet operational, the comparisons will of necessity have to be against the best available design data, against other independent computer models, or against the cognizant expert's engineering knowledge.

Data Visualization

For a model of the IFM's complexity, one concern is how to best comprehend the results of a simulation. Through SPEEDUP's plot capability, any of the 10000 variables in the simulation result may be plotted (as in Figure 2). This is effective if the analyst knows what is important and what to expect. However, in order to obtain a more global understanding of a simulation's results, we are also developing graphical user interfaces (GUI) which will present the output in a dynamic "movie cartoon" fashion, in essence a dynamic representation similar to the form of Figure 1 in which individual streams will take on a color keyed to the current flow rate, in which tank levels will visibly rise and fall, in which "gauges" will display parameter values, etc. We have prototyped one such GUI designed on site and interfaced to the standard SPEEDUP output, and are currently evaluating a commercial package (SL-GMS) which promises even greater flexibility. In addition, the input preparation phase requires a large amount of operations-type information to fully specify multi-year scenarios. To ease this burden, we have developed a simplified, text-based input language to the specifications of the customers for the IFM. This allows the user to focus only on preparing the input in a familiar environment and leaves the preparation of the actual recipe files discussed in the Batch Process Control section to the software.

Our intent is to provide the user full convenience in accessing and understanding the results provided by the long time-scale, comprehensive, high level waste complex modeling capabilities provided in the Integrated Flowsheet Model.

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HLW process overview courtesy of P. D. d'Entremont, WSRC.

SPEEDUP(tm) is a trademark of Aspen Technology, Inc., Cambridge, Massachusetts, USA.

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CHANGES IN CHEMICAL COMPOSITION OF TETRAPHENYLBORON SLURRY DURING PRETREATMENT FOR VITRIFICATION

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ABSTRACT

This paper presents results from a study of changes in the composition of a high-level waste (HLW) stream at two stages of its treatment. The waste stream that was studied was a tetraphenylborate precipitate slurry of a HLW supernate from a storage tank at the Savannah River Site. Changes to the slurry composition were studied prior to treatment (a hydrolysis process) and then again after hydrolysis of the slurry. In addition, the success of the hydrolysis is briefly discussed.

This study showed that prior to hydrolysis, the solubility of cesium-137 and potassium increased significantly under the influence of the beta and gamma radiation from cesium-137 decay. This was due to radiolytic decomposition of the tetraphenylborate precipitate. The concentration of soluble boron was also expected to increase. Although the boron concentration did seem to increase, the change could not be considered statistically significant.

Composition of the hydrolysis product also changed. Concentrations of soluble mercury, calcium, and magnesium all increased by more than 2X in a six month period following the hydrolysis. Concentrations of soluble copper, boron and zinc decreased by more than 2X. The concentrations of the most soluble species in the sample, sodium and potassium, did not change significantly.

Comparisons of results from four hydrolysis demonstrations with radioactive precipitate slurry indicated that the elemental composition of the hydrolysis product did not vary greatly even when the process was scaled down. The one exception was the concentration of soluble copper. This concentration was significantly lower in two of the four studies.

INTRODUCTION

The 130 million liters of radioactive waste currently stored in the Savannah River Site (SRS) Tank Farm consist of caustic sludges and supernates. After treating the wastes, radionuclides from the sludges and supernates will be immobilized into a borosilicate glass at the Defense Waste Processing Facility (DWPF). (1) Before vitrifying the radionuclides, they will be removed from the supernate by precipitating the cesium-137 as tetraphenylborate (TPB) precipitate and adsorbing less soluble radionuclides on sodium titanate. In April 1983, this precipitation process was successfully demonstrated on HLW supernate in Tank 48 at the SRS Tank Farm using the In-Tank Precipitation (ITP) process. (2)

To avoid feeding organic material to the melter, the cesium TPB precipitate will be treated using the Precipitate Hydrolysis Process (PHP) to remove the organic portion of the TPB precipitate. The Precipitate Hydrolysis Process (PHP) was developed to hydrolyze the TPB using formic acid. Hydroxylamine nitrate (HAN) was also added to mitigate production of high-boiling compounds caused by nitrite in the slurry. The late wash flowsheet was later developed when HAN was found to cause ammonium nitrate formation which created a safety concern. (3,4) As part of this development, the PHP has been demonstrated with nonradioactive TPB precipitate slurries expected from the late wash flowsheet. (5)

The SRS operates a shielded cells facility which has the capability to perform each of the DWPF-related processes on a labscale using actual radioactive waste from the SRS Tank Farm. This facility was used to demonstrate the PHP with a sample from the Tank Farm ITP demonstration. In this PHP study, the ITP slurry was successfully hydrolyzed, showing that the process could be used on actual radioactive waste from the SRS Tank Farm.

The purpose of the work described in this paper was to use the results from the initial PHP study and results from more recent studies to identify the effects of radiation on the TPB slurry and on the precipitate hydrolysis aqueous product (PHA). Because this work is ongoing, the results presented here will focus primarily on the elemental compositions of the radioactive slurry and the products from hydrolysis of this material.

At the time of the initial radioactive studies, the slurry had aged for nine years, and some of the TPB precipitate had been destroyed releasing cesium-137 to the supernate. To prepare for the PHP, the slurry was reprecipitated and washed so the

slurry fed to the PHP had the same nitrate, nitrite, sodium, hydroxide, and solid concentrations as slurries expected from late washing. (3) Two PHP tests were performed using this washed Tank 48 radioactive slurry. (6,7)
Later two more PHP tests were performed using the same Tank 48 radioactive slurry which had now aged for ten years. Once again, the slurry was reprecipitated and washed to adjust the concentrations of the soluble salts to the levels compatible with the PHP. The second set of tests were identical to the first except that they were performed in a hastelloy vessel rather than glass, and they were performed with 50 mL of Tank 48 radioactive slurry rather than 200 mL.
Most of the data included in this paper has not been published elsewhere. Some of the data used in this paper has been, but it was included with this work to compare data sets which had not been discussed together previously. This allowed additional conclusions to be made. For example, the effect of radiolysis on the solubility of some species in the Tank 48 radioactive TPB slurry was determined.

EXPERIMENTAL

Information presented in this paper is the result of several studies which have been performed over the last ten years. A summary of significant steps in the studies of Tank 48 radioactive precipitate is given in Table I. The results presented in this paper are primarily from the characterization of the precipitate and of the products of the hydrolysis.
Approximately 1.4 L of radioactive precipitate slurry from the ITP campaign completed in April 1983 was obtained for the demonstrations. To collect the 1.4 L sample, nineteen 100 mL bottles were individually lowered to the bottom of Tank 48, which contained approximately 20,000 gallons of radioactive precipitate slurry. After each bottle was filled, it was pulled out of the slurry and capped remotely. The bottles were delivered to SRTC in two batches, one on September 24, 1990 and the other on November 3, 1990. On November 14, the samples were combined to form the 1.4 L sample. (8)
By the time the first PHP tests were started (in April 1992) the precipitate slurry had received a dose of approximately 1.4 x 108 rad from the cesium-137. The slurry contained approximately 4 Ci of cesium-137/gallon of slurry.
Because the slurry was nine years old, much of the cesium and potassium had been released to the supernate. Prior to washing the precipitate, it was essential that the cesium-137 in the supernate be reprecipitated to avoid losing it in the washing process. Sodium tetraphenylborate was added to the precipitate slurry to reprecipitate the cesium and potassium prior to the hydrolysis tests. To determine the amount of sodium tetraphenylborate to add, a sample of the slurry was filtered and analyzed for potassium and cesium by atomic absorption spectroscopy, and for ammonium by ion chromatography. The resulting mixture was stirred and allowed to settle.
Initially, the sodium concentration in the slurry was 3.8 M, and the nitrite concentration was 0.38 M. These high concentrations were the result of sodium hydroxide and sodium nitrite additions to the waste in Tank 48 at the Tank Farm to inhibit corrosion of the tank. In an attempt to reduce the concentrations of sodium and nitrite to levels acceptable for glass production and for hydrolysis, the precipitate slurry was washed with 0.015 M sodium hydroxide. The late wash flowsheet requirement for nitrite is that the concentration be below 0.01 M. To reach this concentration, the slurry was washed several times. After washing the slurry, the sodium and nitrite concentrations dropped to 0.12 and 0.011 M respectively.
Prior to performing the hydrolysis, the amount of formic acid to be added was calculated. This calculation was made based on an algorithm for determining the amount of acid necessary to hydrolyze the TPB, neutralize the caustic and any buffers in the solution and to produce a solution of 0.25 acid equivalents/liter in the hydrolysis product. This algorithm has been presented elsewhere. (9)
The first two PHP tests were made with a 2 L glass reaction vessel which was heated with a hot plate/stirrer. Mounted on top of the reaction vessel was a decanter, and on top of the decanter was a condenser. During the test, water at 15oC was circulated through the condenser. Water that evaporated from the solution was condensed in the condenser and trapped in the decanter. Organic compounds that evaporated during the PHP were trapped in the decanter and collected in the organic arm. In addition to the port for the condenser, the top of the reaction vessel had three additional ports where a pH electrode, a thermocouple, and an addition tube were located. The purge gas, argon, was introduced through the same port as the

thermocouple and flushed from the reaction vessel through the decanter.
The other two PHP tests discussed in this paper were performed using a much smaller (approximately 200 mL) hastelloy apparatus. Hastelloy was used to allow us to study interactions of the slurries with the vessel material. The only difference other than the size and the material of construction between these two sets of studies was that the hastelloy vessel had a hastelloy "cold finger" which allowed cooling of the vessel upon completion of the PHP.
The procedure followed was identical to that used in larger scale studies with nonradioactive precipitate slurries. (10) The procedure has been discussed in a previous report. (6) The procedure called for the formic acid and copper formate (copper nitrate in the last two studies) to be added to the reaction vessel. The mixture was then heated to 90oC, and the precipitate slurry was added. The reaction vessel temperature was then kept at 88 to 92oC for five hours. After this five hours, the reaction mixture was boiled for five hours. Finally, the apparatus was cooled to room temperature, the purge gas was stopped, and the gas chromatograph was turned off.
For the hydrolysis to be a success, essentially all of the boron-phenyl bonds must be cleaved and the resulting benzene distilled from the solution. Using the conditions described in the previous paragraph, three of the four boron-phenyl bonds could be cleaved. To cleave the final boron-phenyl bond, a catalyst must be used. The copper catalyst was added as a hydrate of copper formate in the original two tests and as copper nitrate in the last two. Sufficient catalyst was added to produce a PHA with 950 ppm copper.
After the tests, samples were taken from the shielded cells, and analyzed by such methods as atomic absorption (AA), inductively coupled plasma-atomic emission spectroscopy (ICP-AES), and inductively coupled plasma-mass spectroscopy (ICP-MS). Analysis of the organic species is in progress for the last two tests. Therefore, this data will not be discussed in this paper.
Because the material used in these studies is very radioactive, the work could not be carried out in a laboratory or even a glovebox. Except for some of the analytical operations, this work was performed remotely using a shielded cells facility.

RESULTS AND DISCUSSION

The compositions of the radioactive TPB slurry and hydrolysis products changed with time. For the TPB slurry, the primary changes were from the radiolytic decomposition of the precipitate. The mechanism for the changes to the hydrolysis product is not as clear, however some of these changes have been reported for nonradioactive product.

Compositions of Tetraphenylborate Slurries

The tetraphenylborate slurries are primarily TPB salts, titanates, sludge solids, soluble salts, and small concentrations of products from the irradiation of the TPB. After ITP and before hydrolysis, the composition of the insoluble material changes only slightly. On the other hand, these small changes in the insoluble material significantly alter the composition of the soluble material. The effect of reprecipitation and washing on the Tank 48 slurry has been shown in previous work (6,7,11) and is outside of the scope of this paper.
After the Tank 48 slurry was reprecipitated and washed, the resulting tetraphenylborate feed was 9.0 wt % solids. Approximately 1 wt % was soluble solids, e.g. sodium hydroxide and sodium nitrite. The remainder was an insoluble mixture of TPB precipitate, sodium titanates, and sludge solids.
As the concentrations in Table II show, the most abundant elements in the insoluble material were

- aluminum and iron from sludge solids in the slurry
- potassium and boron present in the precipitate as the potassium salt of TPB
- titanium from sodium titanate used to remove strontium, uranium, and plutonium

The concentrations given in Table II also indicate that the largest contribution to irradiation of the slurry is from decay of cesium-137. For high-level-waste sludges at SRS, most of the dose comes from decay of strontium-90. Finally, sufficient uranium was present on the sodium titanate to produce 2.4 mCi/g insoluble solids.
The soluble solids in the Tank 48 slurry were made up of

- sodium salts which did not vary significantly with irradiation
- products from radiolysis of the TPB precipitates

As expected, no measurable change was observed in the concentrations of the sodium salts as a function of dose (Table III). Nitrate and nitrite concentrations can be

reduced by radiolysis, (12,13) however the doses of the slurries were not large enough to cause a significant change in these concentrations. The high aluminum concentration after 1.5x108 rad has been included but this value is under further investigation.

Potassium and cesium-137 were released to solution by radiolytic decomposition of TPB salts. The concentrations of these species were expected to increase as the slurry dose increased. Although the composition of the slurry after receiving 1.5x108 rad was not significantly different than it was after 1.4x108 rad, the concentrations of potassium and cesium (Table III) were significantly higher after 1.7x108 rad. The boron concentration was also higher after 1.7x108 rad, but this difference may not be statistically significant. Since potassium, cesium, and boron are all released to the supernate with radiolysis, soluble concentrations are expected to increase with dose. Release of cesium to the supernate is the motivation behind reprecipitating the slurry.

Solids in the Tank 48 TPB slurry were a complex mixture of species from reprocessing efforts at SRS (such as sludge solids, cesium-137, and uranium), corrosion inhibitors (such as sodium hydroxide, and sodium nitrite), material added during the In-Tank Precipitation demonstration (such as TPB, and sodium titanate), and products from radiolysis (such as soluble boron). The insoluble material was primarily sludge solids, potassium and cesium TPB, and titanates. The soluble solids were mostly soluble sodium salts and products from the irradiation of the TPB precipitate.

Compositions of Hydrolysis Products

During hydrolysis, much of the insoluble material in the TPB slurry becomes soluble. Concentrations of soluble species are significantly higher when compared to soluble solids in the original TPB slurry. These concentrations seem to change after the hydrolysis.

The insoluble solids from two of the hydrolysis tests were dissolved at high temperatures using both a nitric acid dissolution method and a sulfuric acid method. As the concentrations given in Table IV indicate, values from the nitric acid dissolutions seem to have been more precise.

Comparisons of concentrations presented in Table III and Table IV indicate that the concentrations of titanium did not change significantly during hydrolysis. Studies have shown that although the titanium may change form (perhaps to titanium dioxide), it is still not soluble in the supernate. (13)

While the insoluble titanium concentrations remained constant, insoluble copper concentrations increased because copper was added as a catalyst for the hydrolysis. Most insoluble species were increased by hydrolysis. Concentrations of cesium are not yet available but the decrease in the amount of insoluble solids, especially in potassium and boron suggests the cesium was released to the solution.

The concentrations given in Table V indicate that the hydrolysis products from the four tests were similar. Variations among the products include

- fluctuations in the manganese concentrations perhaps due to inhomogeneities
- high zinc concentrations in two tests which might also be inhomogeneities
- lower copper concentrations in the last two tests

The difference in the copper concentrations was similar to differences seen in nonradioactive hydrolysis tests. The remainder of the copper in the last two tests was present as insoluble copper as indicated by the high concentrations shown in Table IV.

The concentrations of soluble species significantly changed when the PHA from the first two tests were stored. This can be seen by comparing the concentrations at completion of hydrolysis to those six months after the hydrolysis (Table VI). The concentrations of potassium and sodium ions did not change significantly. Concentrations of mercury, calcium, magnesium, and zinc increased, while concentrations of copper, boron, and cesium decreased significantly.

CONCLUSIONS

In our studies of the Tank 48 radioactive tetraphenylborate slurry and the PHA which results from hydrolysis of this material, much has been learned about the effect of radiation on these materials, about the properties of the materials and about the success of the hydrolysis process. We have determined

Potassium and cesium became soluble as the Tank 48 slurry was irradiated by the decay radiation from cesium-137.

PHA composition changed with time. These changes may be due to a physical process, e.g. adsorption to the glass sample bottle, or to a chemical interaction.

Concentrations of the soluble sodium salts in the slurry did not change significantly as the material was irradiated.

Preliminary indications are that the TPB salts were hydrolyzed during the PHP. Success of the hydrolysis will be determined by analysis of the PHA organic species.

PHP tests using similar equipment and volumes gave very similar results, as expected.

PHP can be performed with as little as 50 mL feed in glass or hastelloy vessels without compromising the process or sample integrities.

Hydrolysis caused the solubility of all the major components in the insoluble phase to increase except titanium. Although the concentration of insoluble copper increased, the copper was added during the hydrolysis process.

Nitric acid dissolution was more compatible with our PHA solids and analytical methods than sulfuric acid.

Insoluble solids in the Tank 48 sample were TPB precipitates, titanates, and sludge solids

Currently, efforts are underway to collect more information on these materials by

Determining the Tank 48 supernate composition at higher doses.

Determining the PHA supernate composition at extended periods after hydrolysis.

Confirming the composition of insoluble material in the Tank 48 slurry using the nitric acid dissolution.

Determining the concentrations of organic species in the remaining slurry and PHP products.

Determining the composition of the insoluble solids in the first two PHA samples.

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THE CIRCULATING AIR BARRIER: INNOVATIVE CONFINEMENT OF LIQUID CONTAMINANT PLUMES

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ABSTRACT

The environmental cleanup task facing the United States Department of Energy (DOE) presents enormous technical, planning and institutional challenges, including the need to develop new technologies that are faster, better, safer, and cheaper, in order to expedite site cleanup. Through its support services contract with the Morgantown Energy Technology Center (METC), K&M Engineering and Consulting Corporation, supported by BDM Federal, Inc., has developed an innovative subsurface barrier concept, the Circulating Air Barrier (CAB), for potential application at the DOE's Hanford Site. Characterization of contaminated sediments resulting from past tank leaks, continued safe operations of the tanks, total confinement of leaking materials, secondary waste minimization, and final closure of the single shell tanks are five of the many facets of the storage tank issue at Hanford and elsewhere in the nation. Each of these issues are considered in the development of the CAB system.

Hanford contains 149 single shell underground storage tanks that have exceeded their original design lifetimes of 30 years, some by more than 20 years. The single shell tanks range in capacity from 55,000 to 1,000,000 gallons. The total estimated liquid volume is 6.8 million gallons, with a maximum drainable volume of 413,000 gallons in a single tank. As of April, 1994, 67 of the tanks have been identified as assumed leakers. K&M was tasked to evaluate current options for establishing a subsurface confinement barrier to prevent the migration of contaminant plumes toward the groundwater.

The Circulating Air Barrier System is a desiccant-type barrier designed to prevent the movement of liquid contaminants toward the groundwater by using an air circulation and processing system to lower the water saturation in a targeted subsurface zone below the saturation level required for liquid flow through that zone. The barrier can be installed using either vertical or horizontal wells, establishing a pattern of air injection and production so that the injected air moves from the injection wells through the target barrier zone to the production wells. The circulating air vaporizes water in the zone and carries the Water vapor and other volatile or entrained contaminants to the production wells. The production stream is then processed in a surface facility to remove the water, contaminants and particulates. In time, the circulating air reduces the water saturation in the target zone, and continues to remove, by evaporation, liquids that move into the zone. In the event of a tank leak, the system serves as a tool for early detection and provides a means to withdraw volatile contaminants to the surface for treatment.

The CAB system offers several important advantages, including the fact that it is a non-physical confinement technology; it has an active monitoring and leak detection capability; it is based on proven, commercially available oil and gas technologies and equipment; it has excellent potential for emergency response and rapid deployment capability; and offers high potential for integration with other remediation technologies. Demonstration- and full-scale CAB systems have been designed for the Hanford Site. This includes chemical and geologic characterization; model development, sensitivity analysis and performance optimization; subsurface configuration and surface processing equipment design; and development of a test program with associated cost estimates.

INTRODUCTION AND BACKGROUND

The Hanford Site, located near Richland, Washington was used by the former Atomic Energy Commission, later succeeded by the Department of Energy, to manufacture materials for use in nuclear weapons systems. Hanford contains 149 single shell underground storage tanks; as of April, 1994, 67 of the tanks have been identified

as
assumed leakers. The single shell tanks range in capacity from 55,000 to 1,000,000 gallons, with a maximum drainable volume of 413,000 gallons in a single tank. C-Tank farm was one of the first tank farms, constructed in 1943, and at the time of construction, functional design life for the tanks was 30 years. All of the single shell tanks are now well beyond the original design life and DOE has established a program to address this problem as quickly and safely as possible.

K&M Engineering and Consulting Corporation (K&M) supported by BDM Federal, Inc. (BDM), completed a study in June, 1993, for DOE's Morgantown Energy Technology Center (METC). In the study, the K&M/BDM team evaluated alternative drilling technologies and subsurface confinement barriers for application at the Hanford Site, specifically targeting the C tank farm in the 200 East area. The drilling assessment focused on the ability to drill in unconsolidated soils, provide good horizontal and vertical position control, facilitate subsurface characterization and monitoring, and permit installation of subsurface barriers beneath waste storage tanks. The study explored the possibility of cross-deployment of commercially available technologies found in the oil and gas industries to accomplish these objectives.

Another facet of the study included investigation and concept development of barrier systems which could be installed beneath and around the tank farms with a minimum of excavation. The barrier concepts included existing, commercially available technologies which were suitable for the variety of soil conditions found beneath the Hanford tank farms. The evaluation of the drilling and barrier technologies culminated in the development of seven integrated subsurface barrier systems and associated implementation plans.

Following the completion of that study, K&M was assigned a task to further evaluate two of the most promising barrier systems identified, including the Circulating Air Barrier (CAB). In this more recent study, K&M was supported by BDM Federal, Inc. and Arctech, Inc. The CAB system is designed to function as a confinement barrier to prevent contamination of the water table located approximately 200 feet below the base of the C-Tank-Farm underground storage tanks. This task addressed the need to design a CAB system for cold testing in uncontaminated soils at Hanford in order to gain regulatory acceptance from the U.S. Environmental Protection Agency and the Washington State Department of Ecology. Objectives of the task included design development and optimization, modeling, selection of drilling and surface processing equipment, and development of test procedures and cost estimates for conducting a cold test demonstration of the CAB process. The demonstration configuration is designed to be scaled to a prototype CAB system designed specifically for the C Tank Farm at the Hanford Site. The design methodology is also applicable to design of the CAB system for application at other sites within the DOE Weapons Complex.

The Circulating Air Barrier System is a desiccant-type barrier designed to prevent the migration of liquid contaminants toward the ground water by using an air circulation and processing system to lower the saturation in a targeted subsurface zone below the saturation level required for liquid flow through that zone. The barrier can be installed using either vertical or horizontal wells (Fig. 1), establishing a pattern of air movement from the injection wells through the targeted barrier zone to the production wells. The dry injected air circulates through the targeted zone vaporizing in-situ water and other volatile or entrained contaminants to the production wells. The production stream is then processed in a surface facility to remove the water, contaminants and particulates. In time, the circulating air reduces the water saturation level in the target zone, and continues to remove, by evaporation, liquids that move into the zone. In the event of a tank leak, the system serves as an early detection tool and provides a method to withdraw volatile contaminants for surface treatment.

The CAB system offers several important advantages, including the fact that it is a non-physical confinement technology; it has an active monitoring and leak detection capability; it is based on proven, commercially available oil and gas technologies and equipment; it has excellent potential for emergency response and rapid deployment; and offers high potential for integration with other remediation technologies. Demonstration- and full-scale CAB systems have been designed for the Hanford Site.

PROJECT DESCRIPTION

The technical approach to designing both the prototype and cold demonstration CAB

systems includes the following elements:
- selection and use of a simulator to predict and optimize performance of the CAB;
- detailed geologic characterization of the sites;
- detailed chemical characterization of the potential contaminants which may leak from the single shell tanks and enter the CAB zone;
- selection of a drilling system for installation of the CAB wells;
- finalizing subsurface configuration and design of the surface processing equipment;
- development of test program and schedule;
- investigation of applicable environmental, safety and health issues; and
- estimation of costs for program/application.

An overview of each of these elements, as applied to design of the CAB system for deployment at the DOE's Hanford Site, is provided below.

The CAB design methodology included detailed analysis of candidate system configurations (well patterns, orientation and dimensions) and issues of scale between the prototype and small-scale designs. The full scale prototype CAB for a single-tank application utilizes three horizontal wells (1 injection, 2 production) to create the CAB zone 100' x 100' x 60', with the top of the 60' thick CAB zone located 60 feet below the surface. The CAB prototype is designed to confine the maximum drainable liquid in any C tank farm tank (48,000 gal) if it leaked instantaneously.

The demonstration design is for a quarter-scale CAB system and utilizes four vertical wells (2 injection, 2 production) to create the 25' x 25' x 15' CAB zone; the top of 15' thick CAB zone is located 15 feet below the surface. The five-spot configuration upon which the design is based was evaluated using classical potential flow theory in addition to the detailed numerical modeling. Extensive analysis was also conducted on a full scale CAB for a single-tank application utilizing four vertical wells (2 injection, 2 production) to create the CAB zone 100' x 100' x 60', with the top of the 60' thick CAB zone located 60 feet below the surface. Each of these configurations are shown in Fig. 2.

The prototype is designed to utilize horizontal wells to establish and maintain the CAB zone. Horizontal wells are preferred for this particular application for the following reasons:
- obstructions between and beneath the tanks such as transfer lines, which will interfere with vertical well drilling;
- improved likelihood of regulatory acceptance, as the horizontal wells can be drilled from outside the tank farm and contact with contaminated soils will be minimized;
- higher efficiency/effectiveness when using horizontal wells; and
- horizontal wells may improve air flow paths if regions of extremely low and/or complex permeabilities, such as dikes, are encountered.

BVAP MODEL SELECTION

After establishing the objectives for barrier design, a validated, commercially available black oil reservoir simulator, BOAST3-PC, was selected, modified, and used to simulate the CAB process. BOAST3 is a finite-difference, implicit pressure / explicit saturation (IMPES) numerical simulator designed for use on a personal computer. It includes options for both direct and iterative techniques for use in the solution of systems of algebraic equations. The program simulates isothermal, multiple-phase fluid flow in three dimensions. It assumes that modeled fluids can be described by a maximum of three fluid phases (oil (in this case, the "leak" phase), gas, and water) with physical properties that depend on pressure only. The program can simulate movement and/or production of fluids by fluid expansion, displacement, gravity drainage, and capillary imbibition mechanisms. Some problems which have been accurately simulated by BOAST3 include primary oil and water depletion studies, pressure maintenance by water and/or gas injection, and evaluation of secondary oil recovery operations.

The design of the CAB required an understanding of air-flow patterns in the sediment layers at the leak site, the estimation of the volume of air required to dry the sediment and the maximum air volume that could be injected without disturbing the storage tanks such as those in the single shell Tank Farms at Hanford. In order to model the specific problems encountered in the non-oily, highly-porous, highly permeable, very dry environment at Hanford, BOAST3 was modified to account for the phase changes that will be encountered at the Reservation problem

sites. The result was a new simulator program, BVAP, with the capability to quantitatively track the vaporization, transport and removal of subsurface moisture through the proper use and control of injection and production wells. The simulator was used to predict CAB performance for each of the three modes of operation:

initial drying: the CAB system operates at up to full capacity to create the CAB zone of reduced water saturation;

leak response: the CAB system operates at up to full capacity in order to contain and produce a leak; and

monitoring and barrier maintenance: the CAB system operates at reduced capacity to provide continuous or intermittent monitoring and maintain the target level of water saturation in the zone.

The key simulator output is the length of time and volumetric air flow rate required to dry and maintain the CAB zone, and the air flow rates and pressures required to confine any simulated leak. These volumetric air flow rates become the critical parameters for design of the subsurface configuration and injection and production stream processing facilities.

GEOLOGIC CHARACTERIZATION

A detailed analysis was then performed of aspects of the depositional history, stratigraphy, and sediment properties which may affect CAB operation at the Hanford Site, using geophysical well logs; tank farm maps and drawings; and RCRA and ERA-VOC well data. For prediction of CAB performance at Hanford, the complex geology had to be characterized and converted to 22-layer BVAP grid format. In the C tank farm, studies of four RCRA wells drilled since 1989 indicate 10 easily-grouped soil units of relatively constant grain size and appearance. These beds are grouped into three stratigraphic units; the upper, middle, and lower Hanford Formation, based on correlation with wells drilled for the ERA-VOC project in the 200 West area, which were cored and extensively sampled and studied. The range of variability of the geologic characteristics are summarized below:

materials range from fine sand and clay to gravel;

thickness of the various subsurface layers ranges from 2 to 85 feet;

porosity ranges from 20 to 55%

vertical and horizontal permeabilities range from 24 to 9785 millidarcies; and

initial water saturation ranges from 6. to 10% of pore volume.

Potential geologic complications were identified, including regions of extremes in permeability such as systems of sedimentary dikes. Effective CAB design, most importantly well orientation (horizontal v. vertical)/placement and pressure differentials, will ensure improved performance in these types of soils.

SENSITIVITY ANALYSIS AND PERFORMANCE OPTIMIZATION

This geologic characterization data was used to predict CAB performance using the BVAP model. A preliminary sensitivity study was performed using a 25' x 10' x 20' simulated CAB cross-section; these results were then used to target specific areas of focus for the detailed, three-dimensional sensitivity analysis.

A detailed sensitivity study was then performed to evaluate the effect on CAB operation of site-and process-specific parameters, based on a five-spot vertical well pattern for a single-tank application. Evaluated parameters included, but were not limited to:

volumetric air flow rate horizontal and vertical permeabilities leak fluid viscosity

leak volume minimum water saturation required for fluid flow soil porosity

initial water saturation high and low permeability zones completion interval

depth to CAB zone leak rate and volume well orientation

CAB thickness

The data obtained from each analysis included:

time history of water production;

maximum vertical and horizontal extent of the leak plume;

maximum injection pressure and the minimum production pressure;

number of blocks with water saturation equal to Swr (irreducible water saturation level) at the end of the drying phase;

number of blocks with water saturation equal to Swr, above Swi (initial water saturation level), and equal to Swm (minimum water saturation level required for flow) at the end of the leak phase and at the end of the analysis;

time required for leak containment (the time required for all grid block water saturation levels to be constant or decreasing); and

evaporation efficiency.

The sensitivity analysis was used to optimize practical operating ranges for volumetric flow rates, operating pressures, and water production levels:

The CAB reduced the maximum vertical penetration depth of the leak plume. The maximum vertical penetration of the leak plume was found to be directly related to soil porosity, initial liquid saturation level, minimum liquid saturation level required for flow, and the volume of soil available for leak containment. The CAB decreased the maximum vertical penetration depth of the leak plume by two methods.

The dry injected air withdrew volatile liquids and reduced the initial liquid saturation level.

The combined effects of the injection and extraction wells induced a horizontal pressure gradient within the subsurface. As a result, modeled leak fluids migrated along the pressure gradient from the injection to the extraction wells and reduced the maximum vertical penetration of the leak fluid.

Figure 3 indicates the significant effects of the addition of a CAB within a soil of porosity 0.43, horizontal permeability of 10,000 md, vertical permeability of 5,000 md, initial water saturation of 8%, and minimum water saturation required for flow of 20%. A 96,000 gallon leak occurring over 20 days remained within 130 ft. of the surface when the CAB was used. Without the use of the CAB, 57,000 gallons of the leak plume remained above the water table with the remaining 39,000 gallons of the leak contaminating the groundwater table. With the use of a CAB, the maximum amount of leak contained above the water table increased from 57,000 gallons to 150,000 gallons.

Following the sensitivity analysis, the input data files were then modified from a homogenous region approximating C Tank Farm soils to the actual permeability and porosity distributions detailed in the geologic characterization. Optimized parameters, including horizontal well configuration, were used to predict performance of both the prototype CAB system and the quarter-scale demonstration.

CHEMICAL CHARACTERIZATION AND PROCESSING DESIGN

The site chemical characterization effort details the chemical evaluation of a possible leak from the C Tank Farm storage tanks in order to design the air production stream processing equipment. Sources of characterization data for the C Tank Farm contents included waste characterization sample data for tanks 103, 104, 105, and 106, TRAC model predictions, and a PNL comparison of waste stream feed characteristics for single-shell tank waste treatment. Potential rates of removal for radon gas, radioactive and non-radioactive particulates, volatile organics, and tritiated water vapor were estimated. While the production processing system is designed to filter each of these potential contaminants, estimated quantities were minimal.

The facility design includes subsurface configuration and surface processing facility for both the full-scale prototype and the quarter scale demonstration CAB systems. The single-tank prototype system (Fig. 4), designed for C tank farm application, consists of a three-horizontal well (one injection, two production) subsurface CAB approximately 100 ft2, 60 feet thick with the top of the CAB zone 60' deep. The surface processing injection components include (two-train redundancy) a blower/aftercooler, a desiccant dryer and associated filtration. The production-side components include (dual train) an initial heat exchanger, roughing and HEPA filter units, vacuum blower/aftercoolers, desiccant dryer/filter, and activated carbon/HEPA units. Both sides are equipped with adequate, redundant instrumentation.

The quarter-scale cold test is of vertical well configuration (two injection, two production), and is 25 ft2, 15 feet thick, with the top of the CAB zone 15 feet deep. Injection-side components are (single train) a blower/aftercooler, a desiccant dryer and associated filtration. The production-side components (single train) include an initial heat exchanger, roughing and HEPA filter units, vacuum blower/afiercoolers, and a desiccant dryer/filter. The carbon filter/HEPA unit is not necessary for demonstration of the CAB concept at a clean site.

DRILLING TECHNOLOGY REVIEW

An assessment was performed in parallel with other design activities to provide an update of the analysis of alternative drilling technologies which was completed during the previous study. Of particular importance was an assessment of the various types of drilling equipment that might be used to install horizontal wells in the

difficult Hanford geologic media while providing safe working conditions for the drill rig employees. Because of the severe problems associated with effluent control for drill cuttings and air returns for air drilling operations, a review of the state-of-the-art in compaction boring equipment and operations was completed.
Other important components of the drilling systems review were:
  directional drilling systems: air rotary, sonic, coiled tubing, compaction boring, river crossing;
  directional control (measurement-while-drilling) systems: hard-wired, mudpulse-telemetry, electromagnetic; and
  modified oil field drilling equipment bits: roller cone, drag, and hammer.
The directional drilling required to support the Circulating Air Barrier system does not require pinpoint accuracy in the location of the horizontal wellbores. The sonic rig can place a vertical wellbore within a one foot diameter circle at a depth of ninety (90) feet. Heavy duty coiled tubing directional drilling units in conjunction with air motors should be able to deploy the drill bit system at the proper elevation and orientation. These systems still have not been tested with directional indexing air hammer bits which would be required to drill through boulders often encountered in the Hanford geologic media. These systems have been tested in deep petroleum and natural gas drilling operations in moderately hard drilling conditions. The study team recommended the sonic drill rig for installation of the cold test vertical wells, and additional testing for compaction boring units for installation of prototype horizontal wells.
TEST PROGRAM DEVELOPMENT AND COSTS
The proposed test program for demonstrating the CAB design concept in an uncontaminated area (Washington Public Power Supply System (WPPSS) site) includes the following objectives:
  demonstrate the CAB system concept and operability;
  validate and update the BVAP model by comparing field performance with model-predicted performance of the demonstration design;
  provide data for scale-up to hot-test and detailed cost estimates;
  demonstrate CAB system capability to comply with applicable environmental health and safety regulations; and
  demonstrate CAB system potential as a monitoring system for potential use in conjunction with other barrier and/or remediation technologies
The Cold Test as proposed consists of drilling a system of four process wells (2 injection and 2 production) and seven monitoring wells that demonstrate the air cimulating and drying process, including demonstration of the equipment required to remove all particulates and liquids that will be removed from the subsurface environment.
The demonstration as planned would verify the ability to contain any material leaked from a tank by vapor phase transport of the liquid component of that material to the surface, where it will be contained and stored in suitable containers until testing of the material can be accomplished and it can be disposed of in accordance with the proper regulations.
The test program duration is two years, including permitting; drilling and soil analysis; model modification and process design verification; detail engineering and procurement specification; procurement; subsurface equipment installation; system commissioning and check-out; cold test operation; and site restoration.
Costs were estimated for installation and operation of the demonstration CAB and capital costs for the prototype. The total cost of the demonstration test program is $3,304,000, with the largest cost components being system equipment installation ($989,000), contingency ($762,000), and cold test operation ($744,000).
Capital cost for the full-scale prototype is $13,772,000. The largest cost components are system equipment installation ($5,862,000), contingency ($3,178,000), and drilling and soil analysis ($1,474,000).
A preliminary overview of the regulatory environment for demonstration testing of the CAB system was performed. A general overview of applicable federal regulations includes DOE Order 5400.0, "General Environmental Protection Program"; the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA); the Emergency
Planning and Community Right to Know Act; the Resource Conservation and Recovery Act (RCRA); the Clean Air Act; the Clean Water Act; and the National Environmental Policy Act (NEPA). An overview of the Hanford Site-specific regulatory issues were

developed from the WHC-CM-7-5, the Hanford Site Environmental Compliance Manual, and it addresses the following as related to the CAB system:

- Environmental Compliance
- Requirements for New Facility
- Regulatory Permitting Requirements

FUTURE WORK

Activities recommended to complement the CAB demonstration at Hanford include laboratory studies of soil characteristics and CAB performance, which will also serve to validate the BVAP model predictions; a survey/characterization of other candidate sites for CAB deployment; continued testing and demonstration of horizontal drilling technologies; and a study of instrumentation and monitoring options for the CAB system in varying contaminant environments. The potential of the CAB system to function as both a monitoring tool and barrier in conjunction with other physical barriers and/or remediation operations is a great incentive for further study.

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17-5

A CASE STUDY: A PROCESS IMPROVEMENT TO SURVEILLANCE AND MONITORING ACTIVITIES AT HANFORD TANK FARMS

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ABSTRACT

This paper describes an improvement to the process of collecting surveillance and monitoring data from high-level waste underground storage tank instrumentation. This process improvement uses conventional operating system (DOS) pen-based computers to improve manual acquisition methods and overall data management for tank farm surveillance and monitoring.

The Department of Energy's (DOE) concerns with demonstrating tank safety, in anticipation of continually increasing regulatory requirements and increased public participation, have resulted in the need for timely, accurate, and valid physical data describing tank farm conditions. In the face of an increasingly constrained annual budget, DOE's challenge is to provide a cost-effective, versatile, safe, and efficient method for collecting, validating, storing, distributing, and analyzing this data.

BACKGROUND

The Hanford Reservation was acquired by the United States government in 1943 as the location for the nation's first full-sized plutonium production reactors. Nuclear fuel production operations began on the 560-square-mile Hanford site in 1944 as part of the United States Department of War's secret Manhattan Project. Since that time, the Hanford Site has operated nine reactors and six processing or finishing facilities.

In 1992, a DOE Secretarial Decision terminated fuel reprocessing activities associated with the nation's weapon complex, and the Hanford mission was formally changed from defense production to waste management and environmental restoration. As a result of prior processing operations, an estimated 60 million gallonsa (230,000 cubic meters) of radioactive hazardous waste from the processing facilities has been stored in 177 underground storage tanks located in 15 different farms at the 200 East and 200 West areas of the site.

The age of and conditions that exist in the waste tanks demand that adequate

surveillance and monitoring of tank and waste physical conditions be performed until final remediation is completed. Tank waste remediation is scheduled to be completed in 2018 for single shell tanks, and 2030 for double shell tanks.

EXISTING METHOD

The existing radioactive waste storage tank surveillance and monitoring system at Hanford was developed before the capability existed for a more efficient and thorough data collection process. The existing tank farm data collection method is completely manual; that is, schedule generation, task assignment, collection, verification, validation, storage and retrieval are done manually (using logbooks, paper files and typed lists) across the site.

Data management in the existing monitoring system begins with scheduling, which is developed based on supervisor data requests and procedural requirements. The operators' schedule to perform all required surveillance and monitoring activities is established manually on a weekly basis.

Data is collected by parameter (temperatures, liquid level, pressure, etc.) and written onto procedurally approved paper forms. Each parameter has an associated procedure, which defines how to collect that type of data. Tank-farm procedures are developed to ensure that all state, federal, and local requirements are met. Procedures require additional data collections one, and sometimes two, when the recorded values are outside an expected normal operating or safety range. These requirements are enforced to confirm the out-of-range condition. Managers must be notified when out-of-range conditions are confirmed. When completed, the data sheets are submitted to managers for review and verification of the collected data.

The review typically includes comparing a sample of collected data with historical records or logbooks. The manager performs a spot check of collected data for validity and then approves and signs the entire data sheet. The approved data is then routed for trending, analysis and manual file storage.

Any subsequent analysis of collected data requires data to be retrieved from paper file storage, manually entered into a computer, and routed to the analyst for use in computations.

See Fig. 1.

STATEMENT OF NEED

DOE requiresb surveillance and monitoring activities to be: timely, legible, accurate/correctable, reviewable, storable for the life of the facility and retrievable.

The need for improved manual data-acquisition methods and better overall data management of underground storage tank surveillance and monitoring has been identified by the Tank Farms Operations and the Tank Waste Remediation System Division. For example, data on paper forms must be transcribed to an electron format for analysis. This process is time-consuming and may introduce errors. The existing system includes many such inefficient and quality-affecting data management practices. Process evaluations consisting of document evaluation and personal interviews with key operations personnel and direct observation of monitoring operations show that the process problems for tank surveillance and monitoring arec,d:

  Establishing task schedules
  Security of data
  Physical data collection
  Managers' primary review
  Access to historical data
  Data transfer and storage

Attributes of a Process Improvement

A valid process improvement should be characterized by the following attributes:

  Simplified operations
- Better support with fewer people
- Simplified maintenance
- Simplified quality assurance
- Simplified process control
- Simplified materials management
- Simplified data processing
- Further process improvements easy to implement
  More involvement by participants
  Increased efficiency

Increased flexibility
Clearly defined responsibilities
Improved quality control and fault cause analysise,f,g.

PROCESS IMPROVEMENT OPTIONS

These process improvement attributes and DOE operational requirements were applied to three potential solutions to Hanford's tank farm monitoring and surveillance problems. Full automation, partial automation, and no automation are considered below.

FULL AUTOMATION: Full automation of the data collection process, including electronic instrumentation and centralized collection of all tank farm data. The system would require new instrumentation and data lines or transmitters to be installed for each data collection point.

It was determined that full automation is not cost effective or desirable. The high expense of designing, developing, installing, maintaining and upgrading a completely automated system is cost prohibited for the foreseeable future. In those few cases where continuous monitoring is required for safety reasons, a fully automated data-acquisition system can be installed without significantly affecting overall complexity and control. Worker action needed to perform monitoring and surveillance activities would be greatly simplified, but at the sacrifice of flexibility and a decrease in oversight and control of the tank waste system. For example, operators would not need to visit the tank farms, and thus not observe the physical condition of instrumentation (corrosion, damage, etc.). The oversight and control is reduced by the complex and sometimes unique equipment needed to achieve full automation, and by the elimination of direct observation by operators. Full automation violates one of the basic principles of automation; do not install equipment in locations where humans are needed to think and solve problems and can ensure sufficient process consistency. Operators are needed to observe tank farms conditions and identify possible trouble. Equipment should be considered a problem and labor an opportunity. The main advantage derived from installing automation equipment is a decrease in process variability. That is, the process is more consistently performed without making changes due to extenuating circumstances.

PARTIAL AUTOMATION: Partial system automation to improve identified problems; schedule, security, data collection, data review, access to historical data and data handling validated by comparison to industry expectations of process improvements. Partial automation was found to be the most cost-effective process improvement solution. Subsequently, a fully implementable partial automation system was developed jointly by BDM Federal and Westinghouse Hanford Company. The resulting system, named the Pen-Based Computer for Handheld Data Acquisition of Tank Farms (PC-HDAT) is briefly described in the following section.

NO AUTOMATION: Revise existing manual system without automating to ameliorate deficiencies.

Revising the existing manual system was determined to be insufficient. Adequate improvements to the existing process could not be determined; the existing process was not designed to meet the existing demands for tank farm data. It is anticipated that this solution would also result in higher overall expense in long-term operations.

SYSTEM DEFINITION

The PC-HDAT system consists of standard PC workstations, dockable pen-based computers, conventional software, the existing Hanford computer local area network, and BDM Federal-developed PC-HDAT software to automate the Hanford tank farm surveillance and monitoring process.

PROCESS COMPARISON

PC-HDAT has been signed to correct the six process problems Hanford has identified. Each section contains a description of PC-HDAT system functions. The comparison is divided into the six Hanford identified problem areas in the existing method. Each section contains a brief description of the existing problems, followed by a description of PC-HDAT system functions developed in accordance with DOE requirements and process improvement attributes (8,9).

SCHEDULING

Existing Method

Procedure based surveillance and monitoring schedules specify the frequency of data collection, and in some cases, the work shift during which work is to be performed. Surveillance and monitoring data collection done by parameter rather than by

physical location is inefficient since more operator time must be spent to reenter a tank farm, which is a radiological controlled area.
In addition, a manually established schedule results in an unbalanced work load. For example, all quarterly mandated monitoring operations occur on the first day of the quarter rather than distributing the monitoring operations across the quarter. Until 1992, all scheduling was performed manually, resulting in data collections at times being delinquent by days or weeks. During 1992, a document was developed to establish a controlled scheduling process. While the written scheduling document reduces the number of missed collections, errors still persist. The daily work load also remains highly variable and the collections are still inefficiently taken by data type.
PC-HDAT Process
The purpose of scheduling surveillance and monitoring operations is to assign qualified personnel to perform specific surveillance and monitoring operations. These operations are scheduled at designated facilities at specific times and with required monitoring frequencies. For Hanford tank farm operations, the scheduling function must ensure regulatory-driven monitoring frequencies are met.
The PC-HDAT system automatically established a daily work list by comparing a detailed list of required monitoring frequencies and collection preferences with the last data collection date for each data point. The task list is sorted by tank farm, tank, location, and instrument to minimize operator exposure time in a radiation area. A "To Do List" is generated listing each location, component and instrument to be monitored. A separate "Urgent" To Do List is generated for any scheduled maintenance that is within a specific time period (grace period) of going overdue. The lengths of grace periods are established to coincide with the monitoring frequency for a given instrument.
The schedule is automatically transferred (along with other necessary configuration) to the handheld unit installed in the docking station. PC-HDAT audits all information exchanges between handheld units and the managers' workstation. The task list is then assigned to the next shift operator that logs onto the PC-HDAT system. The manager may adjust the scheduled monitoring and surveillance in several ways. The manager may balance the schedule by changing the monitoring preference (i.e. collect farm A data on Mondays, Farm B data on Tuesdays, etc.) The manager may easily schedule expedited surveillance and monitoring for either a specific instrument or for all instrumentation in a specific tank. The manager may increase the monitoring frequency above procedural requirements. PC-HDAT will allow the monitoring frequency to be easily changed for an instrument or tank while ensuring that the frequency is not allowed to be less than procedural requirements.
The PC-HDAT scheduler clearly simplifies operations. Automated scheduling eliminates the need for an individual to develop schedule with the attendant human error. The only maintenance required for PC-HDAT scheduling is an update of the required monitoring frequencies list when these are changed. The quality of the scheduling process is improved by decreasing the variability of the schedule building process and allowing balanced work tasking. Scheduling process control is improved by the use of grace periods to warn the operator when monitoring is nearly overdue, reducing overdue data collections. In addition, separate documents to control scheduling are no longer needed. Managers have more control over the periodic monitoring schedule and urgent monitoring requests. The scheduler is more efficient and flexible than the existing manual method while providing clear task responsibilities are greater quality control. Scheduling process quality control is expedited by audit trails built during data modifications and transfers.
DATA SECURITY
Existing Method
Current data security relies on strict procedural compliance. Human interaction is necessary to ensure that only properly trained operators are assigned and allowed to collect tank data. The increasing importance of data security is driven by increased potential of defending the validity of data in a court of law. Data security is also needed to ensure that collected data is not distributed until approved by a manager to ensure all accessible data has been properly reviewed in accordance with DOE Order 5480.19.
PC-HDAT Process
PC-HDAT provides data security through encrypted computer accounts and passwords. PC-HDAT provides seven different user access levels, allowing control of system

function access. Only managers are allowed to perform certain functions, and only registered operators are allowed to collect data using PC-HDAT. To prevent ambiguous or duplicate entries, collected data is maintained in a separate, controlled data file until it is approved by a manager. Logbooks and paper files are no longer needed. All data collections list the name of the operator collecting the data and the manager that approved the data. All applicable surveillance and monitoring procedures are available for viewing within the PC-HDAT system. Operators can access all procedures applicable to surveillance and monitoring at the touch of a button. Operators are automatically notified when procedures have been revised. All data collection entries are automatically date and time-stamped at time of data entry, and automatically transferred from the handheld computer to the manager's workstation upon return to its docking station. Audit trails and conservative data transfer protocols are used to ensure data is not lost during transfers and data can be tracked from the instant of collection.

Under the manual surveillance and monitoring system, data security only established a case-by-case basis when considered necessary. Under PC-HDAT, data security is built into the system from the lowest levels up, simplifying the establishment of data credibility. Data quality is improved by tracking the data from the moment of collection. Robust software design ensures that data quality from the source is not inadvertently lost during loss of power or surges.

PHYSICAL DATA COLLECTION

Existing Method

Manual methods of data collection presents many difficulties. The problems may be classified into three categories: environmental conditions, human error, and lack of complete information.

Environmental Conditions

Operators are limited by what they can carry into radiation areas due to necessary protective clothing. Anti-contamination clothing is cumbersome and may prevent operators from easily carrying in various tools, measurement devices, procedural documentation and other needed information. The operating environment often has high winds, dust, rain, low lighting, and large variations of temperature. Paper forms are prone to being lost or rendered unusable by adverse environmental conditions.

Human Error

Data quality checks cannot be performed in the field, leading to frequent rechecks. Data entries may be illegible due to ink smears or poor handwriting. Manual transcription errors may be introduced during data collection, and data processing, while entering data from paper into an electronic format for subsequent analysis. Additionally, errors may be introduced during data transcription into managers logbooks.

Document control is uncertain during corrections and editing of data sheets. Multiple copies of data sheets may exist with differing editing and corrective marking.

Lack of Complete Information

There is not clearly defined forum for operator feedback. This may result in not communicating important operator observations, or ambiguous or indecipherable operator markings on data sheets. Range checking of data may not be completed until return from the field, resulting in one or two rechecks at a later time. Data range rechecks are inconsistent with ALARA (As Low As Reasonably Achievable) radiation exposure policy.

Operators do not have the flexibility to collect different parameter data. Operators are limited to collecting data only for the assigned data sheet. Additional data cannot be taken since collection procedures are not available. The time lag between data collection and time of data availability electronically is unnecessarily long, delaying corrective actions.

PC-HDAT Process

PC-HDAT allows the operator to carry all necessary documentation in one handheld unit. Current tank farm surveillance and monitoring procedures are available on PC-HDAT at the touch of a button. The system is flexible in that the "To Do List" may be completed in any order desired by the operator. PC-HDAT also allows the operator to collect any additional data determined necessary while in the tank farm. Paper data forms are eliminated. The handheld computer is environmentally rugged, able to operate in a wide range of temperatures, withstand extremely wet or rainy conditions and endure drops onto a hard surface without failure or data loss.

PC-HDAT automatically checks each data entry against procedural normal operating, safety, and data quality ranges. Any range violation will result in one or more automatic data recollections, with indicating messages.
Operator feedback is provided through the use of commenting functions and instrument condition indicators. Operators may provide comments specific to a data collection or instrument. The comments can be selected from an established set of comments, for clarity, or a customized, handwritten comment may be provided. The commenting function serves to provide more detailed, specific feedback about operating conditions and operator observations. Comments are reviewed by managers during data review and approval. Out-of-service instruments or equipment that is operating in a standby mode may be easily indicated as such. Since all data transfers are automatic, transcription errors are eliminated and document control is improved.
PC-HDAT simplifies the data collection process by shortening the path and reducing the amount of extra effort required to make data available to end users. Maintenance operations are simplified by use of comments, and Out-of-Service/Standby indicators. Quality assurance and data processing is improved by the elimination of data transcription and paper filing. Quality is achieved at the source by completing a variety of range checks at the time of data collection. The ruggedness nd amount of required materials is reduced, with a flexibility to eliminate more equipment and improve the process in a full implementation of the system. Maintenance departments can easily access instrument condition information, for determination of optimum maintenance requirements. Process efficiency is increased by the determination of optimum maintenance requirements. Process efficiency is increased by the elimination of unnecessary data handling steps and the elimination of cumbersome data transcription and use of automatic data transfers. Because PC-HDAT uses commercially available hardware and software, a number of improvements and changes can be readily made. Pen-based computers will continue to have the capability to implement existing and emerging technologies, such as Global Positioning Satellite (GPS), digital video systems, and bar-code readers.
PRIMARY REVIEW
Existing Method
A primary review is conducted by the manager to verify data collection results, in an effort to overcome errors introduced during collection. Corrective actions may require recollection of data, resulting in additional time spent in a radiological controlled area (RCA) in violation of ALARA practices.
The primary review also involves cross correlation of a sample of data sheet entries to a number of initial quality-check documents that include: procedural range sheets, historical records, and logbooks. Data sheets are signed by the manager verifying that it was checked. The current manual data review process does not provide a 100 percent check of procedural requirements.
PC-HDAT Process
PC-HDAT performs a 100 percent check of collected data, indicating any data which is outside procedurally allowed normal operating, safety or quality check ranges. The manager is prompted to individually approve each out-of-range data collection. The manager may choose to approve compliant data by page or individually. At the touch of a button, the manager may access all operator comments and historical data (in tabular or graphical form, as desired). If desired, the manager may initiate an additional data collection during this review. The time required to conduct primary reviews is reduced increasing process efficiency by reducing the amount of manager time required to review data. Moreover, primary reviews are simplified, allowing the manager to perform a more thorough review.
HISTORICAL DATA ACCESS
Existing Method
Historical data is only available at the shift operations office, or in file storage. Thus, operators have no access to historical data in the field. Managers must rely upon independent logbooks of historical data to conduct data sheet reviews. The use of logbook data introduces another possible source of error, and data will only be as complete as managers' efforts provide.
PC-HDAT Process
PC-HDAT provides archival access to historical data to all registered PC-HDAT users at the touch of a button, tubular and graphical. A variety of standard and customized reports are available for hard copy or electronic reporting. Recurring reports can be established to automatically build and transfer customized data

reports on specified frequency.
DATA TRANSFER AND STORAGE
Existing Method
The existing data transfer system relies on filing offices and a conventional mail system to ensure data is properly stored and routed. In such a manual transfer system, the time delay for transfer reduces the time available if an urgent response is necessary Additionally, there is a higher likelihood of data error or loss than is achievable in the existing Hanford electronic mail system. Manual filing and routing systems do not provide quick or as reliable access to stored data, and paper records are more subject to misfiling, loss, damage, wear, and destruction than a well-organized and secure data information network.
PC-HDAT Process
Once data has been accepted into the handheld computer, no additional data transfer action is necessary to ensure that approved data is available to all registered data users on the HLAN system. All data transfers are automatic upon completion of the required prerequisites (i.e data is not sent to HLAN until after manager review and approval), thus eliminating unnecessary delays. All data is electronically stored and archived as part of the HLAN system.
PC-HDAT simplifies the data collection process by shortening the path and reducing the amount of extra effort required to make data available to end users. Manually filing systems are eliminated, reducing the number of people required for support while improving support capability. Maintenance is simplified since computer archiving of data is performed in accordance with existing archival systems. Storage facility requirements are greatly reduced since the data can be archived in an electronically compatible format, such as tape or CD-ROM.
CONCLUSIONS
Changes to working, albeit inefficient, operations in the nuclear field must be made with careful consideration and great discretion due to the inherent risk and potential consequences of failures. However, the PC-HDAT system resolves all Hanford identified problems in the waste tank-farm surveillance and monitoring operations, while at the same time providing a simpler, more controlled process that is more efficient to operate, results in lower exposure, easy to implement, and flexible to meet the current and future needs of the DOE.

17-6
QUENCHING OF HYDROGEN-AIR FLAMES BY CRUSHED SALT AS BACKFILL MATERIAL
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ABSTRACT
The emplacement of radioactive wastes in salt leads to the formation of hydrogen due to radiolysis and/or corrosion. A reaction of the hydrogen with atmospheric oxygen (explosion) must be prevented in the repository. One possibility is to backfill the void space left in the drifts and boreholes with crushed salt during package emplacement. Crushed salt acts as a flame barrier.
Test series serve to determine the quenching effect of crushed salt in a hydrogen-air atmosphere. Depending on parameters such as temperature, fine particle fraction of crushed salt and ignition direction, a safe height for crushed salt beds preventing flame propagation is determined.
INTRODUCTION
The current concept for the planned Gorleben repository provides for an emplacement of high-level radioactive waste (HLW) and intermediate-level waste of the upper activity category (ILW-(Q)) according to either the drift or the borehole technique. In the case of the borehole technique, waste packages are lowered into 300 m deep, vertical boreholes drilled into the floor of drifts situated at 800 m depth. The annular space left between packages and borehole wall is backfilled with crushed salt which simultaneously covers the packages.
Crushed salt is the debris produced when driving drifts and drilling the boreholes. It is used as backfill material and, in the event of an assumed local ignition, should also prevent the formation of a propagating flame front caused by the reaction of radiolytic hydrogen released from the packages or hydrogen produced by

corrosion.
The use of crushed salt as a flame barrier in the emplacement of radioactive waste packages in boreholes in salt is described in (1).
The project "ILW-(Q) and HTGR Fuel Element Test Programme" financed by the Bundesamt fr Strahlenschutz (BfS - Federal Office for Radiation Protection) has been dealt with at the Research Centre Jlich (KFA) since mid-1993. A subtask of this programme comprises studies on the quenching effect of crushed salt.

THE QUENCHING EFFECT OF CRUSHED SALT

A bed of granular material can impede or completely prevent the formation and propagation of a flame front (explosion) in ignitable gas mixtures under certain conditions. This effect is known as quenching.
According to present knowledge, the quenching effect is based on two different processes. Chemical heterogeneous surface reactions in the flame front lead to the degradation of free radicals acting in the gas phase reaction (O atoms, H atoms, OH molecules). The uptake of thermal energy by the grain surfaces of the bed causes a near-wall temperature decrease in the reacting gas mixture, which reduces the reaction rate (2,3).
According to Deckardt et al. (2), the chemical effects are predominant in hydrocarbon flame quenching, whereas the quenching effect in hydrogen-air mixtures is primarily based on thermal losses in the flame front. The boundary condition for flame quenching can be described by the Pclet number Pe.

$$Pe = \frac{\text{heat flow dissipated on the wall}}{\text{enthalpy flow of combustion}}$$

Under certain simplifying conditions (4)

$$Pecrit = K \cdot dcrit$$

where K contains physical parameters of the gas mixture and d is the gap width or channel diameter of a flame barrier.
Since the critical Pe numbers do not differ very much for different gas mixtures, d becomes the determinant for flame quenching. In the literature, d/2 is referred to as the quenching distance. The quenching distance depends above all on the combustible gas under consideration, the mixing ratio e.g. with air, gas density and temperature. It is generally specified as a function of the combustible gas fraction in the mixture and reaches a critical minimum value at a certain mixing ratio. The minimum quenching distance is e.g. d/2 = 0.6 mm for hydrogen-air mixtures. It is reached with a hydrogen fraction of about 30 vol.% (2).
A condition for the use of crushed salt as a flame barrier is therefore that the channels left by the grains in the bed do not exceed a certain size, the double quenching distance. This can only be achieved, however, if the bed contains fractions of fine and ultrafine grains.
In order to describe the quenching effect of crushed salt in a hydrogen-air atmosphere, a safe bed height, hs, is defined (2,3,5). The bed height hS is precisely that level which prevents the ignition taking place in an ignition volume from flashing over into a reaction volume, i.e. into an adjacent space, both volumes being filled with the same hydrogen-air mixture and only separated by the crushed salt bed.
The test facility described below was set up to determine the safe bed height of crushed salt.

TEST FACILITY

The test facility is shown in Fig. 1.
Fig. 1. Schematic representation of the test facility
The main components of the test facility are the three tube segments, ignition section 1, bed section 2 and reaction section 3, which are flanged onto each other in such a way that the bed section is located between the ignition and reaction sections. The three sections can be heated by heaters HZ 1 to HZ 3.
The bottom end of the reaction section is connected to vacuum tank 5 by a sealing foil mount 4 (with sealing foil). The sealing foil is destroyed when a defined explosion pressure in the reaction section is exceeded. The gas present in the sections expands into the vacuum tank.
Sections 1 to 3 and vacuum tank 5 can be separately evacuated by vacuum pump 6. The three sections can be filled with defined hydrogen mixtures. Proportioning is effected via the partial pressure display at pressure gauge 7. Homogenization is achieved by recirculating the mixture with pump 8.
Ignition unit 9 ignites the gas mixture in the ignition section.

The test facility described above and shown in Fig. 1 permits the determination of the safe bed height of crushed salt for ignition from above the bed, i.e. in the direction of gravity. The facility can be modified for ignition from below the bed. The experimental procedure is basically the same (see next chapter).

TEST PARAMETERS AND EXPERIMENTAL PROCEDURE

The safe bed height, hs, of crushed salt in H2-air atmosphere is determined as a function of temperature, TG, and fine grain fraction, XF, of the crushed salt.

hs = hs (TG = 25C 150C) and
hs = hs (XF = 0 wt% 90 wt%)

The following was defined on the basis of earlier studies (2,3):

concentration of H2 in air K [vol.%] = 30
grain size distribution of crushed salt D [wt%] = 0 32 mm with about 60 wt%
0 2 mm
ignition volume V [dm3] = 2.5
wall roughness of the ignition section R [m] = "steel wall"
ignition direction Z [-] = from above, from below the bed
moisture of crushed salt Xw [wt%] 0.06

The crushed salt has a grain size of 0 32 mm with about 60 wt% fine grain fraction of 0 2 mm. It consists of approx. 96 wt% NaCl, 3 wt% anhydride, and 1 wt% clays and others. The test material is produced by sieving the fine particle fraction out of the unfractionated crushed salt and adding defined quantities (0 wt% 90wt%) of these particles again to the remaining material.

In order to determine the safe bed height, hs, crushed salt is filled into the bed section up to the level h1, and the section is flanged in between the ignition and reaction sections. A sealing foil is placed into the sealing foil mount so that the volumes of the sections and that of the vacuum tank are separated from each other. The vacuum tank and the sections are evacuated.

The H2-air mixture is prepared by flooding the facility with a defined amount of hydrogen (partial pressure measurement) and subsequent replenishing with air. The mixture is then homogenized by recirculation. This completes the preparations for the actual experiment.

The H2-air mixture is now ignited in the ignition section by means of piezoelectric ignition.

The pressure displayed after combustion (pressure gauge 7) and also the noise during the reaction (explosion in the reaction section, bursting of the sealing foil) reveal whether or not flame propagation through the crushed salt bed has occurred. In the event of a flame propagation, the crushed salt bed was not sufficiently high. The experiment is repeated with a greater bed height h2.

The safe bed height, hs, was defined as that level at which, for the first time, 15 consecutive experiments did not lead to flame propagation. Figure 2 illustrates a test series for the determination of hs.

Fig. 2. Test series for the determination of safe bed height hs.

EXPERIMENTAL RESULTS

Figure 3 shows the results of the experiments determining the influence of temperature on the quenching effect of crushed salt beds in H2-air atmosphere. The fine particle fraction of the crushed salt used is 60 wt%. As can be seen, slightly deviating safe bed heights are obtained for different temperatures, but an influence of temperature on hs cannot be observed in the range 25C TG 150C. The resultant differences for hs can be regarded as general experimental tolerances. It is therefore advisable for a determination of hs from this test series, as shown in Fig. 3, to proceed from the respective highest level determined, so that a safe bed height of hs = 85 mm is obtained for ignition from above and of hs = 375 mm for ignition from below.

Fig. 3. Safe bed height hs as a function of temperature TG

The differences resulting from the ignition direction can be explained as follows. Due to the explosion pressure produced during hydrogen combustion in the ignition section, the crushed salt layer is compacted by ignition from above. Any open channels present in the bed are reduced in size or clogged so that the flame front propagating into the bed is extinguished within a relatively short distance. On the other hand, the bed is loosened in the case of ignition from below. The pore volume is enlarged and the quenching effect decreases correspondingly.

The dependence of the quenching effect of crushed salt on the fine particle fraction

is shown in Fig. 4. The quantity of 0 2 mm grains contained in the crushed salt is defined as the fine particle fraction. Its grain distribution corresponds to that of the unfractionated crushed salt produced when driving the emplacement drifts and drilling the boreholes in a salt dome. The test temperature is TG = 25C.
As already mentioned above and described in the literature (2,3), the barrier effect of a granular bed against flame propagation is the greater, the smaller the grain size of the bed material. With respect to the crushed salt investigated here this means that hs should become smaller with increasing fine particle fraction. Figure 4 precisely shows this trend.
Due to the finite length (1150 mm) of the bed section of the test facility, no safe bed height can be specified for crushed salt beds without any fine particle fraction. However, the results of tests with 10 wt% fine particle fraction suggest that beds without fine particles do not have any or only a very slight quenching effect. Figure 4 also shows that the influence of the ignition direction is of minor significance for fine particle fractions below 10 wt%.
Fig. 4. Safe bed height hs as a function of the fine particle fraction XF in crushed salt
With increasing fine particle fraction hs becomes smaller. The curves show a strictly monotonous decrease. The influence of the ignition direction increases. With a fine particle fraction of 90 wt%, for example, a safe bed height of hs = 40 mm is obtained for ignition from above, whereas hs is higher by a factor of almost 10, amounting to 350 mm, for ignition from below. The corresponding values for a fine particle fraction of 60 wt% are hs = 70 mm and hs = 375 mm.
CONCLUSION
Crushed salt beds can be used under certain conditions as a barrier to the propagation of flame fronts in hydrogen-air atmosphere when emplacing radioactive wastes according to the borehole technique. An important requirement is that beds contain a defined fine particle fraction of 0 to 2 mm grain size as flowable material. Unfractionated crushed salt as produced, for example, in driving repository drifts and drilling boreholes has a fine particle fraction of about 50 to 60 weight percent and thus fulfils this requirement.
The critical ignition direction is that from below the bed, i.e. against gravity, since, in this case, the bed can be loosened. Even then, however, a bed of unfractionated crushed salt less than 500 mm in height will be sufficient to prevent flame propagation.
In the range between 20C and 150C, temperature has no influence on the quenching effect of the crushed salt bed. With a view to preventing the formation of propagating flame fronts in a repository borehole, the same technique can therefore be used for the emplacement of both ILW(Q) and HLW packages in vertical boreholes. The above statements are also applicable to the emplacement of waste packages in drifts backfilled with crushed salt.
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APPLYING MGA FOR WASTE CHARACTERIZATION
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ABSTRACT
There is an increasing need to accurately characterize "alpha contaminated" waste. The aim of this work was to develop an approach using the well known MGA code for waste characterization. To establish what changes are needed in its analysis methods, we have investigated the effects of geometry, matrix, and detectors on the ability of the MGA code to determine the isotopic composition. This has resulted in a new, improved version of the MGA code that is suited for such waste analysis. As a practical limit, we have found that the new improved version of the software is able to accurately determine the isotopic composition of less than 10 mg of plutonium in 200 L waste drums. However, this limit is affected by detector size, source distribution, drum size, plutonium isotopics, etc. This paper presents some data that indicates how important each of these factors are in the accurate determination of plutonium isotopics in waste.
INTRODUCTION
There is an increasing need to accurately characterize the "alpha contaminated" waste that is being generated in fuel reprocessing facilities, nuclear fuel manufacturing plants, and weapons facilities. For regulatory and safety reasons, the presence of plutonium is of most concern, although uranium, neptunium and curium, if present, must also be accounted for. Safe long term storage of "alpha contaminated" waste depends on the predicted build-up and decay of its activity. A reasonably accurate prediction of future activity of a waste drum can only be made with knowledge of its isotopic composition. Of all of the techniques used for waste characterisation, gamma spectrometry is the most isotope specific. However, traditional gamma spectrometric methods are limited for this application due to the complexity of the plutonium spectrum, the low emission probabilities of the even plutonium isotopes and effects caused by high atomic number materials and widely varying matrix densities.
Several other non-destructive analysis techniques exist that allow detection of sub-milligram quantities of plutonium in drums as large as 200 liters. One of these techniques, passive neutron counting detects only the sum of even mass number isotopes of plutonium and curium by measuring the spontaneous fission neutrons (1). Active neutron counting techniques measure only the sum of the fissile plutonium and uranium isotopes by inducing fissions (2). When the fission product content and the matrix density of the measured sample are low, systems based on normal gamma spectrometric techniques can also be used to obtain quantitative estimates of some alpha emitting radionuclides that also produce reasonably strong and distinct gamma lines, such as 239Pu , 241Pu and 235U (3). With calorimetric methods, the plutonium mass is estimated by measuring the total heat output of the sample.
The problem with all of the techniques mentioned above is that they are each only good for certain conditions and can quantify only some of the alpha emitting isotopes. A proper characterization of the waste requires a calculation of the total elemental masses which in turn requires precise knowledge of the isotopic composition of each element that is present. This may be available either through knowledge of prior history of the material being characterised or through measurements. For waste material, determination of the isotopic composition through measurements can be very difficult.
Measurements of isotopic compositions for total elemental mass calculations are also required in safeguards measurements. For such work, both the IAEA and Euratom

inspectors routinely measure their samples for gamma radiation, and analyze the resulting spectra with the Multi-Group Analysis (MGA) software (4). The nature of samples investigated during a safeguards inspection, however, is quite different from waste samples. Waste samples typically have a larger physical size, contain less plutonium, have widely varying matrix compositions, and may contain other alpha, beta, and gamma emitting isotopes. These differences make it very difficult to apply the traditional safeguards version of the MGA software to accurate characterization of waste. Furthermore, safeguards inspections are routinely performed by well trained inspectors. The amount of waste barrels to be characterized in many industrial nuclear sites preclude that operators investigate each drum separately. Waste characterisation is mostly done in automated installations.

The aim of this work was to define the boundaries and limitations of the existing MGA code and to make it more applicable for alpha contaminated waste measurements. In addition, we wanted to investigate how sensitive the accuracy of the results was to the detection geometry, the composition of the uranium or plutonium, the matrix density differences and the amount of uranium and/or plutonium mass in the drum. To facilitate more effective test measurements and permit us to design automated waste characterization systems, the MGA input and output needed to be simplified.

The original MGA software can only establish the isotopics for plutonium. Recently Ray Gunnink, in collaboration with Canberra, has also developed a version of the MGA code for uranium (5). However, this new version of the MGA software was not available at the time the test measurements of this study were made and was not tested for its performance.

EXPERIMENTS

The first tests were done on real plutonium samples at the CEA Cadarache facility in France. However, during the analysis it appeared that the high gamma background encountered at this location influenced some of the results. Moreover, we had some doubts about the declared isotopic compositions of some of the sources. Therefore, it was decided to repeat some of the measurements with very well characterized sources in the (relatively) low background environment of the IRMM laboratories in Geel, Belgium . At IRMM, some other limits for the applicability of the MGA code were examined.

Experimental Set-Up in Cadarache

Experiments in Cadarache were carried out with the plutonium samples in rotating waste drums. For these experiments, the drum turning table of the neutron shuffler installation "Banco" (6) was used. Due to the nearby storage of calibration sources and due to neutron activation of the inside stainless steel lining (^{60}Co), the gamma background was quite high (approximately 5 cps/cm3 of germanium). The characteristics of the high purity germanium detectors that were used for the tests are listed in Table I. Low Energy Germanium (LEGe) detector GL2020 was connected to a second amplifier, ADC and MCA with a calibration of 250 eV/channel. This way the same detector could be used as a 2000mm2 LEGe as well as a 10% high energy detector. For each experiment, 4 spectra were recorded, 2 as low energy spectra and 2 as high energy spectra. Most spectra were taken with 30 min true time. Some weaker sources were measured for a longer time. At Cadarache, some 120 spectra were taken for this work. No peak shift or degradation was observed during the two weeks that these spectra were accumulated.

The plutonium sources used are listed in Table II. All sources are small volume sources, except source 5S and 8S. The former is a 220 L reference drum filled with polyethylene (density 0.206 g/cm3) and with the plutonium distributed over 12 positions throughout the drum. Source 8S is a 110 L reference drum filled with polyethylene (density: 0.297 g/cm3) and with the plutonium and uranium distributed over 7 positions throughout the drum. The details about these plutonium sources have been reported elsewhere (7). For generating background spectra that could potentially produce interfering peaks in the critical 100 keV region, uranium, curium, ^{137}Cs, ^{152}Eu and ^{60}Co sources were used.

Experimental Set-Up at IRMM

At IRMM, two detectors were used (see Table I). The samples were placed at a distance of 25 cm from the endcaps of both detectors. The background conditions were approximately five times better than in Cadarache. We counted a background of only about 1 cps/ cm3 of germanium, mainly due to ^{40}K. The list of sources used are also listed in Table II. Apart from these, a very well characterized (NRM-446-37) uranium

source was measured with and without cadmium filters in front of the detectors.

Experimental Set-Up at La Hague

The new MGA code for waste was also tested using small plutonium sources hidden in rotating 120 L waste drums using a "normal" waste assay system at the AD2 laboratory of Cogema La Hague. The matrix in the drums was fixed at 0.3 g/cm3 (compacted pink vinyl) and the masses of the plutonium sources hidden in the drums ranged from 0.3 mg to 10 mg. The sources that were used had typical reactor grade isotopic compositions (56 - 66 % 239Pu).

RESULTS

After we started analyzing the results of the various tests it became readily apparent that the previous version of MGA (prior to the modifications made during this study) often did not give satisfactory results for the test measurements. It either stopped operating or gave wrong answers with certain sources and/or matrices. Thus it became necessary to change the code as we accumulated test data. Most of all, the code needed to be hardened so that it kept working under as many conditions as possible. Furthermore, the code was changed to tell the operator if and why the detection or source/matrix conditions are insufficient to perform a good isotopic analysis for the cases when successful analysis was not possible. It also now warns the operator that the results given may be suspect if it detects that certain source/matrix or detector conditions are not sufficient.

The error calculation was enhanced. The uncertainty for the 242Pu abundance used to be fixed at 10 %. It was quickly obvious that this uncertainty had to be larger for the smaller plutonium samples.

To accommodate the wide range of conditions likely to be encountered in waste analysis, the new MGA code needed to introduce new set-up parameters. One of these parameters allows the user to choose the minimum number of counts in the low energy spectrum before the MGA code tries to analyze the spectrum. The maximum allowed statistical uncertainty on the 129 keV and the 208 keV peak in the low energy spectrum and on the 208 keV and the 414 keV peaks in the high energy spectrum (in the dual detector mode) can be chosen in the set-up. This way, the operator can choose not to spend analysis time for spectra that would perhaps yield results but with unacceptably large uncertainties.

Other parameters that can be inputted for this new MGA code for waste to direct its calculations include the volume of the low and high energy detectors as well as the attenuators in front and between these detectors (if the two detectors are used in a telescope configuration) and the coefficients for the calculation of the 242Pu abundance.

The influence on the MGA accuracy of the source positioning in the drum was examined by measuring the reference source C45 (778.6 mg of Pu) at 5 different radii in a 110 L drum. Two sets of measurements were done this way, with the drum empty and with it filled with pink vinyl (density 0.3 g/cm3). As can be seen from the results summarized for 239Pu in Fig. 1, the results from the analysis remain correct as a function of the source placement. In the figure, the x-axis simply represents the measurement numbers (with the 0.3 g/cm3 measurement results deliberately offset from the empty drum results for clarity). The y-axis is the ratio of the weight percentage reported by the MGA code in our analysis to the declared value and the error bars indicate the uncertainty estimate of the MGA results as reported by the program. The deviation from the declared value is not statistically significant for either case. Similar results were also obtained for the other plutonium isotopes with the error bars naturally larger for the smaller weight percentages.

Fig. 1. Ratio of MGA results with their uncertainties to the declared value for 239Pu in an empty drum and one filled to a density of 0.3 g/cm3 for a source placement of 5 different radii.

The effect of the drum size was tested by measuring the same reference source at a fixed source to detector distance in a 220 L drum, a 110 L drum and a 25 L drum all filled with pink vinyl up to a density of 0.3 g/cm3. The results clearly indicate that the results are not dependent on the size of the drum and that the uncertainty estimates from MGA remain appropriate.

To determine the influence of the plutonium isotopic composition on the accuracy that can be obtained with MGA, all sources listed in Table II were measured at a fixed position in a 110 L drum. Two sets of measurements were also performed for this test, with the drum empty and with it filled with pink vinyl to a density of 0.3 g/cm3. The results indicate that there are still some problems with certain

isotopic compositions. The deviation of the MGA results from the declared values is sometimes too large. We are still investigating what kind of isotopic compositions are a problem. Source E44 , with 50 % more 240Pu than 239Pu seems to be a particularly difficult combination for the code. Most of the more "normal" compositions were not a problem.

The influence of an increased background due to the presence of gamma sources in the drum was tested in several ways. The influence of a continuous (Compton) background under the plutonium peaks is seen when the results of the IRMM measurements are compared by those from the Cadarache measurements. Also, when the 110 L drum is filled with vinyl, the continuum background under the plutonium peaks in the 100 keV region is about a factor of two larger than for an empty drum. Still lower peak/background ratios were obtained by adding the (Compton scattered) spectra of 137Cs to the plutonium spectra. The general conclusion is that the lower the continuum background the better the results.

The MGA code was also tested on how it handles spectra where photopeaks from other nuclei interfere with the 100 keV energy region where the major plutonium lines used for the MGA analysis are to be found. Therefore, plutonium spectra were summed with a spectrum from a complex source containing 143Cm, 144Cm, 243Am, 239Np and 154Eu. It was observed that even the new MGA code could not analyze these spectra properly.

Some tests were also done with non-uniform matrices. For instance, to examine how the MGA code handles waste drums that have radially inhomogeneous attenuating materials, measurements were done with a reference source in a 110 L rotating waste drum screened at one side and at two sides with a 1 cm thick steel plate.

Also the influence of the atomic number and of the density of the matrix was investigated. Therefore, a 110 L drum with the reference source in a fixed position was filled with homogeneous matrices having a low (polyethylene and paraffin), a high (cadmium and iron) and a mixed (vinyl and iron) atomic number. For these, the average matrix density varied from 0.1 g/cm3 to 1.3 g/cm3. The influence of attenuation was further tested by measuring the stationary reference source with aluminum plates of 2 cm, 4 cm and 8 cm between the source and the detector. MGA results for plutonium samples behind varying steel thicknesses of up to 1.7 cm have been reported elsewhere (8). In that experiment it was shown that the MGA results improve dramatically when strongly attenuated high burnup Pu is measured with a telescope germanium detector. Our results with high Z materials were somewhat inconclusive. One of the measurements produced a result that was not within the allowed uncertainty of the declared value. while the others were. However, the number of measurements was not sufficient to draw any real conclusions.

Spectra of plutonium samples ranging from 1 mg to 1 g (see Table II) were taken to determine the minimum plutonium mass that is needed to do the isotopic analysis with a certain accuracy.

In addition to testing the source characteristics, the variety of detectors used for every measurement in Cadarache allowed us to test several detector types, geometries and shielding arrangements to understand their influence on the MGA results. Unfortunately, it appears that a single detector and
shielding geometry is not optimum for all types of waste streams. For instance, in some cases the use of a telescope configuration or a dual detector system will result in a more precise isotopic composition determination, while in other cases a single low energy germanium detector should be used. For good statistics spectra, the results seemed to be in general agreement regardless of the detector configuration. However, in the case of a spectrum with poor statistics, such as the spectra likely to be encountered in waste applications, the results were not quite as clear.

The measurements performed at La Hague permit us to do a more comprehensive statistical analysis on the MGA uncertainty estimates since there were many more repeat measurements using the same sources. One particular set of results contained 38 spectra where we were able to compare the declared values with the measured values (and their estimated uncertainties) for 239Pu and 240Pu. The calculated test of accuracy parameter for 239Pu is approximately 45 for this set of 38 spectra, with the 5% to 95% fraction of the allowed values of the expected distribution being 25 to 53. For 240Pu, the value of the test parameter is approximately 28, with the same range of acceptable values. The fact that the value of the test parameter falls within the expected range, of course, means that the MGA results are given with appropriate uncertainties. See Fig. 2 for a summary of the 239Pu results and Fig 3.

for a summary of the 240Pu results. Similar to Fig. 1, the Y-axis is the ratio of the MGA results to the declared value, and the X-axis is the measurement number.
Fig. 2. Ratio of MGA results with their uncertainties to the declared value for 239Pu in sources containing not more than 10 mg of plutonium
Fig. 3 . Ratio of MGA results with their uncertainties to the declared value for 240Pu in sources containing not more than 10 mg of plutonium.
Taking all of the results from the La Hague experiment we can deduce the minimum plutonium masses that can be successfully analyzed under such conditions. Assuming that the plutonium source sits in the center position of a 120 L drum filled with a low density matrix up to 0.3 g/cm3, ideal counting and shielding geometries and a 30 minute counting time are used, the minimum masses together with the largest errors that can be expected are listed in Table III.

CONCLUSIONS

This project has resulted in a better understanding of the limitations and capabilities of the MGA code. We observed that the former version of the MGA code was not well suited for the measurement of "alpha contaminated" waste. It required input from an operator in the "unusual" cases that are often encountered in waste conditions that he or she is normally not trained to provide. In some cases, such input was not possible to provide through the user interface of the previous version of the program. The old code sometimes refused to analyze spectra from weak samples, or gave wrong answers and /or the errors quoted by it appeared to be out of bound. This study has led to the development of a the new version of the MGA code that is better suited for the analysis of plutonium bearing waste. Furthermore, some of the improvements also make the new version better suited for automated analysis of large quantities of waste drums.

Our studies show that when this new MGA code provides results, they are statistically accurate relative to the declared values. Furthermore, it has been shown to be able to analyze milligram quantities of both weapons grade and reactor grade plutonium in 100 L and 200 L waste drums filled with a low density (0.3 g/cm3) matrix. The errors at these low quantities are quite high (20 % range, depending on the count time).

This project has led to a better design of the counting geometry and detector configuration for particular waste characteristics, in terms of the expected mass range and composition of the uranium and/or plutonium, in terms of the expected density and composition of the matrix and in terms of the expected internal and external background conditions.

At the present time the new version of MGA that is capable of analyzing waste is available under the Canberra Genie-PC platform. Other programs on the same PC can simultaneously communicate with other computers, perform gamma (or alpha) acquisitions using the ICB NIM or InSpector electronics and do other complex calculations while performing the MGA analysis.

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EXPERIMENTS COMPARING THE UPTAKE OF AMERICIUM FROM CHLORIDE MEDIA USING EXTRACTION CHROMATOGRAPHY

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ABSTRACT

Clean-up of actinide effluent waste steams is of increasing importance at the Los Alamos Plutonium Facility, TA-55, and removing the actinide elements to very low levels allows less radioactivity to go the Los Alamos National Laboratory Water Treatment Facility, TA-50, thus reducing the number of drums of TRU waste. Americium (Am) is a difficult element to remove from chloride media because the +3 state is difficult to oxidize and chelating resins work better with elements such as plutonium which are more readily oxidized to the +4 and/or +6 state. Currently in hydrochloric acid (HCl) media, the acidic liquid waste is neutralized with potassium hydroxide to precipitate the metal hydroxides, before disposal to TA-50. This process is not very efficient.

The removal of Am from chloride media was compared using a series of resins, some commercial and some made in our laboratory, using different percentages by weight of octyl(phenyl)-N,N-diiso-butylcarbamoyl-methylphosphine oxide (CMPO) along with diamyl amylphosphonate (DAAP) or tributyl phosphate (TBP) as diluents. Resins were also made with no added diluent. Early comparisons using small-scale contact studies with 0.5 grams of resin in 0.1M-12M HCl, and subsequent small-scale flow experiments show a trend in which Am uptake is proportional to the amount of CMPO on the resins and the diluent plays a minor role in the uptake of Am from these solutions. Redox chemistry effects were also investigated.

From these studies , it is possible to determine the best conditions for the removal of Am from HCl media thus reducing the gross alpha content of the waste stream by a factor of 10-100 which reduces the number of barrels of waste produced at the Water Treatment Facility.

INTRODUCTION

The use of extraction chromatography is currently a very popular method for the selective removal of various metals from a variety of waste streams. This is due to the availability of selective or metal-specific extractants which can easily be loaded onto an inert substrate without the need for chemically attaching the extractant molecules to the resin backbone. We have been exploring the use of extraction chromatographic resins, both commercial and resins synthesized in our laboratory, for the removal of actinides from both nitric acid and hydrochloric acid media. (1-3) Typically these resins are coated with octyl(phenyl)-N,N-diiso-butylcarbamoyl-methylphosphine oxide (CMPO) mixed with a diluent of either tributyl phosphate (TBP) or diamyl amylphosphonate (DAAP). The removal of uranium (U), plutonium (Pu) and americium (Am) from nitric acid (HNO3) is relatively easy and the extraction efficiency or Kd's obtained from these experiments are quite high. The Kd's for removal of the same elements from hydrochloric acid (HCl) media are much lower but still quite reasonable. In fact standard ion exchange which works well for the separation of Pu from Am in nitric acid and hydrochloric acid also works well for the removal of U from HCl solutions. The most difficult species to remove from HCl solutions is americium.

The commercial (Eichrom Industries)(4) extraction chromatographic resin, TRU-Spec, containing 13% CMPO and 27% TBP, will remove Pu from HCl media down to the microcurie/L level. However removal of the Am from a similar waste stream is difficult and can be accomplished only at higher (>7M) acid molarities. In this

paper, we present some additional studies that have been completed where we examined the effect of increasing the amount of CMPO on the resin to determine if there is a correlation between the amount of Pu or Am removed and the amount of CMPO present on the resin. We have also compared similar resins made using an organic substrate with those made using a silica substrate. For comparison purposes, we have included data on the Am extraction efficiency of TRU-Spec and also in certain cases similar experiments were done using Pu solutions, in order to better define the processes that occur.

EXPERIMENTAL PROCEDURE

Resin synthesis - The procedure we adopted, uses some aspects of the Horwitz recipe found in Analytica Chimica Acta (5) and of the Barney recipe from a Westinghouse Hanford Company report (6) to make different types of solvent impregnated resins. Similar procedures using di-(2-ethylhexyl) phosphoric acid (DEHPA) for making extraction chromatography resins are outlined by Cortina and co-workers.(7) For this work, the extractants/diluents used are CMPO/DAAP or TBP. The two resins used in these experiments were Amberlite XAD-7 and XAD-16. XAD-7 contains an acrylic ester backbone, with a surface area of 160 m2/g, and a 20-60 mesh size. XAD-16 has a polyaromatic backbone, a surface area of 800 m2/g, and the same mesh size.

The first stage in making these types of resins is to clean the resin to remove all of the monomers that could affect our extraction results. Approximately 50g of the resin was first weighed into a flask and 100 mL of E-pure water (18.1 M-Ohms) was added. The flask containing the resin was swirled every 10 minutes for about an hour. The slurry was then poured into a fritted funnel and the liquid was removed using a slight vacuum to speed up the process. The resin was then washed two more times in the same manner using 50 mL of E-pure water. The pH of the rinse water was 8-9. The resin was then washed 3 times with 25 mL of methanol, using a slight vacuum on the last two washes to help remove the liquid. The pH of the wash solution was now usually < 7, which indicates that the basic monomers have been removed. The resin was then placed into a rotovap flask and vacuum dried at 50oC for 16 hours, while rotating at 120 rpm. The resin is now white in color and dry which is very different from the wet, light tan starting material.

To actually make the solvent impregnated resin, 30 grams of clean resin is weighed out and slurried in a rotovap flask with 20 mL of methanol. Specific percentages by weight of the extractants/diluents are added to the slurry. For example, to make a resin with 13% CMPO and 27% DAAP on 30 grams of resin one uses 6.5 grams of CMPO and 13.5 grams of DAAP. Since CMPO is highly soluble in both DAAP and TBP, the CMPO and diluent are first mixed together then poured into the resin/methanol slurry. To make a pure CMPO resin, the CMPO is dissolved in methanol. The slurry is stirred using the rotovap for 30 minutes at 90 rpm. After stirring, the resin is dried by rotating at 35oC and 90 rpm while under vacuum (approximate time to dryness was 24 hours depending upon the concentration of CMPO in the solution). There were no significant losses of material at this stage and yields usually were 95%.

Contact Experiments - The actinide solutions used in all of these experiments were made by dissolving either plutonium or americium oxide (PuO_2 or AmO_2) in 4-6M HCl. For each contact experiment, 0.5 grams of resin was weighed out into 25 mL Bio-Rad polyethylene columns, 5 mL of a specific molarity acid was pipetted into the column and 50-250 uL of the stock solution was added to the acid in each column. An experiment contained either Pu or Am in 1-12M HCl. The columns for each experiment were rotated on a LabQuake rotating shaker for a set period of time, typically one (1) hour. At the end of the specified time, the solution was drained from the column into a scintillation vial, the vial capped and shaken to ensure a uniformly mixed sample. A 10-250 uL aliquot was removed and placed in a scintillation vial containing 6 mL of water and 14 mL of Packard Ultima Gold-XR scintillation fluid. Each acid solution was sampled twice to provide a check on sampling technique and to avoid the complete loss of a sample for an experiment in the event that a sample was spilled. Each sample was counted for ten (10) minutes using a Packard 2550TR scintillation counter. Each experiment was repeated at least once. With each set of resin containing columns, a control column without resin but containing all the solutions was used. The solution column was treated in a similar fashion to the resin containing columns and this control was used to establish the initial actinide concentration.

Upon completion of the sampling, then either hydroxylamine hydrochloride (NH_2OH-HCl) was added as a reducing agent, or sodium chlorite ($NaClO_2$) was added as an oxidizing

reagent. This redox chemistry was done to allow for comparison of the Am data with the Pu data. Each experiment required that the solutions be sampled before the addition of the redox reagents, after the addition of NH2OH-HCl, and again after the addition of the NaClO2.

The liquid scintillation count data was entered into a spreadsheet developed by Louis Schulte. This spreadsheet calculated the Kd for each set of conditions and also plotted the Kd as a function of acid molarity. The Kd is defined as shown in Eq. 1:

$$Kd = \frac{(\text{original solution cpm} - \text{final solution cpm}) \times \text{mL of sample}}{\text{final solution cpm} \times \text{g of resin}} \quad (1)$$

DATA AND RESULTS

Experiments were first conducted with Pu samples in HNO3, using XAD-7 resin without any extractant loading. The Kd's are expected to be very low and indeed this was the case. Additional experiments with the addition of NH2OH (reducing conditions) and NaClO2 (oxidizing conditions) yielded Kd's that were quite similar to those obtained for the "no redox" conditions. (See Fig. 1)

In the next set of experiments using Pu/HCl solutions, the effect of increasing the amount of CMPO on the resin was examined. These resins were synthesized in our laboratory. The resin substrate was XAD-7 with CMPO loadings of 20%, 40% and 60%. In the "no redox" part of the experiment, all three resins gave Kd's of ~750. Upon addition of NH2OH to form Pu(III) which should not extract very well, the Kd's increase as the acid molarity increases and also as the amount of CMPO increases. The Kd for 20% CMPO is 2500, for 40% is 3000 and for 60% is 5000. Similar results were observed after addition of NaClO2 to oxidize any Pu(III) to Pu(IV). The Kd's ranged from 2500 (20% CMPO) up to 6000 for 60% CMPO loading. (See Fig. 1) These results are not very much different from the results obtained with the commercially prepared TRU-Spec resin, and the observed increases in the Kd as the acid molarity increases are consistent with our observations for experiments with TRU-Spec. It is surprising that the Kd for Pu(III) is this high and it would indicate that either the reduction is not complete or most likely that the Pu(III) formed is oxidized back to Pu(IV) during the one hour duration of the experiment.

Experiments using Am in HCl were then done using TRU-Spec. All three experimental conditions ("no redox", reducing, oxidizing conditions) were completed in order to compare the results with those obtained for the Pu solutions. The best Kd's are obtained under oxidizing conditions (addition of NaClO2). From Fig. 2, we observe that the Kd increases as the acid molarity increases (as it did in the Pu experiments).

The highest Kd was obtained at 8-9M HCl, however the Kd is approximately 100X smaller, ~53. This indicates that the Kd for Pu(III) given above is not accurate. In HCl solutions, the Pu(III) should not be as readily oxidized back to Pu(IV) if it can be stabilized with a "holding" reagent, such as ferrous sulfamate.

Fig. 2. Comparison of Am Extraction Efficiency from HCl for Resins Containing 13-40% CMPO with DAAP as Diluent.

Next experiments were performed with two other commercially manufactured "experimental" resins supplied by Eichrom Industries. These resins contained higher percentages of CMPO with DAAP as the diluent instead of TBP. The resins contained 20% CMPO with 20% DAAP (20:20 resin) and 30% CMPO with 10% DAAP(30:10 resin). The 20:20 resin performed in a similar manner to TRU-Spec with increasing Kd as the acid molarity increased. Again the highest Kd was observed after the addition of NaClO2. The Kd at 8-9M is ~200, which decreases as the acid molarity increases up to 11M.(See Fig. 2)

The experiments with the 30:10 resin (organic substrate) yielded even higher Kd's with the maximum Kd of 600-625 occurring at 8-9M HCl. The Kd's decrease as the acid strength increases past this point. (See Fig. 2) Surprisingly, there is no observed improvement upon the addition of NaClO2. Since the addition of NaClO2 should have no effect on the Am(III) in solution, the enhanced Kd's under these conditions are most likely due to trace amounts of Pu in these solutions. It should also be noted that the substrate for these commercial "experimental" resins is organic and is quite similar to XAD-7 although the bead size is smaller.

Experiments with the commercially prepared (Eichrom Industries) 30:10 resin (silica substrate) yielded results that were quite similar to the results obtained with the 30:10 organic-substrate resin. The HCl molarity versus Kd plot was similar in shape with a maximum occurring between 8-9M and bending downward to yield Kd values of 100

at 12M. (See Fig. 3) There were some differences noted, mainly that the addition of NH2OH yielded higher Kd's (850) than the Kd's (500) obtained upon addition of NaClO2.
The removal of Am from the HCl waste stream was negligible until the acid molarity was 6M. The silica substrate was a porous material, however the pore size was different than the typical pore size of the organic substrate.(8)
Four different resins were synthesized in our laboratory. They are 13:27 using 13% CMPO and 27% DAAP on both XAD-7 and XAD-16 resins. This was made to compare with the commercial TRU-Spec resin. The Am extraction data with these resins were quite similar except that the higher surface area XAD-16 yielded Kd's that were ~2/3 those of the XAD-7 resin. The XAD-7 resin had maximum Kd's (under oxidizing conditions) ranging from 75-85, (See Fig. 4) while the Kd's for the XAD-16 resin ranged from 55-60. Once again the highest Kd were observed for 9-9.5M HCl.
The third resin consisted of 35% CMPO with 5% DAAP (35:5 resin). The highest Kd's exhibited by this resin were after addition of NaClO2 and at 9-9.5M HCl. (See Fig. 2) The Kd's ranged from 600-925. Although this resin performed very well and was better than the 30:10 resin, we do not understand why the measured extraction capability or Kd is almost as high as the pure CMPO resin.
The fourth resin contained no diluent and consisted of a 40% CMPO loading on XAD-7. The Kd's for this resin ranged from 850-900 and were highest for 9-9.5M HCl after addition of NaClO2. (See Fig. 2) This resin has been investigated previously by Barney and Cowan(6) as well as the authors. Although this resin extracts Am very well, the kinetics for uptake of Am are very slow.(2,3)

DISCUSSION/CONCLUSIONS

Removal of Am is possible using either commercial resins such as TRU-Spec or using specially prepared resins such as 20:20, 30:10 or 40% CMPO with no diluent. From Fig. 3, we see that better Am extraction is obtained by using a 30:10 resin on a silica substrate. From Fig. 4, it is apparent that DAAP is a better diluent (or co-extractant) than TBP. Perhaps diluent is necessary, and therefore it would be interesting to try a 40:10 or 50:10 resin on a silica substrate.
At the present time we do not have an explanation for the reversed results upon addition of NaClO2 to solutions while using 30:10 silica substrate resin as compared to the 30:10 organic substrate resin. We expect slightly higher Kd's with the addition of NaClO2 if traces of Pu are present in the Am solutions.
The high CMPO loading on these resins should absorb Pu even better than Am. This would imply that a good primary column is required in order to remove as much of the Pu as possible prior to using one of these resins to remove any Am present in the waste stream.
Selective removal of Pu from these various resins (elution) is possible and has been demonstrated in experiments in our lab and at ANL. (1-3,9) The first step is to remove the Am using low molarity (3-5M) HCl, followed by 0.1M HCl/0.1M binoxalate to remove the Pu.

FUTURE WORK

It would be interesting to compare higher CMPO loadings containing a constant percentage of either TBP and DAAP as diluent. Is there really any difference in the extraction efficiency of resins made with these two diluents?
It would also be interesting to compare these CMPO/diluent resins with a 55% CMPO resin synthesized by simply melting the CMPO and allowing the resins to absorb it.(10) This method has been reported by the scientists at the Karlsruhe Nuclear Research Center.
It would also be interesting to determine if simple modifications of CMPO would yield a better extractant for Am from HCl media, and what would be the effect on Pu removal? This has been studied in the past by Horwitz and co-workers, (11) however the choice of CMPO was somewhat dictated by its availability and the ease with which the actinides could be removed from the resin.

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17-9
BEHAVIOR OF THE CONCENTRATED PUREX HLLW FROM A HIGH BURN-UP LWR FUEL PREPARED BY A REDUCED PRESSURE EVAPORATION PROCESS

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ABSTRACT
1.5 kg of UO2 fuel irradiated in a PWR up to 47520 MWD/MTU average burn-up, cooled for 15 years, was dissolved in a nitric acid. The aqueous raffinate from solvent extraction after diluent washing, adjusted to 2.6M HNO3, and to the volume concentration of 9400 liters/MTU was concentrated in a bench scale evaporator with continuous feed under a reduced pressure maintaining the boiling temperature at 60C to give an intermediately concentrated raffinate with ca. 8M HNO3, which was successively concentrated at 50C with water feed to give 211 liters/MTU in 3.8M HNO3 solution. The concentrated active HLLW (High-Level Liquid Waste) solutions, one in 4M HNO3 and another in 2M HNO3, but with the identical volume concentration of 422 liters/MTU, were prepared by appropriate dilution procedures.
Stainless steel blocks were immersed in each concentrated HLLW solution at 50, 60 and 70C for 90 days without agitating. The blocks were weighed before and after the rinsing in a nitric acid solution to physically remove loose solids, then the soaking with a hot alkaline solution to chemically remove firmly adhesive solids. Despite no precipitation had occurred during the concentration by means of a reduced pressure evaporation process, the concentrated HLLW generated solid deposits in some extent during keeping for 90 days at the temperatures examined.
The relative amount of the Mo precipitated from the solution in 4M HNO3 (initial Mo concentration; 6.4 g/liter) was 22% at 50C, 50% at 60C and 79% at 70C. Up to 99% of

Mo precipitated from the solution in 2M HNO3 at 70C. The weight of solid firmly adhered on a unit surface area of the metal block in the solution in 4M HNO3 was 2.3 mg/cm2 at 60C. This was insignificant at 50C. Though solids adhered heavily on the blocks at 70C while the blocks were in the solution, they could be almost totally removed during the acid rinsing. Both 316L and 304L stainless steels revealed clear inter-granular attacks and the weight losses equivalent to the corrosion rate of 8-12 mm/year in 4M HNO3 and 5-8 mm/year in 2M HNO3 respectively at 70C. The corrosion rates were barely detectable, i.e., 0.2-1.4 mm, at 60C and were not depending on acidities either 4M or 2M. Neither weight loss nor local corrosion was observed on the 316L and 304L blocks with previously attached with the solid coating during the immersion in the solution in 4M or 2M HNO3 at 50C. The conceived temperature to operate the storage tanks of the concentrated HLLW, i.e., <60C was justified regarding both corrosion and solid formation points of view for either acidities of 4M or 2M HNO3.

There would be a practical incentive to simplify the reduced pressure evaporation process if the concentrated HLLW in 4M HNO3 could be discharged to the storage tanks, otherwise to produce the concentrated HLLW in 2M HNO3 at the specified fission product concentration should require an additional evaporation to a higher concentration followed by a dilution procedure.

INTRODUCTION

A reduced pressure evaporation process has been developed and established by the BNFL at Windscale in the United Kingdom to concentrate the highly active aqueous raffinate from the PUREX processing for Magnox fuel, but this did not include salt containing solution. In a typical operational condition by maintaining the pressure at 50mmHg in which the raffinate was 3M HNO3 and contained 3kg of fission products (as nitrates) per tonne of uranium processes, together with a small amount of iron, aluminum and others as alloying elements of the fuel. Under the semi-continuous operation scheme such as the continuous feed to maintain at constant volume and the batch taking-off, the acidity rose to 10M in nitric acid before dropping back about 7M as the concentration reached 80 liters/MTU, the acidity could be further reduced (typically to 4M) by water feed in place of the active feed before discharged to the storage tanks. It has demonstrated outstanding reputation to show a high mechanical reliability and a high decontamination factor for the radioactivity, in particular to reach a high evaporation factor without excessive solid precipitation, provided that the entrained and dissolved TBP were to be removed efficiently from the raffinate before the evaporation to avoid excessive precipitation of solid phosphates (1,2,3,4).

An application of a reduced pressure evaporation process to the highly active aqueous raffinate from the PUREX process for a high burn-up LWR fuel could be different from that for a Magnox fuel in the following aspects:

the relative molar abundance of fission product elements to that of residual TBP in the aqueous raffinate from the co-decontamination process increases, as TBP concentration is independent of amount of fission products.

the terminal fission product concentration in the HLLW becomes higher due to higher original concentration,

consequently the other solid forming processes rather than the formation of solid phosphates should likely gain in importance.

D.O. Campbell et al (5) have described that "The dissolver solution appears to be stable to formation of insoluble compounds for several days, provided the temperature is not too high or the acidity too low. Solids form when dissolver solution or extraction raffinate is aged at about 80C or above. From 3 to 4M HNO3 solutions, crystalline solids form which contain zirconium and molybdenum in the ratio of 1 to 2."

M.H. Lloyde (6,7) has described that "Zirconium molybdate precipitation will be more difficult to control. At an aging temperature of 100C, precipitation does not occur at all acid concentrations in the range of 1 to 7M. At a lower aging temperature (60C), precipitation does not occur at the high acid concentrations and the precipitation rate is greatly reduced at 3M HNO3." It has been also shown that the ratio of Mo precipitated after aging for 17 days at 60C was 90% in 2M HNO3, but 20% in 4M HNO3.

P.A. Anderson (8) has described that zirconium molybdate was an exclusive solid species generated during concentration of simulated HLLW under the atmospheric evaporation. T. Izumida et al. (9) have observed that the threshold temperatures to

produce solid zirconium molybdate within 10 hours were depending on acidities adjusted after the evaporation, i.e., 30-50C to 2M, 80-90C and 90-100C to 6 M in their study for the concentrated HLLW solution simulated for 33000 MWD/MTU LWR fuel prepared by a reduced pressure evaporation process.

M. Schmidt (10) has prepared zirconium molybdate ($ZrMo_2O_7(OH)_2 2H_2O$) by reflux boiling of nitric acid solutions containing Zr (1-2.7 g/liter) and Mo (1-4.3 g/liter) for 3-10 days, then measured the solubility of zirconium molybdate crystals in pure nitric acid solutions of various concentrations (1-10M) at 25C. He has described that the solubility increased steeply with increasing acidity, i.e., [Mo] was 0.5mM in 2.5M HNO_3, 2mM in 5M HNO_3 and 7mM in 8.4M HNO_3 respectively. It took more than 320 days to reach equilibrium values particularly in lower acidities than 2.5M. Schmidt suggested that the stoichiometry of [Zr]/[Mo] in his experiments might be attributed to slow kinetics.

A need to know the practical information about the behavior of the concentrated PUREX HLLW from a high burn-up LWR fuel exclusively prepared by a reduced pressure evaporation process was recognized particularly for the effect of acidity on the solid formation and on the corrosion interactions with the structural materials of the storage tank, according to the literature survey. The present study is intending to perceive a harmonized acidity for storing the concentrated HLLW, which is high enough to control the formation of solids and low enough to control the corrosion of material.

EXPERIMENTAL

Preparation of Active Concentrated HLLW

A full length Zircaloy clad UO_2 fuel rod irradiated as a part of a subassembly in the "Jose Cabreta" 153 MWe PWR (located at Zorita, Guadalajara, Spain) at an average heat rating of 9.14Kw/ft for 1334 days to give an average burn-up of 47520 MWD/MTU and the peak burn-up of 51880 MWD/MTU calculated based on the in-core power history over the fuel length, and cooled for ca. 15 years had been available at the AEA Technology's disposal and was used for the preparation of the original HLLW in this work. The fuel is called "Zorita fuel" in the paper. Details of the preparation procedure and relevant information have been published separately (11).

TABLE I, II & III

Table I shows the elementary composition of Zorita fuel calculated using FISPIN (unpublished UK code) based on the irradiation history. Table II shows simplified process flow starting from the fuel to prepare concentrated active PUREX HLLW solution by a reduced pressure evaporation process.

Despite no precipitation had been observed during the concentration by a reduced pressure evaporation process at a constant boiling temperature of 60C to produce the concentrated HLLW solution to the volume concentration of 422 liters/MTU in 8M HNO_3, a certain amount of precipitate was formed during further standing at an ambient temperature for about six months. Amount of solid collected by filtration from the concentrated solution of 36 lcm3 was 4.17g. Assuming that the chemical composition of solid was $ZrMo_2O_7(OH)_2 2H_2O$ (MW=470.2 g/mol based on the atomic weight for fission product isotopes), 41.5% of originally existed Mo could have precipitated by this stage. The solids formed were white fluffy, voluminous and easily dispersed. Gamma spectrometry of the solid showed a relatively high abundance of Te-125m. The Te-125m/Cs-137 ratio in the solid was about 16 times greater than the original ratio found in the raffinate solution (11).

The solids separated from the simulated HLLW solution evaporated simultaneously with the active runs showed the elemental composition such as Mo (55.2 at%), Zr (32.6 at%), Te (7.2 at%) and Fe (5.0 at%) by EDX analysis and the crystalline structure appeared as fluffy needles by an SEM observation as shown in Fig. 1(a), though no detail examination could have been made for the active solids. The chemical composition of the solid formed during this stage was arbitrarily assumed as $ZrMo_{1.76}Te_{0.24}O_7(OH)_2 2H_2O$ (MW=477.9 g/mol). The coordination of Fe in this compound was not understood. This could have been a reason to give a very different crystalline morphology from those previously reported for $ZrMo_2O_7(OH)_2 2H_2O$ (8,9,10). If the above assumption for the Zr-Mo-Te compound is a case, a 4.17g of solid equivalents to 15.4 mmol of Mo (1.50g of Mo) and to 35.9% of originally existed Mo in the irradiated fuel would have precipitated by this stage.

The terminal saturated acidity of the concentrate in a reduced pressure evaporation process operated with the continuous feed of an active raffinate in 2.6M HNO_3 has been known as 8M HNO_3 and it could have been reduced to 4M HNO_3 by water feed

according to the previous evaporation runs. A new mother solution concentrated at least to 211 liters/MTU should have been required in order to prepare the HLLW solution concentrated to 422 liters/MTU in 2M HNO3 by a simple dilution with an equal volume of water. A filtrate of the concentrated HLLW separated from solids was diluted to prepare the feed of 1.9 liters to the reconcentration process by adjusting HNO3 concentration to 2.6M. The continuous active feed to a reduced pressure evaporator operated to maintain at constant volume of 190ml at 50C. Two batches of 200ml water were fed at the final stage of evaporation to reduce the acidity down to 3.8M with the volume concentration of 211 liters/MTU. A vacuum drying of HLLW was also carried out at 47C as an alternative process. It is noteworthy that either reconcentration or even vacuum drying did not form any discernible solids.

Table III shows the elementary composition of the HLLW concentrated to 422 liters/MTU in the manner described above based on the results of Table I with making appropriate corrections relevant to the dissolution and the solid generation during the preparation process considering the mass balance of Mo shown in Table VIII. The total activity was 869 Ci/liter or 2.6 W/liter, though it had decreased substantially after 15 year cooling.

TABLE IV

Stainless Steel Specimens for Corrosion Tests

It has been perceived that 304L stainless steel would be more resistant to a trans-passive corrosion in a nitric acid environment but 316L stainless steel would be more resistant to local corrosion such as pitting. The corrosion tests intended to provide practical information about corrosion interaction between the concentrated HLLW and the stainless steel, both 316L and 304L.

Specimens were fabricated into an identical rectangular form, i.e., 12mm x 6mm x 4 mm, and mechanically ground to finish the surface. A 1.2 mm hole to suspend a specimen by a wire made of the same material was drilled near one end of the front. Specimens for the base metal were taken from a slab of the respective stainless steel and the front face always paralleled to the rolling direction of the original slab and the side face always represented the end grain surface, accordingly the end grain surface occupied one third of the total surface area of a specimen. Specimens for the base metal of 304L were chemically etched using HNO3-HF mixture to remove about 6 m from the surface. This chemical etching had shown effectively to remove a thin layer, which had been proved being prone to the initial dissolution (12). 316L stainless steel could not be etched under the same chemical condition as 304L had been etched effectively. The welded metal specimens were taken from the TIG arc melted section of the filler metal and the melted metal specimens (for only 316L) were taken from the TIG arc melted section of the base metal. Unfortunately, mechanical finishing of surfaces was not a high grade as expected but significant ground marks were left to be observed. Table IV shows the elemental compositions of the base materials and the filler materials for welding.

Preparation of the Specimens Attached with Solid Deposit

Those specimens to be tested at 50C were deliberately attached with solid deposits in the HLLW solution before corrosion tests by reducing acidity at the boiling temperature. The HLLW solution for this purpose was 422 liters/MTU and 1.4M HNO3 prepared by dissolving the vacuum dried HLLW in nitric acid solution. Nitric acid reduction was performed by addition of 2ml of 13M formaldehyde aqueous solution to 25ml of the HLLW under boiling condition in a glass vessel where metal specimens were immersed. This amount of formaldehyde was assumed to destruct all free nitric acid to nitrous oxide and water. The second batch of specimens was placed in the solution from which the previous batch had been removed then 25ml of the HLLW solution and 2ml of formaldehyde solution were added. The third and the fourth batch were processed in the same manner as previous. Table V shows the amount of solids deposited on each batch of process.

TABLE V

It was noted that the amount of solids adhered to the metal surface per a batch decreased along with consecutive batches, though the same amount of Mo was supplied to the solution before each batch and the same amount of solid would have been precipitated from each batch. This could be attributed to the fact in which the ratio of the surface area of metal to the volume of solution decreased along the consecutive batches, i.e., from 737cm2/liter eventually to 171 cm2/liter. Thus, it can be said that the ratio of solid adhered to the metal surface to the total solid

precipitated is proportional to the 0.4th power of the ratio of the surface area to the liquid volume, i.e., [(surface area/liquid volume)0.4].
The solid deposit from the simulated concentrated HLLW solution during the inactive preliminary runs typically showed an elementary composition such as Mo (62.2 at%), Zr (31.2 at%) and Fe (5.4 at%) but no detectable Te by an EDX analysis and showed a granular appearance by an SEM observation as in Fig. 1(b), no detail examination could be made for the active solid. Not only the chemical composition in which [Mo]/[Ze] was exactly two and no Te existed but also the crystalline morphology suggested that the solids formed in this stage were genuine $ZrMo_2O_7(OH)_2 2H_2O$.
Corrosion/Solid Deposition Tests
Experimental apparatus was made of glass and designed so as to minimize the loss of solution by volatilization during the prolonged testing period and to maximize the feasibility to remote operations in an alpha-gamma hot cell. An apparatus could contain 25ml of solution and accommodate up to six specimen blocks suspending by wires (same material as the respective stainless steel, 1mm x 1mm). Two glass apparatuses were kept at the temperature for 90 days in an aluminum block electrically heated and controlled at 1C by a thermocouple embedded in the aluminum block. The temperature of solution in each apparatus was measured by a thermocouple installed in a center tube and was recorded during the test. Fig. 2 schematically shows an apparatus.
FIG. 2 & FIG. 3
TABLE VI
After immersion for 90 days, the specimen blocks were transferred from the glass ware into a 3M HNO_3 solution and soaked for 24 hours then air dried after rinsed in water. This procedure could always enable each specimen block with a wire to move from the hot cell to the radiochemical laboratory in terms of radiation dose rate for observations and measurements. The first step of sequential procedure to measure the amount of solids was shaking each specimen block with a wire in a 1M HNO_3 solution after soaking for 24 hours to remove any loose solids physically, rinsing in water and acetone before air drying and weighing. The weight change of each specimen block including a wire from the initial weight represented the weight of solid firmly adhered on the block, though further corrections for the loss of weight due to corrosion should be made. The second step was soaking each specimen block after removing a wire in a hot 1M NaOH solution for 24 hours followed by shaking to remove adhesive solid chemically. The specimen blocks were rinsed successively in 1M HNO_3, in water and in acetone before drying and weighing. The weight change of each specimen block from the initial weight was assumed as the corrosion loss of weight. All weighing was made at a precession of 0.5mg. Corrosion loss of metal thickness was calculated based on the wetted surface area was 3.07cm2 per a block and the specific gravity of stainless steels were identically 7.9. The precession of weighing was equivalent to 0.2 mm of thickness loss and to 0.4 mm/year of corrosion rate. Fig. 3 illustrates a macrograph taken before the block was treated in a hot alkaline solution. It can be seen in the picture that this specific specimen block was covered almost entirely by the film of solid.
FIG. 4
TABLE VII & VIII
The weight change, the thickness loss and the corrosion rate for each specimen block are shown in Table VI according to the experimental varieties. Each block is identified by a code composed of (temperature)-(HNO_3 concentration)-(base: B, welded: W, melted: M)-(steel species)-(immersed totally: T, partially: P)-(block serial number). Whichever of the stainless steels showed significant weight losses at 70C, which were equivalent to 8-12 mm/year in 4M HNO_3 and to 5-9 mm/year in 2M HNO_3. The corrosion rates at 60C were barely detectable, i.e., 0.2-1.4 mm/year and did not depend on acidity. No weight loss was sensibly detected for stainless steels at 50C either in 4M or in 2M HNO_3. It has been revealed by SEM observations as shown in Fig. 4 that apparent inter-granular attacks occurred at 70C in even 2M HNO_3 solution, though they were more intense on 316L than 304L. The welded metal of 316L showed a particular attack pattern comparing others. Contrastingly no inter-granular attack was observed on any specimen immersed at 60C or 50C in either 4M or 2M. No localized corrosion other than the inter-granular attacks on the specimens tested at 70C were detected by SEM observations on the surface of specimens including end grains or those covered by the solid coating attached before testing. Specimens partially immersed in the solution at 50C did not show any irregular appearance all

over the surface of a specimen.
All solids whichever suspended or settled in the solution in each apparatus and each associated nitric acid rinsing solution were collected using 0.45 mm filter and weighed after air dried for each apparatus batch. Table VII shows the weight of solid classified into that suspended in the solution and that attached on the metal, and the latter is subclassified into that removed by shaking in an acid solution and that removed by shaking in an alkaline solution. The amount of solids totally formed in the solution generally increases with increasing temperature, particularly from 50C to 60C. Such increasing is mainly associated with the solids suspended in the solution. The amount of solids attached with the metal increases sharply from 50 to 60C. The solids attached on the metal at 70C were easily removed by the acid rinse comparing with the case at 60C. Virtually no solid attached on the metal during the tests at 50C but the solids originally attached on the specimens were kept as they had been. The effect of acidity on the total solid and on the attached solid could not be clearly judged due to lack of pertinent data, though there were general tendencies that the amount of solids increased with decreasing acidity.
A series of analysis of Mo in the filtrates separated from the suspended solids was performed using an absorption spectrophotometry method based on a thiocyanate complex formation in order to confirm the mass balance of Mo during solid formation in the solutions. Table VIII shows the results of measurement and comparison with those from gravimetric measurements of solids. Mo concentration in the original concentrated HLLW solution was assumed based on the correlation of data taken in the case of 4M HNO_3 at 50C, in which the analytical value was 4.96 g/liter and the amount of solid, i.e., 86 mg/25ml (equivalent to 1.43 g/liter of Mo). Thus, the Mo concentration in the original concentrated HLLW solution was verified as 6.39 g/liter. Analytical data suggested that very clear dependence of residual Mo concentration upon temperature and HNO_3 concentration. Discrepancies between two sets of values of Mo concentration or of Mo precipitated could attribute to incompleteness of the solid recovery from glass vessels by remote handling procedures for the gravimetric measurement. The ratio of Mo to form solids to the total Mo initially contained in the solution in 4M HNO_3 was 22% at 50C, 59% at 60C and 79 at 70C, but in 2M HNO_3 virtually all Mo tended to form solids at 70C.
DISCUSSIONS
Possibility of Loss of Mo Associated with the Undissolved Residue
Although it had been assumed previously that 41% of Mo would have been contained in the undissolved residue after the dissolution of a UO_2 fuel irradiated to 47520 MWD/MTU (11), the present work assumed that all Mo originally contained in a fuel was transferred into the HLLW after dissolution and solvent extraction. The original concentrated aqueous raffinate in 422 liters/MTU would contain 11.57g/liter of Mo if it had not been lost with the undissolved residue. 4.16g/liter of Mo has been lost from the solution in 8M HNO_3 during a long standing at an ambient temperature. Thus, the concentrated HLLW would have contained 7.41g/liter of Mo, though the Mo concentration assumed based of analysis was 6.39g/liter as shown in Table VIII. This suggested that the loss of Mo associated with the undissolved residue would have been equivalent to 1g/liter (approximately 10% of the original).
Possibility of Solid Formation during Reduced Pressure Evaporation
A fact in which a certain amount of solid precipitated from the concentrated HLLW in 8M HNO_3 during standing at an ambient temperature for six months, despite no solid had precipitated during the initial concentration of HLLW by a reduced pressure evaporation process, implied that the chemical status of the solution was under quasi-equilibrium and there might always be a possibility to initiate solid precipitation by any chance of stimulation. A strong gamma ray irradiation was reported as a stimulant by Kubota et al (13), however, an existence of seed crystals even if some extraneous nature would be effective as well. As some crystals, which have formed previously, may always exist in an industrial evaporator, it is likely that the solid precipitation is always stimulated in situ. Thus, nearly 40% of the Mo originally contained could precipitate under the practical conditions of operation according to the present experimental results.
Possibility of Solid Formation under the Storage Condition
Despite no direct evidence could be available to identify the chemical composition of solids formed in the active concentrated HLLW either in 4M or 2M HNO_3 during the tests at 50-70C, it could be reasonably suggested that the solids were zirconium molybdate and identical to those deposited compulsorily from the concentrated HLLW

at the boiling temperature by reducing acid concentration. In this stage, the solids are obviously easier to form with decreasing acidity and with elevating temperature so far enough Mo does exist in the solution. Formation of the solid exclusively depends on Mo but not on Zr, as stoichiometrically excess Zr does always exist as fission products. This can occur under the storage conditions in which the concentrated HLLW has been reduced in acidity prior to be transferred into the storage tanks. The maximum total amount of solid zirconium molybdate to be accumulated in the concentrated HLLW under storage will depend upon the total Mo contained in the irradiated fuel. The less solids form under the evaporation condition, the more solids form under the storage condition. Existence of solids would cause no notable problem in the storage tanks because they are always facilitated with an agitation capability to make solids being suspended.

Possibility of Solid Deposition and Adhesion on the Metal Surface

It would be a matter of concern if significant amount of solids would sediment from the concentrated HLLW and adhere on the cooling surface in the storage tanks. In compulsory solid deposition by reducing acidity at the boiling for 45 minutes the average solid deposition density on the metal surface was about 1mg/cm2 while all of Mo originally existed were precipitated. The average solid deposition density on the metal surface from the concentrated HLLW in 4M HNO3 after keeping at 60C for 90 days was 2.3mg/cm2 while 59% of Mo originally contained were precipitated under the similar ratio of surface to volume to the above case. The slow kinetics in solid formation seemed to increase adhered solids on the surface of metal. No data about the solid adhering for the solution in 2M HNO3 at 60C were available, therefore no direct statement about a preference of 4M HNO3 to 2M HNO3 could be done in the solid adhesion point of view.

In a conceived industrial storage tank of the concentrated HLLW operating at 60C, 120M3 in volume, ca. 1100kg of the solid is supposed to be formed during storage in addition to ca. 1200kg of the solid transferred from the evaporator, which have been already being suspended in the solution. At a presumed wetted surface area of 400m2, the ratio of surface to volume is 33.3 cm2/liter, which is smaller than the present experimental condition in a factor of 18. Thus, the ratio of solid adhered on the metal surface to the total solid precipitated will decrease to 5% from the experimental result of 15.5% based on the proposed empirical dependency on (surface area/liquid volume)0.4. Conclusively, the possible amount of solid adhered on the metal surface will be 55kg, the solid density is to be 14mg/cm2 and the thickness is to be 0.4mm based on the crystalline density of $ZrMo_2O_7(OH)_2 \cdot 2H_2O$ as 3.25g/cm3 reported by Anderson (8).

Not only gravimetric measurements but also visual observations on the specimens immersed in the HLLW solution at 70C did not show a notable amount of solid firmly adhered on the metal surface. A significant dissolution of the metal surface such as 2-3 mm during the immersion for 90 days might affect to loose solid coating on the metal, and make solids easily removable. No solid adhesion in addition to the previously attached solid was observed at 50C. Thus, to maintain the operating temperature of a storage tank at 50C or less may effectively limit the solid adhesion on the metal surface.

Effect of Acidity and Temperature on Corrosion of Stainless Steels in HLLW

Stainless steels showed notable weight losses equivalent to the corrosion rates of 8-12 mm/year in 4M HNO3, while 5-8 mm/year in 2M HNO3 and anyhow definite inter-granular attacks in either acidity at 70C. At 60C, the corrosion rates were as low as 0.2-1.4 mm/year and no inter-granular attack was detected regardless of acidities. At 50C, weight changes of stainless steels specimens before and after immersion in the HLLW solution were well within an experimental error considering of residual solid materials even after the hot alkaline treatment. Pertinent SEM observations have revealed that the crystalline morphology of residual solids was modified after the alkaline treatment, in which modifications from zirconium molybdate crystals to zirconium oxides crystals were shown.

A particular pattern of local attacks on the melted filler metal section of 316L specimens was observed. This revealed a kind of selective dissolution of metal at the grain boundaries of the ferrite phase precipitated in the melted filler metal of 316L stainless steel. As the deposition of the ferrite phase from melted 316L stainless steel is preceded by the gross solidification of the austenitic matrix phase, segregation of impurities such as sulfur tends to occur along with the grain boundaries of the ferrite phase and this causes a characteristic local corrosion.

Lower content in impurities in the 316L base metal than the filler metal could attribute to the absence of local corrosion along grain boundaries of the ferrite phase.
The environment underneath an intact solid layer could be deficient in nitric acid due to radioloysis to sustain the passivity of stainless steels. Thus, local corrosion such as pitting could have been concerned. It has been demonstrated that a local corrosion due to simple deficiency in nitric acid would not be a case under the present experimental conditions, i.e., an immersion of specimens attached with a layer of solid as thin as 3 mm in the concentrated HLLW solution in either 4M HNO3, or 2M HNO3 at 50%C. Some specimens attached with a solid coating were suspended to immerse partially in the HLLW solution at 50C for 90 days, and showed no irregularity on the observations. There was no distinction between 316L and 304L in terms of the above behavior so far as the present experiments were concerned.
Though any positive fact has not been available, in which 4M HNO3 was preferable to 2M HNO3 for the acidity of the concentrated HLLW before transferred into the storage tanks in either solid deposit or corrosion point of view, the choice of 4M HNO3 will have a practical advantage. The concentrated HLLW in 4M HNO3 can be produced straightforwardly by the water feed in place of the active feed at the terminal stage of concentration procedure in a reduced pressure evaporation process, on the other hand, to produce the concentrated HLLW in 2M HNO3 should require an additional evaporation to a high concentration before diluting with the considerable amount of water.

CONCLUSIONS

The concentrated PUREX HLLW from a high burn-up LWR fuel exclusively prepared by a reduced pressure evaporation process was studied in terms of the solid forming during either the concentration process or the storage process, the solid adhering on the metal surface and the corrosion interaction with stainless steels in the storage process. The concentrated HLLW adjusted acidity in 4M HNO3 was not particularly superior to that in 2M HNO3 regarding the solid formation or the solid adhesion, but it was not particularly inferior to that in 2M HNO3 regarding a corrosion point of view so far the temperature was kept at 60C or less. The solid adhesion on a metal surface was insignificant at 50C regardless of acidities in 4M or 2M. The conceived temperature to operate the storage tanks of the concentrated HLLW, i.e., <60C, was justified regarding both corrosion and solid formation points of view for either acidities of 4M or 2M HNO3. There would be a practical incentive to simplify the reduced pressure evaporation process in the concentrated HLLW in 4M HNO3 could be discharged to the storage tanks, otherwise to discharge the concentrated HLLW in 2M HNO3 at the specified volume concentration should require an additional evaporation to a higher concentration followed by a dilution procedure.

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DEVELOPMENT OF SYNROC MELTING PROCESS

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ABSTRACT

Alternative SYNROC production technology using the melting in high frequency induction melter - "cold crucible" is proposed. SYNROC-A, B and C compositions have been produced. The process of SYNROC-B and C production by melting, phase composition and structure of materials were studied in detail. SYNROC-B and C were produced at 1650-1800 and 1600-1700 oC respectively on lab-scale and scale-up levels. The melt capacity is widely varied (4-15 kg/h). An average specific melt capacity is 690-1062 kg/(m2 h) and depends on melt composition and melter type. It is greater than one for glassmelt at the same condition by a factors of 3 to 5. At SYNROC-C melting the total cesium loss from the crucible was estimated as 3-5%.
Melted SYNROC-B consisted of zirconolite, perovskite, hollandite and minor rutile. Crystal dimensions in quenched and annealed (slowly cooled) SYNROC-B were from 0.05 to 0.7 mm and from 0.05 to 1.5 mm respectively.
Quenched SYNROC-C samples contained zirconolite (45-50%), hollandite (35-40%), perovskite (10-15%) and minor powellite (<1%). The latter was formed due to oxidizing melting condition. Their crystal dimensions were 0.07-0.2, 0.05-0.15, ~10-2 and ~10-3 mm respectively. The mineral composition of the upper part of the slowly cooled block is similar to quenched sample. In the center of the slowly cooled block of SYNROC-C the crystal dimension can reach 1 mm.
In SYNROC-C cesium enters hollandite (58% of total content), powellite (25%) and zirconolite (17%), strontium distributes between powellite (40%), perovskite (36%) and hollandite (24%), REE enters zirconolite (55-70%) and perovskite (30-45%), most of molybdenum enters powellite under existing melting condition. The leach rates of the most of elements are at the same level as hot-pressed SYNROC-C except cesium, strontium and molybdenum. Their elevated leaching are connected to powellite formation.
The main problems observed during SYNROC production through melting stage were as follows: dry batch dusting, melt sputtering, formation of cavities and large crystals as well as undesirable phases such as powellite. A demonstration plant is being under development now. To avoid the above-mentioned problems some methods of waste and batch preparation are proposed. Pilot plant for SYNROC production was suggested to be 160 kW output power operated at 1.76 MHz with expected melt capacity 25-40 kg/h at operating temperature of 1400-1500 oC.

INTRODUCTION

SYNROC is titanate-based ceramics developed in Australia for immobilization of high

level waste (HLW) containing mainly alkali earth, transition, rare earth and actinide elements (1-5). The basic method of SYNROC production is hot-pressing (2). In that case the principal demand is to be ensured a vigorous conduction of solid phase reactions in the batch to form a desirable mineral assemblage. It requires a careful preparation of precursors, application of high temperature and pressure to consolidate the product.

SYNROC melting process was also being considered in Australia (1) and Germany (6). SYNROC melting point has been established to be high (1350oC and higher) (2). There was a problem of an unavailability of small reliable safe and remotely operated melters with long life time being able to operate at high temperatures. This problem has been partially solved in the beginning of 1980's. Developed in France and Russia so-called "cold crucible" - high frequency induction melter does not contain any refractories and electrodes. The crucible is manufactured from metal (usually copper or steel) tubes. Electromagnetic field is induced in the melt. To start up the process it is necessary to form an initial melt by means of heating of conductive material charged in the crucible together with batch. These can be, for example, graphite (as pieces), metal powder, etc. Moreover, the initial melt may be produced by heating of non-conductive batch from external source such as fuel burner, plasmatrone, microwave generator, pyrophore species etc. Principle of so-called "skull melting" excludes the contact of aggressive melt with the crucible metal walls. A layer of the partially melted batch protects the metal pipes from corrosion. The thickness of this layer depends on the melt composition and properties. For the "short" melts such as tintanate-based this layer is very thin. For the "long" glassmelts it can reach of several millimeters. Operating temperatures are limited only the thickness of the protecting layer depended on the cooling water flow rate. A lot of materials such as silica, alumina, zirconia, thoria, spinels, garnets and other minerals were easily melted in cold crucible since the beginning of 1970's (7).

The material corresponding to SYNROC formulation as well as another mineral-type materials potentially suitable for nuclear waste immobilization were produced in small crucibles operated at frequency of 5.28 MHz (8). A detailed investigation of the materials produced has not been performed.

Since 1985 works on an application of high frequency inductive melting are being conducted in SIA "Radon" in cooperation with St.-Petersburg University of Electric Engineering. The works on SYNROC production by the melting in cold crucible were started in 1990. The first samples of SYNROC-A and B formulations were produced in the same year. The melter charged by 1 dm3 of melt, operated at 1.76 MHz and energized from 60 kW high frequency generator was applied. The materials produced corresponded to SYNROC mineral assemblage and their properties were approximately the same as hot-pressed SYNROC (9). A process flexibility has been demonstrated. Any periodical, semi-continuous and continuous modes of operation were realized. Preliminary experience of Radon in the field of SYNROC melting was reported in (9-11). In this work recent advances of Radon in SYNROC melting process development are presented.

LAB-SCALE SYNROC PRODUCTION AND CHARACTERIZATION

The aim of the first step of work is to prove a possibility of SYNROC formation through the melt crystallization stage. So, SYNROC-B and SYNROC-C formulations (2,3) were chosen as basic materials for the study. Calculated materials compositions in wt.% as follows

SYNROC-B: 60.3 TiO_2, 10.8 ZrO_2, 6.3 Al_2O_3, 16.2 CaO, 6.4 BaO (2);
SYNROC-C: 54.6 TiO_2, 11.1 ZrO_2, 6.1 Al_2O_3, 13.5 CaO, 6.1 BaO, 1.0 CeO_2, 2.8 Nd_2O_3, 0.4 Eu_2O_3, 1.0 FeO, 1.6 MoO_2, 0.8 Cs_2O, 0.6 SrO, 0.3 NiO (3).

A basic laboratory works are being performed as follows:

- experimental study of SYNROC production through melting in lab-scale cold crucibles (1-5 dm3);
- evaluation of process variables and operation stability;
- characterization of product to determine the correspondence of phase composition of SYNROC produced to calculated and reference data;
- investigation of radionuclide distribution between co-existing phases;
- determination of the melted SYNROC properties;
- a comparison of phase composition and properties of melted SYNROC and hot-pressed SYNROCs known from reference data;
- modification of SYNROC formulation taking into account radwaste composition in

Russia and former USSR.
Lab-scale cold crucible is vessel of round or elliptic cross-section with 50-100 mm in inside diameter and 70-200 mm in height formed from the stainless steel or copper water-cooled tubes. It is surrounded by copper inductor. The crucible dimensions, a gap between the pipes and the number of the inductor turns are calculated from electrophysical properties of the melt and power and frequency of the generator used. However, because of difficulty of the measurement of the "short" melt electrophysical properties the crucible dimensions are selected experimentally. Commercially available high frequency generators with output power 60 kW operating at frequency of 1.76 or 5.28 MHz was used as power source. Crucible with inductor was placed in process box supplied by off-gas tube connected to gas purification system. This system consists of sleeve filter, glass fiber filter and reactor for catalytic decomposition of NOx.
The SYNROC-B batch was prepared in the vessel equipped by agitator. Screw feeding of SYNROC-B batch was conducted. The SYNROC-C batch preparation was conducted as follows. Simulated HLW components as well as Ca, Ba and Al were taken as nitrate solutions. Titania and zirconia were admixed to this solution as crystalline oxides at the mixing. Suspension prepared was dried in rotary evaporator and fed into the crucible by screw. In both cases the titanium powder was employed for starting heating and initial melt production.
The main process variables (average values) are presented in Table I.
Powdered batch was melted with high rate. However, active hydrodynamic condition resulted not only in the vigorous melt agitation but the sputtering as well, because of very low melt viscosity (<1 Pa s) and surface tension. This is a troublesome phenomenon are able to lead to ejection of the melt droplets from the crucible. Therefore, it is necessary to maintain a certain melt level in the crucible.
Very high process temperature leads to an evaporation of easy volatile components such as cesium. Total cesium loss from the crucible was estimated as 3-5%. Practically all of cesium was trapped in the off-gas purification system.
As seen from Table I, the SYNROC melting process is high productive. SYNROC melt capacity was higher than borosilicate glass capacity by factors of 3 to 5 (10). The melting ratio in both processes is approximately the same (10). So, from technological point of view SYNROC melting process excels glass melting process.
Materials produced were fully crystalline. They were investigated by X-ray diffraction ("DRON-3", Cu K-radiation), optical microscopy ("POLAM L-213"), electron microscopy ("JEM 100c") and electron-probe microanalysis ("Camebax micro-beam", U=20 keV, I=1-3 nA). Leach tests were performed according to IAEA technique (12).
Poured and solidified SYNROC-B material consists of green-gray and grayish-brown sections. The gray-green sections contain microbubbles (0.3-0.5 mm) and tube gas channels (1-2.5 mm). The principal phases are zirconolite (25-30%), perovskite (25-30%), rutile (25-30%) and hollandite (10-15%). Spinel was observed as an insignificant impurity. Zirconolite and rutile crystal dimensions were 0.15-0.7, 0.01-0.05 mm respectively. Compositions of zirconolite, hollandite, rutile and spinel are on the average Ca0.95Al0.14Ti2.22Zr0.70O7.00, Na0.24Ca0.30Ba0.87Al1.86Ti5.92Zr0.04O16.00, Ca0.04Ti1.91Zr0.05O4.00 and Na0.02Ca0.28Ba0.02Al1.94 Ti0.38Zr0.01O4.00 respectively. Sodium presents as impurity.
The grayish-brown sections have microbubble porous texture and fine-grained structure. The large (0.3-0.5 to 1.5 mm) single crystals of zirconolite and microcrystalline assemblage of hollandite and perovskite were observed in the bulk. Hollandite and perovskite crystals measure 0.1-0.5 and 0.05-0.15 mm respectively. These sections contain 40-45% of zirconolite, 30-35% of hollandite, 15-20% of perovskite and 5-10% of rutile. Their formulations are Ca0.93Al0.15Ti2.06Zr0.86O7.00 (zirconolite), Na0.08Ca0.20Ba0.98Al2.00 Ti5.86Zr0.03O16.00 (hollandite), Na0.01Ca0.97Ti1.00Zr0.01O3.00 (perovskite), Ca0.02Ti1.94Zr0.05O4.00 (rutile).
Three samples of SYNROC-C were studied. The sample C-1 was prepared by the melt quenching on metal plate at room temperature. The lower part of sample C-1 is dense and compact. The upper part of the same sample has microporous texture. Zirconolite, hallandite, perovskite and powellite were found in both parts of the sample. The crystal dimensions of zirconolite, hollandite, perovskite and powellite are 0.07-0.2, 0.05-1.5, ~10-2 and 10-3 mm, respectively. Their crystal chemcal formulations are found to be Ca0.83FeII0.03Ni0.02 Al0.19CeIII0.05Nd0.04Ti2.09Zr0.76O7.00 (zirconolite),

$Cs_{0.01}Ca_{0.30}Sr_{0.06}Ba_{0.83}Fe^{II}_{0.19}Ni_{0.18}Al_{1.42}Ce^{III}_{0.01}Ti_{5.93}Zr_{0.15}Si_{0.01}Mo^{VI}_{0.03}O_{16.00}$ (hollandite), $Ca_{0.80}Sr_{0.06}Al_{0.03}Ce^{III}_{0.06}Nd_{0.03}Ti_{0.97}O_{3.00}$ (perovskite), $Cs_{0.01}Ca_{0.66}Sr_{0.18}Ba_{0.26}Fe^{II}_{0.03}Ni_{0.01}Al_{0.16}Ce^{III}_{0.01}Ti_{0.09}Si_{0.01}Mo^{VI}_{0.80}O_{4.00}$ (powel-lite). The debala-nce observed in some cases is obviously due to the presence of the part of iron and cerium in higher valency state. Moreover, the crystal splices of the second generation of hollandite with zirconolite and perovskite (hollandite is predominate) were established. Firstly and secondly generated minerals are differentiated by the crystal dimensions and the intergrowth type and, in some cases, chemical composition as well. For example, hollandite of the first generation contains more Ti, Ca, Sr, Zr and less Al compared to hollandite of the second generation. In total the sample C-1 consists of 45-50% of zirconolite, 35-40% of hollandite, 10-15% of perovskite and <1% of powellite.

The residual melt in the cold crucible was slowly cooled and solidified. The block prepared is inhomogeneous and consists of four zones (Fig. 1). The material in zone 1 is the most dense and fine-grained. The largest crystals are located in zone 3. Zone 2 is intermediate. A lot of gas channels and bubbles are located in zone 4. Two samples from the whole block were cut. The sample C-2 was cut from the upper part of block (zone 1). This sample is not fully uniform too, however, the chemical compositions of its different parts are similar (in wt.%): 0.43 Cs_2O, 17.24 CaO, 2.15 SrO, 2.52 BaO, 1.44 FeO, 1.38 NiO, 12.12 Al_2O_3, 1.83 Ce_2O_3, 2.12 Nd_2O_3, 0.36 SiO_2, 36.25 TiO_2, 19.16 ZrO_2, 3.00 MoO_3. The difference is in crystal dimensions which grow towards the block center. Quantitative minerals ratio in zone 1 is approximately 45-50% of zirconolite, 35-40% of hollandite and 10-15% of perovskite.

The sample C-3 was cut as shown in Fig. 1. The sample is very porous in the near-bottom part. In the central part of the block the sample is dense not containing gas channels although it contains gas bubbles because of gas release was not completed. In this zone 3 the crystal dimensions reach 1 mm. Quantitative mineral ratio is close to one for the sample C-2. Average compositions of minerals as follows: $Ca_{0.84}Fe^{II}_{0.02}Ni_{0.01}Al_{0.19}Ce^{III}_{0.03}Nd_{0.04}Ti_{1.98}Zr_{0.89}O_{7.00}$ (zirconolite), $Cs_{0.01}Ca_{0.06}Sr_{0.03}Ba_{1.02}Fe^{II}_{0.07}Ni_{0.11}Al_{1.94}Ce^{III}_{0.01}Ti_{5.84}Zr_{0.05}O_{16.00}$ (hollandite), $Ca_{0.81}Sr_{0.05}Al_{0.01}Ce^{III}_{0.04}Nd_{0.03}Ti_{1.00}O_{3.00}$ (perovskite), $Ca_{0.01}Fe^{II}_{0.01}Ti_{1.92}Zr_{0.05}Mo^{VI}_{0.01}O_{4.00}$ (rutile) and $Ca_{0.30}Sr_{0.17}Ba_{0.16}Al_{0.02}Ti_{0.02}Mo_{1.10}O_{4.00}$ (powellite). Minor debalance on perovskite and rutile may be connected to transition of the part of Fe(II) and Ce(III) to Fe(III) and Ce(IV) respectively under the air melting condition.

Waste elements partitioning between the co-existing mineral phases from electron-probe micro-analysis data is shown in Table II.

A comparison of the partitioning of waste elements in SYNROC-C produced by hot-pressing (2) and melting in cold crucible is given in Table III.

As seen from Table III, there is some difference in the waste elements partitioning between hot-pressed and melted SYNROC-C probably due to difference in redox conditions. Formation of additional Mo-containing phase of powellite results in the transfer of the significant fraction of strontium and minor cesium to this phase. Unlike hot-pressed SYNROC-C, REE enter both zirconolite and perovskite phases of melted SYNROC-C. Moreover there was no formation of metal alloy at SYNROC melting probably due to mainly oxidizing melting condition. The partitioning of actinides, noble metals and halcogens is now being studied.

Although metal titanium was introduced in the first portion of batch charged into the crucible to produce the initial melt for the start up, its quantity was deficient to ensure the melting reducing condition. Possibly an addition of carbon-based reducing agent must be used to maintain the reducing condition of the melting process.

The results of leach tests are shown on Fig. 2. The leach rate of the most of elements from melted SYNROC-C is at the same level as hot-pressed SYNROC-C except Cs, Sr and Mo. Elevated leach rate of these elements is due to powellite formation. A SYNROC-C doped by Pu has also been produced and the leach rate of Pu has been measured (see Fig. 2). The detailed results of the leach tests will be reported separately.

So, it has been shown from lab-scale experiments that the main problems at SYNROC melting as follows:

dry batch dusting;

low melt viscosity and high melt fluidity and mobility resulting in the sputtering;
formation of gas cavities because of unfinished gas release and cavities at block shrinkage;
formation of large crystals in the central part of the block are being troublesome at alpha-recoil damage and spontaneous fission;
crystal dimension distribution by block height and cross-section;
formation of the undesirable phases because of difficulty of maintenance suitable redox condition.

A work on incorporation of actinide-containing incinerator ash in material similar to SYNROC-A is also being carried out. Incinerator ash oxides contain up to 35 wt.% of SiO2, up to 10 wt.% of mixed alkali oxides (mainly Na2O and K2O) and up to 15 wt.% of P2O5. The rest of oxides are typical for SYNROC (CaO, Al2O3, FeO and other transition metal oxides). The first samples produced have demonstrated a high chemical durability.

INTERMEDIATE-SCALE SYNROC PRODUCTION AND CHARACTERIZATION

First experiments on SYNROC production by melting were performed on simulated intermediate-level waste vitrification pilot plant. This plant was not designed for SYNROC production, however the results reached were quite acceptable. The main results of experiments on intermediate-scale SYNROC production were reported before (9-11). The crucible with melt volume of 10 dm3 and melt area of 0.05 m2 and high frequency generators with output power of 60 or 160 kW were applied for systematical development. The batch preparation and off-gas purification systems were the same type as ones in lab-scale experiments.

Preliminary experiments were conducted at relatively low melt capacity -8-13.5 kg/h (10,11) to try out the batch preparation unit and melters and optimize the process variables. Then it has been shown that melt capacity may be increased to 25-30 kg/h. As before the main problems were the same as in lab-scale experiments.

A demonstration plant is being under development now. Improved batch feeding system is provided. A set of alternatives is being under consideration such as granulation with high carbon binder, low temperature calcination, sol-gel process, wet feeding, direct joining of rotary evaporator to melter. These methods have both advantages and disadvantages. So, melting of precursors prepared by sol-gel method has to increase of melt capacity and produces the high homogeneous material. But, this process is very complex from technical point of view. Wet slurry feeding avoids the dusting, but at the same time increases the sputtering. At the direct joining of rotary evaporator to melter it is difficult to ensure a synchronized operation of these apparata. The development of granulation method is not finished yet. Most probably on the first stage the direct joining of rotary evaporator to melter will be employed. Simultaneously, the development of batch granulation and calcination processes will be conducted.

Current works provide for development and trial of process with existing equipment to eliminate the disadvantages occurring on the melting stage and during the cooling, solidification and annealing (formation of undesirable phases at violation of redox condition, crystal growth, unfinished gas release, formation of cavities at block shrinkage, etc.). To suppress the crystal growth it is necessary to ensure a rapid heat removal from material. It is possible by melt casting as plates or melt granulation.

Moreover a trial of SYNROC compositions is being conducted including uranium-containing compositions such as U-doped SYNROC-C and zirconolite-rich formulation (13). An experiments on production of SYNROC-F is set on 1995.

PROPOSALS ON PILOT PLANT

Proposed process block diagram is shown on Fig. 3. The batch prepared must be fed into the cold crucible energized from generator with output power 160 kW operated at 1.76 MHz. Expected melt capacity is 25-40 kg/h at the operating temperature of 1400-1500oC. Such capacity approximately corresponds to industrial-scale plant. Processing of nitrate solution will require a recovery of nitrogen oxides with nitric acid regeneration, which can be utilized for the interior technical purposes. Off-gas deposit trapped in gas purification system will be recycled. SYNROC melt will be poured directly into containers or charged after preliminary granulation. If the blocks of solidified material will be formed they must be annealed. and then headed for storage.

At the first stage the plant will operate in inactive regime. At the second stage a

simulated SYNROC doped by radionuclides will be produced. The third stage of works an experiments on production of fully active SYNROC is supposed.

CONCLUSION

SYNROC produced by means of the melting and melt solidification is similar to hot-pressed SYNROC. Almost the same mineral assemblage and radionuclides partitioning are observed. The main distinguish of the melted SYNROC from the hot-pressed one is the formation of powellite phase due to the oxidizing melting condition. It is necessary to reproduce the experiments on SYNROC melting under reducing condition to suppress the molybdate formation.

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17-12

SURFICIAL GEOLOGY AND PERFORMANCE ASSESSMENT FOR A RADIOACTIVE WASTE MANAGEMENT FACILITY AT THE NEVADA TEST SITE

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ABSTRACT

At the Nevada Test Site, one potentially disruptive scenario being evaluated for the

Greater Confinement Disposal (GCD) Facility Performance Assessment is deep post-closure erosion that would expose buried radioactive waste to the accessible environment. The GCD Facility located at the Area 5 Radioactive Waste Management Site (RWMS) lies at the juncture of three alluvial fan systems. Geomorphic surface mapping in northern Frenchman Flat indicates that reaches of these fans where the RWMS is now located have been constructional since at least the middle Quaternary. Mapping indicates a regular sequence of prograding fans with entrenchment of the older fan surfaces near the mountain fronts and construction of progressively younger inset fans farther from the mountain fronts. At the facility, the oldest fan surfaces are of late Pleistocene and Holocene age. More recent geomorphic activity has been limited to erosion and deposition along small channels. Trench and pit wall mapping found maximum incision in the vicinity of the RWMS to be less than 1.5 m. Based on collected data, natural geomorphic processes are unlikely to result in erosion to a depth of more than approximately 2 m at the facility within the 10,000-year regulatory period.

INTRODUCTION

At the Nevada Test Site (NTS), one potentially disruptive scenario being evaluated in the Greater Confinement Disposal (GCD) Facility Performance Assessment (PA) is post-closure erosion that would expose buried radioactive waste to the accessible environment (1,2). The GCD Facility is located at the Area 5 Radioactive Waste Management Site (RWMS) in Frenchman Flat, along the eastern boundary of the NTS. This closed basin formed since the middle Miocene (3,4). Alluvium in northern Frenchman Flat is late Miocene to Holocene, with most of the landscape covered with Quaternary alluvium. The RWMS is located at the juncture of three coalescing alluvial fan systems: the Scarp Canyon and Nye Canyon (SC-NC) fan piedmont from the northeast; the southern Halfpint Range and Massachusetts Mountains (SHR-MM) fan piedmont from the north and northwest; and the Barren Wash fan from the west. The Quaternary history of these three prograding fan systems is most relevant to the discussion of erosion and landscape evolution at the RWMS. This paper outlines the surficial geologic history at the RWMS and its implications for future landscape development.

SURFICIAL GEOLOGY

Preliminary geomorphic surface maps have been completed for northern Frenchman Flat. This 1:6000-scale mapping used aerial photographs and field mapping to determine boundaries and composition of map units. Criteria for defining the various geomorphic surfaces include relative geomorphic position, landform morphology, and degree of preservation of surface morphology (5, 6). Seven major geomorphic surfaces are recognized in northern Frenchman Flat. These surfaces range from early Quaternary to historic in age (Table I).

The most dramatic geomorphic features in northern Frenchman Flat are related to evolution of the SC-NC fan piedmont. It is characterized by broad areas of the oldest geomorphic surfaces (S2 and S3). Drainage patterns and clast provenance of the S2 and S3 surfaces indicate that Nye Canyon was the principal source for the alluvium issuing into northern Frenchman Flat in S2 and S3 time. Subsurface investigations indicate these deposition extended to the RWMS (7). Subsequent incision of these old surfaces began in either late S3 or early S4 time. Evidence for this is that S3 surfaces are the youngest fan surfaces entrenched and that terrace remnants with S4 surfaces are the oldest deposits preserved in the canyons. Presumably this incision of Scarp and Nye canyons was climate-driven during late S3 (glacial?) time. The deep fan-head incision means that these canyons continue to distribute water and alluvium to an inset fan (S4-S7 surfaces) downslope of the RWMS (Fig. 1).

Incision of the SC-NC fan piedmont has left the S2 and S3 surfaces as relict fan surfaces. Subsequently, surfaces of S4 and S5 age have formed at the RWMS as a result of: 1) internal drainage on the broad S3 surface; and, 2) drainage along the southeastern flank of the Halfpint Range being directed southwest along the boundary between the SC-NC and SHR-MM fan piedmonts. These surfaces are within the Halfpint Alluvial Fan (Fig.1) of Miller et al. (8). This area of deposition and the erosional landscape upslope is an active drainage subbasin, with most recent geomorphic activity limited to erosion and deposition along small channels (S6 and S7 surfaces).

Geomorphic mapping of the SHR-MM fan piedmont indicates a regular sequence of prograding fans with entrenchment of the older fans near the mountain fronts and

construction of progressively younger inset fans farther from the mountain fronts. At the RWMS, the oldest fan surfaces are of S4 and S5 age. More recent geomorphic activity (S6 and S7 surfaces) has been limited to erosion and deposition along small channels.

Although Barren Wash watershed drains a large part of western Frenchman Flat basin (8), only a small part of this large fan system is within the study area (Fig. 1). The particular fan of interest issues from the west around the southern tip of the Massachusetts Mountains and converges with the other two fan systems at the RWMS. Much of the fan consists of S3 and S4 fan remnants. As with the other two fans, the most extensive surface near the RWMS is of S5 age, with more recent geomorphic activity (S6 and S7 surfaces) being limited to erosion and deposition along small channels.

CHANNEL EROSION

Although geomorphic mapping indicates that the RWMS has been a point of net aggradation since at least S3 time, local incision and local aggradation have been active geomorphic processes even in the depositional portion of these alluvial fans. Detailed mapping of pits and trenches at and near the RWMS provide insight into the type and depth of past channel incision.

At the RWMS, 425 m of pit walls were mapped and sampled to obtain detailed near-surface information about the alluvium to depth of about 8 m (7). This conglomeritic to fine-grained alluvium represents deposition on the lower-middle to distal part of a large fan issuing into Frenchman Flat from the northeast. The alluvium displays characteristics typical of sediments deposited by sheet-floods, in stream channels, and as thin debris flows. Field observations indicate that the landscape at the RWMS has developed by planation by relatively shallow streams rather than by deep stream incision and backfilling. Cross-sectional measurements indicate that paleochannels are less than 1 m deep and typically less than 0.5 m deep.

The four trenches mapped near the RWMS are located at or below the intersection points of the SC-NC and SHR-MM fan piedmonts. Based on measurements of paleochannels and reconstruction of soil profiles at these trenches, the maximum depth of incision was approximately 1.5 m (9, 10). Most of the erosion was less than 0.8 m deep. These depths are similar to data from the pit walls at the RWMS.

CONCLUSIONS

The RWMS Facility lies at the juncture of three alluvial fan systems. Geomorphic surface mapping in northern Frenchman Flat indicates that since at least S3 time, the RWMS has been located in the constructional reaches of these three fan systems. Furthermore, mapping indicates a regular sequence of prograding fans with entrenchment of the older fans near the mountain fronts and construction of progressively younger inset fans farther from the mountain fronts. At the RWMS, the oldest fan surfaces are of S4 and S5 age. More recent geomorphic activity (S6 and S7 surfaces) has been limited to erosion and deposition along small channels. Trench and pit wall mapping found maximum incision in the vicinity of the RWMS to be less than 1.5 m. Based on collected data, natural geomorphic processes are unlikely to result in erosion to a depth of more than approximately 2 m at the RWMS within the 10,000-year regulatory period.

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17-13

CREDIBLE FUTURE CLIMATE FOR THE NEVADA TEST SITE FOR THE NEXT 10,000 YEARS

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ABSTRACT

During the mid-1980's, transuranic wastes were disposed of using the Greater Confinement Disposal (GCD) concept, which consists of 3m diameter by 37m deep boreholes located at the Nevada Test Site (NTS). Federal law (40 CFR 191) governing the disposal of transuranic (TRU) wastes requires the assessment of plausible future events and processes that may affect the performance of the disposal system during the next 10,000 years. Performance assessment indicates that changes in the present climate may impact the performance of the disposal system. The screening of processes expected to be affected by changes in climate indicates that roots associated with deep-rooting plants, such as juniper, may have a major impact on facility performance.

Presently, no deep-rooting plants exist in the immediate vicinity of the GCD facility, but may migrate to the area due to increases in precipitation resulting from climate change. The Climate Change Task was initiated for the purpose of defining possible future climate states, and to estimate the affect of possible future climate change on modeling parameters.

The purpose of this paper is to present background information on global paleoclimate records leading to the definition of possible future climate states based on the results of an analysis performed at the Lamont-Doherty Geologic Observatory at Columbia University. This paper does not attempt to link those climate states to a parameter such as precipitation which will in turn indicate conditions conducive to the growth of deep-rooting plants such as juniper. We intend to develop this link through existing literature which evaluates paleobotanical data collected from pack rat middens in the vicinity of the Nevada Test Site. After establishing the link and identifying the climate state that corresponds to the growth of deep-rooting plants, it will be possible to use the results of this paper to estimate the probability of that climate state occurring during the next ten thousand years at the GCD facility.

The results of this study indicate that climate at the NTS will be gradually changing from the interglacial state we are presently experiencing to a cooler interstadial state in the next several thousand years. However, the variability associated with that future interstadial state will increase dramatically during the same time period. This indicates that although the mean climate is not expected to change radically, any climate state from an interglacial to a full glacial is possible.

THE GREATER CONFINEMENT DISPOSAL PROJECT

In 1981, the DOE's Defense Low-Level Waste Management Program asked the Nevada Operations Office of the DOE (DOE/NV) to demonstrate the feasibility of the Greater Confinement Disposal (GCD) concept for disposal of Low-Level Waste (LLW) and certain high specific activity LLW (i.e. tritium) in an arid region. The GCD concept was so named because it provides greater confinement than near-surface burial. This greater confinement is needed for wastes for which near-surface burial is not an appropriate disposal method. Located in the Radioactive Waste Management Site at Area 5 of the Nevada Test Site, this disposal facility consists of boreholes approximately 3m in diameter and 37m deep. Waste is emplaced in the bottom 15m of the borehole and the remaining 22m from the top of the waste to the land surface is backfilled with native alluvium. The bottom of each borehole is approximately 200m above the water table. The average precipitation at the GCD site is approximately 10 cm/yr. Between 1984 and 1989, high specific activity LLW was disposed of in the GCD facility. In addition to this high specific activity LLW, a small quantity of transuranic (TRU) waste was disposed of in four of the GCD boreholes.

GCD PERFORMANCE ASSESSMENT

The regulation governing the disposal of TRU wastes, 40 CFR 191 (1), requires assessing the performance of the disposal system for 10,000 years. To determine the likelihood that the waste placed in the GCD boreholes will comply with the containment requirements of 40 CFR 191, Sandia National Laboratories has applied an iterative Performance Assessment (PA) based on the high-level waste PA methodology developed for U.S. Nuclear Regulatory Commission (2).

The application of this performance assessment methodology to the GCD facility consists of: i) conducting a PA with respect to the containment requirements of 40 CFR 191; ii) determining the processes and parameters that have the greatest influence on the PA results; iii) conducting site characterization activities and additional analysis to reduce the uncertainty associated with the most important processes and parameters; and iv) updating the PA models based on the results from site characterization activities.

In addition to analyzing the facility's performance under current hydrologic conditions, the PA has also analyzed the effect of three disruptive events. These three events (erosion, human intrusion, and climate change) are thought to have the highest likelihood of occurrence relative to all the events and processes evaluated during the screening process.

CLIMATE CHANGE FOR THE PA OF THE GCD SITE

It was beyond the scope of the preliminary PA to consider the evolution of global climate over the 10,000 year period of regulatory concern. Consequently, a simple and conservative approach to simulating the potential impacts of climate change was taken. If this simulation did not result in a modeled violation of performance objectives by itself or when combined with other conditions such as erosion, no additional analysis or gathering of data would be justified. If the simulation did result in modeled violations, additional data would be gathered and included in the modeling efforts, assuming the additional data would justify the relaxation of the original conservative assumptions.

The conservative approach used in the preliminary PA to estimate the potential effects of climate change consisted of modeling the system under a new steady state that began at the present and extended through the next 10,000 years. Climate change was simulated by increasing the current recharge rate by a factor of 20, and raising the level of the water table by 50m. These conditions, when combined with other parameters, resulted in calculated releases exceeding the limits set in the containment requirements of 40 CFR 191. Because of the potential violation of 40 CFR 191, and because the preliminary PA climate change condition was based on what were believed to be very conservative assumptions, it was decided to better define the credible future climate states at the GCD facility, and the effects of these states on the performance of the GCD facility. The Climate Change Studies task was created to define a range of possible future climate states, and to estimate the potential biologic, geomorphologic, and hydrologic changes that might affect the disposal system to these future climate states. This paper addresses only the definition of possible future climate states.

CHOICE OF METHODOLOGY

Three approaches have been considered to define the future climate at the GCD site: i) use of subjective opinions; ii) use of models to simulate future conditions, and

iii) use of paleoclimate data to infer past climate conditions and then project those conditions forward in time.

The first method of defining the future climate at the GCD is to ask the opinion of "experts." The formal process of asking experts their opinion is referred to as the "elicitation and use of expert opinion." The process typically requires a trained facilitator, experts, some preliminary exchange of information, the formal elicitation process and follow up activities. In the case of climate change, the knowledge which is being elicited has its roots in empirical and/or modeled data (the other two methods of predicting the future). Each expert may synthesize these two inputs into an opinion, but as the final opinion must be based on modeled or empirical evidence it was decided to make direct use of the underlying data, and not use this elicitation process.

The second alternative method of defining the future climate at the GCD is through the use of global climate models (GCMs). The development and execution of a GCM computer code requires understanding of a large number of processes and conditions. As a result of the uncertainty associated with each of these processes and conditions, the predictions made by GCMs vary greatly and are often in question. In addition, the uncertainty of applying a GCM to local conditions also argues strongly against the GCM approach. Consequently, it was decided not to use GCMs at this point of the analysis; but it will be necessary to consider their results to evaluate the impact of anthropogenic climate change.

As a third method, paleoclimate data can be used to identify patterns in past climate, and then project those patterns into the future. For example, on the time scale of hundreds of years, we can plot past average daily temperatures from historical records against time (i.e., a time series). On this plot we would observe average daily temperatures increasing and decreasing in a periodic manner reflecting the seasonal cycles of summer and winter. On the time scale of hundreds of thousands of years we could plot average annual temperature against time. As with average daily temperatures we observe patterns, but in this case patterns of warmer and cooler climates. These patterns cannot be created from historical records as with temperature, but must be developed from analysis of ancient records having some relationship to climate such as: fossilized pack rat middens; ancient lake levels; pollens trapped in sediments; and oxygen isotope ratios preserved in ice, cave carbonates and marine sediments.

We have chosen to use paleoclimate records of this type as the basis for defining possible future climate states by identifying the patterns contained in the records and assuming those patterns will continue into the future. A primary assumption required for the use of paleoclimate records is that the range of future climate states will be bounded by those climate conditions that have occurred in the past. Since these records do not include the effects of anthropogenic climate change, attributed to increased amounts of greenhouse gases in the atmosphere, it may be necessary to adjust the results in some manner to incorporate the anticipated effects of anthropogenic climate change. We have not completed the research required to understand what these effects may be, and have not settled on a method of incorporating these effects into our analysis. Therefore, anthropogenic effects will not be considered in this paper.

RECORDS OF PAST CLIMATE

Long-term, direct records of temperature or precipitation at the GCD facility do not exist. However, there are a number of surrogates for past climate contained in paleoclimate records that are useful. For example the width of tree rings serves as a surrogate for temperature, runoff, precipitation, and soil moisture for periods of several hundred years. Fossil pollen types and concentrations are a surrogate for temperature, precipitation, and soil moisture for periods of hundreds or in some cases thousands of years. Oxygen isotope ratios from layered ice cores give a continuous measure of snow accumulation and a surrogate for temperature for periods in excess of several hundred thousand years. Calcium carbonate precipitated from natural spring water results in a continuous record of temperature dependent oxygen isotopes contained in regional groundwater for periods of hundreds of thousands of years. Marine sediments provide a continuous record of planktonic and benthic fossils, including the oxygen isotope composition of fossil remains, which are used to infer sea-surface temperature and global ice volume for periods of hundreds of thousands of years. In a similar manner, lake sediments yield a continuous isotopic record over hundreds of thousands of years that can be used to infer temperature,

precipitation and wind direction (3).
In an earlier paper (4), a number of paleoclimate records were considered while searching for a record that could serve as a surrogate for past climate at the GCD facility. Three of those records are included here to illustrate information contained in the records that will be important later in the paper. Each of the records uses d18O as the surrogate for climate, which by convention is the relative deviation of the ratio (18O/16O) contained in the sample, to the (18O/16O) contained in standard mean ocean water (SMOW). The first record considered is from marine sediments from locations in the Atlantic and Pacific, and consists of d18O plotted against age (5). The second is a record of d18O versus age contained in ice layers from the Greenland ice sheet (6). The third is a record of d18O versus age contained in carbonates from spring deposits in Devils Hole, Nevada (7,8).
The Marine d18O record, shown in Fig. 1, is a composite record from several sediment cores. These cores are from the western and eastern equatorial Pacific, the Caribbean Sea, and the North Atlantic.
The cores were sampled and dated on an interval ranging from 2000 to 6000 years, correlated using nonofficial biostratigraphy and paleomagnetic stratigraphy, and combined to build a composite record representing the last 1.88 million years (5). As the cores were taken from such varied locations throughout the world the composite record is thought to capture the global nature of past climate change.
Figure 2 illustrates the results of the joint European-Greenland Ice-core Project (GRIP) which during 1990-92, drilled an ice core in central Greenland (6). A preliminary chronology for the core was developed with the first 14.5 thousand years before present based on examination of the ice stratigraphy. The remainder of the core was age-dated using ice-flow modeling. This record yields a higher resolution than the Marine record as a result of a finer sampling interval. Assuming the age dating is correct, these results indicate that at least in Greenland, climate can vary at a much faster rate than indicated in the Marine record.
Figure 3 illustrates the results of the analysis of Winograd et al. (7,8) of two cores containing layered calcium carbonate precipitated from spring water. The oxygen bound in the calcium carbonate provides a continuous record of d18O fluctuations for the time period between 60 to 566 thousand years before present. Although the actual source of the water and travel time from the spring are uncertain, the record is expected to ultimately reflect the d18O of precipitation on the Spring Mountains of Nevada (9). As the GCD facility is approximately 110 km northwest of Las Vegas, and Devils Hole is located approximately 115 km west-northwest of Las Vegas, the Devils Hole record is the closest long term paleoclimate record to the GCD facility.

INFORMATION CONTAINED IN THE RECORDS

Prior to beginning the analysis of the records, it is appropriate to consider what information will be needed to define plausible future climate states. First, we need to establish a relationship between the d18O ratio contained in the records, and a characteristic of climate such as temperature. Second, we need to establish what state, or states, climate will take during the regulatory period, their duration if there is more than one, and finally any associated uncertainty.
Concerning the relationship between the ratio of d18O and temperature, the ratio of oxygen isotopes is strongly correlated with mean temperature (10). But, the relationship between d18O and temperature is not constant for different types of paleoclimate records, and also varies with geographic location. A comprehensive description of this behavior can be found in the paper by Rozanski et al. (11). Ignoring the technical details and simply accepting that mean temperature is proportional to d18O, it is apparent that the paleoclimate records contain a history of past climate, with mean temperatures varying between warm periods (interglacials) and cold periods (glacials).
Examination of the Marine record presented in Fig. 1 illustrates this behavior with a pattern that appears to be remarkably consistent for long periods of time. Milankovitch postulated that variations in climate were correlated to cycles in the Earth's orbit around the Sun; with these cycles having periods of 19, 24, 43, and 100 thousand years (12). Spectral and regression analysis performed on marine records indicates that a large portion of the variability seen in these records agrees with the periods defined by this theory (13). Since the primary frequencies attributed to the theory are several times as long as the regulatory period for the site, it is convenient to classify them as low frequency patterns. We consider the

Marine record of Fig. 1 to be a source of low frequency paleoclimate behavior. On the other hand, the Greenland Ice record of Fig. 2 appears to vary much more quickly, oscillating between upper and lower bounds in periods as short as tens of years. Consequently, we consider the Greenland Ice record to be a source of high frequency paleoclimate behavior.

The combination of these two records illustrates the existence of low and high frequency changes in climate that have occurred throughout the 250 thousand year period they share. We believe the low frequency behavior seen in the Marine record is a good approximation of the average global climate state during the last two of million years, while the Greenland Ice record serves to illustrate the extreme variability or high frequency variation that can occur. As the records indicate that past climate has included both high and low frequency patterns, any effort to estimate future climate states must incorporate both types of behavior. Therefore it will be necessary to combine records, as none of the records we have examined, or have presented here appear to be adequate for our purpose.

Examination of the Devils Hole record of Fig. 3 indicates this record contains patterns that are similar to those seen in the Marine record of Fig. 1. Although this is far from a quantitative proof that the patterns seen in global records are representative of local climate conditions, it does indicate that changes in climate which lead to full glacial conditions in the northern hemisphere also impact climate in the vicinity of the NTS, and the GCD facility.

DEFINING FUTURE CLIMATE STATES

In a recently released paper, Kukla and Gavin (14) document the results of an extensive investigation of paleoclimate records from locations throughout the world. Although the paper discusses many facets of paleoclimate records, the portion that is concerned with the development of a mean or average "glacial climate cycle" from paleoclimate records is of the greatest interest. In that section of their report, the authors chose seven different paleoclimate records considered to have high sampling and time resolutions on the order of approximately 500 to 2000 years. Included in this group are records based on marine sediments, ice cores, and pollen. As none of these records share the same relationship to temperature, they were transformed into a dimensionless measure of climate referred to as the Climate Index (CI) (15).

Recognizing the pattern of glacial/interglacial cycles seen in the paleoclimate records, Kukla and Gavin (14) chose to separate each of the records into individual glacial cycles, each illustrating a glacial cycle from its onset, to its termination. After the terminations of each glacial cycle where identified using the times established by Imbrie et al. (10), the records were broken into individual cycles resulting in a total of thirteen cycles from the seven paleoclimate records. By assuming the time associated with the beginning of each cycle was zero, the cycles were stacked creating an overlay of glacial cycles capturing the variability illustrated by the paleoclimate records. This result is presented in Fig. 4.

In addition, the authors established a relationship between the Climate Index and five principal Pleistocene climate states, allowing the division of the Climate Index, and consequently the glacial cycle into five climate states. These states are: i) interglacial, greater that 190 CI ; ii) temperate interstadial, greater than 163 CI and less than 190 CI; iii) interstadial, greater than 137 CI and less than 163 CI; iv) stadial, more than 109 CI and less than 137 CI; and finally v) full glacial, less than 109 CI.

By assuming that each of the thirteen cycles is equally likely to occur, it is possible to calculate an average climate for each the glacial cycles, and in addition, establish an upper and lower bound corresponding to the warmest and coolest climate index as illustrated in Fig. 5. If we assume the most recent glacial cycle terminated some twelve thousand years ago, we are able to locate ourselves within the overall glacial cycle, and from that point estimate the average climate state for the next ten thousand years. Examination of Fig. 5 indicates that the mean climate corresponding to the our present location within the glacial cycle, corresponds to the border between an interglacial and temperate interstadial. Consequently, the mean climate for the next ten thousand years, which will end at the point corresponding to twenty two thousand years from now, indicates a general cooling trend, but with a very large increase in the variability associated with the mean. This variability implies that in the next two thousand years or so, we may experience the continuation of an interglacial similar to what we are experiencing

today, or we may experience episodes of full glacial conditions. We are unable to reduce this level of variability without further investigation of the corresponding paleoclimate records. Therefore, our only recourse is to assume that each of the climate states within the upper and lower bounds are equally likely. As a result of this assumption, a uniform probability density function is the best choice to model the uncertainty associated with the mean climate state.

CONCLUSIONS

Climate change could impact the performance of the GCD facility. The goal of this activity is to define plausible future climate states for the period of regulatory concern, and estimate the effects on model parameters.

Having decided to use the patterns associated with past climate as depicted in paleoclimate records, we began an examination of the available records for use in our analysis. During that study, we recognized that some records (such as the Marine) contained low frequency glacial/interglacial cycles, and others (such as the Greenland Ice) contained high frequency glacial/interglacial cycles. As a result of this behavior, we concluded that no single record was capable of supplying sufficient accurate information to meet our needs, and it would be necessary to somehow combine a number of different records before any conclusions could be drawn. Kukla and Gavin (13) combined seven different paleoclimate records to create a portrayal of a glacial climate cycle. These results allow us to estimate the behavior of the average glacial cycle, albeit without the influence of anthropogenic climate change. In addition the cycle also allows an estimate of the variability associated with that average glacial cycle. By assuming that the last glacial period ended some twelve thousand years ago, we are able to use this information to estimate the mean climate state for the next ten thousand years along with its variance.

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17-14

A STOCHASTIC COMPARTMENT MODEL FOR NUCLIDE TRANSPORT IN THE FRACTURED POROUS ROCK

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ABSTRACT

A stochastic model using a Markov process is proposed in this study with which the need for solving a set of partial differential for nuclide transport can be avoided. Once the single planar fracture in the porous rock is thought of as a series of finite number of compartments and such geologic system is assumed to have Markov property, since the Markov process requires that only the present value of the time dependent random variable (i.e., time dependent number of nuclides or concentration in certain compartment) be known to determine the future value of random variable, nuclide transport in the medium can be modeled as a continuous time Markov process. Through the study we introduce comparative results of the expectation and the variance of nuclide distribution describing the nuclide transport with chain decay of arbitrary members in the single fractured porous rock. To this end some comparisons between available analytical solutions are made.

INTRODUCTION

It is generally conceived that fractures in the bedrock surrounding a radioactive waste repository can play a very important role in the nuclide transport. In such bedrock, usually the permeability of the fracture is greater than that of the porous rock matrix itself offering a pathway for nuclide transport to the biosphere. Since Neretnieks (1) proposed an analytical model for nuclide transport in rock medium showing that the diffusion into the rock matrix from the fracture is very important mechanism to retard the nuclide, the conceptual model considering the fractured porous rock medium as the porous rock matrix having single planar fracture in it has been regarded as a convenient way to describe nuclide transport and has been discussed by many researchers in recent years (2).

Meanwhile, Nassar et al. (3) have applied continuous time Markov process model to stochastic formulation of chemical reaction in a continuous flow system. Knighton and Wagenet et al (4) recently have applied continuous time Markov process model to describing chemical movement through the soil.

Using the similar concept, the fractured rock medium can be considered as a series of discretized compartments of a finite number and the concentration of nuclide or equivalent number of nuclides can be considered as a time-dependent random variable. A random variable X(t) represents the state of the system at a time t, which is defined as a nuclide transition through a series of compartments of the fracture. Then the event X(t) = j denotes the position of a single nuclide to be compartment of the fracture (or state) j at time t. If the nuclide makes a transition from any compartment i to some other compartment j within a small time increment Dt, then there is a finite probability associated with that event.

The transition probabilities Pij(t) of nuclides, by which the nuclide distribution in the medium can be obtained are then formulated by utilizing the Chapman-Kolmogorov equations with the intensities of transition L(t), which are directly related to the processes affecting nuclide transition between and out of compartments.
A nuclide in a given time interval could make a transit to any compartment by groundwater flow or could form and disappear from any present compartment due to radioactive decay or molecular diffusive loss out of the system, i.e., from the fracture into the rock matrix through the fracture wall. All these processes are obviously conditional only on the present location of the radionuclide regardless of its previous history facilitating Markov conceptualization.
MODEL
As mentioned, in each compartment there are processes that influence the state of the nuclides at time t. Also the transition probability from a compartment to another is affected by the transition intensities. In a short time interval, nuclide transition between and out of compartments can be assumed to be governed deterministically. Therefore, once the flow rate of the groundwater between the compartments is known, the transition intensity of a nuclide from compartment i to compartment j at time t due to groundwater flow can be represented, under the assumption that groundwater is mixed well with nuclides instantly, as
Eq. (1)
where lij(t) and Qij(t) are the transition intensity and the volumetric flow rate of groundwater from compartment i to compartment j at time t, respectively, Vi is the volume of compartment i, and fi is the porosity of compartment i.
The transition probability is governed by the Kolmogorov forward differential Eqs. (5).
Eq. (2)
subject to the initial condition
Eq. (3)
which give the relation between the rate change of the transition probability and the transition intensity. A constraint of transition probability of Markov process is
Eq. (4)
Under the assumption of linear isotherm sorption of nuclides in the compartment i, lij(t) should be replaced by l ij(t) / Rfil where Rfil is retardation coefficient in the fracture. When the diffusive loss into the rock matrix in the direction perpendicular to the fracture from the fracture is considered, we should consider corresponding transition intensities for this loss process. The transition intensity for diffusive loss ldiffil(t) into rock matrix can be expressed as (6)
Eq. (5)
where Dbi is pore diffusion coefficient, bi is half-width of the fracture, Rpil is retardation coefficient in the rock matrix, and ll is radioactive decay constant.
Under the assumption that the transport is considered to be made only between adjacent compartments, the transition intensity matrix coefficients can be finally represented as
Eq. (6)
If we let Xil(t) be the random variable representing the number of lth nuclides in compartment i at time t, the distribution of all nuclides in the fracture and within any compartment can be represented as a function of the transition probability of each lth nuclide. For all N compartments,
Eq. (7)
Random vector Xl(t) can be broken again intoYl(t) and Zl(t):Yl(t) represents the number of survived nuclides by the time t, which were originally in the system at time O, and Zl(t) represents the number of nuclides that have entered the system during the time interval [O,t) and have survived in the respective compartment at time t.
If mil(O) denotes the number of nuclides in i at time O, at time t, each of the mil(O) must be either in one of the N compartments or disappear due to diffusion into the rock matrix and radioactive decay. Therefore, for a given mil(O), the distribution of Yjl(t), the remaining survived nuclides, at time t has multinomial distribution according to Eq. (4).
Now at any time t between time interval [O, t), suppose that nuclides flow into each compartment at the rate of z(t) per unit time. Then,

Eq. (8)
where Qi is volumetric flow rate of groundwater into the compartment i and Coil(t) is source concentration in Qi of the lth nuclide at time t, which is given by the solution of Bateman's differential equations as, with l ranging from 1 to the number of decay chain,
Eq. (9)
where is source concentration of the lth nuclide at time 0.
In case of imposing band release boundary condition at the inlet of the system, i.e., for the first compartment, for band release time q Eq. (8) replaced by
Eq. (10)
where U(t) is Heaviside step function.
Similarly, by analogy, at any time t between time interval [0, t), the change rate of nuclides into daughter component from its parent nuclide xil(t) is represented by
Eq. (11)
where Cil(t) is concentration of nuclide l in compartment i at time t.
As soon as a nuclide which fed freshly by supplying at the inlet of the system (Eq. (8)) or by forming due to the radioactive decay of its precursor enters the system (Eq (11)), it may begin to transfer to one of the other compartments at once or may disappear by either diffusion into the rock matrix or decay. Here we can assume that all nuclides in the system behave stochastically and independently with one another. Therefore, vector Zl(t) is the outcome of sequences of events of input, transition between the compartments, and survival from the decay and diffusion.
When both xil(t)dt, the number of new nuclides which entered a compartment during dt and d(t) xil (t) dt, the number of new nuclides that form in the ith compartment from the precursor nuclides in time interval dt have successfully entered compartment j, it must have a respective probability of Pij(t-t) either entering or remaining there. Therefore a binomial distribution can be formed for these new nuclides.
Therefore, finally we can get the expectancy and the variance of Xjl(t) from the distribution of the number of nuclides remaining in each compartment that have survived, as the convolution of the two independent distributions Yjl(t) and Zjl(t):
Eq. (12)
Eq. (13)
Consequently Cjl(t), the expected value and variance of concentration of the lth nuclides in j at time t can be expressed, respectively, as
Eq. (14)
Eq. (15)
Since the model presented here must be sensitive to the number of compartments of the system, in other words, the numerical dispersion phenomena will increase according to decreasing N, some quantitative estimation in the relation between N and the dispersion coefficient must be prerequisite. By comparing the one-dimensional system which is divided in to N compartments of equal size with the only one process of advective transport due to groundwater flow of constant rate and a common advection-dispersion equation for homogeneous porous medium that approximated by the centered difference equations by means of Taylor expansion, following approximate equivalence will be obtained:
Eq. (16)
Where L is the length of the system.
Since no analytical solutions in closed form to the decay chain transport with rock matrix diffusion are yet available, first we introduce a computational result for the case of no diffusion into the rock matrix, which yields one dimensional model for porous medium with decay chain. For simplicity, we further assumed the retardation coefficients of the lth member in the whole medium are equal, i.e., Ril = Rl for all l and for all compartments.
For the purpose of illustration only three members decay chain is considered and the data used are as follows:
DATA
A result by our continuous time Markov process model was compared against those obtained by Lung (7) through his code UCB-NE-40, designed for general solution for a chain of arbitrary length through the semi-infinite homogeneous medium for infinite release boundary condition.
In Fig. 1, the concentration profiles of the first, second, and third member, normalized to initial concentration of the first member at time 0 as a function of

distance at times equal to 100 and 1000 years, respectively, by the present model are plotted showing good comparison of corresponding analytical solution: one is for the retardation of 9352 at the time of 1000 years and the other for the retardation of 100 at the time of 100 years. Even though the retardation coefficients are the same for all members, the profiles at time 1000 years indicate that the third member of the decay chain can travel further along the fracture than the first and second since the half-life of the third member is approximately 15 and 436 times longer than those of the first and second, respectively. Then it is to be expected that the profiles of C1 and C2 at each time will tend to lag behind those of C3.
Breakthrough curves of the three members are also plotted showing relatively good agreement to corresponding analytical solution in Fig. 2 as a function of time at distances of 20 and 50 m, respectively.
As another specific example illustrates the results in case of consideration of the rock matrix diffusion with constant groundwater flow rate throughout in the fracture. The injection of the nuclide, as a source term takes place only in the first compartment, as is previous case, however, decay ingrowth is not considered in this study.
For single-fracture rock matrix system with process of diffusive loss into the rock matrix form the fracture, the parameter values used are as follows:

DATA

Relatively good agreement of concentration profile is obtained when compared with the solution of the deterministic differential equation governing the nuclide transport in the fracture with rock matrix diffusion, available in many literatures (2). As seen in Fig. 3, the results of the present model for values of the retardation coefficient of 5.0 both for the fracture and the rock matrix and decay constant of 0.023 yr-1 and no retardation and for another case of no decay are compared to the equivalent analytical solution adjusted with the equivalent dispersion coefficient in the fracture. For the latter case, the concentration profile in exact case for no dispersion in the fracture is also shown for reference, because it is also of interest to compare the analytical solution for the case of no dispersion in the fracture with the results obtained by present model in order to show the numerical dispersion due to spatial discretization of the medium.
As for the size of discrete time step T, the accuracy of the solution is shown to be improved when smaller size of discrete time is used (the result is not shown here). Figure 4, which is calculated at the distance of 20 m for three cases with respect to the retardation coefficient and the decay constant, exhibits relatively good agreement between continuous time Markov process model developed in this work and the deterministic analytical solution. However, some discrepancy appears in all breakthrough curves in this figure, which may be probably due to still insufficient discretization of space and time.

CONCLUSION

Through this study a stochastic model using continuous time Markov process for nuclide transport in the fractured rock medium has been developed.
Once the single planar fracture in the rock matrix is considered as a finite number of compartments in series, the medium is continuous in view of various processes associated with nuclide transport but discrete in medium space.
By calculating the transition probability for nuclide from the transition intensity between the compartment utilizing Chapman-Kolmogorov equation, the expectation and variance of nuclide distribution for compartment or fractured medium is obtained. Some comparisons between available analytical solutions have been made showing comparatively good agreement for all cases.
Even though the model developed in this study was shown to be sensitive to the number of discretized compartments showing numerical dispersion as the number of compartments increases, with compensating of dispersion coefficient, the present model is considered to agree well to deterministic analytical solutions. However, some discrepancy appears especially in the breakthrough curves for the case of considering the rock matrix diffusion, which may be probably due to insufficient discretization of both space and time.

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17-16
HUMAN INTRUSION
Some Aspects of Regulation and Assessment of Future Human Actions at Radioactive Waste Disposal Sites

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ABSTRACT
A presentation in some detail of the problem of human intrusion is valuable. The issue is related to fundamental reasons why society disposes of the waste. If these questions are not addressed early in the licensing process, they are likely to turn up later as problems for both the waste management and the regulator.
Intrusion scenarios may include individuals who receive high doses, in excess of dose limits. This is a consequence of the strategy of waste isolation which cannot be rejected solely on the basis of the intruder's dose.
A logical presentation of the interdependence of some factors such as environmental consequences versus area restrictions makes it possible to describe this aspect of the licensing as a balanced decision.
An optimization study of the value of an information conservation system may be helpful in bringing into the open some important assumptions made in the human intrusion scenario. Some issues may be worthwhile to discuss: the problems of a large number of minor disturbances, of repeated intrusions and protection of a hypothetical drilling crew versus protection of the general public at risk as a result of the intrusion.
In licensing a repository with a life-time of thousands of years it is impossible to completely separate the responsibility of the waste management and the regulating system. However, it is the responsibility of the regulator to define its role in the process, where this could influence the environmental impact of the repository. This is also important in order to make the process as coherent and transparent as possible.

FUNDAMENTAL CONCEPTS

In the political and technical discussion of high level waste (HLW) disposal, several fundamental concepts must be dealt with, such as the potential danger of plutonium in the hands of future groups of people, even future governments. The same plutonium, however, can also be seen as a value to groups in the future. Thus plutonium may be treated as a detriment in one analysis, and as an asset in a another. Still another and perhaps predominant view is that plutonium as all other radioactive material is toxic and for that reason should be kept separated from the biosphere.
Some of the fundamentals of radioactive waste disposal may not be possible for a regulating agency to define by itself without a dialogue with political decision-makers. If for instance it is considered important to restrict the use of plutonium not only geographically, which is the rationale behind the Non-Proliferation Treaty, but also to restrict access to plutonium for governments in a distant future, then very deep boreholes (VDH) might be the most important alternative of the deep geological disposal options simply because this option could make the waste most difficult to retrieve.
The considerations above are ultimately political in nature, and it must be up the regulator to determine what decisions are necessary so that the prerequisites for a coherent process of waste disposal are at hand.
It follows that it is not possible to avoid the sense of illusiveness felt by many

about the idea of human intrusion, if some of the underlying concepts are not reexamined and given a stringent treatment.
It may therefore be necessary to have a set of fundamental, explicitly given, assumptions which defines society's motives for disposal of the waste. In defining these motives, a strategy should be presented that includes issues like the need for and the value of retrievability of the waste and repairability of the repository, including a description of the waste's potential future value or detriment as assessed by society today. Other concepts, mainly technical but also philosophical in nature can be expected to be decided upon by the regulator, and a few are presented below.
Waste Disposal Strategies and Protection Principles
For protection against toxic wastes, it is in principle possible to have two mutually excluding strategies exemplified by the expressions:
1 dilute and disperse, and
2 isolate and contain.
A complete radiation protection strategy also includes the three principles of Justification, Optimization and Dose Limitation. The first principle must be addressed already in the licensing process of the practice in which the waste is produced. Most often Optimization, the second principle above, leads to a choice of the second of the above mentioned strategies, isolation and containment in the form of a waste storage facility. The third radiation protection principle ensures the protection of single individuals in cases where the collective dose is deemed acceptable, but it is distributed in such a way that high doses are received by single individuals who may not receive all the benefits of the practice. According to ICRP, the principle of dose limitation should be regarded as a boundary condition to the process of optimization.
The principle "dilute and disperse" may seem provocative but releases to the environment may occur for various reasons:
Unavoidable acceptable releases. Almost any practice which produces waste - as well as waste handling itself - includes releases of radioactive material to the environment. If the practice is deemed justified, the regulator defines the acceptable levels of releases. In this framework, however, releases are not always seen as a conscious waste disposal strategy.
In some circumstances, and for some short-lived radionuclides, dispersion can in some cases (such as certain medical practices), be shown to yield the lowest collective and individual doses.
In some cases releases of short-lived nuclides may not be the option which gives the lowest collective dose but may still be acceptable if i) the practice which causes the release has been considered justified and ii) if it can be shown that alternative disposal options would cost more than reasonable (in the eyes of the regulator).
High individual doses would be avoided if the first strategic view were taken and the toxic waste diluted and dispersed, for instance into the sea. This strategy is often rejected by society on grounds that is it unjustified or that it would represent an under-optimized alternative. In most countries, dispersion of radionuclides with half-lives of many years into the sea is not a realistic alternative.
In the environmental impact assessment of a radioactive waste repository design, the special case of intrusion into the repository, may present a possibility for high individual doses to individual, namely the intruder. However safe the repository is designed, intrusion can always be imagined, sometimes by considering simple practices such as drilling for water, sometimes by taking into account extremely improbable events.
The conclusion is therefore that when society, on grounds of justification or optimization, chooses the option of waste containment in the form of a storage, the hypothetical scenario of intrusion followed by doses to the intruder in excess of dose limits cannot be used as a constraint the same way as in normal standards for building X-ray facilities in hospitals or for determining release limitations for nuclear facilities where doses are continuously monitored. On the contrary, the high dose to the potential intruder flows directly from the strategy of waste isolation in a repository.
Intrusion or other human actions that may influence the performance of the waste repository must still be a part of the total safety assessment. But it follows from

the above reasoning, that the threat of intrusion must be balanced against the probability of the intrusion in the regulation, since otherwise dilution would be the only remaining option.

SOME DEFINITIONS RELATED TO THE INTRUSION CONCEPT

In the absence of a definition of the human intrusion concept, almost any scenario leading to some radiation dose to man can be said to constitute a human intrusion or human action scenario which disturbs the repository. It is therefore worth while to establish a common international basis for definitions and judgement of human intrusion scenarios.

The following sections are dedicated to defining some necessary concepts for human actions at radioactive waste disposal sites. It is assumed that an extended analysis covering human intrusion is carried out in two steps:

a first phase consisting of an assessment of the repository performance in the absence of intrusion.

a second phase where human action disturbs or changes the outcome of the initial performance assessment.

It is assumed that performance assessment criteria exist for the first phase. The present task can therefore be described as finding criteria necessary to cover the additional, and separate, second phase. Human actions are seen as a broad concept, not restricted to human intrusion into the primary disposal. It may also include actions far from the disposal site, but close enough to influence the performance assessment. In some cases, like the Swedish repository for low and intermediate level waste, SFR, intrusion may be an important siting factor. This does not prevent that the analysis is made in two steps as mentioned above.

Definition of an Extended Repository Limit

Radioactive disposal concepts imply that the physical and chemical form of the radioactive waste, the repository design and technical barriers, in addition to natural barriers of the site, guarantee a sufficient protection to compensate for the dangers of the waste.

In this formulation, true for most repositories, part of the environment can be considered an integral part of the disposal concept. Sometimes, it may therefore be useful to define a spatial limit outside which human actions would not influence the outcome of a safety analysis. In theory, this limit is defined as a three-dimensional surface around the repository, where doses would be low enough to meet regulatory criteria for the undisturbed repository function. The limit of the extended repository cannot be expected to have the same meaning for all sites. Even for a specific site it may be difficult to define such a limit, but the concept may be useful as a theoretical reference in the discussion.

Restrictions

It should be observed that the range mentioned in the preceding section coincides with the area, or volume, where society may impose restrictions on various forms of activities. Even if the assessment does not rely on such active constitutional restrictions, it can be assumed that some records will be kept in some form for all high level radioactive waste repositories. Keeping such information available can be seen as a passive form of institutional control.

The human intrusion assessment serves to improve the repository concept. When choices are made between different concepts, the total detriment to society should be considered including the detriment of the restrictions described above. Thus, the performance of a repository cannot be improved simply by including a larger part of the environment in the repository concept. Also information conservation through long periods of time implies a cost to society and must have a weight in the assessment of costs and benefits of various solutions of the total problem, involving both radiation protection and use of the land above the repository. Some problems of optimization of the protection is discussed in the next section.

OPTIMIZATION OF RADIATION PROTECTION FOR HUMAN INTRUSION

Optimization as an important, perhaps the most important, concept in radiation protection was suggested in ICRP 26 in 1977 and in ICRP 60 in 1991 (1). It was pointed out that keeping doses as low as reasonably achievable was a powerful tool in radiation protection and that dose limits were to be seen mainly as a limit for the optimization procedure. Governments were advised that, for the activities where uses of ionizing radiation were justified, regulations should ensure not only that dose limit rules were met, but also that efforts were made to reduce individual doses further below those limits.

Both regulator and waste management could benefit from optimization studies, not least because all the underlying concepts will come under scrutiny in the process of determining the inherent factors in the calculation, the numerical result of which may be of limited value compared to the process itself.

Optimization categories for nuclear repositories

It is pointed out in ICRP 46 (2) that optimization of protection is a broad concept in waste management and it can apply at four levels:

A. Comparison of design alternatives for a specific facility such as a waste repository.

B. Comparison of different disposal options for particular waste streams.

C. Comparison of different overall management systems for particular waste streams.

D. Comparison of complete waste management systems, including conditioning, storage, transport and disposal alternatives for a given source or practice.

The global view should never be forgotten, so that for instance doses from waste handling should be added to doses from the repository in a distant future in the optimization.

Optimization may be used to identify the most cost-effective operation and the best (undisturbed) repository. What is balanced in the optimization described here is a special case of A above: the cost of establishing an information conservation system against the potency of such a system to prevent doses to man by deterring inadvertent intrusion. In principle, other costs could enter the calculation, such as choosing a site for which a repository would cost more, but that has not been taken into account in this presentation.

In this case, actions by individuals or groups must be described as an element in the calculation. Such calculations are always problematic, but they are easier for a shallow land burial where analogues may exist (e.g. experiences from waste dumps).

It is worth noting that, for a given repository, the optimization of the benefit from the information system may not have much in common with the protection optimization of the undisturbed repository itself. The difficulties are not necessarily less but they are different.

A NUMERICAL EXAMPLE OF AN OPTIMIZATION CALCULATION

In the following example, an optimization study of intrusion is described for a high-level waste repository. The calculation is meant as an illustrative numerical example of the calculation, and a demonstration of the uncertainties involved. The time span for the optimization may be assumed to be 10,000 years.

Probability of Intrusion Through Loss of Memory

Some estimates of the probability of intrusion has been given in work presently carried out by Sandia National Laboratories for the US Department of Energy's Waste Isolation Pilot Plant (WIPP) in New Mexico (3). Panels of experts have been asked to estimate the probability of inadvertent intrusion by future societies up to 10,000 years into the future. Some panelists have estimated the probability of intrusion to about 0.1 in 10,000 years. This is for a case where nothing denotes the location of the site. Another panel discussed to what extent information at the site (markers) could be a remedy for loss of knowledge. The effectiveness of markers, that is the ability to deter inadvertent intrusion, was believed by some panelists to be near 0.9 for 10,000 years and higher (0.99) for the first 1,000 years. With these estimates, and for the period 1,000 years the probability of intrusion can thus be calculated to be 0.1 * (1-0.99) = 0.001, for 10,000 years: 0.1 * (1-0.9) = 0.01.

In another site-specific study (4) regarding a repository in Boom clay at Mol, Belgium, the authors themselves assigned values to probabilities for loss of information, interest in drilling and a geometrical factor giving the probability of drillers hitting the repository. They combined these probabilities to arrive at a probability for inadvertent intrusion of less than 0.001 for 2,000 years after closure. In that work, probabilities are attributed to intrusion as far as 250,000 years into the future.

It is assumed, in the example below, that direct intrusion, in the absence of an information conservation system, has a probability of 0.1 over 10,000 years.

Individual and collective doses from intrusion scenarios

In a study of doses from inadvertent drilling intrusion into the repository, made for the Swedish Radiation Protection Institute (5), intrusion scenarios with high doses are presented for direct hits of a fuel waste canister followed by routine examination of cuttings in the field, and even higher doses for close inspection of samples in a laboratory (from inhalation as the sample is cut). No probability is

given for either mode of intrusion, except that intrusion into a waste canister or its backfill material is considered "very low". Doses are received by one or a few members of the drilling crew and laboratory personnel.
The collective dose in the example is taken to be such that the probable outcome of each intrusion or disturbance is one case of fatal cancer (20 person-sievert) for the members of the intrusion (drilling) crew and that the damage done by the intrusion in terms of a dose received by a larger number of people does not exceed nine times this value, so that the total collective dose is taken to be 200 person-sievert.
There can be no strong rationale for this value for the collective dose as a consequence of the drilling scenario. But one line of argumentation could be that if the collective dose was higher, it would be very easily detected (from acute radiation effects in the crew) and knowledge about the repository would be reimposed upon society. Such a reimposed knowledge might limit the damage made by the intrusion to the intrusion crew.
The value 0.05 excess cancer deaths per person-sievert is assumed for simplicity. In a detailed calculation, different risk estimates should apply to members in a drilling crew and a whole city population with a different age distribution. The estimate of this value by international expert organizations such as UNSCEAR (United Nation's Scientific Committee on the Effects of Atomic Radiation) and ICRP has changed over the years, but the estimate has always been thought to lie within the interval 0.01 to 0.1 cancer death per person-sievert.
It is assumed, in the field of radiation protection, that efforts are made to lower the dose as long as such efforts are deemed cost-effective. This assumes that the practice has been approved by the proper authorities and that doses are below the relevant dose limits. The limit where dose reduction is regarded barely cost-effective may be given in USD per person-sievert (1 sievert = 100 rem). This limit is often taken to be about USD 100,000 per person-sievert, although both higher and lower values are employed in individual cases.
If the probability of death is 0.05 per person-sievert, a value can be obtained giving the reasonable cost of increased safety in terms of cost to be taken to save one life (on the average). This reference cost may the be used by individuals and society to direct safety measures where these are most cost-effective. A combination of the death risk of 0.05 per person-sievert and the marginal cost of USD 100,000 per sievert yields 100,000/0.05 = USD 2,000,000 per saved life.
The result: The Cost of the Information System
The result of the calculation presented in Table I is that if intrusion can be avoided by information conservation in some form, it would be worth USD 2,000,000 to establish such a system. It can be seen that if the intrusion probability is set to 1 rather than 0.1 in the above calculation the reasonable effort for information retention would rise to USD 20,000,000. The somewhat arbitrary values in the calculation are discussed in section 3.5.
In view of the conceptual problems mentioned in the preceding sections, it must be kept in mind that the calculations are made assuming certain scenarios, and that maximum values of the doses are scenario specific. With such reservations, it is possible to infer some maximum and minimum values for the product of intrusion probability and the collective dose (i.e. the expectation value of the collective dose as a result of intrusion). The maximum value for the first term is 1 and as mentioned earlier 200 sievert constitutes a possible maximum collective dose to the crew and to the public assuming 20 person-sievert to the crew members which is enough to give life-threatening health effects to all members in the crew if the dose is distributed evenly and with some certain deaths if the distribution is uneven.
The lower set of parameters could be 0.001 for intrusion and 10 sievert for the collective dose to the crew plus the exposed public. It is possible to postulate a lower bound for the collective dose by considering natural analogues, such as doses from radon in ground water from a drinking water well. A dose of one millisievert (0.001 sievert or 100 millirem) per person and year is not uncommon for small wells in Sweden, for example (which does not emanate primarily from drinking, but from inhalation of radon daughters brought to indoor air by other uses of water). It seems reasonable that the family's collective dose: 4 persons * 0.001 Sv/year * 50 years = 0.2 person-sievert can give an example of an intrusion with a collective dose which may only deserve moderate concern.

The lower bound given in Table II represents such a low cost that further calculation seems uninteresting, because in no case would the waste management hesitate to pay this amount if safety could be improved.
Perhaps not even the upper bound given in Table III may seem completely prohibitive for such a large scale project as a high-level waste disposal.
For a repository in salt, such as the WIPP, there is no use of ground water by nearby cities. This may limit the collective dose. If 20 person-sievert to the crew members may be taken as a reference as mentioned earlier, perhaps 40 person-sievert can be taken as the collective dose as a result of a drilling intrusion followed by some additional exposure through some less efficient pathways. If the probability of intrusion is 0.1, the acceptable cost for the planned information conservation system (of markers) is given in Table IV.
If only the crew's dose is considered, the result would be USD 200,000.
A calculation as presented here may be useful for a discussion of the parameters and the assumptions used in the exercise. It may also be valuable to discuss some side issues: should the collective dose to the public be the first concern rather than the dose to the intruders? It is possible that history repeats itself so that several intrusions occurs (followed by restoration) during the ten millennia?

REGULATING HUMAN INTRUSION

There is, in principle, two main methods to inform future societies about radioactive waste repositories:

- Marker systems (such as described in reference 6)
- Archives and other information systems

Markers are created to allow individuals in the future to receive information directly, whereas archives and other information in society must be carried from one generation to the next. This is discussed further in Ref. 7.
It is reasonable to assume that the state must have some responsibility for ensuring the continued information content of the National (and other state) Archives, and it is likewise obvious that the waste management must be responsible for constructing the on-site marker system. It may therefore seem reasonable that the state in its regulation, directed towards the waste management, concentrates on the marker system.
However, the waste management must be responsible for collecting the information in the archives. The IAEA has started work recently in this area related to information conservation for LLW. The author has taken part in a consultant's meeting 25 - 29 April 1994 on "Maintenance of Records and Documents for Near Surface Waste Disposal Facilities". Later, the Agency started working on a similar document for HLW.
The state could influence the markers by its regulation in many other fields: the markers themselves could receive protection through both passive (location marked on maps) and active measures (remediation of markers). The state's and waste management's roles are therefore not completely independent.
It is for this reason, that a separate view of regulating human intrusion may be justified. It is sometime remarked that natural science is more suited for prediction or projection than the social sciences, and that, therefore, human intrusion must be treated differently. There is no scientific justification for such a generalization. It is the role and responsibility of the state that provide arguments for a separate, or special, judgement of human intrusion.

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17-17
THE SCIENTIFIC AND LEGAL BACKGROUND FOR JUDGEMENT OF RELEVANT TIME PERIODS FOR FINAL NUCLEAR WASTE REPOSITORIES

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ABSTRACT
The Swedish Radiation Protection Institute is one of the nuclear regulatory bodies in Sweden. For its preparation of the licensing procedure for a HLW repository, which is planned to commence within a bit more than one decade, it is necessary with a build-up of both the knowledge in general and by this a public acceptance of the time scales involved for the final disposal of the radioactive waste stemming from the Swedish nuclear power production.
Since the key issue in terms of radiation safety is the longevity of the radiation of the waste, the study discusses the meaning of the expressions "longevity" or "long time". In order to grasp the time scales involved, an overview is given for natural large scale processes (earth-quakes, climatic changes, etc) with frequencies similar to those of the life time of a repository for spent nuclear fuel. These natural processes are chosen since they are judged to have a possible impact on the repository. In addition, in contrast to the frequencies of these large scale processes, a review of the history of man is given. With this backgrund in mind, the predictability of these events in a near and far future are discussed. The study has been carried out as a limited literature survey, mainly intended as a background for discussion among decision makers, and as an educational tool for the non-initiated public opinion.
It is concluded that there seems to be limited risks for geological activities in northern Europe, and that the possibilities to predict future earthquakes are limited. A similar conclusion goes for astronomical events like meteors hitting an area close enough to the repository, a conclusion based on surveys of past meteor hits in northern Europe. The natural climatic changes are, however, reasonably well known. By means of knowledge of past climatic cycles, it is therefore possible to make predictions for the future glaciations, the first one of which in Sweden will take place within a time period of about 10,000 years. These predictions do not include the impact from short-term man induced climatic changes, an issue that appears to be unresolved presently.
Conclusively, large scale processes either can be neglected or be accounted for with a reasonable degree of confidence. Contradictory to this, large uncertainties exist when it comes to factors caused by man. Processes like the green-house effect, the reduction of the ozone-layer, etc were identified as environmental impact matters that could affect the pattern of future societies, which in turn, as a second order consequence, can affect the integrity of the deposit. A complicating factor is the short turn-around time for different types of societies of mankind in the history. When trying to project these short cycles into the future, it appears as if the type of future societies is completely unpredictable. However, since some legal principles regarding releases to the environment can be traced more than thousand years back, to Roman civil right principles, one can assume that some kind of civilized society will exist also for the next millennium, particularly bearing the present well-organized society in mind. This assumption leads to the conclusion that it is possible, and also crucial, that the existence of the repository is well

documented by means of marker systems and archives.

INTRODUCTION

A challenging task from a licensing point of view, is the possibility for the authorities to require a safety analysis covering the time spans involved for the spent nuclear fuel to be disposed. At present in Sweden, there are no detailed regulations with respect to technical criteria, long-term aspects and their implications on radiological protection levels. This implies that there are reasons to discuss the possibilities for predictions of future events that may affect the repository, the actual time span denoted "long-term", and the implications these two issues may have on the present and future legislation in Sweden.

In Sweden, electricity has been produced by the country's 12 nuclear power plants during some 20 years. All of the plants are still in operation. The very short-lived low level waste is presently deposited on ground at the power plant producing the waste. Short-lived low and intermediate level waste is disposed in the SFR-facility, an underground facility situated some 50 m below the Baltic sea-bed. The high level waste (about 8,000 tons of spent fuel) and the long-lived low and intermediate level waste are planned to be disposed of in underground caverns at about 500 m below ground surface, a facility that remains to be located and constructed. Prior to the disposal, the spent fuel is planned to be stored for cooling in pools for 30-40 years in the CLAB-facility. The spent fuel is planned to be encapsulated in copper/steel canisters after the cooling period. A funding system for the disposal of the waste is built up as a result of the Swedish legislation, which implies that the waste generator has to live up to the "polluter-pays-principle".

According to the Nuclear Safety Act and the Radiation Protection Act, any generator of radioactive waste in Sweden is obliged to provide all necessary means for the disposal of the waste. As a regulatory body, the Swedish Radiation Protection Institute (SSI) is facing the task of formulating the technical criteria for the licensing of the repository. These criteria have to fulfill the mission of convincing not only scientists and researchers, but also the public opinion, that the strategy chosen is sensible.

Based on scientific considerations, mathematical models are made which in principle can be used to predict a release of radionuclides from a repository for eternity, although substantial uncertainties will exist. This approach is, however, judged to be less meaningful since the confidence that can be expected from such calculations is limited, at least when times scales of millions of years are studied. A more realistic approach is to seek for a suitable Itime span for which a) the radiological impact from the repository is acceptable when damaged for one reason or another, and b) build confidence in the manners by which this is proved. From this we can draw the conclusion that there is a need for a cut-off in time, a limit beyond which it is not worthwhile striving for. This cut-off must, however, not be regarded as an indulgence to safety in a far future. Rather, it must be looked upon as sensible considerations in terms of what can be achieved with an acceptable degree of confidence.

The Swedish disposal strategy complies with the international understanding that those who are consumers of nuclear power generated electricity also should be responsible for the development of a safe disposal handling system, and that the protective limits for man and environment shall be in compliance with those agreed by international organizations in the field on waste management (OECD/NEA, IAEA). One of the issues put forward by the IAEA, states that the waste must be isolated in a manner so that no health risks or environmental risks that are not accepted today can be foreseen in the future. However, this puts us in the situation when we have to face questions like "How can we today state what future generations claim is harmful? What do we know about societal patterns of future generations and their sensitivity to radiation? When does 'future' take place?" Obviously, we have to define what we mean with endurance, persistence, far future, etc, etc.

ENDURANCE - PERSISTENCE - TIME PERSPECTIVES

General

The expected lifetime of constructions like bridges, roads, houses, for example, are based on statistical considerations - accounting for destructive processes with reasonably well known frequencies and magnitudes (rain, wind, erosion, etc). In other words, the sustainability of the construction can be estimated once the destructive processes are accounted for. The time-scales involved, i.e. the planned lifetime, are in the range 50-100 years. Similarly, the environmental impact and the

need for protection, of man-made constructions like, e.g. chemical plants, can be assessed. The plant owner invests in security systems to prevent damages to the environment and his plant. The environmental risk from e.g. an accidental release is, however, put to an end once the plant is shut down.
These two principles of endurance assessment and environmental risk management, respectively, are coupled when designing a repository for radioactive waste, although a repository is a much more complex construction: The repository has to be persistent to destructive processes for enormous time perspectives, and furthermore, the potential release from it has to be considered for thousands of years. The release does not cease due to the termination of the operation of the "plant".
How Long is a Long Time?
A time span of 24,000 years is often mentioned when waste disposal is discussed. This time corresponds to the half-life of plutonium-239. Scientists in climatology seldom dispute that the northern part of Europe, including Sweden, will be subjected to several glaciations, the first major of which will occur in about 10,000-20,000 years. Although a tundra-like climate with endurable conditions could prevail during the interglacials, this part of Europe will probably not be inhabitable to a very large extent for the next 75,000-125,000 years. An upper limit for long time, from a licensing point of view, could according to this discussion be about 20,000 years.
A lower limit for long time can be harder to define. For the low and intermediate level waste the first 500 years is considered the most crucial ones, a time span after which the radiotoxicity has declined to harmless levels. For the high level waste, on the other hand, a direct contact during the first 1,000 years is regarded as dangerous. Although the radioactivity has decreased to 0.1% after that time, the waste is still considered injurious for thousands of years, mainly by uptake in the body by inhalation or food intake.
According to the discussion above, a minimum time span, from a climatic/scientific point of view, for the endurance of the repository is in the range 1 000 - 20 000 years. The question is how this complies with the legislation. Are there means for the authorities to put safety requirements covering these time spans? Are the time spans long enough?
Legal Aspects
The present legislation in the area of environmental protection in Sweden is rather vaguely formulated in terms of acceptable release rates, time limits, etc, and can be characterized as a legal framework. More detailed regulations needed to fulfill the purpose of a specific act can be laid down by the government or an authority appointed by the government. Such an authority could be the Radiation Protection Institute. An act on protection is also characterized in that it contains a certain amount of public influence, e.g. by a concession system implying that certain activities cannot be practiced without being licensed. A license shall contain all the prerequisites to maintain the desired protection, and an official authority is conducting the supervision of the activity. If the conditions change after the license has been granted, the license can be withdrawn or modified. A license can also be restricted in time.
The present legislation in Sweden offers good possibilities for the licensing authority to successively increase the protective demands on the license-holder. Thus, at present there are no incitements for the licensing authorities to carry through a close study of the posed protective demands in a longer time perspective. Deposits for municipal and industrial wastes have thus been regarded as material being temporarily stored awaiting technical solutions for a proper management of the wastes, recirculation or re-use. No time limits or release rates on a national basis have been stated. However, during recent years, the authorities have started discussing the "glaciation perspective" when the endurance of disposal sites for municipal wastes is considered. The implementation of this notion in juridical terms still remains.
However, the nordic countries have suggested in common that an acceptable limit for a highest yearly radiation dose from deposited radioactive waste, should be considered for "a time span reasonably possible to predict over". Accordingly, it is stated that dose rate as a measure of protection limit is very uncertain for time periods exceeding 10,000 years.
The same time span is discussed in the US, although a period of 1,000 years to begin with was regarded as a time during which a repository should provide protection for man and groundwater. This was postulated by the EPA. In court, however, it was found

that the protective measures for radioactive waste, in principle had the same purpose as those for drinking water, long term effects from unsorted hazardous waste, etc. The time span considered for the latter ones is 10,000 years, and therefore the court argued and abolished EPA's statement on 1,000 years. EPA had the time perspective reinvestigated and laid down new regulations implying that the 1,000 years had been changed to 10,000 years [EPA]. EPA claimed that the main reasons for this change were:

A prolongation to 10,000 years would stimulate a better technical design of the repository.

A higher level of confidence can be put into calculations covering long time spans by means of the rapid development in computer hardware.

The marginal cost for investigations on the 10,000 years perspective were considered reasonable.

A harmonization of time limits with corresponding regulations in near-by areas is desirable.

In the US regulations, it is clarified that there are no requirements for an undisputable proof of complete radiation protection, but rather that future predictable courses of events have to be thoroughly investigated.

PAST AND FUTURE EVENTS - PREDICTIONS POSSIBLE BY LEARNING THE LESSON?

In order to grasp the time scales discussed for the longevity of a repository for spent nuclear fuel, it may be worth considering a backward glance on the history of Earth and mankind - what can we learn from history and how much of it is possible to project into future? Apart from geological and climatological events, that seem relevant to discuss here, we also have to look at the implications that life of the modern and future man can have on a repository for nuclear waste. Is the modern man affecting his environment in such a degree and so rapidly that nothing can be taken for granted? Can we by any means predict events taking place in the near future of only a 100 years? As an illustration to these thought-provoking questions, Table I (derived from [NRPB, 1991]) indicates approximate time scales for major events in the history of man, potential future events, and possible different phases of a repository for spent fuel designed for Swedish conditions.

Geological Activity

The oldest type of bedrock in Sweden is found in the northern part, and is about 1.5-2.0 billion years, whereas the more recent (50 million years) types of bedrock are found in the southern part. Volcanic formations younger than about 250 million years have not been found in Scandinavia, although younger formations (about 55 million years) have been found outside the Norwegian coast.

Regional land uplift prior to the last glaciation have amounted to about 1200 m in a series of 13 occasions for about 5-50 million years ago. More large-scale land uplifts of the bedrock amounted to about the same magnitude, but these movements are considered to have ended about 1.6 billion years ago.

Scandinavia has not been subjected to any major earthquakes in historical time. Occurrences of minor earthquakes are on the other hand rather frequent. During the period between 1951 and 1976, some 200 quakes were registered with regional magnitudes lower than 3.8. The quakes took place mostly at depths of about 15 km with minor vertical movements in the bedrock. These movements preferentially takes place at depths far deeper than 500 m where the repository is planned to be located. Furthermore, it is believed that the movements measured in modern time originates from the land uplift caused by the last glaciation, and that the movements thereby will decrease with time.

Science has, during the last decades, found ample evidence for the plate tectonic (or "Pangea") hypothesis. The hypothesis leans on the theory that all continents once were connected, but now drifting apart on the uppermost part of the crust of the earth; movements that have been reconstructed from today's positions of the continents (plates) back to 200 million years ago. The boarders between the plates usually represent areas with a certain degree of geological activity, exemplified by for instance the mid-atlantic ridge and the zone following the west coasts of North and South America (the well-known San Andrea fault forming part of the North American zone). None of the boarders between the drifting continents crosses the northern part of Europe, which in turn would imply that Sweden rests on relatively stable geological formations.

It is of interest to discuss the possibility to predict major events that possibly can affect the integrity of a repository, and to pose the question if there are

means to forecast the next geological activity (earthquake, volcanism) within the Swedish territory for the next, let's say 100,000 years. The answer to this question, taken literally, is of course "no". However, based on earthquakes in historic times (or rather the absence of them), and the present monitoring of quakes of low order magnitudes combined with the understanding of the origin of geological activities due to the plate tectonics, one could possibly state that the probabilities of earthquakes or volcanism of magnitudes large enough to affect the repository, appear to be extremely low in northern Europe.

Climatic changes

During the past two million years (the Quaternary era) the average temperature has fluctuated with about 1-2 degrees (Celsius) at the equator and as much as about 10 degrees in Scandinavia. These temperature variations have given rise to glaciations with large ice-sheets covering parts of the earth. Glaciations means that enormous amounts of water is withdrawn from the hydrological cycle, forming ice instead of flowing water.

It is assumed that ice-sheets with a thickness of 1-3 km covered parts of the northern hemisphere during the last glaciation, while the sea-level was lowered some 100 m. Stagnant groundwater conditions have prevailed down to depths of about 500 m due to permafrost. From these hydraulically stagnant periods, the melting away of the ice-sheet during the deglaciation implies that the water, including the groundwater, is let loose. These both extreme conditions are of course interesting in the context of waste disposal.

The glacial and interglacial periods have been going on for the last two million years. The last glaciation in northern Europe had its largest extent for about 18,000-20,000 years ago with an extent southward to Germany, and its withdrawal took place for about 10,000 years ago. It has been well-known that the glaciations occur with some sort of frequency, but until a few decades ago the origin and frequency of glaciations were disputed.

The possibilities to predict future major climatic changes have improved since 20-30 years. The reason for this is that there seems to be evidence for that variations of the solar radiation causes climatic changes on the earth, and that these variations appear with a certain frequency. The variations occur mainly due to a) changes of the orbit of the earth around the sun (supported by Milankovich's orbital theory), b) periodical changes of the inclination of the axis of the earth, and c) gravitational interaction between the earth and other planets. From today's knowledge of these interacting factors, it is possible to count backwards in time in order to determine temperature variations in the past.

The history of previous temperature variations based on the factors mentioned above, is supported by e.g. sediment samples from deep-sea graves in the Indian Ocean. From the ratio between two isotopes of oxygen (oxygen-18 and oxygen-16) in the samples, it is possible to study the accretion of the sediment (i.e. dead vegetation). The isotope oxygen-18 occurs somewhat more than the oxygen-16 during colder periods, and vice versa for warmer periods. Once the age of the samples has been determined, the history of temperature variations can be studied. It has been found that these measurements support the frequencies of climatic fluctuations in the past determined with the method based on differences in solar radiation mentioned above.

Now that some degree of confidence obviously can be put into the fact that variations in the solar radiation causes climatic changes on earth, it could be possible to predict future major climatic changes based on well-known facts about orbits of planets, etc.

In a report issued by the Swedish Nuclear and Fuel Waste Management Co. (SKB), two studies are referred to, both having investigated the possibility to predict the time of future glaciations in northern Europe [Ahlbom, et al, 1991]. Both aforementioned methods (solar radiation variation and sediment samples) were applied with rather unanimous results. SKB formulates its so called ice-age scenario accordingly: A first minor glaciation occurs within about 5,000 years, covering only the northmost part of the scandinavian mountains. The second glaciation takes place within about 20,000 years from now with a duration of about 5,000 years, the ice-sheet will be about 1.5 km thick, the sea-level will have dropped about 25 m and permafrost conditions are prevailing in the southern Sweden. In about 60,000 years full glacial conditions will prevail, with an ice-sheet of 3 km thickness covering entire Scandinavia, the crustal movement is thought to be about 600 m in the Stockholm region.

Notably though, their predictions do not account for anthropogenic effects on the environment. This means that the lives and beings of present and future mankind societies that may affect the conditions on earth, have to be studied separately.

Societal and legislative perspective

The oldest remains of man or man-like apes are about 1.5-3.5 million years old, and have been found in East Africa. The European continent have been inhabited for about 100,000-500,000 years. The oldest remains of human beings in the northern part of Europe are not older than roughly 30,000 years; although relics of settlements have been found, the majority of them was swept away by the ice-sheet during the last glaciation. Still, we discuss time scales of tens and hundreds of thousands of years.

The first steps of the evolution of man towards modern life, were taken when man started organizing his living in more or less permanent agricultural settlements for about 8,000 years ago. The start of this evolution took place in the middle-east, a point in time approximately coinciding with the withdrawal of the ice-sheet of the last glaciation when the climate went warmer. The agricultural living was spread over Europe within about 4,000 years. At about this time, town-like settlements were formed at places in for example Egypt, India and China. The time scale for this evolution is thousands of years.

Subsequent to the town-like formations, historic "super powers" started to evolve in the Mediterranean and Oriental areas. The Roman empire, and other empires like China, India, Persia, etc, had survival times of less than a thousand years, sometimes far shorter time periods than so.

We see from these figures that the time scales involved are extremely short compared to those of climatic cycles and geological activities earlier discussed. Bearing this in mind, the question arises as to whether or not it's possible to predict forthcoming societies of mankind, their way of living, their potential impact on the environment, their sustainability, etc. The key issue is not to predict the future type of society in detail, but rather to design a repository in such a way that it is persistent to any type of influence - induced by man or by natural causes.

It is possible to identify some actions by future societies which may impair the integrity of the repository:

- planned or unplanned penetration of the repository due to e.g. mining or exploration activities
- political unstableness, sabotage, nuclear war, etc

Although future development cannot be predicted in detail, it is possible to take some actions today which are in the interest of future societies. In a Nordic study [Nordic Council of Ministers, 1993] different mechanisms for informing future societies about the repository, such as markers and archives, are discussed and recommended. This study suggests that a strategy could be built on both markers and archives, and that the relative importance of both strategies should be evaluated for a specific repository. The strategy chosen must involve an assessment of both the durability of a marker system and the retrievability of its information, and the survival of information in a national archive.

CONCLUSIONS

The present study has focused on the predictability of future events that may affect the integrity of a potential repository for spent nuclear fuel. The key issues are the long time periods to account for, the presumed relative lack of sustainability of future societal structures, the present and future legislation, and the necessity of a transfer of the knowledge of the existence of the repository.

Thus, based on the limited literature survey that formed the basis for this study, it is possible to divide the main conclusions into three main categories accordingly:

Geological and Climatic Effects

Apart from anthropogenic effects, the conditions for climatic changes are deemed to be predictable up to the next glaciation. It is therefore reasonable to formulate relatively detailed acceptance criteria for the repository relating to the environmental impact during this period.

Large scale processes like meteors or large earthquakes cannot be ruled out in a longer time perspective. However, these phenomena are associated with so low probabilities that the corresponding risks are deemed acceptable in the shorter perspective. Some processes can be foreseen and described quantitatively (continental movements).

Legal Considerations

The "polluter-pays-principle" implies - because of the extremely long time periods involved - that society must predict environmental impacts from a certain practice in advance in the licensing procedure. This constitutes a new approach in the environmental legislation, which presently allows for a successive increase of the protective demands from the authorities.

Information Transfer

It may be assumed, by considering the climatic development, that societal structures may exist in Sweden in at least a 1,000 year perspective. This implies that there is a basis for an information conservation and transfer strategy to future societies.

The authorities must define criteria for judging the reasonable cost, covering the creation and future maintenance of an archive for a nuclear waste repository, and the measures taken by the waste generators such as constructing passive markers on the site.

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17-19

COST ESTIMATING IS A RELIABLE TOOL TO FORECAST TOTAL PROJECT COST

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ABSTRACT

Accuracy in construction cost estimating is a science that is dependent on specifics such as project definition, accurate construction schedule, accurate material take-offs (MTOs), a reasonable estimating schedule, and experienced staff. Specific information may come from the planning or engineering of the project itself; or it may come from historical information about other similar projects.

Primary costs should be based only on "knowns", for example, soil types, cut and fill quantities, and materials and equipment required to perform the work being scoped based on accurate MTOs. The most accurate estimating method is called the "bottoms-up technique." This method utilizes a Statement of Work and a set of drawings and/or specifications to "take-off" material quantities required to execute each discrete task performed in accomplishing a given operation. From these quantities, direct labor, equipment, and overhead costs can be derived. Other peripheral activities that can be costed with reasonable accuracy include project engineering and oversight costs, with the former being based on workhours to perform, and the latter based on engineering and construction scheduled durations (resulting in workhours).

Waste stream cost is the most difficult cost to estimate. The engineers developing waste stream volumes must work diligently in the preparation of these quantities and must readily provide information relative to the technology used for treatment of materials, storage requirements and durations, and final disposition because the costs for these scenarios remain volatile.

Contingency should be dependent on the status of design, procurement, and construction, as well as the complexity of each project, and should be assigned to incremental components of the cost estimate based on definition, material and equipment, pricing and confidence. Contingency should never be used as a substitute for risk and uncertainty. This analysis should only be performed after completion of the base cost estimate.

INTRODUCTION

Reliable cost estimating is one of the most important elements of remediation or decommissioning planning. As various technologies are evaluated and compared based on their efficiency and effectiveness, so their costs must also be considered throughout the planning process. When the planning process is complete, a thorough cost estimating procedure ensures that a project is economically sound.

Accuracy in construction cost estimating is a science that is dependent on specifics such as project definition, accurate construction schedule, accurate material takeoffs (MTOs), an understanding of the political environment and an experienced staff. Specific information may come from the planning or engineering staffs of the project itself, or from historical information about other similar projects.

The results of an estimate can vary because of changes in assumptions, labor force mixes, assigned escalation values, oversight costs, specific contaminated materials involved, waste stream generated, characterization, and the applicable environmental compliance requirements. Although working within these parameters can be difficult, an experienced staff can prepare a reliable working tool (construction cost estimate) for the majority of projects.

METHODOLOGY

The initial step in preparing a remediation or decommissioning cost estimate should be to gather all available data that may help define what is existing. This may include original design drawings, as-built drawings, construction records, and purchase and installation records for equipment and materials. Some sites have accumulated much of the information pertinent to the remediation or decommissioning task into a single source document such as a System Turnover Package or a Technical Data Summary document. This information provides a preliminary framework for developing an inventory of the materials to be handled in the project. This inventory is necessary to develop the comprehensive cost estimate; that is, the total project cost (TPC). The TPC includes primary cost that are both activity-dependent and period-dependent associated with the intended task and includes direct and indirect construction costs, waste stream costs consisting of both on and off site waste disposition costs, and peripheral costs such as design, engineering, regulatory interfaces, and oversight costs.

The accuracy of TPC estimates depends on the availability of specific information about the activities to be performed. Again, specific information may come from the planning or engineering of the project itself, or it may come from historical information about other similar projects. The cost estimating process may be requested at any stage in the planning, engineering, or design of the project.

It is best to include as few unknowns (contingencies) as possible in a construction cost estimate because as criteria, methods, or peripheral parameters change during the evolution of the project, adjustments can be made more easily and accurately if the original cost figures are based upon known data.

PRIMARY COSTS

The basic elements of primary cost for any task or subtask in a remediation or decommissioning project work sequence include labor, materials consumed, equipment, subcontracts, and other factors such as energy, overhead, and fee.

This primary cost can be estimated in a number of ways. Among other sources, recorded experience from other similar projects, estimating handbooks, and equipment catalog performance data, may be used to develop cost data. The techniques used for preparing cost estimates will necessarily vary with the project's degree of definition; the state of the art of the project; the availability of databases, cost estimating techniques, time, and cost estimators expertise; and the level of engineering detail available. Primary costs should be based only on "knowns". For example, soil types, cut and fill quantities, materials and equipment required to perform the work being scoped are known factors based on accurate MTOs. Direct labor, equipment, and overhead costs can be derived from these quantities.

There are two types of primary cost factors: activity-dependent and period-dependent. Activity-dependent cost factors are further divided into two types: unit cost and fixed cost. Unit costs are expressed in cost per unit output (e.g., $/in. of cut, $/ton, $/yd, $/pump, $/valve). The unit costs may be developed by calculation or from actual field experience. Those derived through calculation are based on a performance parameter such as speed of cut. Fixed cost items reflect those costs fixed by lease, or outright purchase of equipment and materials. As an example, the purchase or lease price of special rigging and handling equipment used

during the removal of a vessel and its internal component parts is a fixed cost. The cost of each activity is calculated by multiplying the appropriate activity parameter (e.g., the volume of heavily reinforced concrete to be removed) by the corresponding unit cost figure. The duration of the activity performance is calculated by selecting the degree of parallel activity considered reasonable for the work; for example, determining how many crews can be effectively utilized for pipe removal during a defined period of work.

Period-dependent cost factors represent costs related to specific phases of a remediation or decommissioning project and are expressed in cost per unit time. Period-dependent costs include such items as administration, insurance, site security, health physics support, and Quality Assurance. It may include certain equipment rentals when their use is common to many activities.

The costs of all period-dependent items are calculated by multiplying each cost factor by its corresponding assigned period. However, some costs that should be captured in the details of an estimate may be insignificant to the cost of a projecttypically less than 1%and should have a dollar amount assigned to that activity based on prior experience. These assigned or "plugged" numbers are typically given to the costs for utilities and consumables required during the performance period of the project.

When both time and project definition are available, TPC is usually best defined by using the "bottoms up technique". To best utilize this technique, a work statement and set of drawings or specifications are used to "take-off" material quantities required to execute each discrete task performed in accomplishing a given operation. From these quantities, direct labor, equipment, and overhead costs can be derived. Using this approach, a project is divided into discrete and measurable work activities. Where possible, the division should provide a sufficient level of detail so that the estimate for a discrete activity can apply to all occurrences of the activity.

Some factors that may affect primary cost calculations include setup time, operating time, required crew size, consumables usage, support services, and energy consumption.

PERIPHERAL COST FACTORS

Peripheral costs include all planning and engineering tasks such as performing radiation surveys; calculating activation and contamination inventories; performing engineering studies; preparing a work plan; preparing major activities specifications and descriptions, designing special tools; characterizations or preparing detailed procedures. Other peripheral activities that can be costed with reasonable accuracy include detailed project engineering and oversight costs with the former being based on workhours to perform and the latter based on engineering and construction scheduled durations (resultant in workhours). All oversight tasks, such as ensuring regulatory compliance, including preparation of permit applications; assisting in negotiations, reviewing the project process to ensure compliance with federal and state regulations; ensuring that the selected technologies comply with administrative and regulatory requirements and attending project status meetings are also considered as peripheral cost factors that are included.

WASTE STREAM COST

Waste stream cost is the most difficult cost to estimate. The engineers developing waste stream volumes must work diligently in the preparation of these quantities and readily provide information relative to the technology used for treatment of materials, storage requirements and durations, levels of contamination, and final disposition, as the relative costs for these scenarios remains volatile.

Because remediation or decommissioning involves the handling of some contaminated materials or equipment, some waste streams will be contaminated. However, these projects generate different wastes which affect the costs differently for each project. Because of changing regulatory requirements, waste facility requirements, and DOT regulations, waste stream costs include both on-site costs for laydown areas, interim storage, processing costs to reach waste minimization, volume reduction, packaging, and transportation, and off-site disposal costs that include special and specific packaging, labeling, transportation, loading and unloading, final burial fee, and surcharges, all of which are principally dependent on the characteristics of that waste. Therefore, identification of the character of waste streams is imperative to reliable cost estimating.

The areas of waste stream generation for a remediation or decommissioning project include the material being dismantled, waste containers, the medium used and its by-products to decontaminate the source material, and the personal protective equipment used by the workers.
Waste stream disposition can be on-site or off-site specific in the cost estimate. The major decommissioning waste stream costs to consider are handling, decontamination processing for waste minimization, volume reduction, packaging and containment, characterization of the waste, shipping and transportation regulations, burial, and any additional fees or surcharges.

CONTINGENCY

Contingency is applied to the cost estimate based on uncertainties associated with the available design information. Contingency is usually applied to discrete elements of a project as a percentage and the contingency for the total project cost is the weighted average of these percentages. The estimate must include all known costs and the identified cost factors for unknowns, and contingency based on the level of design information available, overhead, and fee, thus providing a cost in current-day dollars for each incremental task. Each of these tasks is then merged with the proposed schedule information so that cost can be escalated over the proposed schedule duration (cost increase over time).
Contingency should be dependent on the status of design, procurement, and construction, as well as on the complexity of each project. It should be assigned to incremental components of the cost estimate based on definition, material and equipment, pricing, and confidence. Contingency should never be used as a substitute for risk and uncertainty and this analysis should only be performed after completion of the base cost estimate.

COST ESTIMATING PROCESS

The scope of work can provide a basic foundation for the cost estimate because it specifies the work to be performed (primary cost), ancillary activities required and time constraints (peripheral cost), and known contaminants involved (waste stream cost).
Preparation of remediation or decommissioning construction cost estimates must place emphasis on the basic framework of any reliable costing process including: making a thorough review of the work scope; placing specific emphasis on projected waste stream characteristics (contaminated material); documenting regulatory references and applicability, as well as site-specific or company-specific procedures pertaining to the project; developing a building block approach to establish the total project breakdown by identifying the upper-level requirements and/or specifics for client funding of the project.
From this building block framework, a detailed description of the project in a logical, time-related sequence of series and parallel work activities, broken down into primary cost, peripheral cost, and waste stream cost can be prepared. Once the cost estimate is prepared, a draft project schedule is developed based on the calculated activity durations and the sequential relationship between activities, which can be used to calculate period dependent costs (i.e., project security, or management staff) cost elements.
The three following steps are then needed: estimating each work activity's primary cost and period of performance; generating peripheral costs based on activity-dependent cost data specifics, regulatory requirements, and additional procedural requirements; and estimating waste stream costs.
The TPC is calculated by adding the primary (activity-dependent and period-dependent) cost, peripheral cost, and waste stream cost; the appropriate contingency, overhead, and fee is applied and escalation costs for the scheduled duration of the project are then added.

RISK AND UNCERTAINTY ANALYSIS

A credible cost estimate must take into consideration all the key points described in this paper; however, a Risk and Uncertainty Analysis (RUA) is a task that is rarely requested of the cost estimating group. Many managers feel that contingency included in an estimate has already stretched funding levels to a point that jeopardizes project funding limits; thus they may not want to add confusion by adding the cost associated with possible or probable project overruns. These overruns are usually associated with quantity overruns previously based on engineering judgement rather than on information readily available, and always carry the additional costs associated with lost schedule. These are not, however, the only

risks to a project. The most volatile risks are those associated with omissions and errors, followed by expanded scope, lost schedule, Acts of God, and, to a lesser extent, escalation costs and equipment and materials pricing uncertainties.

DISCUSSION

Should project funding plans fail in this volatile marketplace, blame may be initially cast on the construction cost estimate originally put in place to support project funding. Eventually, though, most projects find that blame for funding shortfalls can be attributed to the lack of a project specific RUA, or discounting the results of a RUA that was performed to support project planning. A RUA is a valuable tool used to support the costs associated with the "what ifs" of a project. Projects can be reliably costed in a quantitative manner, such as costing the removal of contaminated soil/cu. yd. as defined in the project plans. However, when site activities require removal of twice the assumed quantities of contaminated soil as previously costed and the overhead cost associated with a schedule loss, it must be understood that the construction cost estimate is not necessarily flawed; rather, the increased cost is due to a risk associated with the work performed and was (or should have been) identified in a project RUA.

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THE BACK-END OF FUEL CYCLE IN ROMANIA

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ABSTRACT

We shall present in this paper the appropriate variants for the nuclear spent fuel intermediate storage, both for the dry and wet solutions. Also, we shall present the Spent Fuel Transport System from the NPP Cernavoda to the Interim Storage (IS) and to the Nuclear Research Institute Pitesti. Our opinions presented here are based on studies and analysis performed in our Institute, especially in the last 3 years.

INTRODUCTION

In 1992 RENEL initiated a research and development program for the management of nuclear waste produced by nuclear energetics in Romania. CITON, as a subsidiary of RENEL, is the manager of this program.

CITON is involved now in developing the following subprograms:

- a subprogram having the goal to choose and to build an Interim Storage Facility, and
- a subprogram having the goal to select the site and to build a Final Repository.

This year (1995) the first nuclear unit will be put in operation in Romania. NPP Cernavoda is of the CANDU type, and is planned to have in final 5 units of 600MW each.

The spent CANDU fuel will be stored in the NPP Spent Fuel Storage Bay for 7 to 10 years, and after that it has to be removed. The studies performed in our Institute showed that the intermediate storage of the spent fuel cooled for maximum 10 years will be the best solution for the next step of the back-end cycle in our country.

FUEL BACK-END CYCLE

The spent fuel quantity resulting from NPP Cernavoda's 5 units in 30 years of operation is estimated to be 1440OUt. The quantity is large, but the burn-up is relatively low, about 7600 MW day/Ut.

In NPP Cernavoda the spent fuel is manipulated and stored underwater. There are four pools for each nuclear unit, having different destinations: Discharge Bay, Reception Spent Fuel Bay (RSFB), Defective Spent Fuel Storage Bay, and Spent Fuel Storage Bay (SFSB), which is the only having the possibility to be extended with another similar pool. The spent fuel is stored here in trays (24 bundles in a tray).

NPP Cernavoda project doesn't include yet the documentation related to the spent fuel evacuation. However, in the project, areas equipped with some of the facilities are provided for spent fuel manipulation and for cask expedition.

The spent fuel is stored in the NPP for 10 years. The evacuation of the spent fuel from the NPP will be made in one of multiple variants, depending on the Interim Storage type. In the initial project, the spent fuel is transferred in the RSFB, where it will be loaded in a container. We have also studied another variant, in which the spent fuel will be loaded in a basket in the SFSB and lifted in a Loading

Station to prepare the basket for transport and storage.
The next step in the fuel back-end cycle will be the Intermediate Storage. This solution allows to delay the choosing between reprocessing and final repository, and to have enough time to find the optimum solution and the appropriate site.
Also, some defective spent fuel bundles will be sent to Nuclear Research Institute in Pitesti, for research reasons.

SPENT FUEL INTERIM STORAGE

In the studies performed in our Institute, we used the following definition for the Interim Storage main function:

storage of the spent fuel resulted from the NPP's operation after a cooling period in NPP bays, ensuring nuclear shielding, cooling and maintaining the spent fuel bundle integrity for 50 years, so that the impact against the operators, public and environment is within accepted limits.

The Interim Storage is considered to be built in 4 stages, each stage having the capacity for storing 3600 Ut.
The optimum variant for intermediate storage of the spent fuel was not chosen yet. We have already studied two type of IS: wet storage in rectangular pools and dry storage in modules. This year, we will continue the studies started last year with the dual purpose casks and we will start to study the feasibility for using CASCAD solution.

Wet Interim Storage

The first stage of the Interim Storage in this variant represents a complex comprising two main buildings:

the Storage Building, including the storage bays (SB);
the Auxiliary Building, including the reception bay and the auxiliary systems for this unit.

The Interim Storage is planned to be built in 4 stages, corresponding in time with the NPP units commissioning. The facility for the whole quantity of spent fuel will comprise 2 parallel Storage Buildings and one Auxiliary Building, each of them having the length double than the buildings for the first stage (see Fig. 3).
For a wet interim storage variant we think that it is necessary to take into consideration the experience achieved in NPP Cernavoda SFSB design and construction. For this reason, we started to study this solution having in mind the following premise: the Interim Storage SB has the same dimensions as the SFSB from NPP Cernavoda (12m x 20m, with 7.8m in depth). Therefore, for the first stage, 2 bays are necessary. The increasing of spent fuel density compared with the NPP SFSB is done by using modules instead of storage trays and by decreasing the water layer above the fuel, because of the decreased radiation level.
The loading of the spent fuel into the transport cask is done in the NPP SFRB The IS is provided with a RSFB. for unloading the spent fuel from the cask. The bay is connected with the SFSB by means of a transfer channel. The IS RSFB and the main systems serving the facility are located in the Auxiliary Building.
The advantages of using a wet IS in Romania are:

there will be operating experience at IS commissioning time;
it is based on known technologies;
doesn't need modifications in the NPP Cernavoda project for loading the spent fuel into the cask;
allows to keep a lower temperature on the spent fuel bundle;
allows easier access to the spent fuel bundle for inspection and manipulation;
allows a dense storage.

The disadvantages of using this solution are:

needs continuous operation and maintenance;
generates nuclear waste;
can give undesirable effects because of the corrosion;
needs, generally, higher expenses;
it is vulnerable to heavy objects falling or missiles.

Dry Interim Storage - Storage Module Type

The Interim Storage module called MMB is a parallelipipedical construction made of reinforced concrete. The upper part is designed to sustain 20 cylinders filled each with 10 baskets containing 60 spent fuel bundles. There are necessary 16 storage modules for storing the spent fuel for the first stage. In Fig. 4 you can see the layout for the whole IS.
The basket is filled with the spent fuel in the NPP Cernavoda SFSB and lifted in a

Loading Station. In this shielded area placed on the SFSB edge, the baskets are washed, dried, sealed by welding and lifted in the transport flask. Also, we have analyzed a Loading Station comprising two separated and shielded areas (a "dirty" one made of metal and a "clean" one, made of concrete) connected by a shielded channel, with a gate. An auxiliary building is necessary, for both solutions, to prepare the cask for transport.
The loading of the spent fuel into the Interim Storage will be done periodically, according to the meteorological conditions.
The IS module type solution studied in our Institute is based on the AECL's CANSTOR model.
The advantages for using IS modules type in Romania are:
- the construction will be done in stages, according to the IS loading necessities, with minimum capital blocked;
- doesn't need continuous operation, needs only a minimum of maintenance and supervision;
- doesn't generate nuclear waste
- it is adequate for simple decommissioning;
- it is less vulnerable to heavy objects falling or missiles;
- needs smaller expenses than the other variants analyzed by us until now.

The disadvantages of using this solution are:
- changes in NPP Cernavoda SFSB area for building the Loading Station;
- decreasing of the spent fuel storage area by placing a loading table in the bay;
- difficult access to the spent fuel bundle;
- higher temperature on the spent fuel bundle;
- reduced operating experience.

Dry Interim Storage - Dual Purpose Cask Type
We have studied the feasibility of the interim storage in dual purpose casks (DSC type from Ontaryo - Hydro), but a detailed analysis will be done this year, taking into consideration other solutions, too. In this case, the loading of the S.F. into the cask will be done in the NPP RSFB.
The advantages for using dual purpose cask IS in Romania are:
- manufacturing can be done in stages, according to the IS storing necessities, and allows at every moment to change the back-end strategy or the site;
- doesn't affect the storage area;
- the container can be used for transporting the spent fuel to a conditioning facility for the final repository or to a reprocessing facility, too;
- eliminates the spent fuel bundle manipulation between NPP and the conditioning or reprocessing facility; minimum of operation, minimum number of operators, smaller operating expenses;
- doesn't generate nuclear waste;
- simplifies the IS utilities;
- it is less vulnerable to heavy objects falling or missiles;
- may reduce the cost of the back-end cycle, because the cask is both for transport and for storage;

The disadvantages of using this solution are:
- the IS cost is higher than for other solutions;
- higher temperature on the spent fuel bundle;
- reduced operating experience.

THE SPENT FUEL TRANSPORT FROM THE NPP CERNAVODA TO THE IS
We have analyzed the spent fuel transport considering it to be a system. This system represents the ensemble of equipments, devices and tools necessary to do the spent fuel transport according to approved procedures, in good technical and economical conditions and according to the existing laws and standards.
The system composition and its operation are determined, at first, by the IS type and its site (inside or outside of the NPP site), and second, by the NPP spent fuel area design.
In the case of choosing the wet IS or double purpose casks variants, the spent fuel will be stored 10 years in NPP Cernavoda bay. The spent fuel transport and storage will be done probably in modules (2 to 4 modules in a cask), and the loading will be done in NPP RSFB.
In the case of choosing the dry IS in modules, the spent fuel will be stored only 7 years in the SFSB The transport and storage will be done in baskets, and the transport cask will accommodate 1 basket. The basket will be tightened by automatic

welding, and, as we have already mentioned, this operation will be done in a Loading Station. The loading and unloading of the basket into/from the cask will be done with a hoist placed inside the upper part of the cask. Another spent fuel transport type will be done from NPP Cernavoda to the Nuclear Research Institute Pitesti. The transport will be done with defective spent fuel bundles, for research reasons. Therefore, CITON with the Nuclear Research Institute Pitesti are designing a cask for this purpose.

FINAL

In this paper the Romanian variants for the fuel back-end cycle studied until now are presented based on the following:

- the characteristic of Romanian Industry, which has experience in designing and manufacturing utilities for the spent fuel storage bays;
- the NPP Cernavoda site characteristics, which affect the Interim Storage maximum area (if the IS will be inside NPP);
- the NPP Cernavoda SFSB arrangement, which influences the spent fuel evacuation;
- the CANDU spent fuel quantities and characteristics.

All the facilities and equipments involved in the fuel back-end cycle are analyzed as parts of a single system, because the decisions and changes done for one component in one step of the program will influence the other components of the system as well. For this reason, our studies take always into considerations the interfaces between each component of the system and the system.

A sustained activity of information updating about the international practices in the fuel back-end cycle is necessary in this stage of the Waste Management Program. Also, it is necessary to be in touch with designers, manufactures and operators of similar facilities.

The back-end of the fuel cycle is currently under study. The Waste Management Program has the deadline for the Spent Fuel Interim Storage in 2006 if it will be adopted the wet variant, or 2003 if it will be adopted the dry variant. This requires also to bring up to date the legal frame for nuclear materials transport and storage, according to the IAEA standards.

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Fig. 1
Fig. 2
Fig. 3
Fig. 4

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DOCUMENTATION AND CONTROL OF RADIOACTIVE WASTE IN GERMANY

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ABSTRACT
To satisfy the requirements imposed by the German regulations, a system for the documentation of radioactive NPP wastes with negligible heat generation has been developed. This system, called AVK, consists of several database PC-programs and has a tight organizational structure including a quality control office and an active user support, which ensure an appropriate quality level of the database. AVK keeps track of the waste packages from the moment of their creation to the deposition in a final repository. It registers the characteristics, activity inventory, predecessors, relocations etc. of each package and generates the documentation required by the authorities. In addition, AVK includes a module for the computation of the nuclide-specific activities based on the default nuclide assessment procedures. This module can compute the activity inventory of a waste package from a wide variety of input data, such as the experimentally known activities of the representative radionuclides (Co-60 and Cs-137), the total b/g activity of the g emitting radionuclides, the total a activity, the dose rate, the experimentally determined mixing ratios of radionuclides (e.g. in the primary coolant) and the humidity of the waste package (relevant for H-3). AVK is currently in full operation at all NPPs, waste conditioners and interim storage facilities in Germany.
INTRODUCTION
To improve the documentation about radioactive wastes, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety published the "Guideline on the Control of Radioactive Wastes with Negligible Heat Generation" in 1989, which is colloquially known as the "Waste Control Guideline". The objective of the Waste Control Guideline is to regulate the documentation and control procedures for these radioactive wastes.
The Waste Control Guideline prescribes the information which has to be documented of the radioactive wastes and makes the producer of the wastes responsible for the book-keeping. Furthermore, it requires that the waste producer keeps track of his wastes, which means that he must always know their exact location and the current state of treatment. In view of these requirements, a book-keeping system had to be developed, which could keep track of the origin, type, quantity, radioactive inventory, state of treatment and location of the radioactive wastes. Such a book-keeping system helps the waste producer to provide the responsible authority promptly upon request with the latest information about his wastes.
However, even before the Waste Control Guideline was enforced, all operators of the German nuclear power plants initiated with consent of the Federal Ministry of Environment the development of a documentation and book-keeping system called AVK. AVK is an acronym for "Abfallflu-Verfolgungs- und Produkt-Kontrollsystem" which means "waste flow tracking and quality assurance system". AVK has been designed as a decentralized database system consisting of several PC-programs and a tight organizational structure. It keeps track of the radioactive wastes from their origin to their disposal in a final repository and ensures that the exact location and the state of treatment of any nuclear waste can be accurately determined at any time required.
Overview
A particular radioactive waste is registered in AVK as soon as it exists as a closed unit, which is not subject to any further operational changes. The data which have to be registered are
- the general data (mass, volume, package label, package characteristics, type of treatment, location, etc.);
- the radioactive inventory and radiological data;
- data needed for the quality assurance as specified by 1) the "Waste Control Guideline", 2) the "Preliminary Requirements for the Final Disposal in the Repository KONRAD" and 3) the "Acceptance Requirements for the MORSLEBEN Repository".

The data in AVK have to be updated following every relocation or modification of the

waste. Provided that the treatment of the waste results in the production of a new waste package, the latter has to be registered with reference to its predecessors. In this way, AVK provides a complete track of each radioactive waste package.
The participants in AVK are
- the producers of the radioactive wastes;
- the waste treatment plants;
- the external interim storage facilities;
- the central office.

The waste producers are the nuclear power plants and, to a minor extent, the waste treatment plants and the external interim storage facilities. All waste treatment plants, irrespective of their actual location and ownership are the legal responsibility of GNS, the service company of the nuclear power plants seated in Essen (North Rhine-Westphalia). The external interim storage facilities are located in Gorleben (Lower Saxony), Mitterteich (Bavaria) and at the nuclear power plant Unterweser in Rodenkirchen (Lower Saxony). The central office in Essen is also the responsiblity of GNS, but is organizationally independent of the waste treatment plants.
All AVK-participants, except for the central office, use the PC-program AVK and a subsequently developed independent module AVK-ELA. However, some nuclear power plants do not operate AVK-ELA directly, but prefer to use a service offered by GNS to operate this program . In addition, the module AVK-ELA is operated by instituitions which are involved in the final disposal of radioactive waste, but which are not direct AVK-participants. The central office uses a different PC-program described below. The data are exchanged via off-line data carriers.
The main features of the AVK-program are
- a database system with encoded data files and a hierarchical access system;
- data acquisition is possible only after inspection by a second person;
- input support via reference files and list boxes;
- four distinct modules for input of the data about 1) raw waste, 2) storage characteristics, 3) conditioned waste and 4) transportation;
- a report generator to create free lists, standard lists and other documents;
- a module for the computation of the activity inventory (cf. below).

The program AVK-ELA has been developed after the Morsleben repository (Saxony-Anhalt) was reopened for the final disposal of radioactive wastes (It was closed between 1990 and 1994 for legal reasons in connection with the German reunification). The purpose of AVK-ELA is
- the determination of the activity inventory of the radioactive wastes;
- the check for compliance with the "Acceptance Requirements for the MORSLEBEN Repository";
- the recording of the administrative procedures in connection with the final disposal;
- the generation of data sheets, i.e. documentation;

The central office runs its own PC-program for the autonomous control of the AVK-system. The objective of this program is to
- archive all AVK-data:
- inspect the data for mistakes and changes;
- balance the AVK-data between the individual AVK-participants;
- evaluate the complete AVK database of all participants.

The quality assurance measures carried out during the input of data (data input by two persons, data input support, plausibility checks), and completed by the inspection of the data of all the AVK-participants by the central office, ensure an appropriate quality level of the database. Thus, the requirements for a documentation system for radioactive waste tracking are fully met by AVK.
FIG. 1
The exchange of data between the programs and participants of the AVK-system and between the AVK-system and other programs and institutions takes place via several interfaces, cf. Fig. 1. Between two AVK-programs the data are transfered via checksum protected export and import interfaces. The module AVK-ELA and the program used by the central office can import data from the AVK-program but the reverse flow of data is not allowed. Furthermore, the AVK-system possesses interfaces to external institutions and data processing systems, e.g. to ISAR, a PC-program installed at the authorities to support their surveillance activities. In addition, the AVK-program can exchange data with other data processing and documentation systems

and devices, e.g. the dose rate detector, via standard import and export interfaces. Finally, an ex- and import interface between two AVK-ELA modules enables data transfer to external institutions involved in the final waste deposition.

Organizational Structure

The cooperation between all institutions participating in the AVK-system, their organizational integration and responsibilities are regulated in the AVK Organization Handbook. This handbook lists all AVK-participants, describes their tasks, regulates the data exchange and specifies additional requirements where necessary. The tasks of the waste producers, waste treatment plants and the interim storage facilities are for example

- the creation of the waste package and successor records by the waste producers and waste treatment plants;
- the import of externally created waste package records by the waste producers, waste treatment plants and interim storage facilities;
- the updating of the waste package records;
- the input of nuclide-specific activity measurements by the waste producers, waste treatment plants and interim storage facilities;
- the maintenance of the accompanying documentation;
- the export of the AVK-data to the central office;
- the data security measures.

The Organization Handbook demands that all AVK-participants send a back-up of their data, including the program and auxiliary files, to the central office every three months. These are then checked for virus and topicality and compared with the last back-up archived at the central office. Any changes to the data sets are recorded and the AVK-participants are informed in a so-called control message about the result of the comparison and whether any inconsistencies or incorrect files were detected. If there are no complaints, the data is added to a master file containing the waste package records of all AVK-participants, which is then checked for consistency. If inconsistencies are detected, e.g. in the location of a waste package, the corresponding AVK-participants are informed and required to correct their data.

The Organization Handbook also regulates the data exchange between the AVK-participants 1) in connection with the transport of the radioactive waste packages, 2) in the case of updating procedures and 3) for organizational reasons. The data are exchanged via disks or paper. The form and content of the data, the accompanying documentation, the deadlines and other organizational requirements are precisely defined.

The Waste Control Guideline (cf. above) regulates the scope of the reports which the waste producers, waste treatment plants and interim storage facilities have to supply to their authorities. In order to standardize these, the organizational regulations about the form and content of the AVK-reports are specified in the Organization Handbook and are implemented in the AVK-code.

The AVK-Code

The AVK-program is a database system which runs under the operating system DOS. It consists of modules for the input of waste, transport and storage data, for the system maintenance and for the activity calculation.

AVK recognizes three levels of user authorization

- the authorization for the output of data;
- the authorization for the input and output of data;
- the authorization to act as a supervisor for the system maintenance;

The user's identification and password enable the program to recognize the appropriate level of authorization. If the database is not in a correct state, only the supervisor is authorized to log in.

The AVK-program is menu-driven and comprises

1. MADA (Module for the input of raw waste data);
2. MOTRA (Module for the input of transport data);
3. MOZILA (Module for the input of interim storage data);
4. MOKON (Module for the input of treated waste data);
5. MOPRO (Module for the activity determination and quality assurance);
6. MOBERI (Module for the reports);
7. MOSERV (Service module for the system maintenance).

The waste package data are registered using the modules MADA and MOKON. The input is standardized by extensive use of reference files. The AVK-program automatically

supplies each new waste package record with a unique identification number, the AVK-ID. This ensures that a specific AVK-ID occurs only once within the entire AVK-system.
The input fields are either obligatory or optional. The obligatory fields must be filled during the registration of a waste package, whereas the optional fields can be completed later. A waste package record cannot be saved before all obligatory fields are filled. To track the treatment of a waste package, a new data record must be created. In this case, the old data record is automatically saved, so that the complete history of the waste-flow is always consistently documented.
The registered waste package data are saved in a buffer. These data must be inspected by a second user before they can be transferred to the master database . The AVK-program does not allow a single user to register the waste package data and transfer them to the master database.
The module MOTRA makes it possible to register the transport data. A transport data record contains essential transport data together with the AVK-IDs of the waste packages which are transported. The AVK-program automatically assigns unique labels to each transport in a manner similar to the AVK-ID assignment described previously. Following dispatch, the AVK-program registers the waste packages as being underway. The recipient of a transport must inform the dispatcher about the receipt. The dispatcher then registers the date of the receipt and the new location of his waste packages in his AVK-database. Thus, the transport history of each waste package is always documented transparently.
The module MOZILA serves exclusively to supplement the location data.
The module MOPRO computes the radioactive inventory for radionuclides which have to be declared according to the "Waste Control Guideline", but for which no experimental data are available. The experimentally known activities always have higher priority than calculated values and are never overwritten by the computed data. The data which MOPRO uses to compute an activity include
the experimentally known activities of the representative radionuclides (Co-60 and Cs-137);
  the total b/g activity of the g emitting radionuclides;
  the total a activity;
  the dose rate;
  the experimentally determined mixing ratios of radionuclides (e.g. in the primary coolant);
  the humidity of the waste package (relevant for the computation of the H-3 activity).
Based on the available information, MOPRO automatically selects the best calculation method to be used for each radionuclide and documents it together with the result. The module MOPRO computes activities for any arbitrary date and presents the results in any of the several available output formats. In addition, MOPRO determines whether the declaration limits of the "Waste Control Guideline" have been exceeded and whether the preliminary KONRAD waste acceptance requirements are met. Finally, it assignes the waste package to the corresponding waste product category.
The module MOBERI selects the waste package data records and generates the standard lists and documents, displays, prints or saves them in a file. The standard lists are predefined output formats, in which the AVK-user can list the selected waste package records. The standard documents are reports to the regional authorities (cf. above).
The module MOSERV is used for the system maintenance, i.e.
  the import and export of data;
  the transfer of waste package records from the buffer to the master database;
  the defining of the free lists via a list generator (the arrangement of the fields in the list can be predefined, group sums can be formed, etc.);
  the editing of the reference databases ;
  the editing of the master database .
At the end of each session, the user is prompted to save his database on an external data carrier.
Activity Determination
Table I lists the AVK-relevant nuclides.
The radioactive inventory of a waste package can be either based on experimental data or computed ones using default nuclide assessment procedures, representative nuclide distributions and/or surface dose rates. The default nuclide assessment

procedures are based on
  statistical analyses of activity measurements on representative radioactive wastes;
  burn up calculations;
  activating calculations;
  secular equilibrium;
  transient equilibrium;
  other procedures.
Statistical analyses are the preferred approach, provided that a sufficient number of representative experimental data sets is available. The default nuclide assessment procedures require the knowledge of the key nuclides Co-60 and Cs-137, the activities of which can be determined from the total activity or the dose rate. The parameters required for the computation of the activities from the dose rates, e.g. the geometries, the densities, and the shielding characteristics, are determined from the corresponding package characteristics. Thus, the activities can be calculated knowing the waste mass, the dose rate, and the Cs-137/Co-60-ratio or the nuclide distribution.
Figure 2 shows a commonly used procedure for the determination of activities. In this procedure, a sample is drawn from the raw waste before it is processed to waste packages, e.g. drums. The nuclide specific analysis of this sample yields activities, which can be registered in AVK either directly or in form of a representative nuclide distribution or the Cs-137/Co-60-ratio. Independently, the dose rate of the waste packages is measured and registered in AVK together with the general package characteristics, such as the drum type, the waste mass etc. Using these data and the implemented results of the nuclide assessment procedures, the AVK-program computes the radionuclide inventory and generates a data-sheet.
In addition to the common procedure shown in Fig. 2, there are 19 other possibilities for the determination of the radionuclide inventory, which can be combined nuclide-specifically in a multitude of ways.
User Support
In order to optimize the operation of the AVK-system, an extensive user support is arranged
  hotline
The AVK-hotline supports the AVK-participants with respect to organizational, radiological (activity determination) and data processing aspects. Each hotline message is documented on a formsheet and recorded in a database . If any program errors are reported, they are immediately corrected and an upgrade is sent to the users. The frequency and the content of the hotline messages reflect the educational level of the users, the stability of the AVK-program and the consistency of the AVK-system. The analysis of the messages gives hints for the future development.
  user support at the site
Periodically all AVK-users are visited by the members of the AVK-developer staff in order to evaluate their practical experiences. During these visits their problems are discussed and the data processing equipment is checked and optimized when necessary. All results are recorded in questionaires and collected in a database. The collected information is analyzed and used together with that of the AVK-hotline for the optimization of the AVK-program and the AVK-system.
  training
Training is offered to the users twice a year .
  user meetings
The development of the AVK-system, including the programs, is discussed and harmonized during the periodic user meetings.
  newsletter
The current developments, changes, information and hints about AVK are periodically published in an AVK- newsletter.
AVK Project Development and Status
The development of AVK was initiated in autumn 1988. Several weeks later an input module was installed with the users and AVK began with its test operation. The first full version, AVK-V1.0, was introduced for test purposes in May 1989. The experience with AVK-V1.0 was continuously analyzed and used for the development of AVK-V2.0, which was then installed in summer 1992. The latter was upgraded several times. The current upgrade which was installed with the users in August 1994 is AVK-V2.12.

In the opinion of the users AVK-V2.12 is sufficiently comfortable and stable to fulfill their requirements. The module for activity determination is now under expertise by TV-Bayern/Sachsen e.V. Thereafter it will be recognized as a standard. The central office began with the test operation of its first module in March 1992. The second module was installed a year later and since 1994 it is in full operation. The module AVK-ELA-V1.0, which generates the documentation required for the final disposal of radioactive wastes and for the control of the waste packages for the MORSLEBEN Repository, was programmed in 1994. Currently it is in its test phase. With AVK-V2.12, AVK-ELA-V1.0 and the programs for the central office the development of the AVK-system is temporarily completed. Future development is intended only if the corresponding prerequisites, such as the administrative regulations and the requirements for final disposal, change.

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MODELING PRECIPITATION FROM CONCENTRATED SOLUTIONS WITH THE EQ3/6 CHEMICAL SPECIATION CODES
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ABSTRACT
We designed four hypothetical problems in which portlandite (Ca(OH)2), gypsum (CaSO4.2 H2O), or both precipitated from concentrated solutions. The problems are used to compare predictions of precipitate formation from the hypothetical solutions. Precipitates are predicted assuming ideal solution theory as well as various methods of accounting for nonideal behavior. The Davies modification of the Debye and Hckel approximation, the B-dot approach, and two approaches employing virial equations are used to accommodate nonideality. All nonideal modeling is conducted with the EQ3/6 codes, whereas ideal solution predictions are made algebraically.
Three problems involved precipitation of portlandite or portlandite and gypsum. The results of the three problems showed reasonable replication of the amounts of precipitates that formed. Activity coefficients and final solution concentrations differed according to the methods used to model the problem, but comparable Ksp values are calculated for each prediction. The fourth problem modeled precipitation during evaporation. The results showed that the expected amounts of precipitates varied according to the modeling approach that is used. There is also significant variation in activity coefficients and concentrations of different solution constituents after precipitation.
The results of all the tests indicate that the ideal and nonideal approaches to precipitate prediction can be similar when the amount of precipitates is considered. The modeling suggests that ideal solution approaches can be used on occasion to predict precipitation from concentrated solutions. The benefits of ideal solution modeling are that solubility data exist for most species and precipitates of interest. The important drawback to using ideal solution modeling in concentrated solutions is that predictions may not be valid for the system under consideration. Our results show that ideal solution modeling agreed with nonideal solution modeling for the hypothetical problems we designed.
INTRODUCTION
The main purpose of the work presented in this paper is to compare predictions of precipitate formation from concentrated solutions using different chemical modeling databases and different equations that characterize nonideality of solutions. We are interested in precipitation from concentrated solutions and how consistently numerical speciation codes model this phenomenon. Available chemical speciation data are limited for modeling concentrated, nonideal solutions, whereas there are more extensive databases that can be used to model the solution and precipitation if ideality or only weak nonideality of the solution is assumed. Our goal is to use ideal and nonideal approaches and model the solutions, then compare the results of the different approaches. The hypotheses tested in this paper are: 1) ideal and nonideal calculations provide similar predictions of the precipitation processes; 2) variation in activity coefficients depends on how the coefficients are calculated but make only small differences in Ksp and saturation index values; 3) variations in activity coefficients resulting from using different databases or nonideality equations result in significant variations in predicted precipitation during

evaporation campaigns.
We designed four hypothetical problems for the paper, and they are discussed in detail below. Three problems concern precipitation of Ca(OH)2 (portlandite) alone or portlandite and CaSO4.2 H2O (gypsum), and one concerns precipitation during evaporation. The results from modeling the four problems are compared and are used to examine the above hypotheses. We chose the EQ3/6 chemical speciation codes (1, 2) to model the four problems.
Uncertainty in the data from databases or references causes significant uncertainty in the predictions. In general, uncertainty in data and equations used for calculation of activity coefficients, redox states, and ionic relations leads to variation in predictions of the chemical behavior of solutions (3). Users of speciation codes must also be aware of the data used in modeling calculations and the limitations of the databases used for the calculations. Daveler and Wolery (4) discuss specifically the limitations of the EQ3/6 databases, and the results of our calculations illustrate some of these variations and limitations. A note of caution by Wolery (1) is also worth repeating: "Geochemical modeling codes are not black boxes. Much of the usefulness (or lack of) that comes from their use is determined by the level of knowledge brought to bear by the user...Any results obtained by modeling calculations should be weighed against descriptive knowledge of the system being modeled."
METHODS
Overview of Problems
We designed four problems to test the EQ3/6 codes in modeling precipitation of Ca(OH)2 (portlandite) and CaSO4.2 H2O (gypsum) in concentrated solutions. Problem #1 concerns the precipitation of a single substance, portlandite. One liter of a water solution containing 500 g dissolved CaCl2 was mixed at 25C with one liter of a water solution containing 100 g dissolved NaOH. We predict how much portlandite precipitates and the composition of the remaining solution. Problem #1 serves to validate the EQ3/6 codes on an elementary level. In this problem, the solutions are concentrated and calcium is in excess, so most of the hydroxyl ion precipitateds as portlandite.
Problem #2 examines precipitation from a high ionic strength solution that is dilute in precipitating species. One liter of a solution containing 8 g dissolved CaCl2, one liter of a water solution containing 3 g dissolved NaOH, and one liter of a water solution containing 500 g dissolved NaCl are mixed at 25C. We consider if calcium hydroxide precipitates, and if so, how much. We also predict the concentration of different constitutents in the remaining solution.
Problem #3 investigates simultaneous precipitation of portlandite and gypsum. The purpose of Problem #3 is to compare relative amounts of two precipitates, both of which should form because of the concentrated solution. Three solutions are mixed at 25C, and one liter of the mixed solution is modeled. The initial solution concentrations are 167 g CaCl2, 33 g NaOH, 167 g Na2SO4 in one liter of water. We predict how much portlandite and how much gypsum precipitate as well as the composition of different constitutents in the remaining solution.
The fourth problem examines the evaporation of the supernatant of Problem #3. We predict the precipitates after the supernatant evaporated from 1000 g of H2O to 45 g. Problem #4 illustrates possible differences in the predicted results of evaporation campaigns when different thermodynamic databases or different nonideality relationships are used to calculate the solution compositions.
Calculation of Activity Coefficients and the Saturation Index
The "tendency of a species to react (5) is the activity of a substance. Activities are used to calculate thermodynamic equilibrium constants and solubility products (Ksp). The solubility products predict precipitation of solids from solutions and dissolution of solids into solutions. Thermodynamic activity (ai) of a component is calculated from its concentration (Ci) and activity coefficient (gi) by

$$a_i = g_i C_i \qquad (1)$$

Activity coefficients approach one in highly dilute solutions and are defined as unity in ideal solutions. Equation 1 is also called the activity product, Q, of the constituent i.
The first successful approach to estimating activity coefficients for species in ionic solutions was proposed by Debye and Hckel (6). Their approach has been widely used and extended since its introduction. Glasstone (7) and Lewis and Randall (8) give good descriptions of the theory and its applications, and Mazo and Mou (9) give

a sophisticated treatment of Debye-Hckel theory.
Unmodified Debye-Hckel theory may be suitable for modeling solutions only at concentrations less than 0.003 M (10). Thus, unmodified Debye-Hckel equations are not used in the EQ3/6 codes. An extended Debye-Hckel equation proposed by Davies (11) is incorporated in EQ3/6 as is the "B-dot" equation (12, 13). Activity coefficients calculated from high concentration solutions, however, are inaccurate even when these modifications to Debye-Hckel theory are employed. Recent modeling efforts have incorporated virial equations developed by Pitzer for activity coefficients (14, 15, 16). Use of these virial equations allows more exact calculation of activity coefficients for solutions at high concentrations.
The saturation index, SI, is defined as the log of the ratio of the activity product to the solubility product (2). We calculate activity products, Q, from the output values of the EQ3/6 calculations and Equation 1. When Q is greater than Ksp or, alternatively, when SI is greater than 1, we expect a precipitate for the solid of interest. The activity product is particularly useful in explaining some of the results of our predictions.
Chemical Speciation: EQ3/6 Codes
Several chemical speciation codes have been developed since about 1965. The characteristics of some of the codes are compared by Mangold and Tsang (17). We use the EQ3/6 codes (1, 2, 18) because these codes model more aqueous species and more possible precipitated mineral types than most other codes. The EQ3/6 codes also model the evolution of a solution from its starting point toward final equilibrium. The final solution and any intermediate solutions and precipitates can be examined during the course of reaction path modeling. Few other chemical speciation codes have reaction path modeling capability.
The EQ3/6 codes have undergone continuous improvement with expansion of their capabilities and databases. The number of elements included in the most frequently used database now numbers 78, the number of aqueous species 851, the minerals 885, and gases 75 (4). The EQ3/6 codes have their databases in separate files that can be individually addressed and modified if desired. The COM file is the largest of the five databases and draws on many sources of data (1). It encompasses the broadest range of elements and species of any of the five files, taking some of its data from the other four files. Other files have greater internal consistency or apply to specific situations but possess data on fewer species.
The Pitzer equations require specific parameters, and databases developed particularly for using these equations are necessary (14, 15, 19). The PIT and HMW data files were created for using the Pitzer equations with the EQ3/6 codes. The PIT file includes more species, but the HMW file has better internal consistency. The HMW file includes mutual consistency of activity coefficient data and standard-state thermodynamic data. However, the HMW file is limited to the set of components present in the "sea-salt" system, and is restricted to 25C. Wolery (2) and Daveler and Wolery (4) give additional information about these databases and their uses.
The Databases and Activity-Coefficient Relationships Used in this Study.
Employing the Pitzer equations for the concentrated solutions in Problems #1-4 is the first choice of prediction methods. The numbers of solution species and solids available in the PIT and HMW databases, however, are not as extensive as the databases for the Debye-Hckel approach. Therefore, while the Pitzer approach is preferable from a theory standpoint, the data necessary for adequate description of the problems sometimes dictate the use of the Debye-Hckel approach. Because of the uncertainty of the Debye-Hckel approach in modeling activity coefficients at high concentrations, an indication of the errors involved in using the Debye-Hckel approach is desirable.
The Davies and B-dot equations are used to compare the results of the two approaches with the COM database. The Pitzer equations are used with the PIT and HMW databases to compare this approach with different databases. The results of each problem allowed comparison of the Debye-Hckel and Pitzer equations in examining the four problems.
RESULTS AND DISCUSSION
Precipitation Problem #1
The amounts of portlandite that precipitate in the different solutions to Problem #1 range from 44.9 g/l to 46.3 g/l (Table I). The higher value is calculated assuming ideal solutions, the lower value with EQ3/6 and adjusted for nonideality. The amounts precipitates predicted from the ideal solution are expected to be greater

than the amounts predicted from the nonideal solutions because the product of the activity coefficients tends to be less than 1 in nonideal solutions. Thus, the activity product, Q, should be smaller for nonideal solutions than ideal solutions. Most of the Ca++ precipitates because of the small Ksp of portlandite, and there is minimal effect of the product of the activity coefficients on the calculations. The low Ksp gives reasonable predictions of the amounts of Ca(OH)2 precipitation for Problem #1 regardless of ideality or nonideality of the activity coefficients.
The activity coefficients for both the Ca++ and OH- ions show wide variation. The coefficients for the calcium ion range from 0.35 to 2.65, for the hydroxyl ion from 0.035 to 0.57. There seems to be little consistency in the activity coefficients, either within the same database using different relationships for the coefficients, or within the same relationship but using different databases. The listed concentrations are dilute, so they do not differ much in absolute terms. They nevertheless exhibit significant relative differences. The existence of these differences illustrates the uncertainties in using computer codes for chemical speciation as mentioned above.

Precipitation Problem #2

Table II shows the results of Precipitation Problem #2. Ideal solution approaches predict small amounts of portlandite, whereas three of the four nonideal calculations predict no portlandite. Since the solubilities of solids are higher in nonideal solutions than ideal, there is a possibility that precipitates will be predicted in some of the calculations but not in others. The Q for the Davies and B-dot predictions is significantly less than the pKsp for portlandite and no precipitation is expected. The Q for the HMW prediction is also less than pKsp, but the difference between the Q and pKsp is much less than for the COM database. While no precipitate is predicted using the HMW database, a small variation in one of the activity coefficients could result in Q greater than pKsp and portlandite precipitation. The Q calculated in the PIT prediction is equal to the pKsp and portlandite precipitates. Slight variation in either activity coefficient for the PIT calculations could also mean that Q becomes less than pKsp and no precipitation would occur. Alternatively, variation in the pKsp could result in similar changes in predictions.

Since the solubilities are generally higher for solids in nonideal solutions, there should be a possibility that no precipitation is predicted when nonideal solutions are modeled. The Pitzer calculations using the HMW database predicted no precipitation even though a precipitate is predicted when the PIT database is used. The reason for the discrepancy is that the Ksp value for portlandite in the HMW database is significantly less than in the PIT database. Precipitates are considered more likely in the simulations using the Pitzer equations than in the Debye-Hckel simulations. The Debye-Hckel relationships tend to predict activity coefficients that are lower than the coefficients from the Pitzer equations when solutions are concentrated. The expected outcome is reflected in Table II when the low value of the Ksp in the HMW database is factored into the comparison.

The difference in the precipitates predicted from the two ideal solution calculations shows the significance of a small variation in solubility products. There is a 12% difference in the masses of the precipitates from the two predictions, resulting from a variation of about 16% in the listed solubility products. A larger difference in the listed solubility products, such those of gypsum (Table III), could be significant in predicting precipitation or no precipitation from different solutions.

Results of Problem #2 show that there is a significant difference in ideal and non-ideal predictions when the precipitating species are in low concentration and the solution is of high ionic strength (high concentration). Debye-Hckel relationships tend to predict lower activity coefficients in concentrated solutions than do Pitzer relationships. Results from B-dot calculations reflect the tendency to predict lower activity coefficients, but the Davies results show larger activity coefficients than calculated with the Pitzer equations. The reason for larger activity coefficients from the Davies equation is not clear. Results of Problem #2 show the importance of including non-ideality in calculating precipitation from solutions dilute in the precipitating species.

The results of Problem #2 also show the effects of different Ksp values in different databases. The PIT and HMW databases show different values for the Ksp of portlandite. The result is that calculations using the PIT database predicted

precipitation of portlandite and those using the HMW database did not. The results of Problem #2 reinforce the principle that uncertainties for predicting chemical speciation behavior are important and should always be considered when using computer codes.

Precipitation Problem #3

Gypsum and portlandite are precipitated simultaneously in Problem #3. The amounts of portlandite range from 28.6-29.8 g/l, and the amounts of gypsum range 181 to 192 g/l (Table III). The exception is 26.3 g/l portlandite and 197.6 g/l gypsum that result from the unusually low Ksp for gypsum in one of the data sources used for evaluating ideal solutions.

The nonideal and ideal approaches predict similar amounts of portlandite and gypsum. The consistency in the precipitates is illustrated by the following. Let x be the amount of Ca++ that precipitates as Ca(OH)2 and y the amount of Ca++ that precipitates as CaSO4.2 H2O. The ratio y/x ranges from 2.66 to 2.77 for the nonideal calculations and is 2.89 for the ideal calculation, showing the preference for gypsum precipitation. A second ratio is also defined. Let xreal be the amount of portlandite from any of the nonideal solutions and let xideal be the amount of portlandite that is expected from an ideal solution. The ratio of xreal/xideal ranges only from 1.01 to 1.02 and suggests that the nonideal and ideal predictions are similar. The same ratio is calculated for gypsum with similar values of xreal/xideal. Thus, ideal or nonideal approaches predict essentially the same amounts of gypsum and portlandite, and either approach can be used. The nonideality of the solution changes only slightly the amounts of the precipitating species. The results of Problem #3 show the effects of higher solubility in nonideal solutions. In each case the predicted gypsum is less when nonideality is considered than when the solutions are assumed ideal.

Results of Problems #1 and #3 show that predicted amounts of precipitates from ideal solutions can be similar to predictions from non-ideal solutions. The reason for the apparent agreement lies in the calculation of the Ksp and Q values. As calculated with Eq. 1, Q is the product of ion concentrations and activity coefficients derived from the EQ3/6 output. Nonideal activity coefficients tend to be either greater than or less than 1 as shown in Table III. The value of Q is thus a product of large and small numbers, and apparent differences in Q's are minimal when calculated from ideal and nonideal solutions. The activity products, however, do not necessarily cancel in every case, leaving open the possibility of significant divergence of nonideal and ideal predictions. Brown and Ebinger (20) consider the phenomenon of ideal-solution-predicted precipitation agreeing with nonideal-solution-predicted precipitation in more detail.

Precipitation Problem #4

Problem #4 begins with the final solutions from Problem #3. The final concentrations of the different ions of Problem #3 vary in each prediction, so the starting concentrations for each prediction of Problem #4 are not equal. It is expected that the amounts of precipitates predicted from different approaches will not be very similar (Table IV). Water is evaporated to 45 g from 1000 g in each prediction . Four minerals are predicted from the ideal and nonideal calculations: portlandite, halite (NaCl), thenardite (Na2SO4), and glauberite (Na2Ca(SO4)2). Precipitation of no other minerals is predicted.

All starting solutions are concentrated with regard to Na+ and Cl-, and agreement in the amount of Na-containing precipitates is expected when ideal and nonideal predictions are compared. The results show that Ca-containing solids do not agree when the ideal and non-ideal predictions are compared. The discrepancy in the Ca-containing compounds is due to low concentration of Ca++ in the starting solution for Problem #3 and, thus, for Problem #4. Non-ideality is important when the precipitating species are in low concentration. In Problem #3 Ca++ precipitates from the ideal solutions and the Pitzer prediction. In Problem #4 the Pitzer and ideal approaches predict the smallest masses of Ca-containing solids because there is less Ca++ available to precipitate. There is still significant difference between the ideal and the four non-ideal predictions due to the effects of non-ideality on the solubility of Ca-containing solids.

Halite comprises 91-94% of the mass of the precipitate. Halite precipitates because the starting solutions contain large amounts of soluble sodium and chloride ions from the original solution for Problem #3. Calcium, hydroxyl, and sulfate ions have largely been removed in the precipitates predicted in Problem #3. The two

calculations using the Pitzer activity coefficient relationships predict about 10% more halite than the two calculations using Debye-Hckel activity-coefficient relationships.

The reason for the Pitzer calculations predicting more halite than the Debye-Hckel calculations is not clear. In spite of the different activity coefficient relationships and different databases, the ionic concentrations in the water are not greatly different, nor are the calculated activity coefficients. The principal differences in the predictions seem to be the larger number of aqueous species considered in the Debye-Hckel calculations and the solution concentrations of the species considered. This result illustrates the importance of knowing the contents of the databases used for predictions and the limitations of the databases.

The predicted amounts of portlandite, thenardite, and glauberite differ significantly among the calculations. The high value of portlandite is 46% above the low value, the high value of thenardite is 44% above the low value, the high value of glauberite is over twice the low value. There seems to be little correlation of the high and low values with the methods of calculation. The cause for the range of values in the results could lie in the number and kind of species considered, the databases utilized, or the equations used to calculate the activity coefficients. There is also large variation in ion concentrations and activity coefficients. Variations in the results of the Pitzer activity-coefficient relationships are less than the variations in the results derived from the Debye-Hckel relationships. The Pitzer relationships are supposed to be more appropriate for concentrated solutions, and they show that quality here.

Mass Balances

Calcium mass balances in each prediction of Problem #1 are excellent, showing much less than 0.1% error in each case. Calcium mass balance in each prediction of Problem #2 show less than 0.2% difference and are insignificant. We expected more error in the predictions of Problem #3 since the calculations were more complex. The mass balances are satisfactory for calcium and sulfur, the important ions in precipitate formation and in solution after precipitation.

Some of the mass balances for Problem #4 are not as good as in Problems #1-3. Calcium mass balances for each prediction of Problem #4 agree within 1% which is satisfactory for modeling an evaporation campaign. Mass balances for sodium differ by up to 17%, chloride mass balances differ up to 16%, and sulfur mass balances differ up to 7%. The large discrepancies in sodium, chloride, and sulfur raise questions about how well the EQ3/6 codes model evaporation when the volume of the solution decreases by more than a factor of 20. These discrepancies also illustrate the magnitude of the uncertainties in predictions that depend on databases considered very comprehensive in scope.

Solubility Products

We discuss above that variation of solubility products from different databases can be significant and occasionally drastic. For example, the solubility product of gypsum differs by a factor of 7.8 in the two sources used for the ideal solution predictions, and the solubility product of portlandite differs by 17% in the same databases. One test of the consistency of different thermodynamic databases is to calculate activity products from the concentration and activity coefficient information, then compare the calculated Q to the database Ksp values. The EQ3/6 calculations show excellent agreement between calculated and reference (database) values of the different precipitates. The Q values predicted for portlandite from the COM database agree well with the Ksp in the database in Problems #1 - #3, and the same is true for gypsum in Problem #3. The Q calculated from the PIT and HMW databases agree with their respective reference Ksp values. The PIT Ksp values are essentially equivalent to the Ksp values calculated from the COM database. The HMW Ksp values in the databases are 78% greater than the PIT values for portlandite and 26% greater for gypsum. Although large, the differences in Ksp between the PIT and HMW databases is within a factor of 2 which is a relatively small difference for Ksp values.

The free energies of formation for the above constituents are identical in the COM and PIT databases, and it is not surprising that the solubility products calculated using these databases are also identical. However, the activity products calculated from concentrations and activity coefficients in Tables I-IV are slightly different than the solubility products listed in the databases. The solubility products listed in the databases agree with solubility products hand calculated from free energy

data in the databases. The differences between the activity products calculated from the concentrations and activity coefficients and the Ksp listed in the databases are small. The difference is probably caused in part by the convergence criteria used in the calculations.

CONCLUSIONS

The EQ3/6 chemical speciation codes satisfy simple qualitative consistency tests posed by the four elementary equilibrium problems. Usually precipitates show higher solubilities when the solutions are regarded as nonideal than when they are considered ideal. The codes predict almost complete precipitation when solids are of low solubility. Both of these qualitative aspects agree with expectations and chemical theory. The speciation codes also satisfy the simple quantitative consistency tests posed by the four problems. Solubility products are identical when derived from the same database, and numerically close when calculated with values from different databases. Mass balances in the first three problems are consistent regardless of the approach used on a particular problem. The calculations converge to values of concentrations and activity coefficients that give activity products close to Ksp values listed in the databases and those calculated from listed Gibbs free energy data.

The results from solving the four problems using different databases suggest that the amounts of precipitates predicted from ideal solutions are similar to those predicted from nonideal solutions when the initial concentration of the precipitating species is high. The similarity in results between the ideal and nonideal approaches decreases when the initial concentration of the precipitating species is low. While our results show apparent agreement in predictions from ideal and nonideal approaches, extrapolation of our results to other concentrated solutions is not warranted. The predictions of the four problems illustrate well the warnings in the chemical modeling literature about the use of numerical speciation codes. There were large and unpredictable differences in the concentrations and activities of the various ions in the solutions that resulted in significant differences in amounts precipitated in Problem #4.

There are circumstances where a combination of concentrations and activity coefficients allow ideal solution theory to predict precipitation from nonideal solutions with reasonable accuracy. These circumstances can exist even when activity coefficients do not cancel in calculating activity products and/or solubility products.

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17-23

CALCULATION OF PROJECTED WASTE LOADS FOR TRANSURAIC WASTE MANAGEMENT ALTERNATIVES*

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ABSTRACT

The level of treatment and the treatment and interim storage site configurations (decentralized, regional, or centralized) impact transuranic (TRU) waste loads at and en route to sites in the U.S. Department of Energy (DOE) complex. Other elements that impact waste loads are the volume and characteristics of the waste and the unit operation parameters of the technologies used to treat it. Projected annual complexwide TRU waste loads under various TRU waste management alternatives were calculated using the WASTE_MGMT computational model. WASTE_MGMT accepts as input three types of data: (1) the waste stream inventory volume, mass, and contaminant characteristics by generating site and waste stream category; (2) unit operation parameters of treatment technologies; and (3) waste management alternative definitions. Results indicate that the designed capacity of the Waste Isolation Pilot Plant, identified under all waste management alternatives as the permanent disposal facility for DOE-generated TRU waste, is sufficient for the projected complexwide TRU waste load under any of the alternatives.

INTRODUCTION

Projected waste loads are important considerations in evaluating management alternatives for transuranic (TRU) waste generated at sites in the U.S. Department of Energy (DOE) complex. Projected TRU waste loads can be used to (1) determine whether the capacities of existing treatment, storage, and disposal (TSD) facilities are sufficient, (2) estimate the required sizes and expected costs of new TSD facilities, and (3) assess the potential health risks from transportation of the waste and operation of the facilities.

This paper describes the parameters used to calculate TRU waste loads and compares waste loads calculated for the TSD of TRU waste under various management alternatives considered for sites in the DOE complex.

TRU WASTE MANAGEMENT ALTERNATIVES

The 10 TRU waste management alternatives evaluated for DOE sites consist of various combinations of treatment levels, treatment processes, and treatment and storage locations, including no action. The three levels of treatment considered are described below.

The minimum level of treatment reduces TRU components in the waste to levels acceptable for disposal under the current waste acceptance criteria of the Waste Isolation Pilot Plant (WIPP) in New Mexico, currently identified as the permanent disposal facility for DOE-generated TRU waste. Only waste streams that do not already meet these disposal criteria will be treated, usually by solidification and appropriate packaging.

The intermediate level of treatment meets the waste acceptance criteria of WIPP and further treats the waste to reduce gas generation during degradation of organic materials and corrodible metals. Shredding and grouting debris waste and using non-iron-based containers reduces gas generation in the treated waste after disposal.

The highest level of treatment considered would destroy or stabilize all hazardous constituents in the waste to comply with the Land Disposal Restrictions (LDRs) of the Resource Conservation and Recovery Act. For the purposes of this study, incineration is considered a representative process for treating TRU waste to meet LDRs.

Treatment and interim storage sites also vary among the TRU waste management alternatives. TRU waste from individual DOE sites may be treated on the site where it was generated (decentralized configuration), it may be sent to regional centers for treatment (regional configuration), or it may be sent from all sites to WIPP for treatment (centralized configuration). All treated TRU waste would be stored at the treatment site until being shipped to WIPP for disposal.

TRU WASTE LOAD CALCULATION PARAMETERS

The level of treatment and the site of treatment and interim storage impact TRU waste loads at and en route to a given DOE site. Other elements that impact these waste loads are the volume and characteristics of the waste and the unit operation parameters of the technologies used to treat it.

Waste Volumes

The treatment sites identified in a given alternative determine the volume of waste treated at a site. For example, the volume of TRU waste to be treated at Site 1 under Alternative A is derived by combining the TRU waste contributed by all sites whose TRU waste is being sent to Site 1 for treatment under Alternative A.

The annual waste loads contributed include TRU waste inventories plus 20 years of projected TRU waste generation, divided by 10 (assuming treatment will begin 10 years from now). TRU waste inventory and projected generation volumes for each DOE site can be found in the Interim Mixed Waste Inventory Report (DOE 1993) and the Integrated Data Base for 1992 (DOE 1992). At the end of 1991, there were approximately 65,000 m3 of retrievably stored contact-handled (CH) TRU waste and about 4,300 m3 of retrievably stored remote-handled (RH) TRU waste at DOE sites (DOE 1993). (Packaged TRU waste with a surface dose rate less than or equal to 200 mrem/h is categorized as CH-TRU waste, and that with a surface dose rate of greater than 200 mrem/h is categorized as RH-TRU waste.)

An estimated 54,000 m3 of the waste (PNL 1994) could result from environmental restoration (ER) activities under a mixed-land-use scenario (assuming semi-restricted access to sites after remediation).

Waste Characteristics

Information about the characteristics of TRU waste has been obtained through process

knowledge supplemented by x-ray examination, radioassay, analysis of the gas in storage drum headspace, and sampling of the contents of a limited number of waste containers. Most TRU waste exists in solid form (e.g., contaminated protective clothing, rags, glassware, and machine parts), but some is in liquid sludge form. On the basis of its physical and chemical characteristics, TRU waste is grouped into waste stream categories, each of which has its own treatment train to facilitate efficient processing. The waste stream categories include aqueous liquids, organic liquids, solid process residues, soils, debris, special waste, inherently hazardous waste, and unknown.

Figure 1 illustrates treatment trains for five TRU waste streams being treated at a high level in order to meet LDRs. All treatment trains include a pretreatment step to segregate the waste into waste streams by separating liquids from solids or sorting out solids that have different physical properties. Currently, waste load calculations do not include three of the waste streams. The three, special waste, inherently hazardous waste, and unknown, constitute less than 10% of total TRU waste volume and are assumed to be set aside to await special processing.

Technology-Specific Unit Operation Parameters

The unit operation parameters of the technologies used to treat the waste influence waste load as well. These parameters include (1) volume factor, which is the ratio of product output stream volume to incoming stream volume, (2) mass fraction of product stream relative to input mass, and (3) mass fractions of secondary output streams relative to input mass. Unit operation parameters are specific to treatment technologies. For example, the volume factor for incineration is 0.1, for wet oxidation is 0.9, and for solidification is 1.2.

The ratios of specific contaminant quantities in TRU waste after treatment to their quantities in the waste before treatment are also calculated using unit operation parameters. The ratio of mercury in the treated product to mercury in the incoming waste stream, for example, is 0.01 for incineration, 0.1 for wet oxidation, and 1 for solidification. (In this case, solidification does not reduce contamination levels, but reduces a waste's ability to release the contaminant.)

Calculation of waste loads requires identifying the portions or combination of portions of incoming and secondary output waste streams assigned for each of the unit operations in a treatment train. For the incoming waste streams the assignments selected are generally averages of values conservatively estimated from analysis of data in the Interim Mixed Waste Inventory Report (DOE 1993). The assignments of secondary output streams are based on the expected makeups of the incoming streams and, in some instances, on engineering judgements about the expected waste stream behavior in a given process operation.

CALCULATED TRU WASTE LOADS

Total projected CH-TRU waste loads were calculated for the various TRU waste management alternatives using the WASTE_MGMT computational model (AVCI 1995). WASTE_MGMT accepts as input three types of data: (1) the waste stream inventory volume, mass, and contaminant characteristics by generating site and waste stream category; (2) TSD unit operation parameters; and (3) waste management alternative definitions. Some TSD processes generate secondary output streams that are also followed through the treatment process. For example, the primary output stream of incineration is ash, but a secondary stream of high-chloride salt waste is generated in the off-gas treatment of combustion gases (see Fig. 1).

Table I shows the projected annual complexwide waste loads of CH-TRU waste for four representative treatment technologies under three of the alternatives evaluated. As expected, the decentralization/treat-to-meet-minimum-disposal-criteria alternative, which does not use shredding or incineration, has the highest waste load for packaging among the three alternatives. The high waste load for shredding and low waste load for packaging in the regionalization/intermediate-level-of-treatment alternative indicates the effectiveness of the shredding process for reducing the volume of TRU waste, as well as for reducing gas generation to improve performance at WIPP.

Table II shows the projected total complexwide CH-TRU waste loads for storage and disposal, including waste loads from ER activities. The ER waste loads, which constitute approximately 30% of the total, were calculated from ER waste volumes projected to be generated under the mixed-land-use scenario assuming semi-restricted access to the site after remediation.

The maximum total projected CH-TRU waste load for disposal at WIPP under any of the

waste management alternatives is approximately 130,000 m3, which is less than the designed capacity of 170,000 m3 for CH-TRU waste at WIPP.

SUMMARY

CH-TRU waste loads were projected for treatment and storage facilities at each DOE site for all management alternatives considered. Results indicate that the designed capacity of WIPP is sufficient for CH-TRU waste loads currently projected for the DOE complex under any of the alternatives, as well as for projected TRU waste loads from ER activities assuming semi-restricted access to sites after remediation. The projected waste loads can be used to calculate size requirements and estimated costs of new treatment or storage facilities and to assess potential health risks from transportation of waste and operation of facilities. TRU waste management decision making can be facilitated by comparing calculated waste loads, costs, and risk among TRU waste management alternatives.

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DEVELOPMENT OF A PROTOTYPE GAMMA CAMERA (ALADIN) FOR USE IN DECOMMISSIONING NUCLEAR FACILITIES

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ABSTRACT

Mapping the gamma activity of irradiating zones is often an important prerequisite in dismantling nuclear facilities. This operation is necessary to define a suitable decommissioning strategy before any work begins; it is also required during the procedure to measure the residual activity wherever dose rates are too high to allow human intervention. This paper summarizes the work carried out to develop a prototype imaging system designed to display radioactive sources superimposed in real time over a visible light image on a video monitor. This project was developed from an earlier off-line system.

The gamma photons are collimated by a double cone system. The imaging system comprises a transparent scintillator bonded to the fiber-optic window of an ultrasensitive camera. The camera was miniaturized to meet specification requirements: with its radiological shielding, the gamma camera weighs 40 kg and is 120 mm in diameter. The processing system is compatible with a real-time camera, and small enough for use at any nuclear site. The point-source angular resolution is 1.4 for 60Co and 0.8 for 137Cs. The dose rate sensitivity limit is approximately 0.01 mGy.h-1. Process reliability was confirmed by tests in a high-level radiometallurgy cell at an actual decommissioning site.

INTRODUCTION

Existing detectors for absorbed dose rate measurements are based on the ionization chamber principle. When the objective is to localize radioactive emission zones, this equipment is usable only in simple situations. These detectors generally require human intervention, and are inefficient in the presence of multiple sources. They are unable to determine the true location of numerous radioactive sources, in

which case other methods are required.
Gamma mapping is a valuable technique for a variety of operations during decommissioning of nuclear facilities with highly irradiating zones:
prior to dismantling, in order to define the overall decommissioning strategy;
during the dismantling procedure to monitor the residual activity wherever the equivalent dose rates are too high to permit human intervention;
to organize decommissioning tasks in highly irradiating environments, and to define decommissioning procedures.
A photographic technique for localizing radioactive sources was developed by the Commissariat l'Energie Atomique (CEA) during the 1980s. The system uses a pinhole camera, in which the objective consists of a gamma radiation collimator and the visible light image is formed by a simple pinhole in a light-tight body opaque to radiation. An ordinary photographic emulsion is used for the visible light image, and a radiation-sensitive film reveals radioactive sources by differences in contrast. A gamma-transparent shutter is opened only for the visible light photo. After developing the films, the two images (exposed under identical geometric conditions) are superimposed to form a composite image on which radioactive emission sources in the field of view are immediately identifiable.
Numerous experimental measurements in a variety of nuclear facilities have demonstrated the advantages of this technique, notably to optimize radiation protection and dose limitation. Nevertheless, a number of major drawbacks in the initial procedure (uncertain interest of the observed zone, excessively long exposure times, complex processing required before obtaining a result) prompted us to develop a real-time technique (1). Work began in 1988 by the CEA, and a research contract was signed in October 1991 with the Commission of the European Communities to design and build a prototype imaging device meeting the following requirements:
dimensions compatible with telemanipulator penetrations 170 mm in diameter in the target cell;
weight not exceeding 50 kg to allow handling and remote manipulation;
allowance for subsequent installation on a carrier system to allow remote orientation;
maximum sensitivity to radiation emitted by 137Cs and 60Co;
rapid source localization (in no more than a few minutes) without diminishing the angular resolution of the off-line system (1 to 1.5).
ALADIN REAL-TIME GAMMA MAPPING SYSTEM (2)
The double-cone collimator design was maintained. In order to allow real-time operation, the photographic and radiation-sensitive emulsions were replaced by a scintillation screen. The scintillator converts the incident g radiation to visible wavelengths that can be registered by the camera. A transparent screen does not interfere with the visible light exposure of the scene. The camera thus observes the scene through the transparent screen when the shutter is open, and the radiation sources when the shutter is closed. An ultrasensitive camera is used to collect the weak light emitted by the scintillator. The camera supplies a standard CCIR video output (50 Hz, 625 lines). The very principle of this technique ensures perfect superimposition of the gamma sources on the visible light image of the scene.
Figure 1 shows the main components of the system: the objective or collimator, the image converter, the ultrasensitive camera and the computer processing unit. Each subsystem comprises a complete step in processing the initial signal, and the overall system performance depends as much on the inherent performance of each element as on their coherent integration.
Objective
The front part of the device serves as a collimator for g photons, and as a lens for visible light photons. A shutter that is virtually transparent to the observed gamma energy range allows switching between the two operating modes.
In visible light, the pinhole objective forms an image of the scene observed with a definition that depends on the diameter of the opening and the focal length separating it from the image plane. In the gamma range, the collimator limits the gamma photon flux from the radioactive source to a thin beam in order to determine its direction. The collimator determines the spatial resolution, and thus has a controlling effect on the final quality of the "gamma image". The double-cone collimator is a simple device covering a relatively large and fixed solid angle with a suitable collimating power. The spatial resolution depends on the observed energy level; the cone angle represents a tradeoff between the collimation power and the

observed solid angle. The collimator material has a significant effect on the collimating power; the prototype gamma camera allows for interchangeable tungsten alloy collimators with different aperture angles.

Image Conversion

The g-ray converter is a scintillation crystal meeting specification requirements for both visible light and g ray operation. In visible light it must offer maximum transparency and must not deform the image. At the same time, it converts g radiation into visible light, to which the photocathode is sensitive. The major parameters include the light efficiency, the crystal scintillation spectrum (which must be adjusted with the acceptance band of the input photocathode) and the high-energy g radiation stopping power (referenced to 60Co). Other criteria must also be taken into consideration, including the optical refraction index, which affects both the resolution and sensitivity, and the hygroscopic properties of the crystal, which must be low for ease of use.

Signal Amplification

Interactions with g radiation generate very little visible light in the scintillator, and strong amplification is required to detect the signal. The imaging device is a CCD camera coupled with two image intensifiers in series: a "first-generation" unit with a 50 mm input window, followed by a "second-generation" microchannel plate unit. Coupling the two image intensifiers in series provides an overall photon gain on the order of 106.

The input photocathode of each image intensifier tube converts the photon image into an electronic image. It is characterized by its spectrum response and sensitivity. In order to maximize the detection signal/noise ratio, particular attention was given to tuning the crystal scintillation spectrum and the photocathode spectrum response. The second important point is the dimension of the input window, which affects the spatial resolution.

Image Processing

Digitizing the video signal from the camera does not provide a usable image directly. As the gamma image is a statistical phenomenon, the signal must be integrated by summing a large number of corrected images to determine the radioactive emission centers. Dedicated circuitry is required to separate the image acquisition and processing functions.

Acquisition. The images are integrated at the video scanning rate (25 images/sec) by means of a dedicated circuit, which also subtracts the background from each frame to offset the poor black image from the CCD. The image is then sent to the processing circuit.

Display. The system is designed to provide a real-time on-screen display of the visual image from the camera together with the gamma source image as it is acquired: to ensure fully interactive operation, the radioactive source image is displayed in real time as it forms.

Processing. The correlation of a visible light image and a gamma image requires multiple processing: isocontours, filtering, false color enhancement, etc. This is done by a dedicated workstation, using custom application software developed specifically for the station to archive the images, quantize the relative dose rates, provide various superimposition modes and display a color map directly usable at the decommissioning site.

PROTOTYPE COMPONENT OPTIMIZATION

Detection Circuit

Theoretical optimization studies were conducted for the collimator and for the screen and fiber-optic system. Laboratory tests were then carried out to characterize the prototype in various detection circuit configurations.

Double Cone Collimator

The double-cone collimator is defined by three geometric parameters: the aperture half-angle, the pinhole radius and the extension or focal length between the gamma pinhole and the image plane. The collimator material determines the linear activation coefficient mb(E) for the observed gamma radiation of energy E (Fig. 2). The flat portion of the profile is formed by the direct g photon flux passing freely through the aperture. This is the "primary spot", i.e. the zone of highest density, corresponding to the projection of the pinhole onto the image plane P in the direction of the observation. The surrounding region of diminishing density is due to the incidence of the flux attenuated by the shielding, the thickness of which increases with the off-axis distance. The step S in the distribution profile

corresponds to the shielding thickness encountered by the photons outside the aperture radius r0. The collimator material, 97% tungsten alloyed with nickel and iron (s.g. 18.5), was selected to obtain the maximum gamma ray stopping power for the smallest possible dimensions.

The collimator is characterized by two fundamental criteria: its sensitivity (which depends on the sum of the direct and attenuated flux transmitted by the collimator) and its angular resolution (i.e. the ability to discriminate between adjacent point sources). The gamma image obtained with this type of collimator represents a tradeoff between the sensitivity, the resolution and the field of view.

The primary requirement was to ensure optimum resolution for 60Co and 137Cs. A theoretical study revealed the advantages of a modular configuration with interchangeable collimators and a mechanical system for adjusting the extension over a 50 mm travel range between the collimator and the image plane. The two collimators are compared in Table I.

Scintillation Screen and Fiber-Optic Window

The detection system comprises a scintillation crystal bonded to the camera input window. The window must be a fiber-optic design to ensure transparency for the visible light image used to situate the gamma sources.

For a given collimator, the sensitivity and resolution of the unit depend in the opposite manner on the dimensions of the converter screen. A thick scintillator ensures high radiation stopping power, but also diminishes the angular resolution; conversely, a thin screen ensures better spatial resolution, but with poor sensitivity. The optimum tradeoff must also take a number of other factors into account:

- the inherent stopping power of the crystal, depending on its composition;
- the luminous efficiency and optical properties of the crystal;
- optical coupling between the scintillator and the first image intensifier;
- the observed dose rate level and the nature of the incident radiation.

The physical properties of the principal inorganic crystals considered for the scintillation screens are indicated in Table II. The two crystals selected for testing were bismuth germanate (BGO) and thallium-activated cesium iodide (CsI(Tl)). During experimental testing, CsI(Tl) exhibited substantially greater sensitivity for practically the same resolution.

Fiber optics must be used for the camera input window to allow localization of nuclear events. When a gamma photon interacts with the crystal, it loses part or all of its energy to atoms in the screen which in turn emit de-excitation photons in the visible spectrum, isotopically from the point of interaction. A fraction of these photons enters the optical fibers, depending on their numerical aperture and on the crystal refraction index. The optical fibers available on the market were tested at numerical apertures of 0.33, 0.66 and 1.00). The results clearly showed that reducing the numerical aperture from 1.00 to 0.66 enhances the angular resolution; at the present state of development, however, this requires an additional fiber wafer on the camera input window that results in a significant drop in sensitivity. We therefore chose to use fibers with a numerical aperture of 1.00.

EXPERIMENTAL RESULTS

For the collimators considered for our image acquisition system in the observed energy range (0.66 to 1.25 MeV), a scintillator thickness of 2 and 4 mm provides an acceptable balance between sensitivity and resolution. The mean experimental results obtained for observation of 60Co sources are indicated in Table III for various possible configurations.

The results obtained with two 60Co point sources confirmed and quantified the following characteristics: higher resolution is obtained with the 19 collimator; increasing the extension significantly enhances the resolution; the resolution is inversely proportional to the thickness of the scintillator. The point-source angular resolution is approximately 1.4 for 60Co and 0.8 for 137Cs.

Response linearity tests were conducted to evaluate the possibility of using 60Co and 137Cs point source images to quantify the irradiation dose rates. The results observed with different detection configurations showed satisfactory linearity of the response over at least two orders of magnitude of absorbed dose rates. The linearity range may be extended to higher absorbed dose rates simply by modifying the amplification gain. Typical experimental results for a 60Co source observed through a 26 half-angle collimator are shown in Fig. 3.

Amplifier Circuit

The amplifier circuit configuration was determined to meet the compact design requirements and to ensure compatibility with the gamma camera operating principle, i.e. the use of a single digital imaging system to ensure perfect superimposition of both the visible light and gamma images. The system (Fig. 4) comprises a "first generation" (electrostatic lens focusing) intensifier tube coupled to a CCD camera that is in turn intensified by a "second-generation" (proximity focusing) tube. In order to adapt the amplifier spectrum response to the CsI(Tl) scintillation crystal response, the first amplifier stage includes a photocathode with peak response in the blue wavelengths. The maximum diameter of the configuration adopted is 70 mm, and the overall length is less than 170 mm.

Computerized Image Processing System

Generally very few nuclear events occur in the scintillator during an individual video frame, so that simply displaying the incident image on a video monitor is not sufficient to localize the sources in the field of view. When the number of irradiation events in a given direction is too low, they cannot be distinguished from the camera's inherent pulse noise. Moreover, multiple sources are not sufficiently discriminated regardless of the noise level.

Instead, the statistical nature of the events is used to reconstitute "nuclear spots" on the image plane. The camera noise and the ambient background radioactivity are uniformly distributed over the image plane, while interactions due to irradiation sources are statistically localized. The system then simply integrates a suitable number of video frames to generate the source images. The device specifications called for a standard frame rate of 25 images per second. The system comprises a Hamamatsu DVS 3000 real-time acquisition unit and a DEC 5000 UNIX workstation for image processing, display and archival (Fig. 5).

Modular software design was preferred, and three modules were developed: an image acquisition module with provision for real-time control by the operator; the actual image processing module; and a module controlling the final superimposition of the visible light and gamma images.

Radiation Protection of the Detection Unit

The shielding is designed to provide effective collimation of the high-energy gamma photons, while protecting the electronic and optoelectronic components from radiological damage; it also provides mechanical protection, allows handling of the camera assembly and basic photographic functions (shutter, focusing extension, etc.). The specification requirements called for a compact unit. The result is a compact, cylindrical unit 120 mm in diameter and 445 mm long.

TESTING AT AN ACTUAL DECOMMISSIONING SITE

The prototype unit (Fig. 6) was tested in the radioactive cells of the RM2 radiometallurgy facility at the CEA's Fontenay-aux-Roses center, currently being decommissioned by UDIN.

The gamma camera was mounted on a specially designed carrier unit and inserted vertically into irradiating cells through overhead ports. Because of the relatively high ambient dose rates (5 to 30 mGy.h-1) the configuration with the lowest sensitivity and the highest resolution was selected, i.e. the small-aperture (2 19) collimator with a 2 mm CsI(Tl) scintillator.

Figure 7 is a typical image obtained in less than 2 minutes in Cell 2. The photo shows a strong irradiating source (about 2 mGy.h-1 at 1 meter) due to accumulated contamination around the edge of an intercell conveyor access hatch.

CONCLUSION

Since the completion of this work program at the end of 1993, a prototype gamma camera is now operational for mapping of irradiating zones. The unit provides an angular resolution of about 1.4 for 60Co and 0.8 for 137Cs, with a sensitivity of about 0.01 mGy.h-1 at the camera position for 60Co and 137Cs point sources.

Tests under actual decommissioning conditions demonstrated the satisfactory operation of the unit and its advantages, notably for obtaining a radioactive inventory prior to dismantling. The device design and performance allow gamma mapping to be performed in any irradiating room or cell provided with direct access openings, even of small dimensions. The system is compatible with multiple sources and large aperture angles (38 or 52), and is capable of achieving high measurement rates because of its detection sensitivity (for example, a 60Co or 137Cs source delivering 0.1 mGy.h-1 at the camera position can be detected in 60 seconds. Moreover, the system may be remote controlled.

Although the device was initially developed for decommissioning purposes, it may be

applied to other nuclear activities including maintenance, surveillance of high risk zones, process control and waste or fissile material management applications. The primary interest of the gamma camera is in the field of radiation protection. The results obtained are extremely helpful in applying the dose limitation principle.

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A HISTORY OF SOLID WASTE PACKAGING AT THE HANFORD SITE

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ABSTRACT

Since the initiation of the defense materials product mission, a total of more than 600,000 m3 of radioactive solid waste has been stored or disposed at the U.S. Department of Energy's (DOE) Hanford Site, located in southeastern Washington State. As the DOE complex prepares for its increasing role in environmental restoration and waste remediation, the characterization of buried and retrievably stored waste will become increasingly important. Key to this characterization is an understanding of the standards and specifications to which waste was packaged; the regulations that mandated these standards and specifications; the practices used for handling and packaging different waste types; and the changes in these practices with time.

INTRODUCTION

The generation of radioactive solid waste began coincident with the defense materials production mission begun at the Hanford Site in 1944. Since then, the Hanford Site has been managed and operated by several onsite Government contractors. Throughout the years, these contractors developed standards and specifications in the form of waste acceptance criteria for the handling and packaging of solid wastes. As local, state, and federal regulations and requirements evolved, so did the waste acceptance criteria. Knowledge of the changes that occurred in waste handling and packaging will help to assure the safety and health of personnel involved in the restoration and remediation of historical waste and waste facilities, while providing adequate environmental protection.

BACKGROUND

The Hanford Reservation, a 1450 km2 tract of land located in semiarid southeastern Washington State, began operations in 1944 as a major site for production of plutonium and other nuclear materials for World War II. Between 1944 and 1970, approximately 388,000 m3 of unsegregated solid wastes were buried at the Hanford Site, as both transuranic (TRU) and low-level waste (LLW) were disposed in shallow land trenches. In 1970, the Atomic Energy Commission (AEC) issued Immediate Action Directive 0511-21 directing AEC sites to segregate TRU wastes from other waste types (1). The AEC further directed that TRU wastes be packaged and stored as contamination-free packages for at least 20 years. To date, a total of over 600,000 m3 of solid radioactive wastes have been buried or stored at Hanford, including 595,000 m3 of LLW and 15,500 m3 of TRU waste (2). Table I inventories the waste

stored or disposed onsite at the end of calendar year 1992.

EARLY WASTE MANAGEMENT AT HANFORD

Prior to the first waste acceptance criteria, which were released in the late 1960's, early waste management procedures were simple and primarily based on operator safety. Early Hanford procedures define solid radioactive waste as "radioactive waste which is essentially dry, or whose fluids are of small volume and are contained or absorbed to the extent that they are essentially immobile during storage" (3). Since documentation that refers to solid waste is scanty during the early years of Hanford operation, it is assumed that this definition, or one similar, was observed in the 1940's and 1950's. In the mid-1950's, radioactive wastes were classified as either "dry" or "industrial" waste. Dry wastes were described as wastes containing little contamination, including absorbent tissues, rubber gloves, wood, metal parts, broken glassware, small tools, and other small miscellaneous items. Industrial wastes consisted of large items or failed equipment. Radioactive solid waste has been disposed by shallow land burial or stored in underground vaults and caissons since 1943. The Hanford Site is divided into several distinct "areas". The three primary areas of interest in this study include the 100 Area, where the nuclear reactors are situated; the 200 Areas (200 East and 200 West), where the fuel reprocessing, plutonium recovery, and waste management facilities are located; and the 300 Area, where fuel fabrication took place, but which now houses mostly research and development facilities and administrative offices. Prior to 1968, all waste buried in the 200 Areas was generated as a result of the fuel reprocessing operations. After 1968, the waste generated by the 300 Area operations was sent to the 200 Areas for burial. Wastes from the reactor operations in the 100 Area have been sent to the 200 area for burial since 1973.

HANFORD WASTE ACCEPTANCE CRITERIA

In the late 1960's, the first waste acceptance criteria documents were written for the 200 and 300 Areas (4,5). These documents provided specifications and standards for industrial wastes, as well as for chemical hazards control with respect to the burial grounds. Waste generators were required to segregate their waste with respect to compatibility and content. During this time, small materials were packaged in fiber drums, liquid wastes were acceptable only if absorbed by an inert absorbent material, and organic matter had to be sealed in plastic and packaged in wooden or metal containers.

In 1970, a new specifications and standards document, Specifications and Standards for the Burial of ARHCO Solid Wastes, ARH-1842, was released shortly after the AEC directed the segregation of TRU wastes (6). This document stated that generators and operators must segregate and package waste materials containing or suspected of containing, plutonium or other TRU radionuclides for containment and retrievability.

The Specifications and Standards for the Packaging, Storage, and Disposal of Richland Operations Solid Waste, ARH-3032, which was released in 1974, superseded the two documents for the 200 and 300 Areas (7). This document classified wastes into four different segregation groups: nonradioactive, non-hazardous, combustible wastes; low-level, non-TRU wastes; TRU wastes; and high-level wastes (HLW). Packages that contained less than 200 counts per minute beta/gama and less than 500 disintegrations per minute alpha contamination were classified as non-radioactive and disposed in the Central Landfill Facility. Solid wastes containing less than 10 nCi/g of plutonium and/or other TRU radionuclides were considered LLW and were further divided into combustible and noncombustible wastes, which were packaged separately. Solid wastes containing or suspected of containing greater than 10 nCi/g plutonium and/or other TRU radionuclides were considered to be TRU waste. Failed equipment and large items contaminated with TRU radionuclides were also included in this category.

The five revisions of RHO-MA-222, Hanford Radioactive Solid Waste Packaging, Storage, and Disposal Requirements, spanned from 1980 to 1988, and established new definitions for waste classes, placed restrictions on waste contents, provided new specifications for container designs, and included other key elements that directly impacted the waste classification system and segregation requirements. This was the first document that referred specifically to waste classifications, including radioactive solid wastes and TRU solid wastes. Additional requirements mandated segregation of combustible and noncombustible TRU waste. Combustible material was defined as "any material which can be ignited to produce fire through friction,

absorption of moisture, spontaneous chemical changes, or application of an external flame." This manual also included the requirement that waste containers be designed to provide the option of a vent or test connection capable of being fitted with or adapted to accept an air or vacuum hose, or gaseous diffusion vent.

One important change during the early 1980's was the revised AEC definition of TRU waste released in 1982 (8). TRU waste was redefined as waste, without regard to source or form, that was contaminated with alpha-emitting radionuclides of atomic number greater than 92 and half-lives greater than 20 years in concentrations greater than 100 nCi/g at the end of institutional control periods.

The Hanford Radioactive Solid Waste Packaging, Storage, and Disposal Requirements, WHC-EP-0063, was released in 1988 and superseded RHO-MA-222 (9). This document required generators to:

segregate, to the maximum extent feasible, TRU waste, LLW, radioactive mixed waste (RMW), chemically incompatible waste, and uncontaminated waste to facilitate cost effective treatment, storage, and disposal;

segregate radioactive solid waste recognizing waste minimization practices;

segregate TRU, LLW, RMW, hazardous, and non-contaminated waste into separate containers;

treat mixed TRU waste to destroy or remove and segregate any hazardous waste components, where permitted, feasible, and practical;

segregate certified TRU from non-certified TRU waste; and

segregate low-level RMW from TRU RMW.

Since 1970, retrievable storage requirements for TRU waste have mandated interim and long-term storage facilities. Beginning in the mid to late 1980's, changes in the management of dangerous or hazardous waste have mandated storage facilities for these wastes as well. Since 1985, non-radioactive hazardous waste was not accepted for burial in the central landfill. This waste was temporarily stored awaiting shipment offsite for treatment and final disposal. Starting in 1987, LLW containing hazardous constituents was segregated and temporarily stored awaiting future treatment before final onsite disposal.

Subsequent revisions of WHC-EP-0063 further developed low-level waste classifications; expanded the storage and disposal requirements for RMW; and added new sections for handling, storing, and disposing non-radioactive hazardous waste and a waste minimization section. They have also provided updated container design specifications, and included other key elements that directly impacted the waste classification system and segregation requirements. The latest revision of WHC-EP-0063, released in 1993, defines the current waste acceptance criteria for the Hanford Site. This version contains requirements similar to the previous revisions, as well as a new section defining the waste acceptance criteria for disposal of solid sanitary waste at the central landfill.

GENERAL WASTE PACKAGING PRACTICES

Waste packaging practices during the 1940's, 1950's, and early 1960's depended primarily on the size and type of waste being packaged, with special consideration given to dangerous or hazardous wastes. There were several different kinds of waste containers used in the packaging of solid waste. Small materials consisting mainly of dry waste were placed in quart-size cardboard containers, which were than placed in larger cardboard cartons for burial. Equipment was buried in wooden boxes when available and, if a wooden box could not be provided, the equipment was buried without a protective covering. If it was determined that the equipment was too hazardous to bury without confinement, the equipment was wrapped in plastic prior to disposal. Prior to the late sixties, there were no state or federal regulations on segregation requirements for packaging waste for burial at Hanford. There were attempts to package waste to minimize personnel exposure and prevent the spread of uncontained radioactivity to the environment; however, these were not set guidelines and were done at the discretion of the generator.

In the 1960's, low level wastes that were small in size were placed in plastic-lined cardboard boxes and large waste items were wrapped in plastic shrouds; whereas, grossly contaminated mixed fission products were packaged in high integrity containers. The most common method of depositing wastes in trenches during the 1960's was to dump boxes of solid waste directly into the burial trenches. Wood or concrete boxes that contained bulky or highly contaminated materials were dragged from railroad cars into the trench by bulldozers using long cables. Before 1970, the primary concerns during burial operations were to assure confinement of contaminated

materials during transport, minimize exposure to operating personnel, confine radioactive or chemical materials to prevent releases to the environment, and protect public health.

The packaging of waste materials was designed to maintain safety until the material was securely buried; once buried, the containers were considered permanently disposed. Because of the favorable hydrological conditions, concern was not given to whether the containers remained intact after burial. Until the mid-1970's, there were no requirements for venting burial containers to allow for the release of built up pressure. If waste materials were known to generate gases, they were placed within containers constructed of a material known to collapse under the weight of backfilling. Once the integrity of the container was no longer intact, it was considered vented.

Beginning in 1970, in addition to fiber drums and metal containers that were used to bury failed equipment, iron or galvanized steel drums and boxes constructed of fiber reinforced polyester plywood or concrete were used for packaging small materials. The Design Criteria for Transuranic Dry Waste Steel and Reinforced Concrete Burial Containers, ARH-CD-353, released in 1976, stated that burial containers were provided with vents if there was a requirement that they be protected against variations in internal pressure. With the release of RHO-MA-222 in 1980, each container was required to be capable of being fitted with an air or vacuum hose or a gaseous diffusion vent. As of 1980, wood, steel, and/or concrete boxes were used for the burial of process equipment. It was also around 1980, that the DOT 17C 55-gallon galvanized drums were declared to be the required packaging for TRU waste. The 17-C and 17-H non-galvanized drums were used for non-TRU waste shipments (10).

To indicate the segregation of TRU waste from LLW, some facilities used painted drums; for a period, yellow drums were used to package low-level wastes and black drums contained TRU waste. At the 200 Areas, color-coding of drum lids was done to indicate the segregation of hood waste from room waste. Hood wastes were wastes generated inside processing hoods and were considered highly contaminated with plutonium (11). Room wastes were wastes generated from operations outside the processing hoods and were considered potentially contaminated with plutonium. Solid wastes were segregated into combustible hood waste, combustible room waste, and non-combustible room and hood waste. Combustible hood waste was comprised of material such as plastic, rubber, rags, and cardboard. Combustible hood waste was placed in drums with yellow lids, combustible room waste was stored in drums topped with silver domes, and non-combustible hood and room waste was collected in drums topped with red domes.

For safe storage, TRU wastes were segregated into combustible and noncombustible. Small TRU items were also segregated from larger TRU items or equipment pieces. Separate storage facilities and burial trenches were designed for TRU waste storage. Solid TRU waste was packaged, stacked, and stored in trenches with an earth, gravel, plywood, or asphalt pad foundation. Small items were stored on asphalt pads, in underground trenches, or in caissons, whereas larger items were stored primarily in burial trenches. The TRU wastes that were unsuitable for asphalt pad or caisson storage because of size, chemical composition, security requirements, or surface radiation were packaged in reinforced wood, concrete, or metal boxes. High-level (high activity) solid wastes were defined as wastes that emitted high levels of beta and gamma radiation. This waste did not contain TRU radionuclides and typically included failed equipment from B Plant, tank farm operations, etc. The operation of high level (high activity) waste during this time period is different from the current definition of HLW, which refers to waste resulting from nuclear fuel processing. Small HLW items were transported to the caissons or burial trenches, while large items or failed equipment were buried in the industrial waste trenches. In the late 1970's, more specific packaging procedures requirements were introduced. Multiple containment barriers were required in the packaging of waste. In addition, more concern was given to void spaces left in waste packages and the increased used of filler materials. As time passed, the regulations became more focused and the disposal of waste began to follow a set standard of guidelines.

Containment Barriers

In the early years, waste at Hanford was disposed of in the burial grounds using only a single containment barrier. This barrier was the package in which the waste was placed. Typical packages were concrete boxes, cardboard boxes, plywood boxes, or drums. As time passed it was observed that some waste was escaping the single

containment barrier; for example, liquid leaking from a drum. This posed harmful effects for the environment and decreased personnel safety. Therefore, requirements for the number of containment barriers increased as listed below:

In 1968, wastes containing contamination that was easily airborne were contained by an inner container (e.g., sheet plastic) (5).

In 1978, a second polyethylene drum liner was placed inside the first polyethylene drum liner (12).

In 1979, 55-gallon barrels used at Z Plant to store radioactive wastes were lined with a polyethylene drum liner, 99 x 137 cm and 4 mil thick (13).

In 1980, solid radioactive waste containing asbestos had to be packaged within at least one layer of 6 mil polyethylene film. Transuranic solid waste was packaged inside at least two containment barriers, the storage container and an inner sealed liner (10).

In 1981, it was stated that polyethylene liners were to be "horsetailed" and then taped shut before the drum lid was installed (14).

In 1985, all LLW determined to be radioactive mixed waste was packaged with at least three containment barriers (15).

In 1993, PNL determined a 90-mil high density polyethylene inner liner was required for liquid RH waste. A 10-mil nylon reinforced plastic liner was required for solid RH waste. For liquid radioactive mixed waste, inner containers were almost always glass, with a capacity of 18.9 L or less (16).

Filler Materials

Filler materials became important around the early 1980's. At this point a focus was made on the void space left inside some packages and the benefits obtained by reducing this volume. The addition of non-radioactive materials to radioactive waste resulted in improved heat transfer, radionuclide immobilization, and increased physical support. The following list gives an overview of the void space limitations:

From 1978 to 1984, waste package contents were not to exceed 80% of the active volume of the waste container (17).

In 1984, it was stated that to prevent subsidence in Hanford burial grounds interior void spaces within non-TRU packages were to be minimized. However void spaces did not need to be filled in containers which were to collapse during the initial backfilling process (e.g., fiberboard boxes, plastic wrapped equipment) (18).

From 1985 to 1986, interior void spaces for LLW were not to exceed 20% of the active volume of the waste container (15).

In 1987, the list of items which were exempt from being filled was expanded. Items which were not to be filled were high efficiency particulate air filters, which posed hazards to personnel during filling, waste packages with a total internal void space less than 0.042 m3 (1.5 ft3), and any specially designed reinforced-concrete burial boxes with a design life in excess of 300 years under burial conditions expected in the Hanford Site burial grounds. All RMW packages accepted for storage were exempt from requirements for filling void spaces (15).

Prior to 1990, no specific list was provided for approved filler materials. The following list contains materials that were approved for use as void space filler in 1990:

- Diatomaceous earth
- Soil, sand, lava rock
- Tightly packed cellulose matter
- Clay
- Concrete, cement, grout
- Gravel

SPECIFIC WASTE PACKAGING PRACTICES

With an increased knowledge about certain types of waste, new, more specific packaging practices were developed for these waste types.

Process Equipment

Process equipment consisted of equipment used by several of the large plants at Hanford. The equipment caused several problems when it came time for disposal. Due to the large size and odd shape of the majority of process equipment, special measures had to be taken for burial. In the early years the equipment was buried in wooden boxes. Sometimes a wooden box could not be provided and the equipment was buried with no protective covering. When it was determined that the equipment was

too hazardous to bury without confinement, the equipment was wrapped in plastic prior to burial. In addition, large pieces of process equipment were cut into smaller sections and packaged prior to burial. A chronological list of different burial procedures for process equipment follows:

Beginning in 1964, failed process equipment was packaged in concrete boxes. Process equipment from Plutonium Uranium Extraction (PUREX) Facility that was too large to bury was stored in special railroad tunnels adjoining the PUREX Plant (19).

Beginning in 1970, metal containers were used to bury failed equipment from the PUREX Plant and the Plutonium Finishing Plant (6). Some items of failed equipment, such as 12 to 15 meter long pumps used in the transfer of wastes from underground storage tanks, were flushed and packaged in plastic prior to burial (20).

As of 1980, wood, steel, and/or concrete boxes were used for the burial of process equipment.

Beginning in 1981, large radioactive waste items from the PUREX canyon were packaged in burial boxes of precast, reinforced, concrete slabs with a concrete slab lid held in place by its own weight. A steel liner box was sometimes inserted, depending on the waste being packaged. Box configurations varied depending on the waste being packaged, but the most commonly used size had a void volume of 50 m3.

Around 1987, old gloveboxes were packaged in intact burial boxes. For a brief period of time they were sent to the 231-Z Facility to be cut up into smaller pieces. The pieces were then packaged in steel culverts, steel boxes, plywood boxes, and some of the smaller pieces were placed in 55-gallon drums.

In 1993, large process equipment, including leaded glass, fluorescent lamps, and PCB ballasts, were stored in burial boxes of mixed waste.

Class B Poisons

Class B poisons were a main focus of disposal due to the effects the poisons had on the environment and personnel safety. Solid waste containing Class B poisons was packaged in double containment. Small quantities were placed in small containers, which were then placed in storage or disposal containers and the small containers were fixed or surrounded by concrete on all sides. In 1980, it was determined that packaging for larger quantities was to be approved on a case-by-case basis (10). In the mid-1980's mercury, a specific Class B poison, was confined in a concrete culvert and the culvert was then placed in a drum. It was common to fill the space around the culverts with bagged poly bottles and other items. In 1992, PNL packaged liquid metallic mercury in a polyethylene or glass container with a screw-type lid.

Sodium and Alkali Metals

Prior to 1977 there were no documented packaging requirements for sodium and alkali metals. Beginning in 1977, special approval of any waste package containing sodium or other alkali metal was required. Unreacted alkali metals in solid waste was not accepted for disposal. The shipper had to specify quantities, concentrations, and contamination levels of each alkali metal to assure that the appropriate methods of handling, storage, and/or disposal were used (21). The requirements established in 1977 are being observed today.

Oxidizing and Corrosive Materials

Oxidizing and corrosive materials are of special interest because they break down the integrity of the container in which they are packaged. In addition, during the break down of the containers, gases are generated. It was not until the late 1960's that oxidizing material were prohibited from being packaged with combustible wastes or in combustible containers. Rags used to cleanup oxidizing materials had to be well-rinsed to remove all oxidizing materials before they were discarded. Beginning in 1984, wastes containing corrosives were to be treated to eliminated their corrosive properties and to form a chemically stable compound or they were packaged such that the storage container was not exposed to the corrosive agent during its 25 year design life. To enhance the corrosive protection, the interior and exterior of the waste containers were galvanized or painted with a two-component epoxy-polyamide paint system or functionally equivalent paint (18).

Tritiated Waste

Beginning in the early 1980's procedures were introduced for packaging tritium wastes. Tritiated waste, including tritium oxide, in liquid form was to be packaged in steel or concrete containers. Waste containing tritium or tritium oxide was absorbed on silica gel, packaged in a leak-tight 1-gallon metal cans, surrounded by asphalt, and packaged in 55-gallon drums. Waste packages with heat output greater than 3.53 watt/m3 required a special thermal analysis to determine whether special

separation distances for the waste within the burial trench were required. In 1993, the tritium waste was defined as waste containing greater than 20 mCi of tritium/m3 of waste and its disposal requirements changed:

Tritiated waste with less than 100 Ci tritium/m3 in either absorbed liquids or solids was to be sealed in one layers, of 4-mil (nominal) or thicker polyethylene and disposed of in a steel or concrete package.

Containment systems for tritiated waste with greater than or equal to 100 Ci tritium/m3 were to be documented in the storage/disposal approval record.

Liquid and Animal Wastes

Due to the increased knowledge about the waste and the better packaging techniques, the guidelines of liquid and animal wastes have changed throughout time. Table II summarizes the changes in packaging since 1967.

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17-26

ON THE SAFETY OF WASTE PACKAGE TRANSPORT IN A REPOSITORY FOR SPENT FUEL AND HLW

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ABSTRACT

In a deep geological repository, the internal transport of waste packages with vitrified HLW and spent nuclear fuel will include several steps of horizontal and vertical movements from the waste reception station on the surface to the final disposal position underground. During such transport, situations may arise leading to undue mechanical and/of thermal loads acting on the package that could eventually jeopardize its containment. Under very unfavorable circumstances, these loads can even exceed significantly the very tough requirements for type B(U) transport packages. Appropriate measures and/or safety devices must therefore prevent these situations from occur and/or limit their consequences to values within the design limits of waste packages. Using an integral approach, such systems, devices and/or measures have been developed, key issues important to safety being tested under realistic conditions in special purpose installations in 1:1 scale. This paper gives an overview on the design and R&D effort carried out and of the safety relevant results obtained.

INTRODUCTION

In the framework of a project aimed at demonstrating the feasibility of spent fuel final disposal, all transport steps of the final disposal procedure were carefully evaluated from the point of view of safety. Every waste package movement from delivery of the spent fuel to the site until final emplacement underground were considered. The aim was finding out weak point requiring special attention, and developing the measures and/or devices necessary to guarantee the radioactivity containment under all circumstances. The basic assumptions were:

- delivery of the spent fuel and the HLW to the site by road or rail transport in a container complying with Type B(U) requirements;
- conditioning into a Pollux cask for final disposal;
- transport to the underground repository mine via a shaft;
- final disposal in drifts of the repository mine with subsequent backfilling of the emplacement galleries.

As initiating events for potential lost of radioactivity containment during the waste package handling in the surface facilities the following events were identified:

- accidental cask drop from a height in excess of the package design value during crane handling;
- accidental drop during crane handling of a heavy hart object onto the cask;
- fire in the reception or in the shaft dispatching area;
- fall of a loaded cask carrier into the shaft.

As initiating events for potential lost of radioactivity containment during shaft cage charging and during hoisting to the underground the following events were identified:

- fall of a heavy hart object into the shaft and onto the cask during hoisting; hoisting machine overwinding after failure of the machine control system;
- fire in the shaft cage;
- failure of all shaft cage ropes.

As initiating event for potential lost of radioactivity containment during shaft cage discharging or in the underground facilities the following events were identified:

- accidental cask drop from a height in excess of the package design value during handling or disposal;
- accidental drop of a heavy hart object onto the cask;
- fire during underground transport;
- fall of a cask into the shaft sump.

TRANSPORT SYSTEM DESIGN AND TESTING

Taking the listed initiating events into account, a waste package transport and handling system was developed. It consists of a special waste package design, rail-guided transport vehicles, a specially conceived shaft cage, and the corresponding shaft cage charging/discharging equipment. An overview of the initiating events and of the measures and/or devices and/or pieces of equipment designed to prevent them from occurring and to cope with their consequences is given in Table Ia,b,c.
Since in the Federal Republic of Germany only proven technologies and pieces of equipment may be used in a repository, a test programme aimed at demonstrating the safety of the transport system as a whole was carried out. Simultaneously, studies dedicated to single questions, as e.g. a test programme for shaft cage ropes and a Probabilistic Safety Assessment of the complete shaft hoisting system, were carried out. The most important test, the simulation of the shaft transport of payloads of up to 65 metric tons, has already been finished, conclusive results are available. Other ones are in due course, final results will be available by the end of 1995.
CONCLUSIONS
To comply with requirements of the licensing authorities, the transport pathway of waste packages with spent fuel and HLW in a repository has been scrutinized. The objective was to find out situations eventually leading to loss of radioactivity containment. All locations, operations, and movements of waste packages were considered. From the analysis of hazardous situations likely to arise requirements were derived for the design of systems, components and operational procedures. In addition, preventive measures were planned.
Some of the systems and/or components designed were novel. To satisfy regulatory requirements these systems were tested under realistic conditions in 1:1 scale. The complete handling and transport procedures and equipment have been subjected to a final evaluation considering the test results and the results of specific studies, as e.g. a Probabilistic Safety Analysis of the shaft transport. The results of already finished tests, as well as results obtained until now of still running tests, confirm the reliability and safety of the handling and transport systems to the level required by the authorities.

17-27
RISK CONSIDERATIONS FOR THE PACKAGING AND TRANSPORTATION OF PLUTONIUM-CONTAMINATED WASTE FORMS AT DEPARTMENT OF ENERGY NUCLEAR FACILITIES
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ABSTRACT
Plutonium waste streams of various compositions exist in inactive process systems that will be removed during deactivation activities at the various Department of Energy (DOE) nuclear facilities. Also, a number of differing plutonium-bearing waste streams will be generated during the conduct of decontamination and decommissioning (D&D) efforts associated with the site mission of waste management and environmental restoration. The waste streams will include contaminated concrete, metals, soils, ceramics, sludges and liquids that may require processing and must be packaged for transportation and storage/disposal. In many instances, the mixture of waste components may meet the definition of a hazardous/radioactive mixed waste, or a hazardous/transuranic mixed waste.
The packaging requirements for each of these different waste streams is dependent upon the volume of waste material, waste form (i.e., solid, gas, liquid, etc.), waste constituents (i.e., plutonium, acids, uranium, etc.), radiological (Curie) content, dose rate, planned disposition (e.g., storage, disposal, processing, etc.), and preferred method of transportation (e.g., highway, air, rail, etc.). In each instance, the packaging requirements designated in both the DOE and site procedures must be met. In many instances, the DOT transportation regulations (49 CFR) and the waste acceptance criteria (WAC) of the targeted disposal facility will also be factors.
Based upon the considerations of efficiency, cost, disposal space, and risk, support to the program has been structured not only to comply with the applicable regulations, but to minimize the quantity of waste that must be disposed/stored through processing and innovative D&D technologies that generate minimal or no

additional waste streams. Additionally, the DOE nuclear facilities intend to package and transport the remaining hazardous/radioactive/mixed wastes in as an efficient and expedient manner as possible, correlated to the characteristics of the waste stream to be managed.

Risks to the site workers, general population, and the environment associated with the transportation and storage/disposal of the waste materials will be an important factor in the choice of packaging containers and methods of transportation. The use of risk-based analyses and analytical tools which consider the final D&D waste form and volume prior to the choice of the actual D&D technologies is a new and innovative measure which aids in the protection of both human health and the environment. Risk evaluation, management, and mitigation are all elements of the comprehensive planning for handling, transportation, and disposal of plutonium-bearing wastes.

This topic should be of interest to any individual or company that is in the process of, or plans to, remediate or perform D&D activities. As the particular slant of this paper concerns plutonium, it should be of particular interest to those facilities that deal with fissile materials. The anticipated end result of this presentation is to provide an awareness of the risks associated with the packaging and transportation of hazardous/radioactive materials and potential manners in which these risks can be mitigated.

INTRODUCTION

As many of the Department of Energy (DOE) facilities are commencing decontamination and demolition (D&D) activities, the disposition of the plutonium-contaminated wastes is becoming a major concern. In particular, the risks associated with the packaging and transportation of these waste materials must be acknowledged. The volume, waste form, and treatment/decontamination methodologies for the plutonium-contaminated waste must be chosen with the final disposition in mind. For material that is to be disposed of, the waste must meet the Waste Acceptance Criteria of the facility. The purpose of this presentation is to provide evidence that risk must be addressed at the time the decisions affecting the disposition, in particular packaging and transportation, of the plutonium-contaminated materials are made.

PLUTONIUM-BEARING WASTES

Production of Plutonium-Bearing Wastes

The DOE nuclear facilities have been involved in the production of nuclear weapons for the federal government. Some of the key production activities were associated with the fabrication of parts from plutonium, uranium, beryllium, stainless steel, and aluminum. Additionally, some DOE nuclear facilities disassembled and processed components from obsolete nuclear weapons in order to recover the plutonium and americium. DOE nuclear facilities have also operated facilities for the storage, treatment and transport of waste; chemical laboratories; research and development laboratories; and special support operations for the other DOE facilities. Due to both regulatory and environmental concerns, as well as end of the Cold War, the current stockpile of nuclear devices exceeds the requirements of the United States government. To this end, the production of the nuclear devices has been curtailed. Both radioactive and hazardous materials were utilized in association with the activities and processes that took place at the DOE nuclear facilities. Both hazardous and radioactive materials were stored, processed, and transferred throughout the various tank farms and pipe runs. Additionally, over the years, spills, fires, ventilation and leaching problems have also occurred at the site. Because of this, various media, or waste streams, at the DOE nuclear facilities have became contaminated. Plutonium is not the only contaminate of concern at the DOE nuclear facilities. Additional contaminants include americium, uranium, thorium, as well as organics, metals, and pesticides.

Waste Streams

Much of the plutonium-contaminated waste at the DOE nuclear facilities will be generated as a result of the decontamination and decommissioning (D&D) activities. Plutonium-contaminated materials currently reside in many of the tanks (sludges, liquids, and solids); on raschig rings; in ventilation ducts and pipe runs (utilities); within building walls, flooring, and support/bracing; within soils; and as loose contamination within the facility.

During the D&D activities, it is anticipated that the different types of plutonium-contaminated media will generate a variety of different waste streams. The

waste streams will include:
1 Contaminated concrete;
2 Metals;
3 Soils;
4 Ceramics;
5 Sludges;
6 Liquids;
7 Decontamination solutions;
8 Plastics; and
9 Wood.
Each individual waste stream will need to be evaluated to determine its optimum disposition option. Factors that will need to be evaluated include waste form, efficiency, cost, disposal space and risk. Some of the different disposition options that exist include decontamination, material treatment options, unrestricted release, disposal as low-level radioactive/TRU/mixed waste, and storage.
Waste Forms
In some instances, the waste stream may compose the final waste form of the plutonium-contaminated materials, i.e., concrete. In many other instances, some form of treatment or decontamination may be required in order to achieve a waste form that meets the final disposition option requirements. Typical waste forms that would be expected to be composed at least partially from the plutonium-contaminated waste materials include solids, sludges, and liquids.
As an example, generally speaking, one cannot dispose of liquid radioactive waste. Liquids and sludges will need to be treated in some manner such that the waste form meets the criteria stipulated in the disposal facility's waste acceptance criteria (WAC). Additionally, one must be able to transport the waste form to the disposal facility. 49 Code of Federal Regulations (CFR) stipulates shipment criteria for packages, i.e., gas emissions, dose rates, criticality.
Waste Stream Characterization
Waste stream characterization plays an extremely important part in determining the packaging and transportation requirements. Waste that can meet either the criteria for low-level radioactive waste or transuranic waste can be disposed at more facilities than listed (hazardous) or mixed waste. One of the few facilities to accept mixed waste is Envirocare of Utah. Different packaging and transportation requirements may, and often do, exist for each of these types of waste materials.
PACKAGING AND TRANSPORTATION PROGRAMS
Handling Requirements
The handling requirements for the different waste streams will vary based upon the waste stream characteristics. In many instances a worker that is handling plutonium contaminated materials is isolated from the material itself. This is accomplished by the use of glove boxes and/or remote control devices. In addition, the chemical characteristics (i.e., pH, gas production, total dissolved solids) of a particular waste form may require special handling techniques. In some instances, the plutonium-contaminated waste materials will be treated to eliminate chemical constituents. Also, the material may be decontaminated or treated in an attempt to segregate a constituent from the waste such that the material may be disposed as low-level radioactive waste rather than TRU or mixed waste. Waste materials may also be size reduced or compacted to reduce the volume or provide a shape/size/weight that can be more easily handled/treated/packaged. Compaction may also be required to meet the disposal requirements regarding the minimization of void spaces (10 CFR 61).
Packaging Considerations
Packaging considerations include the waste form, volume, dose rate, quantity of the waste constituents, cost, packaging efficiency, disposal space available, the WAC (as appropriate) and the packaging risks. Based upon these criteria, the plutonium-contaminated materials could be shipped in bulk, boxes, drums, casks, etc.

The packaging considerations include not only the onsite packaging concerns, such as personnel exposures and industrial accidents, but the off-site risks to human health and the environment. To address the off-site risks, it must be determined that for the material of concern, what type of package would best protect the contents in an accident situation while minimizing the number of shipments that are required. It should be noted that routine, non-radiological accidents typically are responsible

for the majority of the risk associated with any shipment. This is an example of why efforts should be made to minimize the number of shipments that are required. Packaging concerns, risks, and mitigators are displayed on Table I a, b.

Transportation Considerations

Transportation and packaging considerations are in many respects similar. In both situations the major considerations are the waste form, volume, dose rate, quantity of the waste constituents, cost, packaging efficiency, disposal space available, the WAC (as appropriate) and the packaging risks. Since different modes of transportation (i.e., truck, rail, air, ship, private vehicle, etc.) could be utilized, the different regulations stipulated in 49 CFR for the various modes of transportation must be considered.

As with the packaging considerations, both on-site and off-site transportation concerns must be addressed. Accident scenarios as well as routing considerations (i.e., population density, accident rates, types of roads, etc.), number of shipments, dose rates, etc., must be reviewed when determining the appropriate type of transportation to utilize. The mode of transportation that is chosen should protect the packages in case of an accident situation as well as minimize the number of shipments required. Transportation concerns, risks, and mitigators are discussed in Table II.

Other Factors

Additional factors include the existence and types of contractual agreements with the transportation firms and waste disposal facilities; the availability of both disposal/storage space and of shipping containers; as well as routing and time frame requirements. In almost any specific situation, additional considerations, other than those already noted, will be required. They can range from contractual considerations, to weather, to individual state requirements for transportation to client specific requirements.

RISK CONSIDERATIONS AND FACTORS

Risk-Based Analyses and Analytical Tools

A variety of risk-based analyses may be performed and/or analytical tools may be utilized to determine the risk associated with the packaging and/or transportation of plutonium contaminated waste. Some of the commercially/DOE available codes which look at transportation risk and routing include RADTRAN, HIGHWAY, INTERLINE and INTERSTAT. HIGHWAY, INTERLINE and INTERSTAT aid in the routing of any given shipment. The concerns of these models include population density, accident rates, type of road, etc. The output from these codes is then utilized in the RADTRAN program. The RADTRAN program calculates the risk associated with a particular shipment, or number of shipment, utilizing a particular mode of transportation and designated route.

Parsons developed a computer-based qualitative risk-based analysis program concerning the choice of D&D methodologies for particular facilities or types of materials. This program took both packaging and transportation into consideration. This analysis was comprised of a set of questions to which a designated number of points are assigned to each particular response. The questions concerned the waste volume that would be generated, the types of wastes that would be generated, waste disposal feasibility, the dose rates and the use of Type B packaging. An example form utilized for this screening mechanism is provided as Table III.

Application of Risk Mitigation Programs and Analyses

Risk mitigation analyses and programs should be applied prior to the generation of the waste to be packaged and shipped. If this is not possible, the analyses should be performed prior to the packaging of the waste. If the risks associated with prematurely packaging and transporting waste were considered prior to their generation, much of the exposure, handling, and accident risk - not to mention cost, ALARA, and efficiency considerations - associated with the duplication and "forced-fitting" of these activities could be alleviated.

Risk Evaluation, Management, and Mitigation

Sections 4.1 and 4.2 discuss risk-based analyses and analytical tools and their application. The risk evaluation should include the situation specific information, as much as possible. The evaluation must take into account the form, treatment, packaging, transport and final disposition of the plutonium-contaminated waste. This can be accomplished by utilizing both qualitative, as well as quantitative, methodologies. Once the risks have been evaluated for each step in the process, an intelligent decision concerning the optimum course of action can be made. At this

point, it is possible to begin to manage the risks. Risks in particular that must be managed include limiting the number of shipments, minimizing the volume of material to be transported, criticality, and treatment/decontamination methodologies and their secondary waste products.

The use of mitigative measurement to minimize the risk to human health and the environment is an important concept to accept and understand. While lowering the overall risk of the process, in many instances, mitigative measures may increase the total cost. In cases where the cost versus the benefit is extreme, a cost-benefit analysis should be performed. Simple ways to mitigate the risk include personnel training, the utilization of decontamination/treatment techniques, the segregation of waste types, choosing the optimum package and mode of transportation for each particular situation, and minimizing the volume of waste that must be dispositioned, and thereby limiting the number of shipments required.

CONCLUSIONS

Packaging and transportation risks cannot be adequately or efficiently addressed after the processing, treatment and packaging of the waste materials has been completed. These are considerations that must be addressed at the beginning of the process, not the end or the middle. As previously stated, in order to best utilize the information provided in a packaging and transportation risk analysis, the evaluation should be performed prior to the generation of the waste. In almost any situation, the sooner these considerations are addressed, the more cost efficient and ALARA the final resolution can be. For D&D, as well as other types of waste generating activities, it is extremely important to consider the entire process by using an integrated system engineering approach which considers all of the facets of the task, from the initial action through the material transport and disposal.

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17-28

LARGE-SCALE MACROENCAPSULATION OF FILTERS AND DEBRIS

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ABSTRACT
An environmentally-safe, regulatory-compliant method of macroencapsulating filters, sources, irradiated and contaminated hardware and demolition debris is needed by the commercial nuclear industry and DOE/DOD installations.
Traditional methods of high-activity source disposal have been encapsulating a single source or filter element in grout inside a 55-gallon drum (liner), or cement encapsulation of a larger number of filters in a liner. Debris is typically disposed of in metal boxes or liners without assurance of future integrity of the container. If the activity of the debris requires waste stability, the use of a HIC has been the only disposal option.
In the past, none of the encapsulation technologies employed have undergone testing to demonstrate waste form integrity in compliance with 10 CFR Part 61. Recently, a new encapsulation process has been submitted to the N.R.C. for review and approval. This ENCAPTM process has been used successfully for filter encapsulation at the Diablo Canyon Power Plant.
The ENCAPTM process is the subject of an Addendum to Topical Report DT-VERI-100-NP/P-A, VERITM (Vinyl Ester Resin In Situ) Solidification Process for Low-Level Radioactive Waste (Docket Number WM-105R1)(1). This process utilizes an N.R.C.-approved binder to encapsulate or entomb waste in a free-standing monolith. The incorporation of non-uniform waste products in the monolith generates a heterogenous waste form. This process is applicable in liners as large as 200 ft3. This paper describes the process for encapsulating heterogenous wastes in an entombing monolithic polymer matrix, including steps taken to protect the integrity of the monolith, the design basis to preclude environmental exposure of the entombed waste, and limitations on the applicability of the process. The economic impact of this technology is discussed, as well as typical applications at commercial nuclear and DOE/DOD sites.
INTRODUCTION
The ENCAPTM Process, subject of a Topical Report(1) submittal to the N.R.C., has the potential to minimize the environmental impacts and costs of shallow land burial of waste filter cartridges from nuclear power plants at low-level waste disposal facilities. The reduction in environmental impact and in waste disposal costs would be achieved by reducing waste packages to be handled at the reactor site, shipped on public highways and buried at the disposal facility. The cost of a vinyl ester styrene (VES)-solidified liner is lower than N.R.C. metal HICs and lower than poly HICs in an overpack. Poly HICs designed to accept heterogenous waste are often foamed to prevent damage to the HIC, further reducing the volumetric and cost efficiency of HICs.
The ENCAPTM Process encapsulates numerous filter cartridges and other solid debris in a matrix of VES and spent ion exchange bead resins. The spent bead resins may or may not be contaminated with radionuclides. An ENCAPTM monolith as large as 200 ft3 could encapsulate from 100 to 200 filter cartridges. The number of cartridges or amount of debris to be loaded in a given ENCAPTM liner would depend on a number of variables, including the size of the filter cage in the liner, size and shape of the cartridges, and the concentration of radionuclides on the filters and in the resin matrix.
PROCESS DESCRIPTION
The ENCAPTM process uses Vinyl Ester Resin In Situ (VERITM) solidification to create an enveloping monolith around a caged region of heterogenous wastes (i.e., spent filter cartridges). The ENCAPTM process involves loading waste into an encapsulation liner and solidifying the liner with the VERITM polymer.
An encapsulation liner has dewatering internals identical to a VES bead resin solidification liner. A cage is installed in the encapsulation liner to prevent the waste from coming in contact with any portion of the liner internal surfaces.
A protective layer of ion exchange resin (or other inert granular material) is

introduced to the liner first. This resin layer must be at least 4" deep, and may consist of either contaminated or non-contaminated resins. The resin, sluiced with water or manually poured into the liner, serves a dual function of physically protecting the dewatering internals from damage during introduction of filter cartridges and establishing the lower exclusion zone which consists entirely of VERITM-solidified resins.

When the cage has been loaded with filter cartridges or other wastes to be entombed, the cage lid is closed and the liner is filled with ion exchange resin to a level covering the caged waste by a minimum of 4". The fill resin may be contaminated or non-contaminated, as long as it is of the type identified in the parent TR as being accepted for VERITM solidification.

When the ENCAPTM liner has been filled with resin, VERITM solidification equipment is positioned and VES binder is added to the liner entombing the waste from top to bottom. Upon completion of introduction of the catalyzed and promoted binder to the encapsulation liner, the exotherm is monitored, compliance with acceptance criteria is verified, and the ENCAPTM liner is capped and prepared for shipment.

ENCAPSULATION LINER

The encapsulation liner is a right cylinder of carbon steel or polyethylene plastic with maximum bounding dimensions of 78" in diameter and 72" in height, yielding a maximum volume of approximately 200 ft3.

No credit is taken for the structural strength of the disposal liner in the encapsulation process for waste form qualification.

The key feature of the encapsulating liner is a cage positioned inside the liner to receive the material to be encapsulated. This cage is designed to create an exclusionary zone between the entombed waste and the side wall of the encapsulation liner. Several generic features must be incorporated in the cage to ensure maintenance of the exclusionary zone and prevent buoyant surfacing of the filter elements. A general configuration drawing of the ENCAPTM liner is shown in Fig. 1 at the end of this report.

WASTE FORM

The encapsulating media is spent ion exchange resin solidified in a VES polymer matrix. The waste to be encapsulated does not affect the structural integrity of solidified monolith when the potential void spaces are limited. Such waste form characteristics as compressive strength, resistance to degradation (water immersion), biodegradation, irradiation tolerance and leachability are not influenced by the entombed waste. As a minimum, the requirements 10 CFR 61.56(a) apply to all waste, waste products and waste forms.

Structural Stability

The ENCAPTM entombing of waste within the VERITM-solidified monolith does not physically impair dimensional stability under the expected disposal conditions such as weight of overburden and compaction equipment. Previous testing has shown that the presence of liquid, microbial activity, and radiation (when limited to <1 x 108 Rad) has no negative effect on the structural integrity of the binder matrix and, hence, on the integrity of the monolith.

It has been shown (in the parent TR), that the compressive strength (typically >2000 psi) of the VERITM waste form is several factors greater than the minimum required for stability. Qualitatively, the plasticity of the VERITM waste form is well suited for encapsulation.

Unlike cementious waste samples (which tend to experience acute failure by sheering or spalling during testing), the VERITM waste form tends to bulge circumferentially without failure when subjected to maximum yield compressive forces. It is postulated that this plasticity may tend to protect the integrity of the monolith when subject to compressive forces in the burial environment. It should be noted that no credit is taken for the favorable plasticity of the waste form.

Likewise, no credit for stability is taken for the liner in which the VERITM process is used. In those instances when the entombed waste may be stored on an interim basis before final disposal, consideration should be given to performing the ENCAPTM process in corrosion-resistant containers such as poly containers or lined carbon steel liners. Such corrosion resistance, while not an issue from a waste form perspective, may be important to ensure viability of the waste package a number of years from now, when the waste is moved from its storage site to the transport cask for disposal. No aspect of the ENCAPTM process is affected by, or prohibits use in, such liners.

The waste forms produced by the ENCAPTM process meet the structural stability requirements of land burial without the need for an overpack (per N.R.C.). The minimum separation of the waste-containing cage from the side wall of the liner is intended to ensure sufficient strength to prevent environmental intrusion into the caged waste area, or escape of waste from the area. The filling of interstitial space within the caged waste provides additional strength and stability to the monolith. Some states may require all Class B and C waste to be placed in concrete overpacks. An ENCAPTM liner can readily be overpacked, though there is no technical reason for doing so.

FULL-SCALE EXPERIENCE

In addition to the full-scale testing performed by DTS in support of the parent TR, commercial solidifications have been performed using this process. In 1992, a nuclear power plant successfully solidified spent resin generated from cleanup of a LOMI-CANDEREM decontamination process. The solidified liner was buried at the Barnwell, South Carolina disposal site. Also in 1992, approximately 450 drums of VES-solidified decontamination solution were processed and buried in Beatty, Nevada prior to its closing.

In 1993, a liner of spent mixed bed resin, primarily RCS (Reactor Cleanup System), was solidified using the VERITM process. This liner was also buried at Barnwell. Also in 1993, three liners of spent filter cartridges were encapsulated utilizing the ENCAPTM process. These cartridges were generated from the primary, spent fuel pool and radwaste cleanup systems in a PWR plant. The encapsulations proceeded in the manner typical of a VERITM solidification, though three enhancements were incorporated in the ENCAPTM process as a result of this experience.

First, the vacuum pump drawing a suction on the liner was turned on before introduction of the binder. This initiated a suction on the liner immediately when the binder cap was established on the resin bed, and provided a downward pathway for air displaced by the binder.

Second, a closure was placed on the internal cage to preclude buoyant objects from breaching the surface prior to gelation of the binder. Upon gelation, all objects were effectively locked in place throughout the balance of the cure process.

Third, a low-volume sweep of the air from the freeboard of the liner was procedurally implemented, using ports already available on the fillhead. This air was exhausted to a HEPA-protected exhaust point as a protective measure, though no airborne contamination was expected or found. This air sweep was designed to minimize the condensation that occurs on the cold steel liner during the exothermic cure. A small amount of moisture was present on top of the first ENCAPTM-solidified monolith. Though an on-site assessment of moisture deemed the volume to be insignificant, "0" moisture was desirable, and the described action was implemented. This procedural action was successful in preventing the formation of condensation in subsequent ENCAPTM solidifications.

ENCAPTM solidifications performed after implementation of each of these process enhancements were performed successfully and without incident.

WASTE LOADINGS

Waste loadings (or packaging efficiencies), for ENCAPTM entombing of spent filter cartridges were ascertained from full-scale encapsulations. Filter media loading values for the 80 ft3 ENCAPTM liners referenced in the table below are based on the volume of entombed filters calculated by two different methods.

Loading values for filter media were determined by subtracting the internal void volume of the cartridge from the outside envelope volume, divided by the disposal volume of the liner. Filter envelope volumes are based on the exterior volume of entombed filters divided by the volume of the solidified monolith. Based on this data, typical filter loadings for the ENCAPTM process can be expected to be a minimum of 15% by filter media volume and 30% by filter envelope volume. Table I shows the actual data from three 80 ft3 ENCAPTM liners.

Low activity filters at Diablo Canyon (generally less than 500 mR/hr) are compacted using a drum compactor prior to placement in a liner. The packaging efficiency (on average) of compacted and non-compacted filters has been 1.6 filters per cubic foot. Recently, Diablo Canyon has procured a unit to cut/crush high-activity filter elements, which has increased average loading capacity from 1.6 to 2.3 filters per cubic foot.

VOIDS

A number of qualitative postulations provide grounds for belief that the fraction of

void space in the waste cage is minimal. Void spaces or pockets inside irregularly-shaped objects introduced randomly to the solidified liner cage are minimized by the inherent characteristics of the encapsulation binder and process. The vacuum drawn on the liner during the introduction of binder to the liner in the VERITM solidification process tends to expend the air present in void spaces in and around the waste objects, resulting in a net reduction of air volume in the voids. Upon cessation of the vacuum, the pressure of the downward flowing binder will collapse any remaining air pockets. This collapse tends to pull additional binder into these pockets, thus reducing the net volume of the void space present.

The low viscosity of the VERITM binder (100-200 cps) and its natural wetting properties enable migration of the binder throughout the monolith, while displacing both air and water. The water and air are swept downward toward the dewatering internals by the driving forces of the vacuum being applied and the gravitational push of downward flowing binder. The water and air are preferentially pulled into the dewatering internals, due to their much lower viscosity. The more viscous binder meets resistance in passing into the internals through the restrictive openings.

PROSCRIBED WASTES

Certain wastes and waste forms are not physically compatible with the polymer encapsulation process, or are prohibited for regulatory reasons.

Physical Incompatibility

Included in incompatible waste forms are those materials which interfere with the proper migration of the encapsulating binder throughout the monolith, or which may react chemically with the solidified monolith, negatively impacting its strength and durability.

The variety of physical sizes and characteristics of waste to be encapsulated cannot be precisely defined or quantified. Nonetheless, prudent judgment can be exercised regarding introduction of many objects to ensure maximum monolith integrity.

An example of an inappropriate physical waste form is plastic bags of dry active waste (DAW). Plastic bags, or other objects with a large horizontal cross-section, may result in excessive void space, as well as create a shadow directly below the object which may have deficit binder impregnation. In the case of large flat objects, the shadowing problem can be mitigated or eliminated by positioning the object in a vertical orientation to minimize the cross-section of the object when viewed from above. Likewise, a large sheet of plastic draped in the cage may unnecessarily obstruct the normal binder pathway. The same plastic, when formed into a tight roll, would permit adequate polymer impregnation past and under the roll.

Objects with a potential to expand in size due to chemical or biological reactions which may impugn the integrity of the monolith are prohibited. No objects with this potential have been identified. Likewise, no solid wastes subject to encapsulation are chemically incompatible with the solidified polymer monolith.

Regulatory Incompatibility

10 CFR Part 61 and certain other regulations prohibit solidification or encapsulation of wastes which may physically impair the integrity of the final waste form, or result in a waste form which is not allowed under present regulations. Neither the VERITM-polymerized binder material nor the waste that is acceptable to be entombed contains materials identified or defined as hazardous, biological, pathogenic, or infectious. Numerous TCLP tests of the binder, catalyst, promoter and additives have consistently shown the polymerized waste form to be non-hazardous both with and without the presence of wastes that were initially non-hazardous.

ACCEPTANCE FOR BURIAL

The VERITM process is qualified for stabilization of Class B and C wastes when spent ion exchange resin is solidified in liners no larger than 200 ft3. VERITM solidification of spent ion exchange resin is utilized as the base process for the ENCAPTM process for encapsulation of heterogenous wastes.

Since the mechanical and chemical nature of the in situ solidification process (and the resulting waste forms) are unaffected by the presence of heterogenous wastes, it is presumed that the solidified homogenous spent resin remains an accepted stabilized waste form.

The State of South Carolina Department of Heath and Environmental Control (DHEC) granted approval on June 28, 1993 for burial of resins solidified using the VERITM process. The ENCAPTM filter encapsulation process was reviewed, and approval by DHEC was granted on July 8, 1993. A clarifying letter was issued by CNSI on August 3, 1993 which confirmed that encapsulated filters were acceptable for burial at

Barnwell, with the provision that they would be subject to the same burial restrictions and requirements as all other Class B and C wastes.
The solidified polymer material passes TCLP testing for absence of toxic or hazardous materials, including volatiles. The Department of Energy's Hanford, Washington facility has approved the VERITM solidification agent as meeting its Waste Acceptance Criteria (WAC) for disposal in its land burial facility.

CURIE CONCENTRATION AVERAGING

Present industry practice varies among licensees, but the standard method is to average curie content over the filter envelope. In some instances, when filters are shredded, or when a more conservative approach is desired, curie concentration is averaged over the filter media volume, which yields a higher concentration.
The proposed BTP(2) on encapsulation permits the radionuclides of an encapsulated source to be averaged over the volume of a 55-gallon drum (0.2 m3) container. The practice of encapsulating a single filter element (e.g., 1 ft3) per drum equates to a waste loading of 13% (1 ft3 filter volume 7.5 ft3 container volume).
The waste loading of ENCAPTM liner will be greater than the 13% cited for cement encapsulation of a filter in a 55-gallon drum. As a result, the averaging of radioisotopic activity over the volume of the ENCAPTM solidification liner is less dilutive (more conservative) than that of a filter or source encapsulated in a 55-gallon drum.
A key aspect affecting the utility of the ENCAPTM process is the ability to average the curie content of the encapsulated waste over the volume of monolith. Based on the September 16, 1993 Proposed BTP(2) on Concentration Averaging and Encapsulation, a specific rationale was offered in the TR(1) submittal before taking credit for volumes or masses larger than 0.2 m3 when determining waste concentrations. Such a rationale has been reviewed by the N.R.C., and accepted in principle.
The Ward Valley license issued by the State of California permits curie averaging over the entire package when an NRC approved stabilization media (such as VES) is used for encapsulation. The practical effect of curie averaging is (in some instances) derating or lowering of waste classification from Class B and C to Class A for burial purposes.

CONCLUSION

The ENCAPTM process uses VERITM solidification to encapsulate a caged region of waste cartridge filters in a 200 ft3 monolith. The monolith is composed of a waste form (i.e., spent ion exchange resin beads stabilized in VES) which has been demonstrated to comply with the structural stability requirements of 10 CFR 61 for Class B and C waste forms.
A 200 ft3 ENCAPTM monolith could encapsulate a large quantity (up to 200) waste filter cartridges or other solid waste debris with the capacity being limited by physical volume and radionuclide content of the waste, as well as the radionuclide content (if any) of the spent ion exchange resin beads in the ENCAPTM monolith. The caged region is positioned within the liner so that it is at least 4" from any surface of the monolith. Thus, the cage which contains the filters or debris entombed in VERITM waste form is itself entombed in a 4" thick cylinder of VERITM waste form. Generally, no credible mechanism has been identified for an acceptable entombed waste to impugn the integrity of the monolith, when encapsulated with the ENCAPTM process.
Encapsulation of cartridge filters or debris in VES to provide stability offers a less expensive alternative than packaging such wastes in HICs. Where ergonomically desirable, disposal of sources and activated metals, which qualify as low-level waste, in an ENCAPTM liner would greatly improves upon present packaging techniques for these wastes.

REFERENCES

1. TR DT-VERI-100-NP/P-A, "ENCAPTM Encapsulation Utilizing The VERITM Process", Addendum 1 Docket No. WM-105R1 (May 1, 1994).
2. Branch Technical Position (BTP) on Waste Form, U.S.N.R.C. Office of Material Safety and Safeguards, Revised (September 16, 1993).
3. TR DT-VERI-100-NP/P-A, "VERITM (Vinyl Ester Resin In Situ) Solidification Process for Low-Level Radioactive Waste" (Docket No. WM-105) (April 5, 1993).

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LOW LEVEL WASTE SHIPMENT ACCIDENT LESSONS LEARNED

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ABSTRACT
On October 1, 1994 a shipment of low-level waste from the Fernald Environmental Management Project, Fernald, Ohio, was involved in an accident near Rolla, Missouri. The accident did not result in the release of any radioactive material. The accident did generate important lessons learned primarily in the areas of driver and emergency response communications.
The shipment was comprised of an International Standards Organization (ISO) container on a standard flatbed trailer. The accident caused the low-level waste package to separate from the trailer and come to rest on its top in the median. The impact of the container with the pavement and median inflicted relatively minor damage to the container. The damage was not substantial enough to cause failure of container integrity. The success of the package is attributable to the container design and the packaging procedures used at the Fernald Environmental Management Project for low-level waste shipments. Although the container survived the initial wreck, is was nearly breached when the first responders attempted to open the ISO container. Even though the container was clearly marked and the shipment documentation was technically correct, this information did not identify that the ISO container was the primary containment for the waste.
The lessons learned from this accident have DOE complex wide applicability. This paper is intended to describe the accident, subsequent emergency response operations, and the lessons learned from this incident.

BACKGROUND
The Fernald Environmental Management Project, formerly known as the Feed Materials Production Center, is located about 20 miles northwest of Cincinnati, Ohio. Uranium metal products for the nation's defense programs were produced at the facility between 1953 and 1989. During those years the facility produced slightly enriched or depleted products for use in production reactors to make plutonium and tritium at other DOE sites. Uranium production was suspended in July 1989 to focus on environmental restoration. The Fernald workforce has been dedicated entirely to environmental restoration since 1989. Also in 1989, the site was added to the Environmental Protection Agency's National Priority List of federal facilities in need of remediation. In 1991, the DOE officially announced that production at the facility was ended and the program management responsibility within the DOE was transferred to the Office of Environmental Restoration and Waste management. In February 1992, an Ohio Field Office was established to oversee all aspects of a full-scale environmental restoration and waste management effort which is expected to last several years.
On October 1, 1995 at about 1550 (all times in Central Standard Time), an accident occurred involving a shipment of low-level waste from the Fernald Environmental Management Project, in Fernald, Ohio. The shipment was in route to Mercury, Nevada for disposal. The vehicle transporting the waste for the Fernald Environmental Management Project lost control and rolled over in the westbound lane of Interstate 44, approximately one mile southwest of Rolla, Missouri.
The shipment was comprised of a single cargo container designed in accordance with International Standards Organization (ISO) specifications. The Fernald Environmental Management Project uses ISO containers for packaging contaminated debris for disposal at the Nevada Test Site. Loose bulk radioactive waste is not accepted for disposal at the Nevada Test Site. All bulk waste must be packaged to prevent the spread of contamination at the disposal facility. The use of large scale bulk packaging is more economical and a more efficient disposal practice compared to the use of smaller individual packages.
The container involved in this accident contained radioactive contaminated debris generated as a result of the ongoing remediation activities at the Fernald Environmental Management Project. The material in this package, crushed drums and wood pallets, was loaded directly into the container.
The ISO containers are transported from the Fernald Environmental Management Project to the Nevada Test Site on standard flatbed tractor trailers. A loaded ISO container can weigh between 20,000 and 42,000 pounds gross weight. The container involved in this accident weighed 29,160 pounds gross weight. Containers are fastened to the trailer using chains, binders, and straps.

When the transport vehicle involved in this accident rolled onto its side, the force of the impact, combined with the abrasion of the tie-downs between the trailer and the road surface caused the container to separate from the trailer. The container slid about 40 feet before coming to rest in the median. The transport vehicle continued to slide on its side more than 100 feet further down the pavement. There were no injuries as a result of the accident and the container maintained its integrity thus preventing the release of material.

INITIAL RESPONSE

The Rolla Missouri local emergency organizations provided first response to the accident scene. The container placarding and shipping papers correctly identified the hazards associated with the container and resulted in additional notifications. The State of Missouri Department of Natural Resources and Department of Health were called to assist the local responders. A hazardous materials response team from Fort Leonard Wood Missouri was also contacted for support.

Initially, the Missouri Department of Natural Resources attempted to contact the Fernald Environmental Management Project. Failing to contact the consignor of the bill of lading, they contacted the consignee, the Nevada Test Site. The Nevada Test Site emergency operations personnel contacted DOE-HQ who then notified the Oak Ridge office. The Oak Ridge office was contacted because it was the nearest DOE facility with an established Radioactivity Assistance Program team.

Once the Fernald Environmental Management Project was notified, the Fernald Environmental Management Project Assistant Emergency Duty Officer requested a 100 foot exclusion zone around the container until all the facts concerning the waste were established. The Assistant Emergency Duty Officer then contacted the Fernald Environmental Management Project Emergency Duty Officer to report the event.

The Missouri Department of Health, Department of Natural Resources, and the HazMat team from Ft. Leonard Wood had previously determined that there was no contamination as a result of the accident. After determining that there was no breach of containment, the local response organizations announced their intention to initiate recovery operations.

For unknown reasons, the local fire department attempted to open the ISO container. Had the fire department been successful in this attempt to open the ISO container, the containment would have been breached resulting in a high probability of a material release.

The Fernald Environmental Management Project personnel in contact with the on scene commander requested that recovery operations involving the waste container be postponed until a recovery team from the Fernald Environmental Management Project was on the scene. This decision was based on indeterminate level of confidence in the radioactive material experience and instrumentation available at the scene to truly evaluate the situation. The equipment on the scene was assumed to be civil defense 7000 series portable radiation detection instrument. This instrumentation is not sensitive enough to accurately assess the potential contamination released from this container. Given this situation, the Fernald Environmental Management Project determined that it was best to have representatives from the facility at the scene. Recovery operations would be safer with the assistance of a team of personnel familiar with the waste and supported with the proper instrumentation.

Since initial surveys indicated that no contamination was detected and the container was out of traffic, the recovery operations were postponed until the Fernald Environmental Management Project team arrived. Security of the accident scene was maintained by the Phelps County Sheriff Department. One deputy was at the scene when the Fernald Environmental Management Project team arrived at the scene at 0200, October 2, 1994.

RECOVERY OPERATION

The Fernald Environmental Management Project sent a team of personnel knowledgeable of the waste and equipped with the proper equipment to Rolla Missouri to assist with the recovery operation.

A chartered jet flew the team to St. Louis, Missouri where they rented a car for the last leg of the trip. The accident scene was 80 miles southwest of St. Louis, Missouri. The Fernald Environmental Management Project team arrived at the scene at 0200 CST, eleven hours after the accident occurred. This response time was entirely dictated by the conditions. Because the Fernald Environmental Management Project knew there was no release of contamination, additional time could be used to assemble the necessary equipment for the response team.

Once on the scene, the Fernald Environmental Management Project team immediately started air monitoring and surveyed the site to establish a clean command point. The area between where the container contacted the road and where it came to rest in the median was surveyed first. After confirming that this area was not contaminated, the team began to survey the container. The container survey confirmed that there was no release of contamination. Further surveys of the ground beneath the container had to wait until recovery operations were initiated.

The Missouri State Highway Patrol requested that recovery operation be delayed until daylight. During this time, the Fernald Environmental Management Project recovery team evaluated the accident scene to assist with anticipated internal investigations. The team also used this time to coordinate recovery operation activities with the local wrecker operator. A trip to the wrecker yard was made to inspect the original trailer. This inspection identified minor repairs which were completed while waiting for sunrise. The wrecker operator also located a power unit to pull the repaired trailer to and from the accident scene.

The wrecker operator had contracted two cranes with operators to remove the container from the median. Both eastbound lanes of Interstate 44 were stopped by the Missouri state police for approximately ten minutes while the cranes were positioned in the left lane. Once the cranes were in position, the right lane was reopened for east bound traffic. The west bound lanes remained open for the entire recovery operations.

The Fernald Environmental Management Project recovery team remained at the scene ready to provide radiological support during the recovery. Because the container was lying on its top, the team could not certify the integrity of the entire container. As planned, the wrecker crew attached cables to the container. Using the cranes, the container was rolled 90 degrees exposing the top of the container. The Fernald Environmental Management Project radiological support personnel then completed the survey of the container and the ground beneath it to complete the container integrity examination. The Missouri Department of Health, Department of Natural Resources representatives were then invited to perform their own surveys to confirm the Fernald Environmental Management Project results.

After determining that the container was strong tight, the wrecker operator was directed to attach cables and roll the container into an upright position. After a final survey of the container to confirm the container did not breach during recovery operations, the container was lifted onto the highway. The Missouri State Highway Patrol stopped east bound traffic one final time to allow the container to be loaded onto the trailer. Once the container was on the trailer, the loaded trailer and the cranes were moved out of the left lane on to the right shoulder. Traffic was released to use the entire eastbound interstate.

The entire recovery operation was complete by 1150 CST, October 2, 1994, less than 20 hours after the accident first occurred. The actual recovery operations only required 45 minutes from the time the cranes were in position until the time the container was on the shoulder of the road. Eastbound traffic was restricted only during the actual recovery operation and totally stopped for a total of 20 minutes. The entire recovery operation was completed without further incident.

The trailer with the ISO secured to it was hauled to the wrecker yard to await a properly trained and licensed driver to arrive in Rolla and haul the load back to the Fernald Environmental Management Project. The local power unit was used only to get the container off the highway. An authorized driver with the proper training and permits was required to return the container back to Ohio. The container arrived back at the Fernald Environmental Management Project on October 4, 1994.

ACCIDENT EVALUATION

The root cause of the accident was driver inattention. The driver indicated that he was looking in his right side rear view mirror to determine if he had cleared a vehicle that he was passing. While checking the mirror, he failed to realize that his truck was drifting toward the median. The left front wheel dropped into the median which was six to eight inches below the pavement. The driver immediately attempted to recover from this path but the center of gravity of the load had already shifted causing him to loose control of the vehicle.

The driver fought to regain control over a distance of more than 700 feet before the truck rolled over. The speed limit in the accident area is 65 mph. The police report did not identify the truck speed, mechanical condition, or the physical condition of the driver as possible contributing causes of the accident.

LESSONS LEARNED

The Fernald Environmental Management Project conducts regular emergency response exercises at the facility. These exercises have intentionally focused on more likely potential on-site emergencies or natural disasters. The transportation accident in Rolla, Missouri made it perfectly clear that the facility was not completely prepared for an off-site transportation emergency. During the event, each person involved maintained personal field notes of actions taken or observed. After the event was declared to be secure, the parties involved convened a meeting to critique the response. The following observations were noted during this review.

The Fernald Environmental Management Project Emergency Communication Center was not notified of the accident until nearly two hours after it occurred. The primary cause of this delayed notification was the location of the emergency phone number on the bill of lading. This number was in very small type and not conspicuously displayed. The number that was more prominently displayed and called first by the local responders is a dispatcher office which is not manned 24 hours. The corrective action for this issue is to revise the bill of lading and drivers instructions to clearly identify the emergency phone number.

After the Fernald Environmental Management Project was notified of the incident, initial response efforts were dedicated to working with the on scene commander to determine the magnitude of the situation, and assemble the appropriate personnel. Once the situation was clearly communicated to the Fernald Environmental Management Project Emergency Operations Center, the task of classifying the event was initiated.

The Emergency Duty Officer could not find an appropriate reference in DOE Order 5000.3b that addressed transportation accidents with no release of material. This order primarily addresses transportation accidents where there is a release. The Emergency Duty Officer then referred to the "Fernald Environmental Management Project Emergency Plan". In the chapter entitled "Emergency Action Levels", he identified the classification for transportation accidents involving non-release of material. This plan and event classification is applicable to all site transportation accidents involving shipment of hazardous or radiological material coming from the Fernald Environmental Management Project in which the integrity of the shipment is in doubt or can not readily be determined. This plan determined the incident classification as Emergency/Alert. This classification required additional notifications of DOE-HQ and USEPA Region 5. The delay in classifying the event and subsequent notification of DOE-HQ resulted in a temporary confusion as to ownership of the response actions. The Fernald Environmental Management Project Emergency Operations Center was working independently of the DOE-HQ. Until the Fernald Environmental Management Project contacted DOE-HQ to notify them of the incident, neither had any indication that the other was involved. Once DOE-HQ was apprised of the situation and knew the Fernald Environmental Management Project Emergency Operations Center was involved, the response was turned over to Fernald.

The corrective action for this issue will be addressed by the revised shipping papers. Clearly identified emergency phone numbers will prevent the delayed notification of incidents to the Fernald Environmental Management Project. This issue will also be addressed by issuing the completed Fernald Environmental Management Project Transportation Emergency Response Plan. The plan was in rough draft and out for review at the time of the accident. The plan will address the lessons learned from this incident, and provide a ready resource to assist in classifying incidents and identifying the notification requirements.

The delay in contacting the Fernald Environmental Management Project was complicated when specific information concerning the waste in the container was not readily available. The Fernald Environmental Management Project shipment certification files are stored in locked files as required by DOE orders. To protect the integrity of the files, only authorized personnel can open them. No authorized personnel were available to retrieve them at the time of the accident. The on scene commander had the bill of lading but the description on this document is based on DOT proper shipping name. The DOT shipping name does not specify that the container was full of contaminated metal and wood. As a result, the on scene response team, unaware that the ISO contained loose bulk waste, attempted to open the container. There was no indication in the shipping documentation that this was a bulk waste container.

The corrective action for this issue is to revise the drivers instructions provided with every load to better identify the contents of the container. Additionally, a

daily shipping record is provided to the Fernald Environmental Management Project Assistant Emergency Duty Officer and available if needed.
After the Fernald Environmental Management Project Emergency Operations Center committed to send a response team to the accident scene, additional time was required to assemble the people and equipment. There was no established response team for off-site transportation emergency responses. The team that was assembled for this response was well informed but there was no guarantee that these individuals would have been contacted in different circumstances. Once on the scene, the team realized that the emergency communications equipment on hand was limited. The cellular phones provided to support the team were not fully charged and there was no adapter to charge the batteries or operate the phones using the rental car electrical system. Use of these phones was limited to short calls and resulted in some calls being disconnected in mid-conversation. The phones could not be left on due to the weak batteries and the "sky-pager" provided for the trip did not function as expected. The team was basically out of direct contact with the Emergency Operations Center which stood down soon after the team departed for the airport. The corrective action for this issue is to staff an emergency response team with knowledgeable personnel and provide them with the proper equipment and training. This issue will be included in the Fernald Environmental Management Project Transportation Emergency Response Plan.
The wrecker operator involved in the accident recovery operation was extremely cooperative and well organized. The operator assembled the necessary equipment and personnel to quickly and safely recover the container. This situation is more likely the exception than the rule and therefore, better planning will be required in the event such a wrecker operator is not available. In the Rolla incident, the wrecker operator was very instrumental in the success of the operation. As a corrective action for this issue, the Fernald Environmental Management Project secured funding from DOE-HQ for an outreach program for communities along the route. This funding is intended for training first responders along the transportation route but this stakeholder interaction will enable the Fernald Environmental Management Project to identify potential resources for future reference.
The recovery operation was delayed while the wrecker operator repaired the original transport trailer. This was necessary because the transporter of record could not provide another trained driver with truck and trailer in the time frame required. The container could not be placed on a trailer that was not permitted to haul the waste back to Ohio. An alternative plan would have been required had the original trailer been damaged beyond repair.
The corrective action for this issue is to identify transportation services along the route that can provide emergency support and include them in the Fernald Environmental Management Project Transportation Emergency Plan.
CONCLUSION
The Fernald Environmental Management Project has been shipping low-level waste to the Nevada Test Site for disposal since 1985. During this time, more than three-thousand truck loads have been successfully shipped without a serious transportation accident. This performance has resulted in more than 6.4 million loaded miles traveled before the first serious accident was encountered. This performance exceeds national highway transportation safety statistics and the Fernald Environmental Management Project risk assessment projections.
Never-the-less, a thorough review of the occurrence was conducted to enable the Fernald Environmental Management Project to improve future transportation emergency response.
During the critique of this incident, several commendable aspects of the response was documented. One of the more notable highlights of this review is the recognition of the professionalism exhibited by the people involved in this response. The local response organizations performed well and were augmented by competent Fernald Environmental Management Project individuals. The successful integration of the multiple response organizations at the scene resulted in a completed response in a relatively short period of time.
The actual circumstances surrounding this accident dictated the response actions and time frames. The Missouri Department of Health, Department of Natural Resources, and the HazMat team from Ft. Leonard Wood had previously determined that there was no contamination as a result of the accident. Because the container did not fail and was not in the traffic lanes, the actual response permitted some degree of

deliberation. Had the container failed or if it was lying on the highway, the two hour delay in notification to the Fernald Environmental Management Project would have had greater consequences.
Recognizing that the primary cause of the delayed notification was due to the format of the shipment documentation, the Fernald Environmental Management Project has revised these documents to better delineate the emergency response information. This communication will be duplicated on a pocket emergency response instruction card to be provided to the drivers. These corrective actions are intended to prevent recurrence of the destination facility receiving first notification of the transportation accident and to prevent emergency response actions from proceeding without Fernald Environmental Management Project involvement. It is imperative that the facility generating the shipment be involved with the response actions because that facility has the most personal knowledge of the shipment contents. This involvement is most important in the event that the driver's paperwork could not be retrieved from the power unit. The originating facility has the only other copy of the shipment documentation for deciding response actions.
An accidental breach of the container, which nearly happened in this incident, must be addressed. Even though the shipping papers clearly identified the hazards associated with the shipment and the container was properly marked, local first responders attempted to open the container. While it is not known why they tried to open the container, it is clear that the communication to local emergency response organizations must be better. The Fernald Environmental Management Project recognizes that the many local communities along the transportation route must be provided better information concerning the waste. The Fernald Environmental Management Project is pursuing an outreach program to promote awareness for communities along the transportation route, but the effectiveness of this program will be too limited in scope. The primary communication must be provided in the shipment documentation and driver training. The Fernald Environmental Management Project is in the process of implementing these changes.
Instrumental to the success of this recovery operation was the local wrecker operator. This individual secured the required equipment to hoist the container back onto the truck. The wrecker operator also made repairs to the original trailer making it road worthy and able to be used to reload the container. This support was crucial to the recovery operation and cannot be taken for granted. Had the original trailer been rendered unfit for reuse, a local transporter would have been required to support the recovery operation. Finding a driver permitted to haul radioactive material for the return trip to Fernald, Ohio, would have resulted in a longer delay before the operation would have been completed. As it occurred, the original transporter was used to return the container to the Fernald Environmental Management Project. The corrective action for this issue will be for the Fernald Environmental Management Project to contact the transporter to initiate transportation support earlier in the response action.
The final improvement developing as a result of this incident is an improvement of the Fernald Environmental Management Project off-site emergency response capability. Once on the scene, the Fernald Environmental Management Project team performed well but it took too long to assemble the equipment and the communications equipment did not function as required when the team arrived at the scene. The Fernald Environmental Management Project is in the process of organizing a trained and knowledgeable staff of personnel to participate on a facility response team. The lessons learned from this incident will improve the readiness of this team in the future. The team will be assigned Emergency Operations Center pagers to directly contact the team members on duty. This corrective action is intended to prevent recurrence of the obstacles encountered by the team involved with this response.
The Fernald Environmental Management Project is the first DOE facility to be declared a total remediation project. As the remediation of the Fernald Environmental Management Project continues, the volume of off-site transportation will increase. Because the root cause of the accident was driver inattention and no mechanical or driver physical conditions were identified as contributing factors, it is safe to assume that the incident was not the result of a system failure. In actuality, the proper preparation of the shipment contributed to the success of the response. However, despite the best efforts of the generating facility to properly package the waste, an accident will occur involving the release of material. A review of the current notification communication process by the generating facility

prior to an accident could minimize the impact of the incident. The primary revisions required for improved response notifications at the Fernald Environmental Management Project have been identified and implemented. This paper is intended to share the Fernald Environmental Management Project lessons learned from this low-level waste transportation accident to provide other radioactive waste generators an opportunity to be better prepared for a transportation accident.

17-32
AN EXPERT SYSTEM FOR ENHANCING COMPLIANCE WITH HAZARDOUS MATERIALS PACKAGING AND TRANSPORTATION REGULATIONS
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ABSTRACT
The Transportation Management Division (TMD) of the U.S. Department of Energy has supported the development of an expert system for enhancing compliance with hazardous materials packaging and transportation regulations at the Oak Ridge National Laboratory. This system, the Hazardous Material Expert System (HaMTES), was developed with the objective of providing a computerized expert system which a) is easy to use, b) will provide straightforward and consistent application of the hazardous material transportation regulations, and c) will reduce the potential for human error in applying the regulations to both packaging and transportation of hazardous material activities.
HaMTES is based on an analysis of what an expert in hazardous materials shipping regulations could and should do when preparing a shipment of such materials. The system's proof-of-concept was demonstrated using the radioactive material regulations, and it was then expanded to include regulations for all hazardous materials. It has incorporated, as much as possible, multi-media capabilities so that the user can access audio/visual demonstrations of explanations and applications of regulations, the descriptions of package designs, and the actual loading and preparation for shipment of specific package designs. It can be used to determine for a given set of contents, the proper shipping name, hazard class, packing group, whether the material is a hazardous substance as defined by the U.S. Environmental Protection Agency, and whether it is a marine pollutant as defined by the International Maritime Organization. Specific modal requirements, packaging requirements, marking, labeling, and placarding requirements are also identified for the specified hazardous materials contents. It takes a user beyond simply "browsing" through the regulations to using embedded logic that allows the computer to make decisions and ensure all requirements have been scrutinized. The system consists of the program, a few full-motion video files, and a special search engine that allows access to a regulation scanning data base. The system has been developed based upon U.S. regulations, has been tested by experts, and is being upgraded based upon the results of these tests.
Coupling of HaMTES to other transportation operations management software is underway, and methods for maintaining and upgrading the program have been developed. The details of the capabilities of HaMTES, its computer hardware requirements, and the plans for implementing HaMTES within the DOE shipping community will be presented in this paper.
INTRODUCTION
The ability of the computer to simulate and make decisions as a subject matter expert (expert system) has greatly expanded potential computer applications in complex decision-making situations. With this type of computing power available in an increasingly user-friendly format (using object-oriented programming and Hypermedia capabilities), the development of a transportation packaging expert system that is based on the hazardous material regulations was possible. Under the sponsorship of the U.S. Department of Energy's

(DOE's) Transportation Management Division (EM-261), the Transportation Technologies Group at Oak Ridge National Laboratory (ORNL) has designed and developed an expert system application of the U.S. hazardous materials transportation regulations.
The strategy to develop the expert system prototype was to, first, develop modules to capture the knowledge of different areas of the transportation regulations; and, second, append these different modules into one final package. The individual module development focused on one prototype for transporting and packaging radioactive material and another for transporting and packaging hazardous chemical materials.
It was found that the capability to scan the pertinent regulations in an interactive mode is invaluable. The regulations are always at hand for further consultation using a regular menu or a hypertext search mechanism. Graphic files with package information (such as dimensions, authorized contents, etc.) can also be accessed. Additional assistance is provided by means of graphic files and/or video images.
From the analysis of the different features required for the expert system prototype, it was concluded that the developmental efforts should be directed to a WindowsTM 3.1 Multimedia environment. Multimedia technology usually works as an interactive software system that gives personal computer users the ability to organize, manage, and present information in a number of formatstext, graphics, sound, and full-motion video.
The two prototype modules have been developed and have incorporated within them the expertise for the transportation and packaging of radioactive and hazardous materials. The two modules are integrated into a single system being implemented to support DOE-wide transportation operations.
Work is in progress to include HaMTES as the major portion of the hazardous materials module of the automated transportation management system being implemented to support DOE-wide transportation operations.
The verification and validation of HaMTES has been performed by comparing the operation of the HaMTES system with known shipments and packages. Model validation was performed by trained and experienced traffic specialists at ORNL using 49 CFR. HaMTES was also sent as a beta-test version to transportation experts in several DOE and contractor locations who then tested whether the computer model accurately represents the real system. In addition, experts from the U.S. Department of Transportation (DOT) and the Nuclear Regulatory Commission (NRC) reviewed and commented on the logic diagram.

DEVELOPMENT

The strategy to develop the expert system was to, first, demonstrate the feasibility of developing an expert system prototype by developing modules to capture the knowledge of different areas of hazardous materials transportation based on the shipper's perspective; second, select an appropriate environment in which to deploy the expert system; third, analyze the feasibility of appending these different modules in one final full package; and fourth, develop the full-scale expert system.

Knowledge Acquisition

The DOT and the U.S. Environmental Protection Agency (EPA) have stringent regulations regarding the shipment of hazardous materials which must be complied with before these materials are shipped. These regulations are extremely complicated. To achieve compliance, the shipper must sift through hundreds of regulations, search large tables, and perform calculations, depending on the applicable regulations. Only when all the applicable information has been determined, depending on the regulations and the tables, can the shipper properly prepare a package for shipment. Transportation specialists and packaging engineers must have the expertise to know which regulations to apply to a given shipment. It is this knowledgethat of the correct and complete path through the regulationsthat this expert system contains.
The initial requirement for proper knowledge acquisition was to identify the regulations pertinent to hazardous and radioactive material transportation. From those identified, it was decided to implement in the HaMTES the 49 CFR 100-173, the transportation sections of 10 CFR, and the transportation sections of 40 CFR. When this project began, HM-169A and a corresponding NRC rule making proposed changes to the radioactive materials portions of the regulations; these were expected to be finalized by 1994. The decision was made to develop the prototype to these proposed regulations to minimize obsolescence. HM-169A has not yet been finalized. In order to complete and release version 1.0, the radioactive module was modified recently to

reflect the current DOT and NRC regulations.
These regulations state the mechanism by which the transportation of hazardous and radioactive materials must be done. There is also room for interpretation of the regulations. It is also agreed that understanding the mechanism of how these regulations should be applied may signify a large task and representing them computationally undoubtedly poses a formidable endeavor. Consequently, the first obstacle faced by this project was to choose an appropriate way of representing the regulations and how to navigate through them. The solution was found in the use of logic diagrams. Logic diagrams allow the representation of a knowledge base by means of a graphic tree where a decision point is found at the beginning of each branch. Figure 1 illustrates a simplified number of decision points found in the logic for determining packages for the transportation of radioactive materials.
An IF-THEN rule mechanism applied to the decision points in the logic diagram made it possible to convert the logic tree into a computer program.
Environment to Deploy the Program
To determine the appropriate deployment environment, some developmental stages were necessary. The first stage was to produce a proof-of-concept version of the program. The preliminary program to support the proof-of-concept process was developed using Prolog (Programming in Logic). This preliminary program helped determine that it was possible to develop such an expert system. At the same time, this preliminary prototype was an aid in recognizing the basic requirements for development and the required features of the program. The goal of the proof-of-concept stage was to determine if it was feasible to develop a more comprehensive expert system.
The basic requirements of the prototype development were that the expert system would run on the personal computer (PC) platform. It was clear that one requirement of the expert system was that it should provide the ability to access the regulations from the commercial programs that update the regulations on regular basis. The friendliness of the user interface along with the ability to navigate throughout the regulations and display graphics and full-motion video information on the screen were other important requirements. The first concern of the knowledge engineer was to develop an interface between the user and all the specific program(s) that were transparent regarding the manipulations that are required to go from a) one set of input data, to b) calculation programs, to c) decision-maker programs, etc. Thus, a straightforward system of a question-answer relationship between the computer and the user was highly desirable. The environment selected had to be able to implement a rule-based system that represents the regulations.
Other requirements, including access to several types of information sources, were suggested by the logic diagram. Behind every question asked of the user there is a set of regulatory requirement(s) which may influence the answer given by the user. Thus in some cases, it was obvious that the user had to access the regulations before an answer could be given to the question. This feature provides the less knowledgeable user with enough information to give an appropriate response. It is well-known that regulations normally refer to other regulations or parts of regulations, which in turn may refer to other regulations and so on. In such complex cases, the expert system not only needs to access regulations, but also it must be able to browse through them. Additional explanations about regulations, interpretation of regulations, or any other aspect of the decision mechanism to determine types of packages were required to be available in a form of video images or audio. Consequently, multimedia elements, such as Hypertext navigation, visual aids (whether as graphic or full-motion form), and mouse-driven interface elements, were considered to be essential parts of the expert system.
To decide what software tool would be used to create the expert system prototype, five possible tools using six criteria were rated. These criteria were 1) multimedia capabilities, 2) rule-making capabilities, 3) flexibility of the environment, 4) user interface provided by each tool, 5) data-handling capabilities of each tool, and 6) ease of use of each tool. The following tools were rated: OWLTM Industries GuideTM (a multimedia document presenter), the C/C++ programming language (a general programming language), the Prolog programming language [a logic-based disk operating system (DOS) programming language], general expert system shells (tools used to create expert systems), and the Visual BasicTM programming language (a general Windows programming language).
Ratings on these criteria suggested that Visual Basic was the best environment in which to create the expert system. Although a Prolog-based code had already been

created during the proof-of-concept stage, it was clear that putting the Prolog version together with multimedia features would be difficult. Consequently, this option was abandoned. The solution found for the prototype stage was to translate the Prolog code into the Visual Basic code, which is a Windows application. Figure 2 illustrates the general architecture of HaMTES and the various interactions of the system with sources of information.

The prototype expert system that covers radioactive materials shipment preparation has been subjected to a review and validation process by experts in the field. Minor changes have been suggested, the program has been revised, and implementation of this module within the complete system has been undertaken. The development of the expert system prototype has demonstrated that it is possible to represent knowledge about radioactive material transportation using a logic diagram that has been translated into a rule-based system. The mechanism to access updated regulatory information from RegScanTM, a commercial program, has also proven to be technically feasible.

Appending Modules

The radioactive and hazardous chemicals modules have been finalized and integrated. Consequently, the final version has incorporated both modules with the appropriate links so the user can prepare shipments involving radioactive, hazardous, and most importantly, mixtures of both types of materials. The system developed here promises to help transportation management not only with routine daily shipping duties but also, and more importantly, with those unusual cases requiring more regulatory analysis to ensure compliance.

Implement the Full Program

The full program development has occurred in several stages and will continue to develop depending on the funding resources allocated to the project. Version 1.0 of HaMTES will help the user to prepare the shipment and the shipping papers. The HaMTES output includes information on the proper shipping name, markings, labeling, and placarding as well as package type. Version 1.0 does not include accessing a data base of Type A packages, UN packages or Type B packages since such a data base has not been built. However, it will indicate the specific paragraph(s) of the regulations where the type of package is referenced and the special provisions indicated. HaMTES Version 1.0 has been prepared for the shippers' most pressing needs.

It is anticipated that later versions of HaMTES will include hazardous wastes and/or a completed package data base. Further releases will include better audio and video features that have not been included inversion 1.0. These Multimedia features were demonstrated in the beta test release with positive comments from the reviewers.

VALIDATION AND VERIFICATION

The following subsections describe the validation and verification of HaMTES which has been complete.

First Phase of Verification and Validation of HaMTES

During the first phase of verification and validation, experts from ORNL reviewed the program and submitted their written and oral comments. Their comments varied from changes to the screens to changes in the logic diagram. All these comments have been incorporated into the computer program.

Second Phase of Verification and Validation

During the second phase of verification and validation, the HaMTES program and a review package were sent to experts at several DOE and contractor sites. The review package consisted of an installation manual, an operating manual, and a set of eight disks containing the program, full-motion video files, and the special search engine that allows access to regulation scanning (RegScan) for those who have this commercial program. For those who did not own a copy of RegScan, the beta test release version of HaMTES simply provided an error message if access to the non-existent regulations data base was attempted.

Presentation of HaMTES to a JAD Session

The DOE Transportation Management Division has different software applications under development. Joint Application Development (JAD) sessions are being held to analyze, improve, and discuss the integration of these software packages for transportation purpose. The presentation and discussion of HaMTES during a JAD session resulted in suggestions on user interface as well as changes in logic. As DOE funding allows, it is anticipated that HaMTES will be the front end for the hazardous module of the Automated Transportation Management System (ATMS) being

developed under the sponsorship of DOE's Transportation Management Division.

Logic Diagram Analysis with NRC and DOT

A meeting was held with regulatory experts from the NRC and the DOT. At these meetings, the HaMTES logic diagram for transporting materials according to the current regulations was thoroughly reviewed. During the NRC review of the logic diagram, the work was concentrated in the radioactive module of HaMTES. The review with the DOT experts was essentially focused on the hazardous materials portion of HaMTES. The findings of these two meetings were incorporated into HaMTES Version 1.0.

FUTURE OF HaMTES

As previously mentioned, DOT regulations determine the shipping requirements for the transportation of hazardous materials, including radioactive materials. The logic involved in determining these requirements for a shipment has been successfully implemented as a decision tree, using the Revised Radioactive Materials Transportation Regulations as proposed in HM-169A and the collateral proposed changes in NRC's 10 CFR Part 71. Since HM-169A has not yet been published, HaMTES Version 1.0 has been built on a modified logic diagram which reflects the current regulations. The version of HaMTES based on the proposed regulations has been filed for use when the regulations are finalized.

Version 1.0 of HaMTES, which is a stand-alone version, will be distributed to DOE facilities during early 1995. Concurrently, methods for making HaMTES available to users outside of DOE are being investigated. Future versions of HaMTES are expected to include more detailed data bases of package designs (Type A, Type B, UN packages, etc), waste material shipment capabilities, and audio and video help to users. Eventually, when resources are available, HaMTES will be integrated into ATMS. Modifications to fit the ATMS systems requirements will be necessary. Mechanisms of integration have already been analyzed and tested.

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NEW BOX-CONTAINER SYSTEM FOR WASTE DRUMS: DYNAMIC TESTS AND QUALIFICATION

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ABSTRACT

In this paper we present a new 20' box container, designed by the firm CORROBESCH for the transport of radioactive and other dangerous wastes, and having a carrying capacity of 22 tons. The container itself weighs only 4 tons, and it incorporates a proprietary corrosion finish that is highly resistant to mechanical wear, deformation, and radioactive contamination. Recently, the container was subjected in Germany to a series of collision and drop tests, as specified by the International Atomic Energy Agency (IAEA) , and it passed these tests to complete satisfaction. As a result, the container received certification by agencies such as the Germanischer Lloyd (GL), the Deutsche Bahn A.G. (DB), and the Bundesanstalt fr Materialprfung (BAM). At the same time, the writers have also developed a mathematical model of the container to predict its dynamic behavior during service loads, and have been able to make motion predictions that are in good agreement with signatures recorded during tests.

BOX CONTAINER

The German firm CORROBESCH, with corporate offices in Hamburg, has recently developed a 20 foot ISO box-container of type M-CB 49, intended for the transport of highly radioactive wastes and other dangerous products stored in liquid and granular form within steel drums. While the container weighs only four tons, it can carry up to 22 tons of cargo in various configurations. The container is made of stiffened steel plates, and is closed at one end by two doors with latches and bolts. All interior and exterior surfaces are covered with a highly resilient paint developed

and patented by CORROBESCH, which is resistant to chemical corrosion, to mechanical deformation and wear, to scratches and tear, and to radioactive contamination. Indeed, the anti-corrosion finish can sustain radiation levels as high a 30 MGy without damage or permanent contamination, and painted steel plates can be bent by up to 140 degrees without obvious fissures in the corrosion finish. As a result of these characteristics, the container can easily be cleaned, decontaminated and reused.

SUPPORT TRAYS AND MOTION-RESTRAINING DEVICES

One of the aims in the development of this container was to achieve a system that could transport cargoes with variable numbers of drums stacked in changeable configurations -including an only partially filled container- while ensuring the safety of the container for accidental dynamic forces and vibrations that could conceivably arise during transport. This goal led CORROBESCH to the design of an interchangeable tray system for the drums that can easily be handled and loaded into the container, together with passive restraining devices that prevent this assembly from moving inside the container when solicited by dynamic forces. Another important consideration was the fact that after loading and sealing of the container, it is not possible to inspect or adjust the contents until after it has arrived at its destination. Since the transport can last days, weeks or even months, it was deemed necessary to avoid the use of restraining devices on the drums that rely on initial set forces, such as tension straps, because with time such devices can loose their tension, and thus their effectiveness. Hence, CORROBESCH developed passive restraining devices that adjust themselves as the drums vibrate back and forth, see Fig. 1.

The drums are stacked in two layers, and rest on light flexible trays within grooves that have been carved out in the shape of the drum heads. There are three layers of trays: one on the bottom surface, one between the two drum layers, and one on the top covering the drums. These trays are not continuous at a given elevation, but consist of individual trays having the width of the container, with space for up to three drums; these are placed, one at a time, into the container. Between each individual tray lies a removable restraining bar that is anchored elastically to the side walls of the container. A remaining clearance space above the top tray layer of about 50 cm, which is needed to load or unload the drums using forklifts, is then secured with passive restraining devices consisting of masses and springs that lock the trays into place, and which adjust themselves as the container is shaken by vibrations during transport.

DYNAMIC TESTS

On June 29, 1994 the fully loaded container with a total weight of 26 ton was subjected to a series of controlled collisions tests (1) at the experimental facilities of the German Railroads (Deutsche Bundesbahn, DB) in Minden, FRG. The container was loaded and anchored onto a 20 ton flat car, which was subsequently rammed by an 80 ton car at varying speeds. The speed was adjusted in steps so as to achieve prescribed levels of acceleration of up to four times the acceleration of gravity at the anchoring points of the container. Acceleration signatures were recorded at various points in the container, and extensive measurements were made of the motions experienced by the drums. The container withstood these tests without external damage, no drums were broken, and no fill material was spilled. Following these test, the container received certification by the Deutsche Bahn, the Germanischer Lloyd, and the Bundesanstalt fr Materialprfung as being fit for the transport of dangerous goods.

Six months later, on December 6, 1994, a fully loaded container was subjected to a drop test at the port facilities of Blohm+Voss in Hamburg (2). The container was hung by one edge 30 cm above the ground and dropped with some inclination onto a concrete mat overlain by a thick steel plate. The container crashed first onto a corner, then pivoted about this point and slammed onto the surface with great violence. The motion signatures were recorded at various points with triaxial devices; additionally, the test was captured on video. While the container suffered some external damage, no fill material was spilled to the exterior.

MATHEMATICAL MODELS

Clearly, full scale dynamic tests are very expensive and time consuming. Even though a number of essential tests are prescribed by the licensing agencies and must unavoidably be carried out, it is also true that these test may often not be exhaustive enough. One can easily imagine-or postulate-scenarios not covered by the

tests. Thus, it is desirable to supplement the physical tests with mathematical models that can be used to make predictions about hypothesized accidental conditions. Such tools can be invaluable for the prediction of the behavior of the container during such conditions, and can also be used to interpret any measurements made during actual tests. These considerations moved the writers to develop a discrete numerical model for the analysis of the container when subjected to dynamic conditions (3). In essence, the authors implemented a finite element model with discrete elements that allow for large motions of the components. The technical details will not be presented in this paper, however, not only because of the limited space available, but also because the theoretical details of such discrete models are generally well understood. Thus, we shall describe only in very general terms some of the characteristics of the model used to predict the behavior during collisions.

Consider a container loaded onto a flat car, which is in turn impacted from the rear end by another heavy car (Fig. 2a). The container is loaded with two layers of drums on trays, and there are eight rows with three drums in each (Fig. 2b). Thus, the container contains a total of 2x3x8=48 drums. To a first approximation, such a collision elicits mostly longitudinal horizontal forces within the container, so it suffices to represent this system in terms of a plane (two-dimensional) geometry. In such model, we have made the following assumptions:

Both the ramming and flat cars as well as the container itself ---but not its contents-are infinitely rigid separate bodies of known masses.

The shock absorbers between the cars are modeled as a linear spring-damper system, which may neither exceed a maximum elongation, nor be subjected to tension. Upon reaching its maximum elongation, the ensuing collision is assumed to be of negligible duration (the time required for a shock wave to travel through the flat car is between one and two orders of magnitude smaller than the response time of the container with the drums inside).

The drums in the container are rigid bodies connected by elastic springs (the trays and the restraining devices); each drum can both translate as well as rotate about a transversal axis.

The longitudinal motion of the bottom edge of the inferior row of drums is impeded by friction against the bottom of the container. Upon exceeding its maximum frictional value, each drum may slide independently.

Inelastic damping forces are neglected.

Using these assumptions, we have then made motion predictions for tested configurations considering three mechanical models of increasing sophistication:

First, the two colliding cars were assumed to be perfectly rigid and obeying the physical laws governing an elastic-plastic collision (conservation of momentum and energy balance). This is the very simplest model that we used to make predictions about the abrupt change in velocity that takes place when the shock absorbers between the cars have reached their maximum deformation (21 cm).

Secondly, we considered the collision of two rigid bodies separated by a spring. This model can be evaluated in closed form, it supplements the previous model, and allows one to make simple predictions about the orders of magnitude of the response (accelerations, forces, etc.).

Finally, we developed a complex numerical model in which the contents of the container are discrete masses separated by springs that can move, rotate, or slide relative to each other. The resulting system of non-linear equations were then solved numerically by means of a special computer code written in the FORTRAN language. Motion signatures were then evaluated at various points and compared to those recorded at the Minden facilities.

After the two cars make contact, momentum begins being transferred through the shock absorber to the flat car with the container, which in turn transfers some of its kinetic energy to the drums in the form of a wave propagating through them in both a horizontal as well as vertical direction. If and when the maximum elongation of the shock absorbers is reached, an elastic plastic collision ensues, which changes abruptly the momentum, and thus the velocity of the flat car. This collision is associated with very large accelerations, which do not immediately affect the drums, since the compliance of the trays and the sliding capability of the drums on the bottom of the container limit the forces that can be transferred to them. A shock wave then develops which again propagates through the system eliciting translations, rotations, and sliding. Each drum experiences then different motions, and must be

analyzed in turn.
Figure 4 depicts the maximum acceleration of one of anchor points of the container as computed with the aid of the second elementary model (the spring-mass system), and shows also values recorded during the actual tests at the DB Minden facilities (where the actual mass ratio was 0.625). As can be seen, the simple model provides excellent estimates of the maximum acceleration, even though the actual system exhibits a more complex (non-linear) behavior due to the changing participation of the contents of the container with the impact velocity.
Figures 5 and 6 show a comparison of the motions in the front left row of barrels in the container, as computed with the discrete model and as recorded at the Minden test facilities, for a collision velocity of 10 km/hr. Figure 5 gives the motion at the bottom of the lower layer of barrels (similar to the container's motion), while Fig. 6 provides the acceleration at the top of the upper layer of barrels. In both cases, the synthetic signatures have been low-pass filtered to match the filtering used during the tests (although this made little difference in the computed motions). While the computed and recorded motions are clearly not the same, they exhibit similar overall trends, the orders of magnitude of the acceleration are similar, and there is general concordance in the duration of the process and in the frequencies of oscillation. Indeed, the differences in these figures is no worse than the variations that were observed in actual motions recorded at the left, center, and right barrel columns (not shown), which on account of symmetry could have been expected to be very similar, or even the same. This experimental observation indicates that the trays supporting the barrels may not have had symmetric stiffness/geometric properties, and/or that non-linear processes (e.g. local yielding of materials etc.) played an important role. Nonetheless, the numerical model provided useful results concerning overall physical behavior.
CONCLUSION
Discrete numerical models for the dynamic analysis of box containers intended for the transport of drums filled with radioactive products, such as that briefly described herein, can provide valuable insight in the behavior of such systems during hypothesized accidental conditions. More importantly, they can be used to supplement and expand the results obtained in actual testing, to help in their interpretation, and above all, to test for design changes in the computer without the need for expensive experiments. The model considered in the context of the new 20' box container designed by CORROBESCH has yielded prediction numbers that were in reasonable agreement with the signatures recorded during actual tests.
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17-34
ADVANCED TRANSPORT SYSTEM FOR DANGEROUS MATERIALS BASED ON A NEW 20' ISO CONTAINER WITH COMPUTABLE LOAD BEHAVIOR
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ABSTRACT
Transports of dangerous materials have to meet increasing safety requirements. An essential prerequisite in the licensing procedure for a new transport system is the demonstration of its integrity even under accident conditions.
The transport unit presented here is based on the 20' ISO steel container. It has been provided with a new system for reliably fixing the loaded material, which is self-locking and does not have to be retightened and whose behavior under load can be mathematically precalculated due to its well-defined geometry and known material properties.
The new container for dangerous materials has been designed for an optimum payload capacity of 22 tons with a low dead weight of 4 tons. A positive self-retaining cargo securing system was developed so that no tightening belts are necessary.

Vibrations during transportation (whether by road, rail or ship) are largely neutralized by the self-securing system. Moreover, this cargo securing system can be computationally and technically adapted to all load cases. The precalculated load capacity of the transport unit was practically confirmed by a crash test with an impact velocity causing fourfold gravitational acceleration at the loaded material. Loading and transport times are uncoupled by the application of this new safety system because no retightening of the securing elements is required after loading, unlike the case of tightening belts, even after an extended service life.
The container is internally and externally provided with a coating which is easy to decontaminate. The radiation resistance of this coating has been demonstrated in practice up to 30 MGy. The coating is extremely corrosion-resistant and considerably prolongs the container life.
The necessary certifications by the competent institutions Germanischer Lloyd, Deutsche Bahn AG and Bundesanstalt fr Materialforschung are available.

INTRODUCTION

The newly developed FHT freight securing system is a reusable, elastic system consisting of support trays serving to secure drums with dangerous goods in a 20'I SO box container during road and rail transport. The geometry and material of the support trays can be varied within broad limits so that the system can be adapted to the various levels of demand in the transport regulations for dangerous goods corresponding to the classification of the dangerous goods. In the following, this is shown in detail on the basis of restraining devices for steel drums used for the transport of low level radioactive wastes.
To date, steel drums with dangerous freight have often been anchored by tension straps to secure the load in the container. The tension straps were applied around several steel drums and attached at the bottom to the anchoring points in the container. Securing the load in this way often proved to be disadvantageous in practice since steel drums do not have any special anchoring facilities, such as eyelet attachments, shackles or clips, for reliable restraining with tension straps. Due to the inevitable vibrations and movements of the steel drums in the container during transport, the tension straps may slip and thus fail to fulfill their task of securing the load.
The FHT freight securing system avoids this disadvantage since an interchangeable tray system takes up the forces emanating from the drums and transfers them to the container walls or container roof and floor. Wedges between the individual trays ensure that the system automatically adjusts itself.
The system can be described mathematically. It can thus be precalculated and designed for different load cases, e.g. collisions at various speeds or drops from different heights. Expensive and time-consuming tests to demonstrate the safety of different load variants can be dispensed with.
The container, interchangeable tray system and drums form one package, which fulfills the requirements for the transport and storage of radioactive substances. Not only is the tray system of significance, but also the jointless interior design of the container and is paint finish. this paint finish is radiation- and corrosion-resistant, provides good protection against mechanical damage and can be easily decontaminated.

CONTAINER

The container is a 20' I SO piece goods container according to DIN ISO 668:1 CC, closed on all sites and of an all-steel design. The manufacturer's model code is M-CB 49. It has a double-wing door at one end. The door wings each have a rubber seal all the way around and in the closed state they are secured by latches and bolts. the container floor is shaped like a trough. The inner surfaces of the walls and their transition of the floor are jointless. The container is coated on the inside and outside with a paint finish whose surface is easily decontaminated pursuant to DIN 25415 Part I. Inside, the container is equipped with anchoring points to secure the load. The permissible total weight is 26,000 kg, and the payload 22,000 kg.

FREIGHT SECURING SYSTEM, PAYLOAD

The FHT freight securing system is a dynamic system whose main elements are the support trays shown in Fig. 1. These shaped pieces serve as elastic top and bottom shock absorbers, which in this example comprise the top and bottom sections of the drums. They are contiguous to each other and the container walls, to which they transfer the forces.

Figure 2 shows a layer of 24 x 200-l drums in a 20'container. The support trays of the second and third level have grooves at the side (Figs. 1 and 2), to accommodate the restraining and tensioning devices. The restraining elements are bars anchored to the side walls of the container to effect load transfer. The tensioning elements are wedges which slide into the wedge-shaped grooves due to their own weight and thus continuously ensure passive tightening of the entire formation.
The drums are attached to the container roof by passive restraining devices consisting of masses and springs that lock the trays into place and which adjust themselves as the container is shaken by vibrations (Fig. 3).
Figure 4 shows a view inside a container with secured drums.
The geometry, mass and physical properties, such as spring and damping behavior, of the materials are known. The formation consisting of the drums and restraining devices represents a system of coupled oscillators, whose essential properties can be described mathematically and may thus be precalculated. The mathematical modeling of such a system for the collision was undertaken by H. Flessner and E. Kausel (1).
The advantage of this freight securing system is its calculability. Different configurations can be designed computationally, thus enabling their reliability to be predicted. Expensive and time-consuming collision and drop tests, which have to be performed for systems not determinable mathematically in order to demonstrate transport safety, could be dispensed with.
Standard configurations with the FHT freight securing system are:
  48 200-l drums in two layers with a maximum drum weight of 437.5 kg;
  42 200-l drums in two layers with a maximum drum weight of 500 kg;
  24 200-l drums in one layer with a maximum drum weight of 875 kg;
  42 280-l drums in two layers with a maximum drum weight of 500 kg;
  21 280-l drums in one layer with a maximum drum weight of 1,000 kg.
The maximum drum weight depends on the weight of the freight securing system, in which connection the permissible payload of 22,000 kg per container may not be exceeded.
Figure 5 shows the container before loading.
LOAD TESTS
The load tests necessary for granting the qualification certificate for the design as a package pattern type IP-2, IP-3 and type A for the transport of radioactive substances by the Federal Institute for Material Research and Testing (BAM) were successfully performed on a container fully loaded with 48 200-l drums.
For these tests, the support trays and restraining devices were made of pasteboard.
The following tests were performed:
  tests Pursuant to DIN-ISO 1496 Part I
with the following supplementary tests:
  front and door wall load: 51,000 kg
  side wall load: 33,000 kg
  anchoring point tensile test with 1.5-fold rated load.
  Collision tests with delayed impacts at the advancing lower corner fittings of the container with low pass filtering of 16 Hz
- up to 40 m/s towards the rear door and
- up to 32 m/s towards the front wall
  Free-fall tests, primarily from 0.3 m approx. along the principal diagonal onto the roof edge of the container door side and secondarily on the roof surface.
On the basis of the positive test results, BAM certified that the package type
  fulfilled the requirements specified in Exception 49 of the Exception Regulation for Dangerous Goods for the scope of the German Ordinance on the Transport of Dangerous Goods on the Rhine (ADNR), the Dangerous Goods Ordinance for Inland Waterways (GGV BinSch), the Dangerous Goods Ordinance for Rail Transport (GGVE) and the Dangerous Goods Ordinance for Roads (GGVS) for the transportation of Substances with low specific activity and of surface-contaminated objects and
  corresponds to the requirements made on Type A packages for the transportation of solid radioactive substances laid down in the Regulations for the Safe Transport of Radioactive Material, 1985 Edition (as amended in 1990) of the International Atomic Energy Agency in Article 524.
The following data were recorded for the collision tests:
  collision velocity v km/h
  longitudinal acceleration velocity m/s at the advancing restraining fittings of the container to evaluate container strength

longitudinal acceleration velocity m/s of the bottom, central and top layers of drums in rows 1, 3, 6 and 8 to evaluate the vibration behavior of the drums inside the container.
The loaded container was placed on a flat car with a net weight of 20,500 kg. A ramcar with a weight of 80,000 kg ran down a shunting slope against the free, breakless flat car standing on the track. Changes in the direction of load (rear door or end wall in the direction of collision) were effected by rotating the flat car. Before beginning the tests and after changing the direction of load, a setting blow was first executed in order to exhaust the axial play between the holding pins and the restraining fittings in the direction of collision.
Table I shows the results of the collision tests with the rear door in the direction of collision.
After test no. 5, damage to the support trays was found upon opening the rear door. No obvious damage to the drums and container was established. No permanent deformations occurred at the container door and in the roof region. Tightness of the rear door was maintained. On the basis of the container stability, which was sufficient for the evaluation criteria, a further test with an acceleration of 4 g (test no. 10) was carried out at the client's initiative (CORROBESCH Vertriebsgesellschaft mbH), after the variants with the end wall in the direction of collision.
The interior of the container was not inspected after test no. 10. However, no external damage was apparent.
Three tests were performed with the end wall in the direction of collision and the results are shown in Table II.
After test no. 9, the interior of the container was not examined. No external damage to the container was detected.
Figure 6 shows the interior view of the container before unloading after the collision tests. The support trays and passive restraining devices display damage, whereas the drums and container remain undamaged.
The free-fall test was performed on a crash plate consisting of a concrete mat of 18.0 m x 15.4 m with a thickness of 1.5 m (approx. 800 t total weight), which was overlain by a steel plate 35 mm in thickness. The container was hung in such a way that it was dropped from 0.3 m on to its upper corner on the doorside and then pivoted about this point and slammed onto its roof.
The container was equipped with a total of 14 acceleration sensors. The acceleration signatures display values of up to approx. 140 g.
Inspection of the container after the drop revealed the following:
No damage occurred which would permit any escape of the radioactive contents or would result in reduced shielding.
Control measurements of the upper door was corner fittings at the corner pillar, the longitudinal roof bar and at the side wall in the direction of the was diagonals did not indicate any measurable linear deformations.
The front corner pillar directly above the corner fittings displayed a compression, which may in part be attributable to the fact that the other corner fitting did not crash onto the steel plate during the drop test.
ANTI-CORROSION FINISH
The containers are coated with the special anti-corrosion finish CORROBESCH-DF-Nuklear. This paint coating is smooth, glossy and tight to water vapor diffusion. Its base component consists of tar epoxide polyurethane and inert additives as well as color components. The second component is a cycloaliphatic amine.
Its good radiation and corrosion resistance and ease of decontamination was experimentally demonstrated according to the relevant DIN and ISO regulations (2). Anti-corrosion finishes were tested in the colors black, yellow, blue, red and white.
After exposing the test pieces to radiation of 107 Gy discolorations of various intensities resulted (greatest discoloration of the white finish, least of the black finish), as well as very slight decreases in gloss. Other changes were not detected. In order to evaluate possible changes after the action of chemicals and decontamination solutions, ethanol, 20% phosphoric acid as well as sodium hydroxide solution, materials testing mixture A 20/NP II and additionally two decontamination detergents frequently used in practice were applied to the test pieces which had been irradiated with various energy doses.

After an exposure time of five minutes, both unexposed surfaces and also those exposed to a radiation dose of up to 107 Gy displayed only insignificant changes, irrespective of coloring agent.

After an exposure time of 24 hours, all surfaces displayed very slight changes after the action of the materials testing mixture A 20/NP II and the two water-diluted decontamination detergent solutions.

After an exposure time of 24 hours to ethanol, phosphoric acid and sodium hydroxide solution the color change increased with rising radiation exposure of the surface and the hardness of the finish temporarily decreased (the white finish had the poorest values).

Other changes such as blisters, fissures or swelling due to the action of the above-mentioned solutions were not detected on any of the specimens irradiated with up to 107 Gy.

All anti-corrosion finishes, whether black, yellow, blue, red or white, are easy to decontaminate before irradiation.

The following results were obtained after irradiation:

with 3 x 105 Gy = very easy to decontaminated,
with 3 x 106 Gy = easy to very easy to decontaminate,
with 6.8 x 106 Gy = easy to decontaminate,
with 107 Gy = moderately easy to easy to decontaminate.

CERTIFICATION

To date, the following certification has been granted for the container or for the package:

By Germanischer Lloyd

CSC license,
certification of the successful free-fall test in accordance with the requirements of the Regulations for Safe Transport of Radioactive Material, 1985 Edition (as amended in 1990) of the IAEA in Article 524 (622).

By BAM

license as a IP-2, IP-3 and type A package

Deutsche Bahn AG

certification pursuant to UIC 592-1

Main Customs Office Bremerhaven

license for the transportation of goods in bond.

CONCLUSIONS

The ISO piece goods container MCB 49 coated with CORROBESCH-DF-Nuklear with the FHT freight securing system represents an advanced system particularly suitable for the storage and transport of radioactive substances.

It offers economic benefits due to the fact that

the computational demonstration of the safety of the box-container system can replace expensive tests,
the container shows a very favorable ratio between payload and net weight, and
the anti-corrosion finish of organic base material displays, contrary to expectation, a high radiation resistance, so that it can be used instead of electrodeposited coatings.

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17-36

RADIOLOGICAL CONSEQUENCES OF SHIP COLLISIONS THAT MIGHT OCCUR IN U.S. PORTS DURING THE SHIPMENT OF FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL TO THE U.S. IN BREAK-BULK FREIGHTERS*

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ABSTRACT
Accident source terms, source term probabilities, consequences, and risks are developed for ship collisions that might occur in U.S. ports during the shipment of spent fuel from foreign research reactors to the United States.
INTRODUCTION
In support of U.S. nuclear weapons nonproliferation policies, Foreign Research Reactor spent nuclear fuel may be shipped to the U.S. in break-bulk freighters. The risks associated with such shipments were examined qualitatively by an Environmental Assessment (EA) that was published in 1994 (1). Accidents that occur in ports and thus near population centers were the principal contributors to accident risks. Because EA results were obtained using source term probabilities that did not decrease as the severity of the postulated accidents increased, the EA may have significantly overestimated the risks posed by severe accidents that occur in ports. Therefore, to support the preparation of the Foreign Research Reactor Spent Fuel Environmental Impact Statement (EIS), event trees were constructed that allowed the decreasing probability of increasingly severe ship accidents to be estimated. Next, source terms were developed for those very improbable long-duration, high-temperature, fire scenarios that might lead to melting of research reactor aluminum-uranium alloy fuels, or the burning, if exposed to air, of TRIGA research reactor fuel. Finally, the radiological consequences that might result from these improbable accidents were estimated using the MACCS site-specific accident consequence code.
EVENT TREES
Figure 1 presents an event tree that depicts the course and probability of a severe ship collision that leads to the release of radioactivity from a radioactive material (RAM) cask. Because ship collisions are not high-speed events, impact forces are not large enough to damage the cask or its contents. Therefore, release occurs only if the cask is subjected to crush forces. Crush forces are most likely to cause seal failure. If only seal failure occurs, release must occur by diffusion and consequently will be small. However, if the collision leads to an engulfing fire, expansion of cask gases will transport radioactive aerosols and vapors from the cask. Finally, if an unusual configuration of collapsing ship structures occurs that leads not only to seal failure, but also to a second failure due to shearing or puncture, then differential heating by the fire could induce convective flow through the cask and thus transport to the environment of all materials released from the fuel to the cask interior.
Consistent with past EAs (1,2), accidents that lead only to seal failure are placed in Severity Category 4, accidents that lead to seal failure and a long-duration engulfing fire are placed in Severity Category 5, and accidents that add shear or puncture failures to Severity Category 5 conditions are placed in Severity Category 6.
In Fig.1, the ship collision probability per port call was developed from accident data (3-6). The probability that the hold that is struck contains the cask is 1/7 as the prototypic break-bulk freighter (3) used in this analysis has seven holds. The probability that the cask is subjected to crush forces was estimated using the Minorsky method (7) and distributions of ship displacements, speeds, and collision angles developed by ORI, Inc. (3). The probability that a ship collision leads to a long-duration fire that engulfs the failed RAM cask was estimated from ship fire data (4-6,8,9) and from the number of locations (holds and decks within holds) where a fire might occur. The probability that an engulfing fire would heat the cask to temperatures above 900 K where aluminum-uranium alloy fuels melt and, if exposed to air, TRIGA fuels burn, was developed by constructing an oxygen availability event tree. The probability that the collision leads not only to seal failure but also to a second failure due to shearing or puncture was estimated to be no larger than 0.1.
Oxygen Availability Event Tree
Figure 2 presents the oxygen availability event tree. The probability that the hold is open to the air when the fire occurs was estimated from the fraction of port-call

time spent loading or unloading the ship, and from the fraction of time that a hold is open while the ship is loading or unloading. Ship data (10) suggest that the chance that the break-bulk freighter will be equipped with CO2 fire suppression systems is about 0.6. The chance that these systems fail on demand was conservatively estimated at 1 time in 5. Thus, the probability that an engulfing fire will heat a cask and its contents to 900 K was estimated to be 0.087, which was rounded up to 0.1 in Fig. 1.

Impact Forces

Modeling of RAM cask drop tests (11) indicates that 100 g accelerations may sometimes damage fuel rods contained in the cask. For ship collisions, the acceleration experienced by cask contents is the time derivative of the transverse speed of the struck ship. The acceleration varies with the distance of penetration of the striking ship prow into the hull of the struck ship. Average acceleration is obtained by integrating the instantaneous acceleration over the penetration distance and dividing by that distance. Average accelerations were calculated for a range of striking ship displacements (5 - 75 kilotonnes) and speeds (4 - 10 m/s), and four cargo loading cases: no additional cargo, light cargo shipped in cartons (packing fraction = 0.6, crush strength = 6.9 MPa); medium cargo shipped in wooden crates (packing fraction = 0.2, crush strength = 34.5 MPa); and heavy cargo, for example machinery (packing fraction = 0.5, crush strength = 1500 MPa). The calculations showed that cask contents would be subjected to average accelerations no larger than 0.2 g, which suggests that damage to cask contents or cask failure will not be caused by ship collision impact forces.

Crush Forces

By applying conservation of momentum and energy to ship collisions, Minorsky derived a set of equations that express the distance that the prow of the striking ship penetrates the hull of the struck ship in terms of the displacements of the two ships, the mass of water that moves with the ships during the collision, the speed of the striking ship, and the collision angle (7). Using collision data for nine severe accidents, Minorsky showed that the energy expended penetrating the struck ship to a given distance was a linear function of the sum of the volumes of the deformations in important deformed structures (hulls, decks, bulkheads).

Compaction of cargo around the RAM cask due to collapse of ship structures will subject a cask to crush forces only after all of the empty space in the hold has been used up by the penetration of the striking ship prow into the hold in which the cask is stowed. Although collision speeds are not high (typically less than 25 knots, much less in ports), because collision masses are very large, ship collisions are always high kinetic energy events. Therefore, if a collision subjects a cask to crush forces, it is certain that the cask will be damaged.

The probability that a ship collision will subject a cask to crush forces due to compression of cargo around the RAM cask was examined using the three cargo cases described above. If the RAM cask is assumed to be stowed on the ship centerline, cargo compression always causes the hold to go material solid before the prow of the striking ship penetrates to the centerline of the struck ship. Material solid conditions were predicted to occur most often for the medium cargo case. Figure 3 presents a histogram of the penetration distances calculated for this case. The figure shows that the hull of the struck ship is not penetrated about half the time. By comparing penetration distance to the critical penetration distance needed to cause the hold to go material solid due to cargo compression about the RAM cask, the probability that the collision will subject the cask to crush forces was estimated to be about 0.25 for the light cargo case, 0.4 for the medium cargo case, and 0.25 for the heavy cargo case. Thus, 0.4 is a reasonable estimate for the probability that a ship collision involving the RAM hold will subject the RAM cask to crush forces.

Thus, the impact and crush calculations suggest 1) that impact forces are too small to be of concern, 2) that crush forces are applied to the spent fuel cask when the collision uses up all of the empty space in the hold by collapsing and compressing the other cargo stowed in the hold with the cask, 3) that the very large kinetic energies that typify ship collisions ensure that the cask will be damaged if it is subjected to crush forces, and 4) that the likelihood of crush can be substantially decreased by restricting the amounts of cargo stowed with the cask.

CASK INVENTORIES AND RELEASE FRACTIONS

This study examined four spent fuels: the fuels used in the BR-2, RHF, OSIRIS, and

TRIGA research reactors. ORIGEN calculations provided inventories at three years after reactor shutdown. Screening calculations showed that 34 of the ORIGEN nuclides accounted for at least 99 percent of the dose that would result from the full set of ORIGEN nuclides. Cask inventories for these 34 nuclides were developed assuming that each spent fuel would be shipped in a Pegase cask (12). Table I presents the four inventories used in the study.

Three of the fuels examined, BR-2, RHF, and OSIRIS, are aluminum-uranium (Al-U) alloy fuels that melt at temperatures above 900 K (13). The fourth fuel, the TRIGA fuel, contains zirconium hydride which burns spontaneously and exothermically if exposed to air while at temperatures above 900 K (14). Because melting or combustion will greatly increase radioactive release, two sets of release fractions had to be developed for Severity Category 5 and 6 accidents, one for fires that do not heat the RAM cask and its contents to 900 K (Severity Categories 5A and 6A), and one for accidents that do (Severity Categories 5B and 6B). Table II presents the release fractions used in this study, summarizes in footnotes the basis for the fractions, and compares the fractions to those used in the Foreign Research Reactor EA (1).

CONSEQUENCES AND RISKS

Accident consequences were estimated using the site-specific consequence code, MACCS (16,17). The MACCS code was used because port accidents are most likely to occur at specific locations, either dockside or at some unfavorable location in the port channel. Use of MACCS also allowed the effects of nonuniform population distributions and of variable rather than constant weather conditions to be examined.

Footnotes to Table II

a. EA Category 4 release fractions are those developed by Wilmot (15) for release from power reactor UO2 pellets clad in zircaloy when the cask that contains the fuel is subjected to a severe impact. Wilmot assumes that the impact fails 10% of the fuel rods, that release fractions from the rod to the cask cavity are 0.2 for Kr and 2E-6 for fuel mass released as fuel fines, and that release fractions from the cask cavity to the environment are 0.6 for gases and 0.05 for fuel fines (particulate). Thus, release fractions from the fuel to the environment for commercial pelletized UO2 fuel are 1E-2 for Kr and 1E-8 = 0.1 x 2E-6 x 0.05 for all radioactivity contained in fuel fines (i.e., Cs, Ru, and Particulate).

b. Because TRIGA fuel is fabricated from several large U-Zr-H pellets that are clad in stainless steel, as long as the fuel is not exposed to air while at temperatures above its ignition temperature, release fractions should be quite similar to those used in the EA that were developed by Wilmot for power reactor UO2 fuels. Even if exposed to air while above its ignition temperature, release to the environment will be significantly increased only if there is an efficient mechanism for transport from the cask to the environment of fission products released to the cask by fuel oxidation. Thus, release fractions for TRIGA fuel for Category 5B conditions should be similar to those assumed for Category 5A conditions.

c. Because Al-U metallic fuels are stacks of plates or sets of nested cylinders, most fuel bundles are likely to fail during a ship collision that causes 5% of the cask volume to be lost by cask compression due to crush. Therefore, all of the bundles here are assumed to fail. However, because Al-U fuel is constructed from metal plates, the fraction of fuel present as fines should be substantially less than the fraction of UO2 pelletized fuel that is in fines. If the fraction of Al-U fuel that is present as fines is 10% of the fraction of UO2 pelletized fuel that is in fuel fines, the release fraction for fuel fines for Al-U metallic fuels should be quite similar to the value developed by Wilmot for UO2 fuel fines. Accordingly, the release fraction for fuel mass from plates to the cask cavity for Al-U fuel is conservatively taken to be 2E-7 = 1.0 x (0.1 x 2E-6).

d. Because diffusion in metal plates is negligible, until the metal fuel plates melt, only Kr trapped in fuel fines that become suspended in the cask interior volume will be available for release to the environment. Thus, the release fraction for Kr is the same as that for Cs, Ru, and Particulate, which are all released only as components of fuel fines.

e. Release fractions (FR) for Al-U fuel released as particles for Severity Categories 5A and 5B are calculated using the following equation:

$$FR = FBFFC[1.0(FCE) + (1.0 - FCE)(1 - RT)]$$

where

FB = fraction of fuel bundles that fail

FFC = fraction Al-U plate mass released to cask cavity as fuel fines
FCE = fraction of fuel fines released from cask cavity to environment
RT = TA/TP
TA = temperature of cask gases before the fire (ambient temperature)
TP = peak temperature reached by cask gases due to fire.
Because TA/TP = VA/VP = fraction of gas left in the cask after expansion due to heating from TA to TP, 1 - RT = the fraction of the cask gases that are expelled from the cask due to heating from TA to TP. For Al-U fuel, the release fractions given in the table for Categories 5A and 5B were calculated using the following parameter values: FB = 1.0, FFC = 2E-7, FCE = 0.05, TA = 300 K, TP = 600 K for Category 5A where 600 K is the midpoint of the softening range for carbon steels, and TP = 1023 K for Category 5B where 1023 K is 100 K above the melting point of Al-U alloy fuel.
f. Once fuel plates have melted, release fractions for Kr and Cs from Al-U alloy fuels are calculated as follows:

$$FR = FMC(1 - RT)$$

where
FMC = release fraction for Kr or Cs from melt to cavity
RT = TM/TP
TM = melting point of Al-U alloy fuel
TP = peak temperature reached by cask gases due to fire.
For Category 5B releases for Kr and Cs for Al-U alloy fuels, the following values were used: TM = 923 K, TP = 1023 K, FMC = 1.0 for Kr, and FMC = 0.1 for Cs.
g. For Category 6, all materials released to the cask cavity are released to the environment because convective flow through the cask is assumed to be considerable. Therefore, for Category 6, FR = FFC where FFC = the release fraction from fuel to the cavity for Category 6 conditions. The following values for FFC were assumed: For Al-U alloy fuels, 2E-7 for Kr, Cs, Ru, and Particulate for Category 6A; and for Category 6B, 1.0 for Kr, 0.1 for Cs, and 2E-6 for Ru and Particulate since melting of Al-U alloy fuel is not expected to be violent. For U-Zr-H TRIGA fuel, because it burns at temperatures significantly above 900 K, for Category 6B, 1.0 for Kr, 0.3 for Cs and also for Ru due to oxidation to volatile RuO_4, and 0.01 for particulate on the assumption that burning of the fuel will aerosolize 1 % of the fuel mass.
Given an inventory, release fractions for the accident severity category being examined, a population distribution constructed on a compass sector polar-coordinate grid out to some specific radial distance (here 50 miles), one year of meteorological data (hourly readings of wind speed, atmospheric stability, and accumulated precipitation) recorded in the site region, and a site wind rose, MACCS calculates plume dispersion and deposition; individual and population radiation exposures for the cloudshine, inhalation, groundshine, resuspension, and ingestion exposure pathways; acute injuries (e.g., prodromal vomiting) and fatalities; cancer fatalities due to radiation induced cancers; and accident costs caused by emergency response actions, and by the decontamination, temporary interdiction, or condemnation of contaminated property.
Each MACCS calculation performed develops consequence estimates for about 150 weather sequences selected by an importance sampling routine. Each sequence is examined using each of the sixteen compass sectors in the population distribution (plumes wider than a compass sector spill over into the two neighboring sectors). Thus, about 2400 trials are performed during each MACCS calculation. The product of the accident severity category probability, the weather sequence probability, and the population sector probability (the inverse of the wind rose) gives the probability of the predicted consequences for a given trial. By cumulating the trials, a Complementary Cumulative Distribution Function (CCDF) is constructed that depicts the probability that a consequence of a given magnitude will be equaled or exceeded.
Figure 4 presents CCDFs for cancer fatalities for the channel accident location at each of the 12 representative ports of entry examined. Each of these calculations used the BR-2 inventory, the largest of the four inventories studied, and Severity Category 5A release fractions. Category 5A results are presented because this is the more likely large accident, as fires that reach 900 K and accidents that lead to shear or puncture failures are problematic.
The probabilities displayed in Fig. 4 are conditional on the occurrence of a Severity Category 5A accident, that is they reflect the wind direction probability

and the weather sequence probability, but not the accident severity category probability which is 4.6 x 10-9 per port call for this severity category. The leftmost CCDF in Fig. 4 was obtained for the port with the smallest surrounding population (265,000) out to a distance of 50 miles. The rightmost CCDF in the figure was obtained for the port with the largest surrounding population (1,672,000) out to a distance of 50 miles.

Table III presents a full set of results at the dock of the hypothetical port of entry having the largest surrounding population. The calculation for this port used the BR-2 inventory and Severity Category 5A release fractions. These results are typical of all of the calculations performed. Table IV presents expected (mean) cancer fatality results for the 0 to 50-mile distance range for the BR-2 and TRIGA inventories for all five accident severity categories (Categories 4, 5A, 5B, 6A, and 6B). The expected cancer fatality results are conditional on the occurrence of an accident representative of the indicated severity category. The table also presents the probability of occurrence per port call of each severity category and the absolute cancer fatality risks that are obtained by multiplying the conditional expected result for each severity category by that category's probability of occurrence.

In summary, the MACCS calculations showed that:

Similar consequence predictions are obtained using two quite different types of weather data (a) constant weather conditions measured at the port, and (b) variable weather data measured at the National Weather Service station nearest to the port.

No acute health effects (e.g., prodromal vomiting) are predicted for the few quite improbable accidents that release the larger source terms studied, even when the accident takes place in a port located in a highly populated metropolitan area.

Total population dose is almost entirely due to long-term groundshine exposures.

The expected (mean) number of cancer deaths that might occur several decades after the accident due to the radiation exposures are small compared to the number of cancer deaths (200 per 100,000) normally expected to occur in the exposed population (18).

Finally, for the 12 hypothetical ports studied, the cancer death risk per port call ranged from 1.0 x 10-10 for the hypothetical low-population port to 1.7 x 10-8 for the hypothetical high-population port. If the high-population port is the destination for all shipments and each shipment makes two intermediate stops at high-population ports, then the cancer death risk for the entire shipping campaign would be about

2.9 x 10-5.

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STUDY OF EVACUATION TIMES BASED ON RECENT ACCIDENT HISTORY

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ABSTRACT

A key parameter in the calculation of accident dose-risks by the RADTRAN 4 code is the time assigned for evacuation of the affected area surrounding the accident. Currently, in the interest of assured conservatism, this time is set at 24 hrs. Casual anecdotal evidence has indicated that this value is overly conservative and results in assignment of overly conservative estimates of accident dose-risk. Therefore, a survey of recent truck accidents involving various hazardous materials which required evacuation of surrounding populations reported in various news media was undertaken. Accounts of pertinent scenarios were gleaned from databases citing newspapers and other periodicals, and the local authorities involved in each were contacted to get details of the evacuation including time required.
This paper presents the data obtained in the study and the resultant mean evacuation time plus limits and factors influencing specific results together with conclusions regarding the appropriate value to be used in the RADTRAN 4 code.

INTRODUCTION

The RADTRAN 4 computer code, which calculates estimates of accident dose-risk corresponding to specified transportation scenarios, ascribes doses to potentially exposed members of the public. These persons are modeled as not being evacuated from the affected area for 24 hours following a release of radioactive material. Anecdotal evidence has suggested that this value may be unnecessarily conservative; consequently risk estimates are unnecessarily high. A survey of recent trucking accidents, reported in newspapers and other periodicals (1988 through 1994), that involved evacuation of the general population in the affected areas was undertaken to establish the actual time required for such evacuations. Accidents involving hazardous materials other than those which are radioactive (e.g., gasoline, insecticides, other chemicals) but also requiring evacuations of nearby residents were included in the survey in order to obtain a statistically significant sample size. A total of 25 references (including some duplications) to incidents in the United States was found; approximately half of them could be sufficiently documented for inclusion in the analysis of evacuation time reported here.

DATA COLLECTION

An initial collection of abstracts was obtained from searches of a computer database, available through CompuServe, which included abstracts of newspapers and periodicals published in the U.S. and internationally. From this group, a subset of 25 abstracts was identified which met the criteria of the study: evacuation of the general public from areas surrounding truck accidents involving hazardous materials. Using the information included with the abstracts, local authorities were contacted

to obtain details of each incident including the amount of time required, the number of people, the size of the area, and other details as they were volunteered. In some cases, no record of the evacuation time was available, but verbal accounts were obtained from involved agency personnel to corroborate or correct the press accounts. Generally, it was found to be essential to verify or correct press accounts of incidents by contacting local authorities, e.g., the number of people evacuated was often a factor of 2 greater in press reports than the number given by authorities. Also, the data included in official accident reports do not consistently include details of the evacuation, e.g., highway accident reports primarily provide information on traffic/roadway details, injuries/fatalities, and responding personnel. The final list of incidents analyzed qualitatively and statistically in this study consisted of 13 accidents for which data judged reliable were obtained. Table I lists these incidents, pertinent data obtained, and derived parameters.

Population divided by the area defined by Evacuation Radius.

In the course of discussions with local authorities, information relating to non-truck accident emergencies requiring evacuations was obtained. All except one occurred at a fixed location such as a chemical plant; the exception was a train accident in Mississauga, Ontario, Canada. These additional cases are tabulated in Table II.

ANALYSIS

Inspection of the data revealed that the time required for an evacuation was influenced by population density, location of the population (e.g., residential, business, industrial, school) and perception of the urgency by the affected public. This can be seen from a comparison of Case A1 (suburban setting) with Case Q (rural setting) listed in Table I or comparison of Case A1 with Case --, in which 3200 of the 5200 evacuees were students at four schools. In addition, some evacuations occurred in stages spaced over times which were longer than the actual time required to evacuate groups once the decision to clear an additional area was made. For this initial evaluation of evacuation data, total number of people and total time were tabulated; differences in details of the evacuations were ignored.

Averaging the evacuation times in Table I yielded a value of 2.0 hours with a standard deviation of 2.6 hours; including the values in Table II changes these values to 3.9 6.8 hours. If the exceptionally large evacuation of Mississauga, Ontario, Canada (250000 people) is excluded, the average is 2.8 hours with a standard deviation of 5.0 hours.

A histogram of the evacuation times listed in Table I is plotted in Fig. 1. Adding the times from Table II yielded a histogram which is not significantly different in shape over the region where they overlap. The general shape of these plots suggests a lognormal distribution, as may be seen from Fig. 2 which displays normalized (at 10 hours) graphs of the two histograms and an approximate fit of a lognormal distribution. The roughness of the fit is attributed to the sparse data in the two histograms; more precise definition of the actual distribution of evacuation times using a larger sample of data would make a more precise fit feasible.

CONCLUSIONS

Three conclusions were drawn from this preliminary study:

1. Adding data describing evacuations resulting from incidents at fixed sites to trucking accidents does not significantly alter the distribution of evacuation times.
2. Average evacuation times derived from the data obtained to date are substantially shorter than the value of 24 hours currently used in RADTRAN 4 analysis.
3. The longest evacuation time (23 hours) obtained indicates that the 24-hour value is in fact a properly selected conservative value.

In addition, since the histograms in Fig. 1 do not suggest normally distributed evacuation times, a better definition of the distribution, allowing more precise statistical inferences to be drawn, is needed. Calculation of risk estimates based on values of evacuation time randomly sampled from such a distribution would provide an improved representation of accident-related risks. Since fixed-site and rail-accident data appear to have the same type of distribution, a broader search of news abstracts, including evacuations for incidents other than truck accidents, will be undertaken to increase the statistical significance of the derived distribution of evacuation times.

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RADTRAN 4 TRUCK ACCIDENT-RISK SENSITIVITY ANALYSIS

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ABSTRACT

The sensitivity of calculated dose estimates to various RADTRAN 4 inputs is an available output for incident-free analysis because the defining equations are linear and sensitivity to each variable can be calculated in closed mathematical form. However, the necessary linearity is not characteristic of the equations used in calculation of accident dose risk, making a similar tabulation of sensitivity for RADTRAN 4 accident analysis impossible. Therefore, a study of sensitivity of accident risk results to variation of input parameters was performed using representative routes, isotopic inventories, and packagings. It was determined that, of the approximately two dozen RADTRAN 4 input parameters pertinent to accident analysis, only a subset of five or six has significant influence on typical analyses or is subject to random uncertainties. These five or six variables were selected as candidates for Latin Hypercube Sampling applications.

To make the effect of input uncertainties on calculated accident risk more explicit, distributions and limits were determined for two variables which had approximately proportional effects on calculated doses: Pasquill Category probability (PSPROB) and link population density (LPOPD). These distributions and limits were used as input parameters to Sandia's Latin Hypercube Sampling code to generate 50 sets of RADTRAN 4 input parameters used together with point estimates of other necessary inputs to calculate 50 observations of estimated accident dose risk.

Tabulations of the RADTRAN 4 accident risk input variables and their influence on output plus illustrative examples of the LHS calculations, for truck transport situations that are typical of past experience, will be presented.

INTRODUCTION

The transportation risk analysis code, RADTRAN 4 [Neuhauser, 1992], computes estimates of incident-free dose consequence and accident dose risk. The output of the code can include a tabulation of sensitivity of the result to variation of the input parameters for the incident-free analysis. The values are calculated using closed mathematical expressions derived from the constitutive equations, which are linear. However, the equations for accident risk are not linear, in general, and a similar tabulation has not been available. It is, nevertheless, important to know how the accident risk is affected by uncertainties in the input parameters. A direct investigation of the variation in calculated accident dose-risk with changes of individual parameters was undertaken. Initially, a limited, representative group of transportation scenarios was used to determine which of 23 accident-risk input parameters or arrays affect the calculated accident dose risk significantly. Many of the parameters were observed to have minimal effect on the output while others were judged as "fixed" either by regulation, convention or standards. The remaining 5 input arrays were selected for further study through Latin Hypercube Sampling (LHS) (1). LHS yields statistical information from observations (risk calculations) resulting from multiple input parameter sets compiled from "random" sampling of cumulative probability distributions. The LHS method requires fewer observations than classical Monte Carlo methods to yield statistically significant results. This paper presents the preliminary input parameter studies and their results together with some initial LHS investigations.

ANALYSIS

A list of the RADTRAN 4 input parameters and arrays employed in accident-risk calculations was compiled and is presented in Table I. RADTRAN calculations of accident risk were performed for transportation scenarios drawn from actual experience or special cases designed to emphasize a parameter of interest (e.g., non-dispersal accidents). The baseline values used in this study were adapted from archived input files supporting published Sandia analyses, e.g., shipment of spent nuclear fuel by highway from Norfolk, VA to the Savannah River Site, SC. Actual input files used in the examples presented here are listed in the Appendix. Either eight-category or six-category accident-severity schemes are typically employed in RADTRAN analyses depending on the specific isotopes being shipped and the type of

packaging used. Thus, the SEVFRC (all parameter names are defined in Table I) arrays in the test files have six or eight baseline values for each population density category (rural, suburban, urban). The RFRAC, AERSOL, and RESP arrays have six- or eight-element arrays defined for each physical-chemical group (only 1 in the examples listed in the Appendix) required to describe the materials being shipped. Relationships between various possible severity schemes are discussed in Whitlow and Neuhauser (4,2).

The results of discrete calculations, in which values other than the RADTRAN-supplied value for a particular input parameter were tried are listed in Table II. The "Results" are values of the "Total" accident dose-risk calculated by RADTRAN 4. Although some of the input variables listed had a larger effect on component risks such as the "Urban", "Inhaled" and "Resuspended" dose-risks, the "Total" risk is the value most often quoted in applications of RADTRAN 4 analyses. Therefore, it is the basis for sensitivity evaluations reported here. Current practice in reporting results has been to tabulate values to two significant digits, in keeping with the uncertainties of the analysis.

Discussion of each of the five input variables and the test calculation results listed in Table II follows:

BRATE (breathing rate) -- The suggested point estimate is based on the maximum rate for humans and a reduction to half of that value did not change the risk estimate.

BDF (fraction of respirable aerosol inside buildings) -- Changes of a factor of 10 up and down yielded the same results as the point estimate.

RPD (ratio of pedestrian to resident population density) -- An increase by a factor of two resulted in the slightest reportable change in the result: 2.7E-08 to 2.8E-08.

RU (Urban building shielding factor) -- As noted in Table I, it only affects accident risk results in cases of non-dispersal accidents. The calculated change in risk (3.0E-11 to 3.1E-11) occurred for an increase of RU by a factor of 10.

CULVL (cleanup level) -- This has a significant effect on calculated risk upon being lowered by a factor of 10, but such a change in regulations is not presently known or expected.

In summary, none of these five input parameters was included in subsequent LHS analysis since, for reasonable distributions of their values, their effects on risk calculations would be hidden by the effects of other input parameters or arrays. Some of the variables listed in Table I were not subjected to these tests because they were viewed as not being subject to random uncertainties, or clearly would affect output significantly (proportionally). The list of input parameters and arrays from Table I and the sensitivities of total risk to them, are presented in Table III together with an indication of which (five input arrays) should be studied further with LHS.

To date, two of the input arrays identified in Table III as suitable for LHS analysis have been investigated through use of LHS in conjunction with RADTRAN calculations: Pasquill Category Weights (PSPROB) and Link Population Densities (LPOPD). These inputs were selected for initial study because justifiable distributions of their values could be estimated easily and because LPOPD and PSPROB array sizes are not affected by the choice of accident-severity scheme. Accident-severity and related variables are to be addressed in the future.

Use of LHS requires definition of probability distributions for each variable of interest. In the case of the six Pasquill Stability Class variables (PSPROB array), a uniform distribution between 0.0 and 1.0 for each of the six classes was sampled independently and the sum of each set of six samples was normalized to 1.0, as required by RADTRAN 4. For the link population densities (LPOPD values) normal distributions centered on the point estimates for each of the links describing a

particular route were used. The widths of the normal distributions were chosen such that s (standard deviation parameter of the normal distribution function), was 10% of the mean for one set of calculations and 25% of the mean for a second set of calculations.
Table IV summarizes the results from 50 observations (RADTRAN accident-risk calculations) employing 50 samples from either the PSPROB or one of the LPOPD distributions in the input files. Also, Table IV includes results from calculations employing 50 samples from the PSPROB and one of the LPOPD distributions, together, in the input files. The risk value calculated for the same transportation scenario without LHS, 5.32E-05 person-rem, is higher than values for the three cases in which PSPROB values were varied. This is because the default set of PSPROB values in RADTRAN leads to a more severe dispersion situation than the "random" mix of all six conditions. When each Pasquill category was applied individually, the results varied by more than a factor of 10.
CONCLUSIONS
The rather long list of input parameters and arrays required to calculate accident dose-risk estimates with RADTRAN 4 has been pared down to five arrays that have proportional effects on the estimate of total risk. A few additional input variables have proportional effects on component risks such as risk in urban areas or the respirable component, but negligible effect on the total risk. Use of Latin Hypercube Sampling on two of the input arrays, PSPROB (6 array elements) and LPOPD, revealed that quite reasonable selections of distribution type and limits yield risk estimates and uncertainties which are in accord with expectation based on the input array means and the randomness of the input values. In the case of Pasquill atmospheric stability classes, random mixes of the 6 class weights result in lower risk estimates than those calculated with the RADTRAN 4 default set. Treating link population densities (LPOPD's) derived from the HIGHWAY (3), 1992] routing code as normally distributed variables with 25% standard deviation led to risk estimates having an average value equal to the risk estimate calculated without LHS; the standard deviation of the 50 risk estimates was 11%.
Further studies of the rest of the input variables to which accident risk estimates were found to have proportional sensitivity must be investigated further, individually and in combinations, to determine the range of uncertainty to be expected in typical RADTRAN 4 accident dose risk calculations.
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APPENDIX
RADTRAN 4 input file listings.
Sample of RADTRAN input files used for evaluation of sensitivity to BRATE, BDF, RPD and CULVL:

```
         && _TRUCK_CASE_NORFOLK_TO_CHARLESTON_
         FORM UNIT
         DIMEN 1 6 1 10 18
         PARM 1 3 2 1 0
         PACKAGE
            LABGRP
              STUFF
         SHIPMENT
            LABISO
               CO60
         NORMAL
```

```
      NMODE=1
         1.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2.000E+00 1.000E+01 0.000E+00 1.100E-02 0.000E+00 0.000E+00
         0.000E+00 5.000E+01 2.000E+01 0.000E+00 0.000E+00 0.000E+00
         2.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         0.000E+00
   ACCIDENT
      SEVFRC
         NPOP=1
          NMODE=1
          6.23E-01 3.74E-01 3.00E-03 3.00E-06 5.00E-06 7.00E-06
       NPOP=2
         NMODE=1
         6.22E-01 3.74E-01 4.00E-03 4.00E-06 3.00E-06 2.00E-06
       NPOP=3
         NMODE=1
          6.24E-01 3.75E-01 3.80E-04 3.80E-07 2.50E-07 1.30E-07
   RELEASE
      RFRAC
         GROUP=1
          0.00E+00 0.00E+00 2.50E-03 5.00E-02 7.50E-01 1.00E+00
      AERSOL
         DISP=5
          0.00E+00 0.00E+00 0.00E+00 1.00E+00 1.00E+00 1.00E+00
      RESP
         DISP=5
          0.00E+00 0.00E+00 0.00E+00 5.00E-02 5.00E-02 5.00E-02
   OTHER
      BDF 8.600E-04
      EOF
      ISOTOPES 1 1  1.00 0.100 1.00 0.00 SAMPLE
           CO60 1.00E+00  STUFF 5
      LINK 1 1.74E+01 8.86E+01 9.38E+00 4.70E+02 2.33E-07 R 1
      LINK 1 2.90E+01 8.86E+01 5.62E+02 7.80E+02 2.33E-07 S 1
      LINK 1 6.76E+00 4.80E+01 2.11E+03 2.80E+03 1.37E-07 U 1
      LINK 1 8.06E+01 8.86E+01 1.16E+01 4.70E+02 4.23E-07 R 2
      LINK 1 1.40E+01 8.86E+01 3.19E+02 7.80E+02 4.23E-07 S 2
      LINK 1 1.34E+01 8.86E+01 1.10E+01 4.70E+02 2.33E-07 R 1
      LINK 1 4.34E+00 8.86E+01 4.12E+02 7.80E+02 2.33E-07 S 1
      LINK 1 2.25E+02 8.86E+01 1.61E+01 4.70E+02 2.30E-07 R 1
      LINK 1 7.27E+01 8.86E+01 1.55E+02 7.80E+02 2.30E-07 S 1
      LINK 1 2.21E+02 8.86E+01 1.65E+01 4.70E+02 1.61E-07 R 1
      LINK 1 4.05E+01 8.86E+01 3.08E+02 7.80E+02 1.61E-07 S 1
      LINK 1 7.08E+00 4.80E+01 2.13E+03 2.80E+03 3.16E-07 U 1
      PKGSIZ
           SAMPLE  1.00
      EOF
- - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
- - - - - - - - Special RADTRAN 4 input file used to evaluate sensitivity to RU
(required an Urban, non-Interstate LINK *):
   FORM UNIT
   DIMEN 1 6 1 10 18
   PARM 1 3 2 1 0
   PACKAGE
      LABGRP
         STUFF
   SHIPMENT
      LABISO
         CO60
   NORMAL
      NMODE=1
         1.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
         2.000E+00 1.000E+01 0.000E+00 1.100E-02 0.000E+00 0.000E+00
```

```
            0.000E+00 5.000E+01 2.000E+01 0.000E+00 0.000E+00 0.000E+00
            2.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
            0.000E+00
         ACCIDENT
            SEVFRC
               NPOP=1
                NMODE=1
                6.23E-01 3.74E-01 3.00E-03 3.00E-06 5.00E-06 7.00E-06
               NPOP=2
                NMODE=1
                6.22E-01 3.74E-01 4.00E-03 4.00E-06 3.00E-06 2.00E-06
               NPOP=3
                NMODE=1
                6.24E-01 3.75E-01 3.80E-04 3.80E-07 2.50E-07 1.30E-07
         RELEASE
            RFRAC
               GROUP=1
                0.00E+00 0.00E+00 2.50E-03 5.00E-02 7.50E-01 1.00E+00
         EOF
         ISOTOPES 1 1  1.00 0.100 1.00 0.00 SAMPLE
                CO60 1.00E+00  STUFF 1
         LINK 1 1.00E+03 8.86E+01 2.70E+01 4.70E+02 2.33E-07 R 1
         LINK 1 1.00E+02 8.86E+01 6.69E+02 7.80E+02 2.33E-07 S 1
         LINK 1 1.00E+02 4.80E+01 1.93E+03 2.80E+03 1.37E-07 U 1
         LINK 1 1.00E+01 4.80E+01 1.93E+03 2.80E+03 7.40E-06 U 2 *
         PKGSIZ
                SAMPLE  1.00
         EOF
- - - --- - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
- - - - -  Base RADTRAN 4 input file used in LHS calculations:
         && _Shipped_by_Truck_from_SNL/NM_to_NTS_
         && _Package_Dimension_Approximates_a_55gal_Drum_
         TITLE _LABORATORY_WASTE_FROM_AREA_V _DRUM_
         FORM UNIT
         DIMEN 5 8 1 10 18
         PARM 1 3 2 1 1
         PACKAGE
             LABGRP
                   A
         SHIPMENT
             LABISO
               CS137  CS134  CE144  NB95  ZR95
         NORMAL
            NMODE=1
               1.000E+00 0.000E+00 0.000E+00 8.856E+01 4.032E+01 2.416E+01
               2.000E+00 6.760E+00 0.000E+00 1.100E-02 0.000E+00 0.000E+00
               0.000E+00 5.000E+01 2.000E+01 0.000E+00 1.000E+02 1.000E+02
               2.000E+00 0.000E+00 0.000E+00 1.000E+00 4.700E+02 7.800E+02
               2.800E+03
ACCIDENT
            SEVFRC
              NPOP=1
               NMODE=1
               4.62E-01 3.02E-01 1.76E-01 4.03E-02 1.18E-02 6.47E-03
               5.71E-04 1.13E-04
              NPOP=2
               NMODE=1
                4.35E-01 2.85E-01 2.21E-01 5.06E-02 6.64E-03 1.74E-03
                6.72E-05 5.93E-06
              NPOP=3
               NMODE=1
                5.83E-01 3.82E-01 2.78E-02 6.36E-03 7.42E-04 1.46E-04
                1.13E-05 9.94E-07
```

```
RELEASE
  RFRAC
       GROUP=1
       0.00E+00 1.00E-02 1.00E-01 1.00E-01 1.00E+00 1.00E+00
       1.00E+00 1.00E+00
EOF
ISOTOPES -1 1  1.00 10.000 1.00 0.00 MTAP
       CS137 1.81E-01   A 7
       CS134 9.40E-03   A 7
       CE144 6.70E-02   A 7
        NB95 3.46E-02   A 7
        ZR95 1.90E-02   A 7
LINK 1 1.40E+00 2.42E+01 9.16E+02 7.80E+02 2.07E-07 S 2
LINK 1 1.80E+00 2.42E+01 2.68E+03 2.80E+03 7.40E-06 U 2
LINK 1 2.38E+02 8.86E+01 7.20E+00 4.70E+02 2.69E-07 R 1
LINK 1 2.01E+01 8.86E+01 4.98E+02 7.80E+02 2.69E-07 S 1
LINK 1 7.20E+00 8.86E+01 2.10E+03 2.80E+03 3.24E-07 U 1
LINK 1 4.74E+02 8.86E+01 1.70E+00 4.70E+02 2.60E-07 R 1
LINK 1 2.03E+01 8.86E+01 3.40E+02 7.80E+02 2.60E-07 S 1
LINK 1 1.60E+00 8.86E+01 2.14E+03 2.80E+03 2.81E-07 U 1
LINK 1 1.16E+02 8.86E+01 1.90E+00 4.70E+02 2.69E-07 R 2
LINK 1 1.17E+02 8.86E+01 3.00E+00 4.70E+02 4.51E-07 R 2
LINK 1 2.59E+01 8.86E+01 5.69E+02 7.80E+02 4.51E-07 S 2
LINK 1 1.47E+01 4.80E+01 2.46E+03 2.80E+03 4.51E-07 U 2
PKGSIZ
       MTAP  1.00
EOF
```

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ASSESSMENT OF RISKS TO INDIVIDUALS FROM THE TRANSPORTATION OF RADIOACTIVE MATERIALS*

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ABSTRACT

The radiological impacts to individuals from the transportation of radioactive materials must be assessed when evaluating alternatives for major federal actions as required by the National Environmental Policy Act. Public comments on past environmental impact statements indicate that the public is concerned about the risks of radiation exposure to individuals along a transport route from radioactive materials shipments. Individuals may be exposed during routine, incident-free, transport of radioactive materials or, potentially, as a result of transportation accidents. This paper discusses the computer model RISKIND, which was developed at Argonne National Laboratory to estimate the potential radiological risks to individuals and population subgroups from the transportation of radioactive materials. The code was designed to use site-specific data to provide a detailed analysis for each receptor location. This type of analysis complements the traditional collective-population transportation risk analyses conducted for radiological transportation risk assessments.

The RISKIND program features an easy to use WindowsTM interface with on-line help for all input parameters. The majority of these parameters are user-accessible to provide the flexibility needed to model a variety of distinct individual circumstances. In addition, a complete set of input data for spent nuclear fuel transportation risk assessments is included with provisions for assessing other radioactive material shipments. A detailed analysis of incident-free and accident risks is performed on the basis of site-specific input such as receptor location, weather, and agricultural data. RISKIND incorporates a number of advanced features, including cask size correction for external exposures, buoyant plume rise, terrain height adjustments, wet and dry deposition, plume depletion for atmospheric dispersion estimates, and acute and long-term models for estimating health risks. RISKIND can be used to estimate consequences to individuals under a given set of

accident and weather conditions, or the program can be used to perform a probabilistic risk assessment encompassing the entire spectrum of potential accident and weather conditions.

INTRODUCTION

As required by the National Environmental Policy Act (NEPA), the radiological consequences to individuals, in addition to the risks to the collective population, from transportation activities must be considered when evaluating major federal actions that could adversely impact the environment.

Individuals may be exposed during routine, incident-free, transport of radioactive materials or, potentially, as a result of transportation accidents. During routine transport, individuals in the vicinity of a radioactive materials transport vehicle receive an external exposure dose from the low-levels of radiation emanating from the shipping package. Should a transportation accident occur, individuals could be exposed, via a number of exposure pathways, to radiation from the material that might be released to the surrounding environment following an accident. Predicting exposure risks from accidents requires a complex model and detailed information on accident parameters. Such information should include data on accident severity, radioactive releases, probabilities of occurrence, weather conditions, and the relative proximity of the individual receptors to the release point.

Early NEPA documents used different approaches to assessing individual risk that were sometimes too simplistic, incomplete, or inconsistent with previous analyses. This paper discusses the computer model RISKIND (1,2) that was developed at Argonne National Laboratory (ANL) to provide a complete individual risk analysis that complements the traditional collective population risk analysis performed with RADTRAN (3).

A comprehensive transportation risk assessment can be performed using RISKIND's emphasis on site-specific individual scenarios and RADTRAN's more generic route-specific collective population characteristics. Several major environmental impact statements (EISs) (4,5) are presently using this approach.

RISKIND is a self-contained, highly flexible program that is able to model most location-specific scenarios. Features of the program that are discussed in subsequent sections are as follows:

- Scenario-specific ("what if") conditions involving individual risks;
- Easy to use graphical interface with on-line help;
- User-accessible input for most routine and accident model parameters;
- Waste package shape adjustment and shielding options;
- Standard datasets for spent nuclear fuel (SNF) analysis;
- Gaussian puff atmospheric dispersion model with waste package geometry considerations, thermal conditions (fire with buoyant plume rise), terrain height considerations, plume depletion, and wet and dry deposition;
- Site-specific population, weather, and agricultural productivity input; and
- Acute and long-term health risk models.

BACKGROUND

The RISKIND computer code was developed by ANL for the Office of Civilian and Radioactive Waste Management within the U.S. Department of Energy (DOE) to assess the radiological consequences and risks to individuals associated with the transportation of SNF (1). This development was in response to public comments on the NEPA documents concerning the repository environmental assessments (6,7,8,9). These comments expressed the public's concern over specific risks to individuals near the transportation route. The RISKIND program was designed to implement site-specific data to provide a detailed analysis for a wide variety of local situations.

The original release of RISKIND contained radionuclide inventories of SNF for commercial boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) (10) and release fractions for 20 accident response regions. The response regions (severity categories) were developed specifically for spent fuel casks in a Lawrence Livermore National Laboratory (LLNL)/U.S. Nuclear Regulatory Commission (NRC) study (11) referred to as the modal study. In response to user requests, the RISKIND code is being updated (RISKIND 1.00, forthcoming). The older DOS-based menu system has been replaced with a WindowsTM-based application that provides a user-friendly interface. Also, the spent fuel inventories have been updated (12), and users have the option of inputting their own radionuclide inventories and release fractions to assess the transportation risk associated with other radioactive materials.

USER INTERFACE

The new RISKIND user interface was designed for ease of use and relies on the point-and-click familiarity and graphics capabilities of the Windows™ 3.1 environment. The main user screen is shown in Fig. 1. The main screen provides for selecting the major input options, provides quick access to more detailed input and output options, and serves as a brief overview of the analysis to be performed. Once the problem to be analyzed is defined, the selections can be saved to an input file for later use, if desired, and the analysis can be performed. On-line help is also available to assist the user with input parameter descriptions.

If a large number of similar analyses are required, RISKIND can be run in batch mode under the DOS operating system. This mode of operation foregoes the Windows™ interface and uses previously generated input files.

INCIDENT-FREE RISKS

Individuals would be exposed to external radiation from the radioactive materials shipment under various incident-free (normal) conditions (Fig. 2). Pedestrians along the transport route would receive a dose from the passing shipment, whereas others may be exposed for longer periods of time, such as the truck crew and motorists beside the shipment in stop-and-go traffic, or service attendants at refueling stops and persons at rest stops. These examples are typical of the situations for which RISKIND can be used to assess the incident-free risks to individuals.

Routine analysis with RISKIND entails the assessment of specific gamma and neutron dose rates at various receptor distances. External exposures and health risks are determined by scaling dose rate curves, as a function of distance, on the basis of the user input values for the external dose rates of the shipping package at a given reference distance. Waste package geometry and shielding effects are taken into consideration.

The reference gamma and neutron dose rate curves in RISKIND are based on a three-dimensional Monte Carlo dose model. The model incorporates an energy spectrum representative of commercial spent fuel and considers air and ground scattering. An option may be selected to employ a user-supplied dose rate curve if the appropriate data are available for a nonstandard type of SNF or other radioactive material of interest.

Conventional numerical methods are used to estimate gamma and neutron doses to individuals, both from a passing shipment and for a fixed period of time and distance from the shipment. Up to 20 individual locations relative to the shipment route may be specified for a single analysis. Shielding options are also available. For subpopulations, doses from a passing shipment to groups along the route and sharing the route are estimated. Doses from a stationary shipment to subgroups at up to 10 locations may also be calculated in a single run.

ACCIDENT RISKS

The RISKIND code assesses the doses to individuals via a number of pathways from potential accidental releases of radioactive material during transportation (Fig. 3). Individual dose and health risks can be calculated with RISKIND for the median (50% probability) and highly unlikely (less than 5% probability of being exceeded) accident cases through consideration of the entire spectrum of potential accidents, the entire range of weather conditions, or both. These results provide a comprehensive probabilistic transportation risk assessment.

Radionuclide inventories may be selected from PWR or BWR SNF options, or they may be user-specified on a radionuclide-by-radionuclide basis if nonstandard SNF or other types of radioactive materials are shipped. The SNF inventories are extracted from the latest spent fuel database (12) and characterized by the amount of uranium present, the fuel burnup, and the fuel cooling time.

For risk analysis, default accident rates are available for population zones (rural, suburban, and urban) by state. The default accident model from the modal study contains 20 accident response regions (accident severity categories) categorized by the level of strain on the shipping cask (impact force) and thermal stress (fire considerations). Each region is assigned a conditional probability of occurrence for truck and rail transport in conjunction with release fractions of the radioactive inventory. The regions cover accidents with high probability and no release to accidents with very low probability and some release. Users may specify their own accident rates, conditional probabilities, and release fractions if the data are available for a custom analysis.

An accidental release of radioactive material would generally result in a plume of

radionuclides being discharged to the air; the material would be dispersed by atmospheric transport to various locations. RISKIND uses a Gaussian puff model to estimate dispersion of the plume and calculates the time-integrated ground-level air concentrations at each receptor location. Pasquill-Gifford dispersion coefficients (13) are used for releases near ground level, while Briggs dispersion coefficients (14) can be used for elevated releases involving fire and subsequent plume rise. The dispersion model in RISKIND accounts for buoyant plume rise should the accident scenario under consideration involve fire, for differences in elevation between the release point and receptor, for plume depletion, and for dry and wet deposition of radionuclides as the plume travels.

Dispersion of the plume is dependent on the prevailing weather conditions. For a consequence assessment where conditions are known, a given wind speed for a specific weather stability category is used to determine the plume dispersion and subsequent receptor exposure. For a risk assessment where conditions are unknown at the time of an accident, all possible combinations of stability categories, wind speeds, and joint frequency data are used. RISKIND now has the capability to read STability ARray (STAR) files containing the joint frequency data. Over 200 STAR files from the U.S. National Oceanic and Atmospheric Administration covering the 48 contiguous states are included in RISKIND. These files are a superset of the STAR data provided with the U.S Environmental Protection Agency CAP88-PC code (15). CAP88-PC is the code that DOE facilities are required to use to demonstrate compliance with radionuclide emission standards (16).

Exposure pathways considered are external exposure from the shipping package, inhalation from the passing plume, external exposure from the passing plume, external exposure from radionuclides deposited on the ground, inhalation of resuspended radioactivity, and ingestion of contaminated foodstuffs (Fig. 3). As was also the case for the incident-free analysis, shielding options are available with RISKIND, and up to 20 individual locations relative to a potential accident scene are possible in a single analysis. Location-specific weather and agricultural productivity data are used for the most relevant assessment. Results for population consequence assessments are provided on an isopleth-by-isopleth basis to better define the outcome.

An example of RISKIND's graphic output capabilities is a concentration isopleth contour plot (Fig. 4) that is available when performing a population consequence assessment. Such a plot gives the analyst a better perspective on the problem. Other plot options include cumulative dose distribution function curves, depending on the input options selected.

HEALTH RISKS

RISKIND estimates health risks in terms of chances of fatalities and genetic effects for both acute (short-term) and latent (long-term) exposures. Acute risks are estimated on the basis of the NRC health effects model (17). Latent risks are estimated on the basis of the Biological Effects of Ionizing Radiation (BEIR) V Report (18) and International Commission on Radiological Protection Publication (ICRP) 60 (19).

CONCLUSIONS

With an easy to use graphical interface, the RISKIND code provides a comprehensive transportation risk assessment for individual receptors near a radioactive shipment route. A complete set of input data for SNF transportation risk assessment is included with provisions for assessing other radioactive material shipments. A detailed analysis of incident-free and accident risks is performed on the basis of site-specific input data. This type of analysis complements the traditional collective population transportation risk assessment conducted for radioactive material shipments.

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THE GERMAN CASK CONCEPT FOR INTERMEDIATE AND FINAL STORAGE OF SPENT FUEL

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ABSTRACT
On the direct disposal of spent fuel elements development work has been done since 1979 based on a governmental resolution. In the period 1981 - 1984 the R&D program "Alternative disposal technologies", sponsored by the Ministry of Research and

Technology (BMFT), was established to study the technical feasibility and safety and to demonstrate, that direct disposal can be an alternative to reprocessing.
From 1985, the German nuclear industry continued to develop a final disposal cask and planned a pilot conditioning plant (PKA).
As a result of this work two containers have been developed:
the final disposal cask POLLUX (for final storage in drifts),
the final disposal canister (for final storage in bore holes).
These containers and the status of their development are described in more detail in the following report.

DESIGN REQUIREMENTS FOR THE FINAL DISPOSAL CASK POLLUX

The following requirements have to be met by a final disposal cask:
Safe enclosure of the radioactive substances
- after treatment in the pilot conditioning plant, during handling, transport and intermediate storage
- during the operating phase of the final repository
- after close down and sealing the final repository

Appropriate shielding
- during handling above ground
- during handling in the final repository

Transport and handling ability
Suitability for final disposal
- in drifts (reference procedure)
- in bore holes.

These requirements, the properties of the handled fuel and the specific data of the final repository led to the following planning conditions:
barrier resistance in accordance with the requirements for a final repository 500a

temperature in the final repository 200C
maximum rock salt pressure 300 bar
maximum mass of cask 65 Mg
type B(U) requirements in accordance with IAEA transport regulations
approval for storage of radioactive sub-stances in an intermediate storage facility.

DESIGN CONDITIONS--FINAL STORAGE CASK

A standard PWR fuel element was used as the design fuel element.
10 PWR fuel elements with a fuel equivalent of approx. 5.5 tHM were taken as maximum load for the final disposal cask.
PWR and BWR fuel elements are conservatively covered by this type of fuel element with respect to the radioactive inventory and the geometric fuel dimensions.
It was considered that the PWR fuel element could contain uranium or two types of MOX fuel, respectively.
The two selected Pu vectors were taken to be representative for MOX fuel elements to be expected in the future.
In accordance with the practiced mixed loading of nuclear reactors with approx. 2/3 uranium fuel elements and 1/3 MOX fuel elements, the same load variants were considered for the final disposal cask as envelope.

DESIGN CONDITIONS--FINAL DISPOSAL CANISTER

The final disposal canister is designed to be loaded with uncut fuel rods of 4 PWR or alternatively 12 BWR fuel assemblies.
The outer diameter of the final disposal canister is the same as for the glass canister for vitrified high-level wastes from fuel reprocessing. Thus, a joint handling and final disposal of both canister types in bore-holes are possible.

CONSTRUCTION OF THE FINAL DISPOSAL CASK

Fig. 1 shows the basic construction of the final disposal cask design. The cask consists of a shielding cask with a screwed on lid and an inner cask with a screw-fitted primary lid and a welded secondary lid.
The radioactive material is enclosed in canisters, the canisters itself are loaded into the final disposal cask.
Fig. 2 shows the design of the final disposal cask in more detail.

INNER CASK

The cylindrical shell and the bottom of the inner cask Fig. 3 are made from fine-grained construction steel 15 MnNi 6.3 pressed from a single piece without seams. The wall thickness of the cylindrical shell is designed to comply with the

mechanical and shielding requirements and has a thickness of approx. 160 mm. The weight of the complete inner cask is approx. 21 metric tonnes.
Primary lid The primary lid of the inner cask is made from the same materials as the inner cask. It provides for leak tightness and shielding before and during welding of the secondary lid.
Below the lid a plate is attached that is made of materials which moderate and absorb neutrons.
Secondary Lid
The secondary lid of the inner cask is also made from the same material as the inner cask (see Fig. 4). It is welded with the body. The welded seam joint, which is approx. 50 mm thick, is made using a special technique - the narrow-gap welding. It ensures leakproof and durable barriers against radioactivity releases during transport, storage and final disposal of the fuel elements.
Inner Structures
The inner cask has a basket structure. It consists of an inner square box connected by sheets at each corner with the cask and thus forming four cavities for canisters with consolidated fuel rods. The sheets center the basket structure and ensure good heat transfer from the center to the outer cask wall.
The central square position can be filled with the fuel rods of two PWR fuel elements or with compressed structural parts of fuel assemblies.
Fuel rod canisters with consolidated fuel can contain rods from two PWR fuel elements.
To guarantee subcriticality even in case of a rather hypothetical flooding of the final disposal cask, boronated steel plates are screwed at the outer side of the canisters. This measure assures neutron decoupling between the canisters.
After remote-controlled loading, the canisters and the central basket position are sealed dust tight with lids.
CORROSION PROTECTION
The complete inner cask can be coated to avoid corrosion, if the final storage conditions should require it.
SHIELDING CASK
Body
The body of the shielding cask is cast in a single piece from nodular cast iron GGG 40. It is designed to comply with the mechanical and shielding requirements and has a wall thickness of 265 mm. The weight of the body is approx. 34 metric tonnes.
There are two circles of drilled holes with moderating material in the wall of the shielding cask. Each one has 36 holes with a diameter of 75 mm each.
This arrangement of the moderating material inside the shielding wall corresponds to the previously tested concept used for the CASTOR casks-family.
The primary function of the shielding cask is to reduce the gamma and neutron dose rate at the surface, so as to comply with the envisaged limit for intermediate storage of 0.5 mSv/h.
Furthermore, in the final repository, the shielding cask has to withstand the isostatic salt pressure of approx. 300 bar.
The shielding cask lid is screwed down using multiple-threaded trapezoidal threads.
CONSTRUCTION OF THE FINAL DISPOSAL CANISTER
The final disposal canister is a cylindrical cask. Its design and components are shown in Fig. 5.
It consists of:
- a cylindrical canister piece made of fine grained construction steel;
- a welded lid plate with a grapple device and
- two holes for the loading and ventilation.

The holes are shut by a primary plug and ventilation valve after loading.
By welding the disposal canister after loading, the gas-tight containment of the radioactive material during transport, storage and final disposal is ensured.
An anti-corrosion coating of the canister offers further protection against corrosion inside the repository.
The loading of the final storage canister with fuel pins from spent LWR-assemblies will be done in the PKA using hot cell technology. After completion, the canisters are inserted into a transport/storage cask.
RESULTS OF THE DESIGN CALCULATIONS--CALCULATED DOSE RATES
Based on the shielding geometry and burnup calculations using the computer code OREST, shielding calculations were performed for the three types of fuel under

consideration with a mean burnup of 55 GWd/tHM. The calculated dose rates at the cask surface are in compliance with the envisaged value for intermediate storage of 0.5 mSv/h.

CRITICALITY CALCULATIONS

The final disposal cask is designed in such a way that even during unfavorable operating events and assumed accidents, the neutron multiplication factor keff maintained below 0.95.

The criticality calculations show that the subcriticality of the loading arrangements is guaranteed by design, also in the case of

- assumed fresh fuel and
- assumed penetration of moderating water without salt content.

THERMAL CALCULATIONS

The maximum permissible thermal output in the final repository under normal conditions is derived from the following requirements:

- the integrity of the cask and the enclosure of the radioactive materials, must be guaranteed for the long-term storage
- the integrity of the fuel cladding must be preserved.

In order to fulfill safely the demand for the integrity of the fuel cladding and of the leakproof enclosure, it is necessary to determine - as precisely as possible - the maximum quantity of heat transmission by the cask.

Using the burnup calculations made with OREST, the fuel rod temperatures were determined as a function of the decay time for the three types of fuels. The calculations proved the design to be in accordance with the requirements.

In addition it could be shown, that the surface temperature of the cask was below 85C as demanded by transport regulation for a type B(U) package.

MECHANICAL DESIGN

Stress Analysis for 9-meter Drop Test

The verification that the final disposal cask withstands the load conditions to be taken into account in its design is provided by FEM calculations and drop tests with a 1:1 prototype cask.

The final disposal cask is initially designed to withstand the rock salt pressure under final storage conditions. The resulting cask structure is than used to show its compliance with IAEA-transport-related design requirements.

The calculation of temperatures proceeds from stationary temperature profiles with a maximum decay heat of the fuel rods and 38C ambient temperature. Characteristic values of the material and admitted tensions are used in each case for the most unfavorable temperature. The following calculated temperatures for transport and intermediate storage are used as the basis of the calculations of tensile strength to provide a conservative safety margin.

- Shielding cask 135C
- Inner cask 250C

Taking into account the maximum possible effects on the inner and shielding casks, the impact positions described below have been selected as representative for the analysis and drop test.

Impact on the Shell Line, Drop Test No. 1

The final storage cask falls onto the shell line of the shock absorber with its longitudinal axis parallel to the impact plate.

To examine this drop test, shielding and inner casks are analyzed by means of a three dimensional FEM calculation and taking into account non-linear material characteristics. The shock absorbers are also calculated with a non-linear characteristic.

The structure has, as starting condition, the drop speed of 13.3 m/s, which results by being dropped from a height of 9 meters.

The positions of the strain gauges and the accelerometers were selected for areas of the cask with the greatest impact and expected strains.

Impact on the Lid, Drop Test No. 2

The final disposal cask falls on the shock absorber with its longitudinal axis rectangular to the impact plate.

Impact on the Edge of the Lid, Drop Test No. 3

The final storage cask is dropped onto the corner of the shock absorber with the diagonals of its center of gravity rectangular to the impact plate.

Stress analysis for 5-meter drop tests without shock absorbers

Impact on the Trunnions, Drop Test No. 4

The final storage cask falls onto both trunnions simultaneously with its longitudinal axis parallel to the impact plate (concrete foundation), without shock absorbers.
The method of calculation corresponds to that of drop test no. 1, but with the difference that the trunnions are modelled as "shock absorber" with non-linear material characteristics.
Impact on Cask Bottom, Drop Test No. 5
The final storage cask falls with its bottom to the impact plate (concrete foundation), without shock absorbers.
THE PRESENT STATUS OF DEVELOPMENT
The safety analysis report and the approval documents FOR THE FINAL DISPOSAL CASK were submitted to the authorities (BAM and BfS) to obtain a type B(U) license and the approval for the storage of radioactive materials in an intermediate storage facility.
For demonstration purposes, the final storage cask was built in the model dimensions 1:10, in order to demonstrate the functional interaction of essential cask components.
Then a prototype cask on the scale 1:1 was built. To achieve leak tightness of the inner cask in accordance with final storage conditions, 1:1 welding tests were performed for the secondary lid. The narrow-gap welding technique proved on nuclear power plant components was selected as the suitable welding technique.
The tests for the welding technique including ultra sonic tests were performed under the supervision of the German Technical Inspectorate (TV) and the authority (BAM), and received a positive evaluation.
To prove the theoretically determined heat transfer of the final storage cask, heat load tests are planned with the prototype casks. These tests are to determine the temperature fields of the cask in a horizontal and vertical position.
To prove the analytically performed verifications and to provide verification of the ANSIS FEM program used for calculation of the mechanical tensile strength in accidents (e.g. drop from a height of 9 meters onto an inflexible base), 1:1 drop tests were performed by BAM. On the basis of these tests we expect no difficulties in obtaining approval for the final storage cask.
We expect to receive the type B(U) license for the final disposal cask at the end of 1995 and the approval for storage for the intermediate storage plant in 1996.

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DOE SPENT NUCLEAR FUEL DRY STORAGE DESIGN REQUIREMENT: A COMPARATIVE ANALYSIS OF NRC REQUIREMENTS WITH DOE ORDERS FOR DESIGN OF A DOE SPENT NUCLEAR FUEL DRY STORAGE FACILITY
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ABSTRACT
The Department of Energy (DOE) Office of Spent Fuel Management, EM-37, is currently proceeding with actions to implement safe and cost effective long-term interim storage for DOE-owned spent nuclear fuel (SNF) prior to its final disposal. These actions include evaluation of alternatives for design, construction, and operation of new storage facilities in accordance with either Nuclear Regulatory Commission (NRC) requirements, DOE Orders, or both. This paper presents the results of an evaluation of DOE Orders pertaining to the design of a dry storage SNF facility to analogous NRC requirements. This evaluation included identification of differences between DOE and NRC requirements, as well as identification of DOE requirements for which no analogous NRC requirements exist. Requirements that govern all safety related aspects for design of these facilities including quality assurance, safety analysis and review, technical safety requirements, and radiation protection were considered.
INTRODUCTION
DOE's responsibility for establishing requirements for safe and efficient operation of its SNF facilities and ensuring compliance with those requirements is derived from the Atomic Energy Act of 1954 (as amended), the Energy Reorganization Act of 1974 and the DOE Organization Act of 1977. The primary means of discharging this

responsibility and for specifying the associated nuclear policy and requirements with respect to health, safety, and the environment are the DOE Orders issued by DOE Headquarters in Washington, D.C.

The increasing concern of the Nation, in both government and private quarters, in regard to the disposition of SNF has resulted in a growing constituency favoring subjecting DOE SNF storage facilities to the same management standards imposed upon commercial spent fuel. Furthermore, these facilities have been designed, constructed and operated under DOE self-regulation, which has created the public perception of a credibility problem. The House of Representatives of the U.S. Congress has considered legislation that would require DOE to submit to external regulation of its defense nuclear facilities. In an effort to be proactive, EM-37 has undertaken an evaluation of approaches associated with NRC external oversight of future DOE SNF dry storage facilities.

A key aspect of future NRC external oversight is compliance with the body of NRC regulations and guidance. This study provides the technical validation for this proposition that compliance with NRC minimum requirements will ensure compliance with all DOE technical requirements for design of a SNF dry storage facility.

Objectives

The goal of this evaluation was to review the DOE Orders pertaining to the design of a DOE-operated SNF dry storage facility, and to compare them with similar requirements developed and imposed by NRC upon licensees of SNF storage facilities. The specific objectives of the analysis comprised:

- identifying and evaluating areas of differences between DOE and NRC requirements, and;
- identifying DOE requirements for which no analogous NRC requirements exist.

Scope

This evaluation focused upon the DOE's design requirements for a SNF dry storage facility. These requirements govern not only the structural design criteria but also quality assurance, safety analysis and review (i.e., Safety Analysis Reports and Unreviewed Safety Questions), Technical Safety Requirements, radiation protection and fire protection concerns. Therefore, this study includes related DOE orders (e.g., DOE Order 5480.28, "Natural Phenomena Hazards Mitigation" and related DOE Standards for its implementation), as well as the primary DOE Order for design criteria 6430.1A "General Design Criteria".

In many cases, the requirements and guidance contained within the DOE documents were of a general nature and applied to design of all facilities (e.g., Plutonium processing, fuel fabrication). Furthermore, requirements contained within DOE Order 6430.1A specific to SNF storage (i.e., Section 1320, Irradiated Fissile Material Storage Facilities) required evaluation for applicability to dry storage. For example, requirements pertaining to water pool design were not considered.

The evaluation reviewed only those DOE requirements that pertained to safety class equipment or impacted facility safety. For a dry storage SNF facility only the physical confinement systems, such as storage cask and cement pad, would meet the DOE's definition of safety class equipment. Hence, ancillary support systems, (i.e., instrumentation and control systems, constant air monitors, etc.) would not be subject to the additional constraints of a safety class system. Whereas, 10 Code of Federal Regulations (CFR) 72 states that these types of ancillary support systems are important to safety and thus, subject to a more stringent set of requirements. Therefore, in all of these cases, the NRC requires more attention to the design of these types of systems, that is, the NRC requirements and guidance are more restrictive than the corresponding DOE requirements. Additionally, DOE and NRC definitions of safety-related equipment as it pertains to natural phenomena hazards (NPH) differ in several aspects; this difference was considered in the analysis. Requirements associated with the Environmental Protection Agency (EPA) and other government agencies' documents prepared by licensees for review and approval by NRC, and requirements pertaining to the design and handling of SNF storage casks were not considered in this review because they generally did not provide requirements applicable to existing DOE SNF storage facilities.

EVALUATION

The methodology identified and considered all safety related aspects for design of SNF dry storage facilities including quality assurance, safety analysis and review, technical safety requirements, and radiation protection design-limiting criteria. The analysis identified all requirements and guidance contained in DOE Orders that

pertained to the design of SNF facilities. It then identified all analogous NRC requirements and evaluated the differences as they would impact the technical safety basis.
The resulting 50-page report was subjected to a peer-review by the SNF Dry Storage Demonstration Working Group. This group included representatives from sites storing DOE-owned SNF, as well as an independent national laboratory. Although the final report was modified to provide a more detailed discussion on the differences in seismic design constraints, no substantive comments on the technical validity of the report were received.
The evaluation recognized that implementation of the DOE Order system results in the development of individual site specific criteria for DOE facilities. This development requires application of engineering judgment to ensure that the site criteria will result in meeting the DOE requirements. Thus individual site criteria and corresponding conservative margins vary from site to site. Due to this variation, the evaluation considered only those DOE requirements established at the Headquarters level.
DOE Orders
The requirements from the following DOE Orders were evaluated:
DOE Order 5480.7A Fire Protection
DOE Order 5480.11 Radiation Protection
DOE Order 5480.21 Unreviewed Safety Questions
DOE Order 5480.22 Technical Safety Requirements
DOE Order 5480.23 Nuclear Safety Analysis Reports
DOE Order 5480.24 Nuclear Criticality Safety
DOE Order 5480.28 Natural Phenomena Hazards Mitigation
DOE Order 5700.6C Quality Assurance
DOE Order 6430.1A General Design Criteria
NRC Requirements and Guidance
Requirements and guidance from the following NRC Regulations, Regulatory Guides (RGs), and NRC Nuclear Regulations (NUREGs) were evaluated:
10 CFR 20 Standards for Protection Against Radiation
10 CFR 50 Domestic Licensing of Production and Utilization Facilities
10 CFR 72 Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)
RG 1.13 Spent Fuel Storage Design Basis
RG 3.43 Nuclear Criticality Safety in the Storage of Fissile Materials
RG 3.48 Format and Content for License Application for an ISFSI (dry type)
RG 3.49 Design of an ISFSI (Water Pool Type)
RG 3.50 Format and Content for Application
RG 3.53 Applicability of Existing Regulatory Guides to the Design and Operation of an ISFSI
RG 3.60 Design of an ISFSI (dry type)
NUREG 0800 Standard Review Plan
CONCLUSION
The results of this evaluation verified that DOE requirements for the design of a SNF dry storage facility are analogous to NRC requirements with few exceptions. Notably, NRC generally provide greater specificity than corresponding requirements contained within the DOE Orders. However, meeting NRC requirements would not be a significant obstacle since the DOE Orders are similar in intent to their NRC counterparts.
ACKNOWLEDGMENTS
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17-44

ISOLATION PERFORMANCE OF NORTH-EAST GRANITES IN TURKEY

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ABSTRACT

In this study, locations of granite formations and main fault system of Turkey are mentioned and shown in maps. Simple isolation performance tests are applied on granite samples. These samples are taken from north-east region of Turkey. For this reason, most of these samples include fractures and filling minerals. Dimensions of the tested samples are; diameter:50 mm. and width: 3 mm.. Plastic columns (diameter:50 mm.) are used for isolation tests. Both ends of the plastic columns are opened. Each granite sample is emplaced into the plastic column tightly. Each column is filled by Cs-137 solution (Initial specific gamma activity:260 bq/ml). Transported solution that passed through the fractured sample is taken periodically from the bottom part of the column. The amount of this solution is determined and analyzed by a gamma spectrometry during the test period (test period:100 days). Results are shown in regression graphs.

INTRODUCTION

High-level nuclear wastes should be permanently isolated from the environment and remain safe for very long periods. Migrations of radionuclides from the waste form, is prevented by surrounding rock. For this reason, host rock has been considered as a final barrier in multibarrier designs (1). Radionuclide migration in geologic media is occurred by groundwater. If radionuclide transferred directly by the flowing groundwater, this is called advection. Furthermore, dispersion in the pores of host rock is occurred with the interactions of minerals, this is sorption effect. In this study, isolation performance of cylindrical granite samples are analyzed. The dimension of these granite samples are; diameter = 50 mm., width = 3 mm. These samples are prepared from the granite blocks that are taken from the north-east region of Turkey. These granites naturally includes fractures and fissures. Therefore permeability of these granites is greater than massive types.

NORTH-EAST GRANITES

General tectonic conditions of our country can be described according to the basic fault zones and previous earthquakes. Our country can be separated into five earthquake levels. Especially North Anatolian Fault is more effective in this separation (2). Several candidate host rocks are available in Turkey. General locations of main formations of these host rock types are; rock salt and tuff formations are in the middle region of Turkey, Granite formations are in the north-east region of Turkey (Fig. 1). In fact, north-east granites has not any priority to take into consideration among from the other candidate host rocks. Tectonically, locations of rock salt and tuff formations are more convenient than granites. Since this is not a site selection study, north-east granites have been chosen as a fractured media for this type of study. The aim of this study is: taking information about the isolation performance of these granites. Main fault system of Turkey is shown in Fig. 2.

Fig. 1. Locations of main granite formations in Turkey (2).

Fig. 2. Main fault system of Turkey and lateral stress directions (3).

RADIONUCLIDE TRANSPORT

Radionuclide transport can be described for the two geological media; porous media and fractured media. In this study, radionuclide transport for fractured media is taken into consideration.

Fractures are natural pathways for groundwater flow and also radionuclide migration. During the groundwater flow through the fractures, radionuclides diffuse into pores from the fracture surface to the interior of rock sample. At the same time, diffusion into infilling minerals is expected. This is called matrix diffusion and it cause retardation of radionuclide transport in rock sample.

Radionuclide transport in fractures (advection and dispersion) is expressed by the

following Eq. (4).
Eq. (1)
Eq. (2)
Radionuclide transport in the matrix (matrix diffusion) is expressed as follows;
Eq. (3)
Eq. (4)
In these equations, the subscript i denotes the i th radionuclide in a decay chain, the subscript f denotes a variable within fractures and the superscript m denotes a variable in the rock matrix. Variables in the equations are as follows;

Rdif :Retardation coefficient through adsorption onto fractured surfaces (-)
Rdim: Retardation coefficient through adsorption into rock matrix (-)
Cif: Radionuclide concentration within fracture (LT-1)
t: Time (T)
V: Groundwater velocity within fracture (L)
z: Distance in the direction of fracture (L)
DL: Dispersion coefficient (L2T-1)
Dpm: Diffusion coefficient in pores of rock matrix (L2T-1)
Cim: Radionuclide concentration within pores of rock matrix (ML-3)
x: Distance in a direction perpendicular to the fracture (within rock matrix) (L)
ll: Decay constant (T-1)
Kai: Distribution coefficient related to adsorption onto fracture surface (L)
Kdi: Distribution coefficient for rock matrix (L3M-1)
2b: Fracture aperture (L)
em: Porosity of rock matrix (-)
rm: Theoretical density of rock matrix (ML-3)

ISOLATION PERFORMANCE TESTS

Recent results have demonstrated that direct measurements of unsaturated transport parameters on subsurface materials and engineered systems, e.g., hydraulic conductivity, vapor diffusivity, retardation factors, thermal and electrical conductivities, and fluid potential, are essential for defensible site characterization needs of performance assessment and restoration or disposal strategies. Predictive models require the transport properties of real systems which can be difficult to obtain (5).

This study illustrates the results of a simple method. Isolation performance tests are applied on granite samples in a plastic column. Granite sample (diameter=50 mm., width=3 mm.) is fixed back end of this column. Between the column circumference and side surface of the granite sample, seepage is prevented by using PVC glue. After this isolation operation, column is filled by Cs-137 solution, which has approximately 260 Bq/ml. specific activity. Test time is considered as 100 days. During this time period, solution that passed through the granite sample is taken under the column periodically. Passed solution amounts determined and their final specific activities measured by a gamma spectrometry. Final specific activities and volumes of passed solutions are recorded and plotted during the test period. Power regression is applied on these graphs. According to the test results, isolation performance of these granites are determined (Fig. 3 and Fig. 4). Therefore, regarding to the mass transport and radionuclide transport, isolation performance of these fractured granites is not convenient for long term isolation.

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17-45

EVALUATION OF STORAGE TECHNOLOGY ALTERNATIVES FOR FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL

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ABSTRACT

The U.S. Department of Energy (DOE) has been evaluating alternatives for managing spent nuclear fuel (SNF) from DOE and research facilities. Two environmental impact statements (EIS) are in progress: the first covers all of this fuel programmatically (i.e., generally), while the second one specifically addresses foreign research reactor (FRR) SNF. Safe storage technologies are an integral part of each EIS, and these technologies are briefly summarized and evaluated in this paper.

FRR SNF consists primarily of metallic fuels with aluminum cladding, which are more susceptible to degradation than the zircaloy-clad oxide fuels used in power reactors. FRR SNF also utilizes higher Uranium-235 (235U) fuel enrichments, in some cases up to 93 percent assay. Consequently, both wet and dry storage technologies were evaluated for potential use, with emphasis on the following criteria: chemical compatibility, subcriticality assurance, shielding effectiveness, structural integrity, thermal performance, ease of application, economics, accident risks/consequences, and regulatory bases. The evaluations concluded that current wet and dry storage technologies were appropriate for FRR SNF, with a preference for approaches that allow monitoring and SNF retrieval and examination. FRR SNF storage is approximately equivalent in size to a large commercial SNF storage site. Utilization of existing facilities (i.e., for staging) favored wet and individual unit (e.g., cask) dry storage, whereas "greenfield" approaches favor integral dry storage (e.g., vault designs). For FRR SNF, either route would require approximately 2 hectares (5 acres) of land for the storage areas and fencing.

INTRODUCTION

In the 1950's, the United States began providing assistance for the peaceful application of nuclear technologies to countries that agreed with its policy on the nonproliferation of nuclear weapons. The assistance included the supply of highly-enriched uranium (HEU) fuels for FRRs. HEU consists of uranium enriched to 20 percent or more in the fissile isotope 235U. After irradiation in the research reactor, the SNF was returned to the United States for recovery of the HEU by reprocessing and, subsequently, manufactured into new fuel. This provided a closed cycle for the HEU fuels, minimized SNF storage requirements, and maintained U.S. control of the HEU.

Research reactors represent the principal civilian use of HEU fuels. Thus, in 1978, the United States refined its nonproliferation policies to minimize HEU civilian use by instituting the Reduced Enrichment for Research and Test Reactors program. This program had the objective of replacing HEU fuels with low enriched uranium (LEU) (up to 20 percent 235U assay) fuels in research reactors, with the ultimate goal of eliminating HEU use. The United States continued accepting FRR SNF until the policy expired, in 1988 for HEU SNF and in 1992 for LEU SNF.

FRR SNF has accumulated at the overseas sites, forcing many operators to expand storage facilities or seek offsite alternatives. DOE has decided to pursue an EIS addressing options for management of the FRR SNF and alleviating FRR storage concerns(1)(2). This paper discusses the storage component and technologies of the proposed action.

A BRIEF DESCRIPTION OF FRR SNF

Figure 1 compares power and research reactor fuels. Unlike power reactors, most research reactors use metallic fuels, usually based upon uranium-aluminum alloys, with a thin metal cladding. The cladding is usually aluminum. A small fraction of

the FRR SNF consists of Test, Research, Isotope, General Atomic (TRIGA) elements, which differ from the all-metal element. TRIGA SNF contains HEU as a uranium zirconium hydride, clad in incoloy, stainless steel, or aluminum. The element typically contains around 1 kilogram (kg) of uranium (LEU or HEU) and has a total weight a little over 5 kg. High density, LEU FRR fuels contain slightly more uranium. Many variations of fuel exist, but the elements usually are within 1 meter (m) in height and 10 centimeters in overall diameter. Thus, at least five FRR elements can fit within the dimensional envelope of a commercial reactor assembly. The FRR SNF has very high burnups, typically exceeding several hundred thousand MegaWatt Days per Metric Tons of Heavy Metal (MTHM); but, isotopically, the fission product inventory is at least an order of magnitude below a commercial SNF assembly. Relatively small quantities of transuranics are produced due to the enrichment level in the research fuels.

The potential exists for up to approximately 23,000 FRR SNF elements, containing a little over 19 MTHM. The storage period might be as long as 40 years, and followed by ultimate disposition, either by processing or disposal in a repository. The storage of FRR SNF is approximately equal in scope to a large dry storage area at a multi-unit, commercial plant.

EXISTING STORAGE METHODS FOR COMMERCIAL SNF

The commercial utility industry has addressed the storage of SNF using both wet and dry technologies(2). Figure 2 provides an overview of storage approaches and a partial list of commercial designs. Wet storage of SNF in a stainless-steel lined concrete pool is the historical choice. Assemblies are stored vertically on racks mounted on the floor of the pool and can be visually inspected at any time. Cranes and long-handled tools are used to handle and move SNF within the pool. Water provides the shielding and heat transfer medium. Ion exchange columns, filters, and heat exchangers maintain water quality within specifications. The entire pool is housed within a seismically-qualified facility that includes Heating, Ventilating, and Air Conditioning and High-Efficiency Particulate Air filters. A wet storage facility for SNF contains active mechanical components and requires staffing and maintenance.

In the past decade, the accumulated quantities of commercial SNF in wet pools at nuclear power plants have provided the incentives for the development and implementation of dry storage technologies. Dry storage methods are inherently passive in nature and, once loaded, require very little attention and operating costs. Dry storage involves the placement of SNF in a metal canister. The canister is subsequently dried, inerted (usually with helium), and sealed. Subsequently, two different approaches can be followed. In a vault design, a bridge crane transfers the canister in a shielded container to a charging face, which is inserted into steel tubes within a concrete vault. Natural convection of outside air through the vault, around the tubes, and up a stack provides for the cooling of the fuel. The vault is also a self-contained facility that includes an inspection cell and a shipping cask loading/unloading area. Currently, there is one vault licensed and operating with commercial SNF in the United States.

The second approach utilizes either metal or concrete casks, and multiple licensed designs have evolved. With metal casks, the loaded SNF canister is usually designed as the storage cask itself and, thus, provides the necessary shielding. Heat transfer is accomplished by conduction through the thick metal walls of the cask. Alternatively, the loaded SNF canister is placed within the cavity of a concrete cask for shielding and storage. The canister can be either horizontal or vertical. Natural convection of the air inside this cavity provides heat transfer; and, thus, shielded labyrinth air inlets and outlets exist in the concrete design. Cask approaches are inherently modular in design and rely upon the use of existing facilities (e.g., a wet storage pool) for loading, unloading, and inspection (i.e., "staging").

Most commercial utilities have analyzed the SNF storage alternatives, and concluded that cask approaches represent a logical, operational extension of their existing wet pools for rapidly providing safe and cost-effective SNF storage. Many of these same utilities have also concluded that concrete casks offer the best combination of parameters. Several installations already exist at commercial reactor sites. Programs within DOE are also developing a multi-purpose canister (MPC), with the intention of reducing the handling and operating costs from reactor storage, transportation, and ultimate disposal in the proposed SNF repository. The MPC is

essentially a standardized SNF transfer canister for dry storage. The MPC concept is still evolving and will not be available for several years.

EVALUATION CRITERIA FOR FRR SNF STORAGE METHODS

SNF storage approaches and designs have been compared and evaluated, with an emphasis upon the use of designs already in use for commercial SNF. The evaluation criteria and the results are briefly discussed in the sections that follow. Overall, wet and dry designs were found to be comparable from a program perspective, although significant differences exist in some areas.

Chemical Compatibility

The storage environment must be compatible with the cladding material and mitigate, or preferably prevent, its degradation. Aluminum cladding can deteriorate under both acidic and alkaline conditions in water, and this effect has been observed in older fuel pools with poor water chemistry and lacking a concrete coating or liner. However, continuously filtered and demineralized high-purity water has been observed not to result in any degradation of FRR elements. Thus, a modern fuel pool design is expected to provide satisfactory long-term storage. In contrast, dry storage uses an inert gas (usually helium) within the canister, and this is not expected to react at all with the cladding. HEU FRR SNF has been dry-stored for up to 30 years without any cladding degradation in Australia, Japan, Denmark, and Canada(2).

Subcriticality Assurance

Both HEU and LEU FRR fuels have higher enrichments than fresh commercial fuel, even after discharge from the reactor. Both wet and dry storage methods maintain subcriticality principally by geometry. Water pools use racks to maintain spacing and sometimes include neutron absorbing materials in the spacers and racks (e.g., borated steel). Dry storage methods would probably utilize an additional geometric barrier by individual containerization of each element, thus avoiding reconfiguration concerns from slumping. The total fissile material in a dry storage canister of FRR SNF is expected to be comparable to or below that of a canister loaded with commercial SNF.

Shielding Effectiveness

FRR SNF shipped to the United States will contain fission product inventories at least an order of magnitude lower than commercial SNF. Thus, since the radiation energy spectrum is approximately the same, shielding designed for commercial SNF can easily accommodate the FRR SNF. This translates into approximately 8 feet of water cover in pool storage and 30 inches of concrete in dry designs.

Structural Integrity and Confinement

The two principal structural effects of FRR SNF relate to its weight and strength in dry storage methods. FRR SNF weighs considerably less than commercial fuel. On an equivalent volume basis, 5 FRR SNF elements have a total mass of approximately 25 kg, compared to around 800 kg for a typical commercial assembly. For a typical storage transfer canister, this corresponds to weight savings exceeding 30 tonnes, thus increasing the design margins.

The lower strength of the FRR SNF will require additional analyses to demonstrate that operational and postulated accident events do not impact the integrity of the cladding. However, the principal confinement remains the canister surrounding the fuel, and this has already been qualified for the heavier loads due to commercial fuel. Thus, commercial design margins are expected to be more than adequate and meet DOE requirements(3).

Thermal Performance

Wet storage of SNF utilizes water as the heat transfer mechanism, which provides for effective cooling and low cladding temperatures. Dry storage utilizes convection cooling from the fuel to the metal canister, conduction through the canister, followed by natural convective heat transfer to the atmosphere. Thus, when compared to wet storage, it is less effective and results in higher cladding temperatures for a given heat load. For commercial fuels, the cladding temperature must be maintained below 350oC during routine operations, and concrete temperatures must be kept lower (around 150oC maximum, depending on the design). Most of the FRR SNF elements have aluminum cladding. Aluminum undergoes a phase transition just above 200oC which greatly reduces its strength and results in highly undesirable slumping of the fuel(4)(5)(6). Consequently, a conservative 175oC temperature limit has been defined for FRR SNF. This effectively derates existing commercial designs by approximately 60 percent [e.g., 24 kilowatts (kw) down to 10 kw]. Analyses further indicate that this only impacts FRR SNF with a heat load above 40 Watts per element, a situation

that only exists within the first 3 years or so after discharge from the reactor. Therefore, dry storage would entail the use of a small wet pool facility for up to 3 years prior to dry storage.

Ease of Use

FRR SNF storage would utilize existing designs which are modified to accommodate the FRR SNF(2)(7). Wet storage allows for easy periodic fuel inspection, but requires the greatest amount of maintenance and operation of all the storage designs and generates the largest amount of waste. The dry vault and dry metal or concrete casks generate less waste and inherently demand less maintenance and operation, but do not offer convenient periodic visual inspection of the SNF. The vault's design includes many operational functions (i.e., loading and fuel inspection) that require a separate pool and transport cask for the metal or concrete cask designs. Casks allow for incremental increases in capacity without large outlays. The passive nature of the vault and cask heat removal and confinement systems results in minimal operation expenses; whereas, the wet storage design demands monitoring and maintenance of active safety systems for heat removal, water purification, and maintaining water inventory.

Accident Risks/Consequences

Analyses were performed of postulated accidents at wet storage and both vault and cask dry storage facilities containing FRR SNF at five DOE sites (Savannah River Site, Idaho National Engineering Laboratory, Oak Ridge Reservation, Hanford Site, and Nevada Test Site). Five different conservative accident scenarios were developed: fuel element breach, criticality, dropped fuel cask, and airplane crash with or without a resulting fire. Source terms specific to FRR SNF were calculated using the ORIGEN2 computer code(8), and the GENII code(9) was used to calculate atmospheric releases and doses. Using site-specific population distribution and both 50 percentile and 95 percentile site-specific meteorology, doses to the workers, maximally exposed off-site individual, nearest public access individual, and the 50-mile-radius general population were calculated. Calculated individual doses due to these accidents at the 5 sites varied from 0.00034 millirem (mrem) to 2,300 mrem, which are all within regulatory limits. The comparable population dose range was 0.11 to 2,900 person-rem. The highest risk for any accident, accounting for the probability of each accident, was 0.0044 latent cancer fatalities. The accident risk and consequence analysis results did not show that any one storage technology was safer than any other, but did substantiate the very small radiological risks associated with FRR SNF storage.

Economics and Costs

Typically, wet designs tend to be more capital and labor intensive and generate more waste than dry storage designs for the same quantity of SNF. This is principally due to the use of active systems in wet storage. Also, dry concrete cask systems have lower capital costs than dry metal cask systems(7).

Regulatory Basis and Licensing

Wet and dry storage facilities for commercial SNF have been built and licensed domestically. FRR SNF would occupy the same physical space as approximately 4,500 commercial assemblies, which is comparable in size to large SNF facilities at multi-unit commercial sites. DOE sites are required to meet specific requirements for SNF storage facilities(3), which are generally the same as 10 CFR 72.

Table I summarizes the FRR SNF storage parameters for several approaches.

SUITABLE APPROACHES FOR FRR SNF STORAGE

The analyses identified three generic designs that could be utilized for the construction of new facilities for FRR SNF storage. These are:

- The Modular Vault Dry Store (MVDS),
- Dry Cask (Metal, Vertical Concrete, or Horizontal Concrete) Storage, AND
- Wet Pool Storage.

The MVDS would occupy an area approximately 213 m by 61 m (700 ft by 200 ft), and include its own receiving and staging area for transferring SNF. The storage vault itself would contain up to 29 modules with up to 44 tubes per module. The Dry Cask Storage would utilize up to 94 casks with up to 260 elements per cask. Cask storage would occupy an area approximately 91 m2 (300 ft2). The generic wet storage approach would use 2 pools, each 16.5 m by 10.4 m by 9.5 m (54 ft by 34 ft by 31 ft) deep. Four smaller pools would provide for unloading, cutting/canning, and leak checking operations(10).

The analyses also identified existing facilities that fit within these categories

and which could rapidly implement FRR SNF storage. MVDS analogs include the Engine Maintenance and Disassembly (E-MAD) facility in Nevada, the Test Area North (TAN) facilities at Idaho, and the Fuel Materials Examination Facility (FMEF) in Hanford. The E-MAD has sufficient capacity for all of the currently identified FRR SNF, while TAN and FMEF would require some supplemental storage capacity (e.g., by an adjacent small vault or casks). Dry cask technology could be implemented incrementally at any of the DOE sites with SNF handling capabilities, including Savannah River, Idaho, Hanford, Oak Ridge, and Nevada. While several sites have existing wet pools that could be used for short-term storage of FRR SNF, the Barnwell Fuel Receiving and Storage Station (FRSS) adjacent to the Savannah River Site is seismically qualified, suitable, and available for long-term wet storage. Analyses indicate that the FRSS can accommodate all of the expected FRR SNF.

CONCLUSIONS AND RECOMMENDATIONS

The applicability of SNF storage technology was evaluated as part of DOE's ongoing EIS on foreign research reactor fuels. All currently available technologies, including wet pools, dry vaults, and dry casks, were found to be amenable for the safe storage of FRR SNF. A well developed commercial industry indicates ample availability of designs and hardware, and implementation time should be minimal. The analyses also identified existing facilities that could be used in whole or in part for FRR SNF storage.

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ACCIDENT ANALYSIS FOR TRANSURANIC WASTE MANAGEMENT ALTERNATIVES IN THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAM*

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ABSTRACT

Preliminary accident analyses and radiological source term evaluations have been conducted for transuranic waste (TRUW) as part of the U.S. Department of Energy (DOE) effort to manage storage, treatment, and disposal of radioactive wastes at its various sites. The approach to assessing radiological releases from facility accidents was developed in support of the Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS). The methodology developed in this work is in accordance with the latest DOE guidelines, which consider the spectrum of possible accident scenarios in the implementation of various actions evaluated in an EIS. The radiological releases from potential risk-dominant accidents in storage and treatment facilities considered in the EM PEIS TRUW alternatives are described in this paper. The results show that significant releases can be predicted for only the most severe and extremely improbable accidents sequences.

INTRODUCTION

Transuranic waste (TRUW) is waste contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g of waste at the time of assay (1). Packaged TRUW with surface dose rates less than 200 mrem/h is categorized as contact-handled (CH); that with surface doses greater than 200 mrem/h is categorized as remote-handled (RH). TRUW results from a variety of activities, including the processing and handling of plutonium and plutonium-contaminated materials. Principal sources are research and development, special nuclear materials recovery, weapons manufacturing, decontamination and decommissioning (D&D), and disposition of plutonium-bearing residues. Most residue is in solid form such as contaminated protective clothing, paper trash, glassware, tools, and machine parts. At the end of 1991, there were approximately 65,000 m3 of CH TRUW and 4,300 m3 of RH TRUW. Most TRUW is stored in 55-gal drums or standard waste boxes on asphalt pads within weather-protective structures, earthen berms, or concrete structures. Estimated inventory and generation rates for major storage sites, derived from data in the Interim Mixed Waste Inventory Report (2) and the Integrated Data Base for 1992 (3), are provided in Hong et al. (4).

To support analyses for the U.S. Department of Energy (DOE) Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS), Argonne National Laboratory (ANL) has developed an integrated risk-based approach for evaluating the source terms from radiological releases from postulated facility accidents (5). The methodology has been developed in accordance with the latest DOE guidelines, which consider the spectrum of possible accident scenarios in the implementation of various actions evaluated in an EIS. This approach allows comparisons of facility accident impacts on EM PEIS strategies for consolidating the storage and treatment of TRUW at sites throughout the country. Accidents considered in this analysis include operational (scenarios such as handling accidents, facility fire, and facility explosion) and severe external events (such as earthquakes, tornadoes, and airplane impacts). This paper analyzes the radiological releases from potential risk-dominant accidents in storage and treatment facilities considered in the EM PEIS TRUW alternatives.

TRUW MANAGEMENT ALTERNATIVES OVERVIEW

Under DOE's current management program, retrievably stored and newly generated TRUW are to be prepared (characterized, segregated, packaged and/or certified for transport and disposal) and stored pending transport to the planned Waste Isolation Pilot Plant (WIPP) facility in New Mexico. The DOE has entered a WIPP test phase to evaluate long-term repository performance and subsequent acceptability for disposal. During this phase, limited quantities may also be located at the WIPP. Additional storage facilities may be required, depending on the timing of retrieval operations and the WIPP schedule. Other sites that generate TRUW, including that from D&D will have to store on-site, contract for commercial storage, or transport to one of the nine current storage sites. All sites are responsible for minimizing the quantities of waste generated. The current strategy is to treat TRUW to meet WIPP waste acceptance criteria (WAC) (6) and dispose of it at the WIPP under the Resource Conservation and Recovery Act (RCRA) "no migration" determination rule. However, treatment might be required, under RCRA, Title 40, Part 268 of the Code of Federal Regulations (CFR) (7), to remove or reduce the hazardous components to acceptable levels to meet the hazardous land disposal restriction (LDR), or to meet the radiological requirements of 40 CFR Part 191 (8), before WIPP disposal. These requirements depend on results of the test phase, a WIPP disposal performance

assessment, and the establishment of U.S. Environmental Protection Agency (EPA) criteria for compliance certification.
Four alternatives are considered in the EM PEIS: no action, decentralization, regionalization, and centralization. Decentralization and regionalization each have several cases that may result in distinct inventories for treatment at each site. In addition, three treatment options are considered. In the first, TRUW is treated to meet the minimum requirement for WIPP-WAC namely, liquid absorption, compaction, immobilization, and repackaging. The second option considers an intermediate treatment level beyond WIPP-WAC that includes shredding, grouting, and changing containers to reduce gas generation. In the third option, TRUW is treated to meet LDRs. Detailed descriptions of TRUW treatment processes are provided in Hong et al. (9).

METHODOLOGY AND MODELING CONSIDERATIONS

The volumes, physical characteristics as defined by treatability categories, and radiological composition of the sites' TRUW inventories have been compiled in the ANL WASTE_MGMT computational model (10,11). Each site's facility throughput used for determining source term information was obtained directly from the ANL computational model. The data used included unique volumetric inventories and physical, radiological, and chemical compositions for each waste treatability category at each site for each alternative. Information on accident sequences, such as initiating frequencies, damage fractions (DFs), probabilities, respirable airborne release fractions (RARFs), and leak path factors (LPFs), are also compiled in the ANL accident analysis computational model (12).
The DF is defined as the fraction of the total inventory of waste in a facility or particular operation at risk involved in the accident sequence and actually susceptible to airborne release. The LPF is the fraction of the airborne inventory that passes through the containment barriers and filters to escape to the atmosphere. The RARF is the fraction of the potentially available inventory rendered airborne and having particulates with aerodynamic equivalent diameters below 10 micrometers at the point of the accident. The values of RARFs are a function of the physical form of the material rendered airborne, which varies by the treatability category of each waste stream, and are based on the work of Mishima (13).
Preliminary consequence factors from unit-radionuclide releases at all sites were obtained from ORNL and incorporated into the accident analysis computational model to screen the accident sequences on the basis of risk, defined here as the consequence times the frequency of the accident sequence. Chemical releases for mixed TRUW were considered in the EM PEIS but are not discussed in this paper.

Storage Facility Accidents

TRUW is typically packaged in drums or canisters and stored in concrete structures, weather-protective sheds, below-grade caissons, or earthen berms. Most TRUW is stored in facilities with minimal containment. Accordingly, in this analysis, the use of a generic storage structure was assumed to represent facilities with minimal confinement or with confinement that would likely fail under severe external challenges. To ensure conservatism, no credit was taken for filtration or containment integrity in the accidents postulated for storage, although DOE sites are increasingly moving toward development of qualified TRUW storage. Because special provisions have been made for storage of RH wastes that involve much more robust containment (e.g., underground caissons), the storage accidents investigated here cover only CH wastes.
Although the inventories, physical forms, and radiological and chemical compositions of waste stored at each site were characterized for the EM PEIS and the data then used in the ANL WASTE_MGMT model, compilation of analogous information for individual facilities on each site is beyond the scope of the EM PEIS. A unit inventory approach was used to develop source terms on the basis of waste generation and inventory data at each site. All storage facility accidents reported here assume an inventory of 2,000 m3 (10,000 drums) with a site-dependent radiological and physical composition derived by volume-weighting the inventories of the treatability categories within each waste type at that site. Scaling of these unit source terms will be required to account for actual facilities.
The storage accidents investigated include handling accidents, operation-induced facility fires, and external-event-induced fires and explosions. Representative handling accidents involve a single drum and assume that 25% of the drum inventory is affected and subject to stresses capable of rendering the contents airborne. The

representative operation-induced fire scenario assumes that 10% of the facility inventory is affected. The earthquake was selected as a limiting surrogate for other natural phenomena because of its overriding damage potential. All external-event source term parameters vary according to the particular sequence. Aircraft (small or large) impacts were analyzed at selected sites, depending on their importance to risk.

Treatment Facility Accidents

In the minimal treatment (WIPP-WAC) option, retrieval drums and packages are opened and inspected in a glovebox (CH) or hot cell (RH). Absorbers are added for any free liquids, and compaction, immobilization, and/or repackaging are performed as required. The intermediate treatment option involves essentially the same operations except that all waste is treated and repackaged. The LDR treatment option includes incineration, evaporation, and various RCRA contaminant treatments.

Although several treatment operations may present potential exposure hazards, incineration was assessed to be the treatment technology most likely to dominate risk to facility and site staff and the surrounding general population. This is because incineration has key process characteristics affecting the potential for airborne release, including high temperature; the presence of fuel and fuel feed lines; the presence of combustible input feed waste (combustible solid waste, organic liquid waste, and organic sludge); the potential for overpressurization or explosion; and the high dispersibility of the ash by-product, which has radionuclide concentrations two orders of magnitude higher than the input feed waste. The focus here is on radiological accidents with sequences involving fires and explosions capable of producing large airborne releases of the ash present in storage or filtration systems.

A generic treatment facility was defined for all options for assessing a range of radiological releases from treatment process accidents. Each generic facility consists of a series of linked process modules that provide a specific treatment process. A DOE Hazards Category of 2 and concomitant performance requirements were assumed for the generic facilities. Double high-efficiency particulate air (HEPA) filtration systems are assumed to be in place.

Treatment facility accidents analyzed include (1) a fire in the baghouse area of the incineration facility, failing the filtration systems completely (LPF = 1.0) and affecting 3% of the total amount of ash existing in the facility (DF = 0.03); (2) an incinerator ash explosion caused by combustible gas buildup that affects the existing ash in the rotary kiln (12% of the total in the facility [i.e., DF = 0.12]) and partially degrades the filtration system of the facility (LPF = 0.001); and (3) external events leading to a fire. Aircraft (small or large) impacts were analyzed at selected sites, depending on their importance to risk. The earthquake was selected as a limiting surrogate for other natural phenomena because of its overriding damage potential. All external-event source term parameters vary according to the particular sequence.

RESULTS AND CONCLUSIONS

Preliminary results of the accident sequences for various site consolidation cases for each EM PEIS alternative were reviewed for risk importance in terms of the frequency-weighted dose (risk) to the maximally exposed individual. These sequences were then grouped into four annual frequency categories: anticipated (greater than 1.0E-02), unlikely (between 1.0E-02 and 1.0E-04), very unlikely (between 1.0E-04 and 1.0E-06), and extremely unlikely (less than 1. 0E-06). Representative source terms for the important sequences were then selected as the bases for health effects calculations, which are now being considered as part of the risk impact calculations performed for the EM PEIS. Generic CH TRUW storage facilities were analyzed at the nine major sites. Representative estimated total releases for each accident and its frequency group are provided in Table I for selected storage facility accidents. These accidents assume a generic 10,000 drum facility with site-specific waste composition.

Incineration facility accidents were analyzed for three cases under regionalization: treatment at five sites (ANL-E, Hanford, INEL, LANL, RFETS, and SRS), treatment at three sites (Hanford, INEL, and SRS), and treatment of RH waste at two sites (Hanford and ORNL); and one case under centralization: treatment at one site (WIPP). Representative estimated total releases for each accident and its frequency group are provided in Table IIa,b for selected incineration facility accidents. Detailed radionuclide release data for all accidents can be found in Mueller et al. (14).

These accidents assume the generic source term parameters discussed above, with facility inventories defined by the EM PEIS alternative.
Uncertainties in the inventories, source term parameters, and frequencies of accidents imply that absolute source terms are highly uncertain and should be used cautiously. In addition, the results presented here are based on the unit storage facility size or treatment throughput inventories. Nevertheless, with appropriate weighting by the ultimate inventories to be stored or treated at the various sites, these results allow the relative importance of accident source terms to be calculated with sufficient accuracy to provide a measure of comparison among the EM PEIS alternatives.
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RADIOACTIVE WASTE MANAGEMENT IN LITHUANIA
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ABSTRACT
Lithuania generates more than 80 per cent of power energy based on nuclear fuel. Nuclear fuel as well as the greater part of fossil fuels are imported. This shows that nuclear energy is a highly significant energy sector and the overall economy

branch in Lithuania. Two RBMK-1500 type reactors operate at the Ignalina Nuclear Power Plant. The first one was launched in 1983. Plenty of measures have been implemented and will be undertaken in future to increase reactors safety operation. A technical and financial assistance is rendered from the Western states, especially from Sweden, European Bank for Reconstruction and Development, IAEA, others. It is projected Ignalina Nuclear Power Plant to operate a designed life time. In Ignalina low enrichment uranium is used. Spent nuclear fuel is stored at water basins built at the vicinity of reactors. While designing this power plant it was considered to export spent nuclear fuel to a Russian reprocessing plant, however, no refineries were constructed thus causing spent nuclear fuel from Ignalina Nuclear Power Plant to remain at the site. At present basins are almost totally exhausted. The international tender had evaluated 8 companies proposals and the German company GNB was accepted to store spent nuclear fuel at metal containers within power plant's site. Currently these containers are underway of manufacturing, however, they are highly costly. It would be acceptable to use technology of concrete transportable containers or concrete module storage, which could enable to reduce costs for spent nuclear fuel management and to use local production and engineering potential.
Low radioactive waste is stored in two places in Lithuania, i.e. in the Ignalina Nuclear Power Plant and in a storage for non-energy radioactive waste.
Sufficient reservoirs are installed for low radioactive waste at the Ignalina Nuclear Power Plant where cementation and bituminization undergo. The non-energy waste storage was built more than 20 years ago and it should not be used any longer.
As well as elsewhere in the world, Lithuania has not solved yet a radioactive materials and first of all spent nuclear fuel final disposal issue. Obviously, this issue is a concern of every country, despite that trans-boundary aspects of this problem cannot be neglected.

ENERGY CHARACTERISTIC

Only 3 to 4 percent of Lithuanian primary energy resources in the energy balance is local, including hydroenergy, peat, fire-wood. Other energy resources as nuclear fuel, oil, gas are imported. If total power plants within Lithuania capacity is used, around 30 billion kWh of electricity can be produced, whereas 50 percent goes from the Ignalina Nuclear power Plant. However, due to 1991-1993 economical crisis impeded by industrial products decrease, primary energy resources, especially fossil fuel, drastical increase of prices, electricity export reduction to Russia and Byelorussia as a reason of economical disorder, electricity production in Lithuania decreased in 1993 by 14,1 TWh, including 12,3 TWh generated in the Ignalina Nuclear Plant. Thus, more than 80 per cent of power energy produced in Lithuania is based on nuclear fuel.
Unit 1 at Ignalina NPP was launched at the end of 1983. Today two RBMK-1500 type units are operating at the plant. These reactors have been designed and produced in the former Soviet Union. In chanal type reactors uranium of low enrichment is used as fuel, moderator - graphite.
Before Lithuania regained independence in 1990, regulations, standards and licensing procedures for nuclear facilities and activities were the same as valid elsewhere in the Soviet Union. The system was strongly centralized and local authorities were not consulted or even informed. This means that Lithuania after having declared its independence, has to face the task to built up a new national legal and regulatory system for radioactive waste management.
When Lithuania restored State power and independence, all operational issues at the nuclear plant had been transferred under the auspices of the Lithuanian State. Ignalina NPP itself became on operator, the State Atomic Energy Safety Ispectorate - VATESI was established, scientific institutions dealing with nuclear energy issues had been strengthened. Though earlier RBMK reactors safety operation was doubtful, the Chernobyl accident forced to review their safety issues.
Striving to improve units safety, at Ignalina NPP the total capacity was reduced to 2 500 MW, as well as other safety measures were accomplished. The construction of the Unit 3 was ceased regardless the used investment of approximately 80 per cent.
In the former Soviet Union nuclear power plants were very closed factories. After collapse the Soviet Union with western countries assistance studied the Ignalina NPP project in detail as well as its operational status, safety improvement possibilities. Programs for further safety improvement have been prepared. Assistance was provided by many European and American countries; a particularly great contribution allocated by the Government of Sweden should be mentioned.

Meanwhile 18 projects for safety improvement are underway, financed from the Nuclear Safety Account through EBRD, total grant sum of 33 ml. ECU. When these and other projects are completed, Ignalina NPP safety requirements will comply with Western standards in an acceptable level.
Following the Energy Strategy approved by the Government in 1994, it was anticipated that both Ignalina NPP units would operate the designed life time if measures for safety improvement were undertaken and the license received. It is not expedient to allocate huge investment seeking to prolong the designed retirement.

RADIOACTIVE WASTE MANAGEMENT

Ignalina NPP is the dominant generator of radioactive waste in Lithuania. A minor part of radioactive waste comes from nuclear applications in research, medicine and industry. The dominating part of the short-lived waste comes from the Ignalina NPP. During operation and maintenance the following types of low and intermediate level waste form: evaporation concentrates, ion - exchange resins and solid scrap and trash.
After evaporation the concentrate is temporary stored in tanks. From the storage tank the concentrate is transported to a bituminization facility. From there the solidified product is pumped to storage basins made of reinforced concrete. The inner surfaces of the concrete are covered with 4 mm thick steel plates. Average annual production of bituminized concentrate in the two reactors is of about 750 m. No arrangements for new basins are necessary.
Granual ion exchange resins with the inorganic filter aid Perlit are used for treatment of the reactor water. So far the resin Perlit mixture has been transported to a storage tank in a liquid form. No treatment measures have been taken except dewatering after sedimentation. As the sludges have been stored in the tank for several years without any agitation, a fairly thick layer of sediments with a very low viscosity has assembled on the bottom of the tank. No arrangements for emptying the tank are made, and special measures for making the sediment slurry transportable will have to be taken. The storage tank will be completely filled in the near future. Additional storage capacity is planned to be obtained by using one of the tanks adjusted for evaporation concentrates. This, however, is a provisional solution awaiting the installation of a solidification facility probably using cement as solidifying material. The procurement process of such a facility is presently in progress. The annual production is estimated to about 200m/year.
The solid waste generated at Ignalina NPP is divided into three groups according to surface dose-rate. Waste with dose-rate below 2.6 Sv/h is considered as non-radioactive. Presently the solid waste is dumped in reinforced concrete troughs with removable roofs. The design of the troughs are somewhat different depending on the type of waste to be dumped. There is no conditioning of the solid waste before dumping. The intention is to install a bale compactor in the near future and thereby reduce the volume of this kind of waste. The procurement process is in progress. The annual quantities of solid waste are about 1500 m3 and mostly of group I with low surface dose rate.
Radioactive waste generated after use of radionuclides in research, medicine and industry are subjects of radiological center of Health. Most of those, however are probably not generating any radioactive waste but using X - Ray equipment. On the medical side using radioisotopes for diagnostic purposes and for therapy are all the RMI waste generated up to 1989 is disposed in the repository for low and intermediate level waste at Maishiogala. The repository is disposed in a concrete vault, designed in the early 1960s. Repository in Maishiogala is absolutely filled. Since 1989 all RMI waste is collected by the Institute of Physics and transported to the Ignalina NPP and is stored in the vaults used for the waste from the power plant. During this time volume of solid waste transported to Ignalina NPP was 8m3.

SPENT NUCLEAR FUEL MANAGEMENT

The enrichment of the fuel is 2 per cent and the average burn-up is given 20.000 MWd per tonne. Spent fuel elements are stored in their full length in a water pool at the reactor unit for about one year. Then the elements are taken to a hot cell, where the central support rod as well as two end pieces are cut off. This leaves two fuel bundles and some metallic radioactive waste to be taken care of. The fuel bundles are placed in cages, each one of 102 bundles, which are taken to a pool for continued storage.
These pools at the moment are almost totally exhausted. Pursuant to the preliminary project it was anticipated to construct a centralized reprocessing plant in the

Soviet Union for spent fuel from RBMK type reactors, however, later this project was rejected and every plant individually had to solve a spent fuel storage and disposal issue. The project prepared by the Russian engineers on Ignalina NPP spent nuclear fuel wet storage technology was rejected. When 1993 international tender was announced, a Germany company GMB proposal was accepted for Ignalina NPP stock fuel stockpiling at the metal containers in the open site. This project is underway now. Each container can store 51 fuel bundles; these containers are transportable, they are safe and licensed in Europe. However, containers are expensive. Currently possibilities to reduce spent nuclear fuel management costs are under consideration. From that standpoint attractive proposals happen to be while placing fuel into the concrete transportable containers or module fixed concrete facilities. The basic prerequisite is that when the final disposal issue is solved for spent nuclear fuel, these filled containers could be taken out from the plant site without any complicated fuel bundles overloading to transportable containers procedures. A very similar technology is offered by Canadian AECL.

A plant, operating under present loading, needs approximately 1000 fuel bundles per year. Taking into consideration the previous yearly generation of 16 TWh of electricity and that designed life time is between 2004 and 2007, there will accumulate around 35-40 thousand fuel bundles of spent nuclear fuel. Many states, including Lithuania, have approved laws which ban to import radioactive waste. Economically it is not attractive to reprocess low enrichment spent nuclear fuel, thus it is hardly possible that in the nearest future in Lithuania or any other country it might be processed. Provisory appraisals showed that a direct nuclear fuel disposal from the economical point of view is more than two folds as efficient as its reprocessing.

DISPOSAL OF RADIOACTIVE WASTE IN LITHUANIA

The short-lived wastes will lose most of their radioactivity, the requirement on reliable demonstration of long-term good function and stability of the barrier system are much more demanding. Among other things this implies, that the host rock for a spent fuel must be selected with a higher degree of discrimination and, that the choice may by fairly limited. It also implies, that substantial efforts will be needed for characterization work and evaluation of possible disposal concepts. The geology of Lithuania is characterized by thick sedimentary rocks overlying a submerged crystalline basement. The bedrock surface is generally flat and covered by quaternary deposits consisting mainly of till and sand with a thickness between 10 and 300 m. The basement rock is dipping at depth of 200 m while the depth is more than 2000 m. The geological conditions in Lithuania seem to favor the choice of repository for the short-lived waste. As there is only one nuclear power station it is natural to try to find a suitable site for a repository at or in the vicinity of the Ignalina NPP site. The site investigations should include geological mapping including geomechanical and geohydrological properties, recording of the ground water level and its seasonal variations and geohydrochemical conditions. Repository for long term waste (spent fuel) will be needed after 30 or more years. On the other hand it is of interest to get an apprehension of feasibility and costs as soon as possible. Lithuania primarily works on a national solution but at the same time attentively follows up the possibilities for international solutions. Lithuania with its limited financial resources can not afford to build up a full competence of its own at the same extent as a bigger country with a solid economy. A large portion of the necessary knowledge and experience is today from various countries. Spent nuclear fuel disposal technical as well as legal problems shall be solved by every state, notwithstanding that in compliance with inter-governmental aspects of nuclear threat high level radioactive waste disposal should be treated as the world-wide objective in future. In the IAEA yearbook 1992 was mentioned "Interest is expected to grow regarding co-operation between countries having small nuclear programs in developing regional or international management facilities for high lever waste and spent fuel".

LEGISLATIVE BACKGROUND FOR RADIOACTIVE WASTE MANAGEMENT

Lithuania today does not have the accomplished legislative base which regulates nuclear use. However, Draft laws of the Republic of Lithuania: Energy Law, Nuclear Energy Law, Radiation Protection Law are under preparation. Some regulations are issued by the Government. In 1991 Lithuania accessed to the Nuclear Weapon non Proliferation Treaty, has co-ordinated the Safeguard Agreement with IAEA. Lithuania has accessed or is intending to access to international conventions and agreements:

Vienna Convention on Civil Liability for Nuclear Damage and Vienna and Paris Joint Protocol (1993), Convention on Physical Protection on Radioactive Material (1993). Convention on Early Notification (1994), etc. We signed or are going to sign bilateral agreements with Canada, Denmark, Norway, Poland, Russia, Byelorussia etc. Lithuania has to face the task to built up a new legal and regulatory system and requires consultative assistance while it implements the legislative establishment programs.
Implementation of a national program for the management of radioactive wastes requires a careful planning. A radioactive waste management program is not only a matter of solving technical issues, it must investigate and demonstrate to the decision-maker and the public how the demands on safety and radiation protection are to be satisfied both in a short and in a very long term perspective. The Lithuanian Ministry of Energy and Swedish Nuclear Fuel and Waste Management Co SKB signed an agreement on the preparation of an Overall Plan for Radioactive Waste Management in Lithuania. This Plan must be flexible and maintain a preparedness to modify the plan for best adaptation to current circumstances. To implement different projects have been and will be prepared tenders. With regard to radioactive waste management there should be made a clear distinction between the implementing party and the regulatory, supervising and licensing authority. The restrictive factor for an optimal and timely accomplishment of radioactive waste management a plan in Lithuania is to create such funds controlled by a national authority.
Experience of 1986-1991 anti-nuclear movement in Lithuania demonstrated that public opinion has big influence on the governmental decision-makers. The lack of understanding may seriously disturb the activities related to radioactive waste management. An open and consistent information to the public as well as comprehensive consultation with local organizations primarily concerned will therefore be an important issue in radioactive waste management program. The information should be open, reliable and consistent, should be given an early stage, initiated by the disseminator of information. Information must be passed to Seimas (parliament) political parties, local politicians and public organizations, the scientific community, teachers and young people at schools.
CONCLUSIONS
Intensive surveillance is undertaken in Lithuania and Ignalina NPP targeting to upgrade radioactive waste management.
There is prepared overall Plan for Radioactive Waste Management in Lithuania.
The key assumption for a successful solution of this issue is the national legislative background.
Seeking to achieve accurate radioactive waste management it is indispensable to study overall states experience within mentioned scope. Long term waste, especially spent nuclear fuel disposal is an inter-government issue, which requires co-ordination of appropriate activities.

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THE DISPOSAL FACILITY MOCHOVCE - STATUS AND ISSUES
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ABSTRACT
Low-level Radioactive Waste Repository in Mochovce was planned and constructed since end of seventies under unfavorable conditions: On-site storage after waste treatment during all plant operational time as the basic approach for waste management on power plants, no existence a quality assurance system applicable to the waste management, no legislative oriented specially to waste management, economical, social and political conditions. Nuclear Regulatory Authority of the Slovak Republic (NRA) has been requested for facility licensing to operation. Paper contains a description of repository various aspects: hydrogeology, seismicity, design and construction, acceptation of waste. NRA approach, IAEA Waste Management Assessment and Technical Review Program peer review team recommendations and their implementation to NRA requirements is discussed below.
BRIEF HISTORY
The initial strategy for nuclear power plants radioactive waste management in former Czechoslovak federation valid to the end of seventies envisaged for a storage of waste after treatment in inside basins and vaults for all living time of plants.

Decision for shut-down of NPP A-1 (HWGCR, 150 MWe) after accident in 1977 was the first motive, which induced a state to interesting in disposal of low-activity waste. A basic goal of activities was following: disposal of low-level waste from NPP A-1 and also operational waste from constructed and planned WWER (PWR, 440 MWe) power plants.
Selection of locality for disposal was done by state bodies for land planning from cca 30 alternatives. They decided for localize a repository close to Mochovce Nuclear Power Plant building site.
It is necessary to mention a political, economical and social climate of that time. Decision and information from nuclear energy area fell among extremely secret. All of activities were done under dictate of state and political administrative. Public had not any possibilities to express of opinion about nuclear programs. State was sole owner of all facilities, thus a competing business, which would have been an inherent guaranty of maximal work quality did not exist. A strategy of use of nuclear energy and radioactive waste management basically copied east model; any other alternative was not possible from political reasons.
The basic legislative existing within the given period which should have been met during siting were:
  Regulation of the Federal Ministry of Technical and Investment Development No. 85/1976 on detailed procedure for siting
  Edict of the Czechoslovak Atomic Energy Commission No. 4/1979 on general criteria of nuclear safety assurance in siting of nuclear power facilities.
On September 1, 1981, after having been assessed of Introductory Safety Report prepared by the repository designer (1), Czechoslovak Atomic Commission issued an approval with decision on siting. An underestimate of detailed hydrogeological investigation was a fundamental deficiency of this license procedure step. In next it has been an important source of problems.
The application for a construction license attached with Preliminary Safety Report was submitted to Czechoslovak Atomic Commission at 1984. The Czechoslovak Atomic Commission issued an approval with the construction license, but with many conditions. The final stage of complementary safety report was being prepared during the construction, which started at 1985 and finished at 1990. The disposal facility design assumpted a compacted clay layer as a fundamental engineering barrier and disposal of 200 l drums with bituminized waste into concrete vaults (2).
The quality assurance system underestimate besides existing political and economical relations caused recurring problems with quality during construction works.
During repository construction, several changes have been proposed and after having been approved also implemented. Some of them resulted from solving partial problems and impreciseness of the design, another were initiated during inspections of Czechoslovak Atomic Commission. Following changes were important:
  the change in technology of compacted clay sealing placement
  the changes in drainage systems
  the change in the technology of radwaste placement
  the change in control system.
When the works on Preoperational Safety Report (PSAR) started, the new Regulation No. 67/1987 (about safety assurance on radioactive waste management) had been already valid 3).
Detailed investigation of locality started during the PSAR preparation. The content of PSAR revised version based on above mentioned regulation, preceding discussions with Czechoslovak Atomic Commission representatives as well as American recommendation (4), was approved on November 1992. A fundamental change in disposal waste form was done approximately at the same time. Waste would be disposed in fibre-reinforced concrete containers (inner volume 3.1 m3). Waste conditioned into the containers will be produced in Bohunice conditioning centre; its operation has beginning at 1996-97.
Third version of PSAR was submitted to NRA in October 1993. This version contains the evaluation of detailed hydrogeological and seismological investigation. A` posteriori determination of waste acceptance criteria is the most important contribution of PSAR safety analyses.
DESCRIPTION OF REPOSITORY
Surface type low-level radioactive waste repository is built-up in northeast direction, approx. 1.5 km from nuclear power plant Mochovce (PWR, 4 x 440 MWe, under construction) - see Fig. 1.

Site Geology and Hydrogeology (5)

The Mochovce repository was built at the end part of the headwater area of nameless dextral affluent of the Telinsky stream, directly on the flood plane situated on both banks of the intermittent stream bed. The flood depression is filled with deluvial and flood loams (good or semi-permeable quarternal soils of the upper pleistocene). The loam thickness is somewhere between several decimeters and 4-6 m, in the repository southern part up to 10 m.

Brackish sediments from the earliest Miocene period, i.e. middle to upper Sarmat in the underlying Quaternary loams appear. Pelite and even aleuropelite layers (slightly consolidated in some places) are found between layers of silty and fine-grained sands. All the above rocks are primarily lenticular in shape.

The deeper underlying rock at the repository site and its surroundings is formed by volcanic andesite, agglomerated andesite and andesite tuff to tuffite. The andesite complex is analogous to farther places of occurrence of lower Sarmat origin.

In the Mochovce repository locality, permeable and impermeable layers (aquifers, aquitards and isolators) interchange in Sarmat sediments. There is often a smooth transition between these layers in the vertical direction and also sidewards. The layers dip generally coincides with the gradient and direction of stream bed. The conditions for appearance of partial hydrodynamic systems with groundwater free surface as well as artesian groundwater table arise. The only contact with the earth surface and surface or rainfall water occurs in places, where the aquifers comes out to the surface.

On the valley axis where the repository is situated, under Quaternary earths a fine and even silty sand layer with considerable areal spread and small thickness changes are found. This layer of permeable sands is identified as the main H aquifer. In the repository southern part under H aquifer another 2-3 aquifers are found without a stable stratigraphic level and with rapid pinching near at least some wells. The aquifer close under the H aquifer is identified as the P1 aquifer and the underlaying as the P2 aquifer. Both underlying aquifers (P1 and P2) merge southward and they form so called unified P1 + P2 aquifer.

It was found, that contingent groundwater contamination by leaching radioisotopes from the repository vaults might be spread only through the H aquifer. Contaminants in water solution would have to pass through unsaturated zone at first, its thickness is varying from 8 m (western part of vaults) to 4 m (eastern part). Infiltration through the unsaturated zones place is passed in vertical direction with minimum hydrodynamic dispersion.

After reaching a groundwater free surface the infiltrating contaminated water would mix with the uncontaminated water from the layer and they would flow together in the direction of maximum hydraulic gradient, thus causing hydrodynamic dispersion of contaminants. Groundwater from H aquifer is passing to the surface stream southern of the repository site.

Seismic Conditions and Geotechnical Characteristic

No tectonic fault has been proud to exist directly at the repository site. The nearest faults have been found west from the western repository border about 250 m in the S-N direction and parallel one about 600 m east from the eastern border. Recent movements have not been proud at these faults. In conclusion it can be assumed that the confined floe is compact and no faults are occurred under the repository within this floe.

The seismic risk estimated on the geological and seismological basis results the maximal intensity 6 - 6.5 oMSK-64 with return period once through 10 000 years and maximum developed peak ground acceleration is not overcrossing the value 0.06 g. A related risk of the sand liquefaction in the subsoil is not imminent. Risk resulting from soil properties in the subsoil and artificially installed sealing clay layers round the repository have not been proved. As conclusion of geotechnical studies and assessment of concrete structures of the repository including embankments it was follow that repository is safe and can guarantee a functional capabilities during waste disposal as well as after final coverage.

Design and Construction

The basic design of repository is a shown on Fig. 2 & Fig. 3. The structure of repository consists itself from 2 concrete double-rows with vaults. Each single row consist from 20 vaults with inner dimensions 17.4 x 5.4 x 5.5 m. The vaults in single row are grouped by 4 into dilatation units. Ninety fibre-reinforced containers can be situated to each vault, on three level.

The reinforced concrete skeleton, in this case, is not taken as barrier, which is able to prevent radionuclide release into environment. This function is fulfilled by the compacted clay sealing installed under repository (1 m thick) and on both sides (3.5 m thick). The clay bath layers were compacted to the Proctor Standard 96 % and hydraulic conductivity 10-9 ms-1.
The drainage system has been built to enable testing in case of an accident and radioactive material penetration into groundwater and subsequent radioactive waters collection. The drainage is divided into 2 systems:
"checking" drainage for drains collection and monitoring in drainage layer (space between concrete bottom of vaults and clay layer)
"observed" drainage has been installed with the aim to catch contingent infiltrating water from the space outside clay bath.

Acceptation of Waste

Limits and safety parameters of accepted waste were derived from various limiting requirements:
from performance assessment
from way of disposal
used waste form
design solutions
requirements from geotechnic stability evaluation
packaging of waste
recommendation of international and abroad bodies and operational experiences of similar type repositories
"classical" regulatory requirements (free liquids, content of phyrophoric, toxic materials, etc.)
meteorological conditions at the disposal operation
radiation protection.

Some of limits and safety important parameters are connected with operational procedures, some of them are the acceptance criteria of waste. Total activity and specific activity limits are the most important and discussed acceptance criteria formulated in PSAR.
They were computed by using of various radionuclide pathway scenarios. The general overview of calculations is shown in Table I.
The results of calculations in total acceptance activity, mean specific activity for single vault and maximum specific activity terms are shown in Table II. Because in Slovak Republic the institutional control period is not done in any regulatory requirements, the calculations had been done in two alternatives: 300 and/or 500 years. However, results are different only for Cs-137 and Sr-90.

WATRP MISSION

In December 1993 the NRA requested the International Atomic Energy Agency, Vienna, to review the Mochovce Radioactive Waste Disposal Facility within the IAEA's Waste Management and Technical Review Program (WATRP). A review meeting between international WATRP team and Slovak specialists was held at the Mochovce in May 1994. NRA obtained the official version of WATRP team report in December 1994 (6).
Some notable recommendations of WATRP team were:
a more integrated approach to safety of the disposal facility
a program should be instituted to ensure that there is a thorough identification, characterization and categorization of waste streams
the list of radionuclides considered in the safety analysis should be expanded
safety assessment for the operational phase of repository should be done
additional scenarios should be considered in the safety analysis
the maximum duration of the institutional control period which the applicant take credit for in the safety analysis should be determined by the regulatory body
a separate licensing process should be established for the disposal facility closure, on a basis of preclosure safety report
additional geotechnical tests should be made to ensure the reliability of the design assumptions
particular attention should be given to the design and construction of the various cap clay layer to ensure proper performance.

The summary number of recommendations (some of those are oriented also to the national waste management system) is 70.

NUCLEAR REGULATORY AUTHORITY STANDPOINT

The NRA standpoint to operator request for permission of operation based on the PSAR

reviewing, has been prepared at the last days. The WATRP Report recommendations and also recommendations of other Slovak specialists, who evaluated the separate chapters of PSAR, were generally adapted to the this standpoint.
NRA standpoint is divided to five parts:
- requirements concerning to repository stability
- requirements concerning to design and construction improvement
- requirements concerning to a safety analyses
- requirements for another revisions of PSAR
- quality assurance area.

In the first part NRA requires additional analysis of data to ensure of repository stability from geotechnical, seismic, erosion points of view. Principal requirements in second group are:
- drainage system improvement to obtain a possibility to drainage of individual vaults during the institutional control
- change in placement of waste containers and backfilling to obtain an extractability of containers.

In the safety analyses area NRA requires more systematic approach to selection of radionuclides, scenarios and pathways to the environment.
Requirements in fourth group are oriented to better and critical evaluation of all aspects of repository quality. The necessity of detailed procedure for waste control on disposal facility place and/or on producer part and analyses for extend of repository is also required.
Requirement for independency between producer of waste and disposal facility is principal in the last group of these.

CONCLUSION

Mochovce repository operation safety beginning has been evaluated as a basic condition for acceptation of nuclear activities by public. Overworking of PSAR and better ensure of disposal facility stability and safety is required. It is possible to assume that the first radioactive waste will be accepted to dispose at 1997.

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RADIOACTIVE WASTES STORAGE IN ROMANIA

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ABSTRACT

The objective in wastes management in Romania is to ensure that there will no significant effects to man and environment from such wastes. In this paper storage and disposal for the long term of low and medium level radioactive wastes generated from diverse applications of nuclear material in industry, medicine and research in the repository from Baita-Bihor are described.

INTRODUCTION

The appearance and development of nuclear energetics together with the wide penetration of nuclear technologies and techniques in the economic and social life led to the on ever increased production of radioactive wastes.
These wastes are produced along the whole cycle of the nuclear fuel starting with the uranium extraction and ending with the used fuel reprocessing. An important problem is therefore to find the adequate methods to keep nuclear wastes away from

disturbing or affecting human activities. The most dangerous ones are the long lived radioisotopes.
The main purpose of nuclear wastes handling and processing is to protect the environment and the population by a continuous monitoring of the nuclear activities and their pollutants production.
Final deposit of radioactive wastes was subject of many papers and communications at international conferences in the field (1-5). A valid solution is far from being found, each case presenting its specificities.
The difficulty of the problem comes from the large amount of the actually produced nuclear wastes and from the large variety already applied solutions.
In order to protect the population and environment against nuclear radiation effects, in Romania has been prepared a repository in geological formations for long term storage of low and medium radioactive wastes.

SITE CHARACTERIZATION

The repository for the long term storage of low and medium level radioactive wastes has been sited in consideration of geological, hydrogeological, seismic, meteorological and mining studies at the premises of a currently extinct uranium mining enterprise Baita, Bihor county (6). The site is at 840 m above sea level and at geographical coordinates X=46020', Y=22023'.
Geologically, the site features crystalline rocks of high toughness, coming in bulky, compact formations that provide a solid ground and good vertical shielding. The rocky compound has a low porosity (<3%), good chemical homogeneity and impermeability and keeps these qualities over considerable horizontal and vertical spans.
The site is at about 8 km from the nearest community. Within its perimeter there is no shallow underground water and practically no risk of floods (8oo m height).
From the seismic standpoint, the zone's history is quite scarce in events. According to Romania's seismic map, the maximum expected intensity at the site would not exceed 6 (Mercalli).
The underground compound of the repository was sized for 20,000 standard containers, 200 l each. This would ensure a 30-year storage capacity for radioactive wastes originating from scientific research and nuclear techniques applications in industry, health care and biology (sources other than the nuclear plants).
The repository was committed in 1985. Proper approvals were secured at the time, from the institutions and bodies that had a legal say to this effect (the National Commission for Nuclear Activities Control-NCNAC, the Departments of Health and of the Environment, the Bihor County Authority and the Municipality of the local town-Nucet).
The geological environment and the mining opportunities technically approachable at the site would allow in principle an extension of the repository's capacity-if and when deemed appropriate-up to 150,000-200,000 standard containers.
In the year 1990 the repository at Baita-Bihor was inspected by an IAEA Expert Team under a WAMAP (Waste Management Advisory Programme) Mission in Romania. The Mission issued a favorable appraisal on the Repository's technical design and storage strategy implemented by the Operator-the Institute of Nuclear Physics and Engineering-INPE Bucharest. Recommendations were also made in regard to in situ tightening of the already deposited containers, geomembranes and upgrading of access routes.
It is of great importance to chose some optimal indigent sorts of natural ion changers that can be used as effective barriers against the migration of radionuclides deposited in natural geological formations. In this respect, the volcanic tuff and bentonites existent in a series of ore deposits from Romania were studied (7).
For the studied sorts of ion changers it turned out that, for the same working conditions (pH, mass to volume ratio, time of contact, etc.) the retention capacity has the following trend: 137Cs>60Co>51Cr>65Zn>131I>82Br with equilibration times ranging between 48 and 72 hours. An increase of the mass to volume ratio leads to a decrease of sorbtion-desorbtion processes. It was also observed that:

- the retention capacity of the studied ion changers is higher for cationic radionuclides (Zn2+, Cr3+, etc.) than for the anionic ones (Br-, I-),
- the higher the zeolites content in volcanic tuff and the sodium content in bentonites, the higher is the retention capacity of these indigent ion changers.

On the basis of these data, for the confinement of nuclear wastes were chosen the

volcanic tuff with 65% zeolites from Salaj region and bentonites with 90% sodium.

EFFECTS ON THE ENVIRONMENT

With the purpose of monitoring the radioactivity level in the protection zone of Baita-Bihor radioactive wastes repository, samples of soil, water and vegetation were prelevated and analyzed (8,9). The prelevation places and frequency were agreed with the local authority for environment protection.

The analysis of data gathered since the commissioning of the repository in 1985 shows that the radioactivity level is in limit of natural background determined before 1985 and that increases of radioactivity level due to the depositing activity were not observed in the zone of geological formation at Baita-Bihor.

The gamma spectra of the samples reveals the presence of U, Th and their descendants. Their presence is explained by the fact that the geological formations at Baita-Bihor were former uranium mine (10).

CONCLUSIONS

By realizing of this repository all the radioactive wastes coming from the nuclear units in Romania other than Cernavoda Nuclear Plant are collected and processed at the Radioactive Waste Treating Unit and finally deposited in the National Repository.

This measure leads to the diminishing of the risk of radioactive contamination in Romania due to technical and technological applications of nuclear energy in social and economical life.

The studies concerning the introduction in the technologic flux of radioactive wastes depositing of mineral absorbants that have the role of retaining possible uncontrolled migration of radionuclides from the deposited containers as well as for in situ consolidating of containers lead to the conclusion that using for this purpose bentonites and volcanic tuff has the benefit of diminishing the risk factors for the environment in the repository region in our particular case the geological formations at Baita-Bihor.

This fact was demonstrated also by the evolution of radioactivity level in the repository's surroundings as resulting from the analysis of soil, water and vegetation samples a long seven years of the mentioned repository.

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MANAGEMENT OF HIGH AND INTERMEDIATE LEVEL RADIOACTIVE WASTES IN INDIA

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ABSTRACT
In India, today, three types of Nuclear Reactors, i.e. research rectors with metallic fuel, power reactors with oxide fuel and a fast breeder test reactor with mixed carbide fuel, are in operation. Besides this, to attain self sufficiency, necessary facilities have also been established to cover all aspects of the nuclear fuel cycle including the back-end. Even though a wide spectrum of wastes are generated from these activities, and technics for their management have been developed, special attention was required on the design and development of immobilization process technology for the safe ultimate disposal of high level liquid wastes.
On the basis of indigenously developed matrices and processes, the first waste immobilization plant was constructed at Tarapur. This plant which is currently in operation, uses borosilicate glass as matrix in a pot melter for the immobilization of HL waste. A second plant at Trombay is in an advanced stage of completion wherein a calciner has been incorporated followed by a metallic melter. A third plant is under construction at Kalpakkam which not only incorporates some design changes based on feed back from operating plants but also has a modified process scheme using ceramic melter.
For the management of intermediate and low level liquid wastes, the process technology has already been successfully demonstrated and is in continuous use for over three decades now and has adequately met the regulatory requirements with respect to effluent discharges. The treatment techniques adopted include chemical precipitation, ion-exchange and evaporation.
Extensive research and development work is currently in hand on the development of various other matrices for the fixation of high level waste and for the separation of actinides from high level wastes. Result obtained in these studies are very encouraging. This paper deals in detail the processes and technologies adopted in the management of high and intermediate level radioactive wastes in India.

INTRODUCTION
In India, today, three types of nuclear reactors, i.e. research reactors with metallic fuel, power reactors with oxide fuel and a fast breeder test reactor with carbide fuel are in operation. Total installed capacity of nuclear power is about 2000 MWe. Besides reactors, to attain self sufficiency, necessary facilities have also been established to cover all aspects of nuclear fuel cycle including the back-end. Considerable expertise has been achieved in all these steps vis-a-vis uranium mining & milling, fuel fabrication, reactor technology, reprocessing technology etc. The success of nuclear technology lies in the safe management of various radioactive streams. Top priority has been accorded to waste management since the very inception of nuclear program. Annual waste generation rates from various stages of nuclear fuel cycle for generation of 1000 MWe are presented in Table I. The philosophy adopted for radioactive waste management has been to concentrate and contain as much radioactivity as possible and activity releases to environment are kept far below limits prescribed by regulatory bodies. Strategies adopted for management of high and intermediate level wastes are described in this paper.

MANAGEMENT OF HIGH LEVEL WASTE (HLW)
A three stage program has been devised for management of aqueous high level waste generated during reprocessing of spent fuel containing more than 99% of the radioactivity associated with entire nuclear fuel cycle. In the first stage the HLW is immobilized in inert solid matrices. A wide range of matrix formulations are developed to suit different waste characteristics, which include ceramics, titanates and vitreous (borosilicate) matrices. The borosilicate matrices selected for immobilization on industrial scale are presented in Table II. Some of the early HLW streams contain considerable amount of sulphates for which special lead borosilicate matrices have been developed.
Flowsheet adopted for first waste immobilization plant at Tarapur envisages concentration in thermosyphon evaporator by factor of two. The concentrated waste and inactive slurry containing glass formers are metered to an inconel process pot heated in a multizone induction furnace. Off gas clean up system consists of line cooler, condenser, wet scrubber, Ru adsorber, prefilter and HEPA filter.
It is well known that metallic melters have throughput limitations. The maximum energy consumption step in vitrification is evaporation. Hence, for the second immobilization plant being set up at Trombay, a modified flow sheet has been adopted

to isolate evaporation/calcination from glass melting. This is achieved by incorporation of a rotary calciner for HLW and dry frit feed system for inactive glass formers.
Operation/Design feed backs of Tarapur and Trombay plants were used while developing flowsheet for third immobilization plant to be set up at Kalpakkam. The simplified vitrification process flowsheet is presented in Fig. 1. In this plant joule heated ceramic melter will be employed in place of metallic melters.
A prototype ceramic melter being tested at Trombay is shown in Fig. 2. Indigeneously available fusecast AZS refractory has been employed as glass contact refractory. However for actual melter, crome bearing refractory will be employed.
For the first time acid recovery system has been planned with intention of reusing the decontaminated acid for reprocessing. This serves dual purpose of resource recycle and reduce waste generation. Unlike earlier plants, this plant has provision of treating in-house generated wastes and what goes out of the plant is within exempt limit for direct disposal.
In the second stage of HLW management, the vitrified high level waste is stored in stainless steel (SS 304L) canisters. Two such canisters after remote welding of caps are decontaminated and enclosed in secondary carbon steel overpack for storage in air cooled vault for interim period. A stack induced air cooled vault has been commissioned at Tarapur. Schematic of air cooled vault is shown in Fig. 3.
In the third stage, these high level waste canisters are planned to be disposed of in deep geological repositories. A conceptual design of such repository in deep granitic formation is shown in Fig. 4. Investigations for site selection and other thermomechanical, geohydrological, geochemical studies are going on.

MANAGEMENT OF LOW & INTERMEDIATE LEVEL WASTES (LLW & ILW)

For the management of low and intermediate level liquid wastes, the process technology has already been successfully demonstrated and is in continuous use for over three decades. Our earlier preferred matrix for fixation of intermediate level waste had been bitumen. Extensive R & D efforts have resulted in development of modified cement formulations with superior product characteristics. Cementation being simpler and safer process, future plants are being designed for fixation of intermediate level waste in modified cement matrices. As an alternate to cement matrix, a program is evolved for fixation in polymer matrices. A campaign had been undertaken in Tarapur, some time back wherein about million liters of low heat generating waste was immobilized in situ in polyester styrene matrix. Prior to immobilization, various treatments such as chemical precipitation, evaporation are imparted to reduce the volume of the ILW to be solidified.

RESEARCH AND DEVELOPMENT

Extensive R & D activities are also on hand with respect to long term durability of high level waste forms in view of the extended period of their isolation from biosphere. For vitrified waste products, temperature and radiation induced degradation are being evaluated by microstructural characterization. Alternate waste forms with improved long term durability such as synroc, ceramics are also being investigated.
Partitioning of actinides from HLW using newly developed solvents like CMPO is also under investigation as an alternate strategy. Studies on development of ion exchangers for selective removal of specific radionuclides like Cs137, Sr90 and Ru106 from liquid effluents are also being pursued.

CONCLUSION

Much of our future program on the treatment of waste will relate to high level and alpha bearing wastes. One of the problem areas to receive attention concerns the development of improved matrices for incorporation of high level waste, particularly in respect of long term stability of the solidified product, in the face of radiation and ageing. Another aspect of work to be undertaken in the field of radioactive waste management would be to develop techno-economically viable processes to deal with alpha bearing radionuclides present in the high level wastes; this work will assume greater significance in the near future when India's fast reactor program gets underway.
Our efforts in future will also be directed towards the demonstrations required, with field experimental data, to establish an adequate degree of confidence in deep geological disposal of vitrified products of high level wastes.
Our experience in managing low and intermediate level wastes has been good and it is expected that our present schemes of management of high level wastes will also be

a step forward in fulfilling our objectives.

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RADIOACTIVE WASTES IN THE REPUBLIC OF KAZAKSTAN AND SOME PRINCIPLES OF ITS ISOLATION

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ABSTRACT

Radioactive wastes (RW) in the Republic of Kazakstan are generated from the following sources:
- uranium mining and processing enterprises;
- raw material or wastes (from ore mining and processing enterprises) which are characterized by increased contents of natural radio nuclides
- locations of nuclear explosion tests;
- nuclear power plant and research reactors; and
- enterprises using sealed radioactive sources.

Enterprises of the Republic of Kazakstan have ability to store all RW located on its territory with safe isolation. After preliminary cooling, spent nuclear fuel is shipped to Russia for storage.
There are safety rules for handling RW, and now the legal system of regulations is being created. Safe storage of RW is provided by plant safety services, and will be supervised by national control bodies.
For the creation of a national system for RW disposal, three main approaches will be used. The experience of the uranium industry will be used and developed. Regional collection centers for RW of low and medium activity with shortlived radio nuclides (less than 30 years) will be created. Highly radioactive RW and RW with long lived radio nuclides (more than 30 years) will be stored in special storehouses until further decisions about their disposal are made.

INTRODUCTION

After the collapse of the USSR a sovereign state - the Republic of Kazakstan - was formed. This new state inherited on its territory portions of an economic complex which included processes which used radioactive materials. For disposal of radioactive wastes the complex included:
1) along with advanced systems of safety for handling RW , there are separate elements of a national system of collecting, transporting and disposal of RW;
2) there are rules for safe handling of RW and control of the execution of those rules, a legal base is being created and a state control system is being formed.
The project "Concept of RW disposal in the Republic of Kazakstan" and the laws of the Republic of Kazakstan "About handling with RW" are now being developed. Acceptance of these documents will be a legal base for the creation of a national management system with supervision in this area as well as creation of a national system of handling for RW. This article is written as a result of discussion of the mentioned documents.

RW SOURCES

The Republic of Kazakstan contains more than 230 millions of tons of radioactive wastes (RW) with an activity which exceeds 13 million Curies. They are characterized by origin:
- uranium mining and processing plants;
- ore mining and processing enterprises, raw material or wastes of which are characterized by increased contents of natural radio nuclides.
- locations of execution of nuclear explosions;
- nuclear power plant and research reactors;
- enterprises using sealed radioactive sources [1].

There are 529 declared locations of storage and disposal of RW on the territory of Republic. These locations include:
127 - in the uranium mining and processing plants;

76 - ore mining and processing enterprises;
16 - locations of nuclear explosions execution;
5 - in the nuclear facilities;
301 - in the plants involving sealed radioactive source production
According to the regulations of the territory of Kazakstan, all locations used for storage or disposal of RW should be registered and supervised.
CHARACTERISTICS OF SOURCES AND EXISTING FORMS OF RW ISOLATION
RW from uranium mining and processing plants
During prospecting, mining, and processing more than 219 million of tons of RW with activity more than 250 thousand Curie were created. RW from the uranium mining and processing industry have low activity levels, the most significant volume (93 % of the total RW volume) and are widespread on the territory of Kazakstan. Two storehouses were created in the plants of the uranium mining and processing industries for RW storage. They contain:
- rock wastes; and
- special surface storage of low level activity wastes (tail storages).

The project for tail storage does not provide for the construction of special engineering buildings. The isolation of the stored radionuclides is provided by choice of location (location with low level of ground waters), availability of waterproofing and anti erosive protection. The surface of the tail waste floor is covered by a layer of waterproof ground or special waterproofing cover. The surface of the tail waste is covered by road bitumen or other waterproofing. Above this surface, the tail store is covered by earth, the thickness is sufficient for protection from plant life root system penetration.
After the tails storage is filled and ready for reclamation, it should be made even, planed, and then covered by binding material (clay, etc.), a road bitumen layer, and a layer of clean earth (fill). Waste sites positioned far from populated places (not closer than 5 km) are not required to be made level.
At the present such forms of disposal are not unequivocally equated as the final form of safe disposal for RW. The locations of tail storages and recultivated waste are fenced in or are posted depending on the distances to populated areas and they remain under supervision. The supervising bodies impose restrictions on uses of these grounds.
RW from non uranium ore mining and processing enterprises
In Kazakstan, we have a series of deposits containing uranium. During mining assorted byproducts are extracted and often go to waste or tails. On some coal deposits, the top parts of the coal layer are accompanied by uranium radionuclides. This coal is not used as fuel and is subject to storage as RW. In the oil industry there are also present soils, equipment, and pipes which are polluted by natural radionuclides. The RW weight of this group is 1.57 million tons with an activity total of 519 Curie. These wastes (98 %) are formed by the oil trade. It stores them on specially allocated platforms under the supervision of the enterprises which produced them.
RW from nuclear explosions
The RW produced from nuclear explosions is estimated to be 12 million tons with an activity of 12,9 million Curie. Of this total, 6.5 million of tons is attributed to underground explosions. Wastes in cavities of explosions classified as medium activity and wastes on the earth's surface classified as low activity add to activities of 11.6 thousand Curie, they consist of less than 0.1 % of the RW activities of this group. Though the wastes of underground explosions are isolated in melted rocks, it now should not be unequivocally equated to their final safe disposal. It seems that the locations of nuclear tests will be restricted areas, supervised and investigated for a long time. RW from nuclear explosions are located within the limits of nuclear test site areas and are under the supervision of government agencies.
RW from nuclear power plant and research reactors
On the territory of the Republic of Kazakstan, we have one power nuclear plant of the BN-350 type in Aktau and four nuclear research reactors (three in Kurchatov and one in Almaty).
The weight of medium activity RW of this group is about 9 thousand tons with an activity of 26.6 thousand Curie. The weight of low activity RW is about 3 thousand tons with an activity of 50 Curie. The spent nuclear fuel departs for storage in Russia after preliminary cooling with subsequent reprocessing in Russia. RW from

regeneration of spent fuel will be returned in the Republic of Kazakstan. Besides that by 2005 occurrence of high active RW is expected as a result of scheduled decommission of power nuclear plant BN-350. All of these nuclear facilities are supplied with special storehouses for average and low activity RW. These storage facilities are surface concrete buildings with technological capacities for RW storage.

RW from enterprises using sealed source production

There are to 100 thousand assorted sealed sources in Kazakstan. Annually more than 10 thousand pieces go to disposal. For example in 1991, enterprises in Kazakstan have sent to disposal 11603 pieces of sealed sources with an activity of 1274 Curie. Surplus declarations on the first of January 1992 identified 11362 sealed sources with an activity of 32389 Curie.

According to regulations of the Republic of Kazakstan, enterprises do not have the right to store spent sealed sources and are obliged to declare locations of permanent disposal. Due to administrative reasons in 1992 the work on utilization of spent sealed sources were not executed according to schedule.

CONDITIONS OF NATIONAL SYSTEM OF COLLECTING, TRANSPORTATION AND DISPOSAL OF RW

In the former USSR there were specialized enterprises which worked with RW, nowadays they belong to Russia. The Republic of Kazakstan has not had regional points of disposal. Till 1992 the work was conducted by specialized enterprises of the Russian Federation. After the Russian Presidential Decree which prohibits disposal of non-Russian RW on the territory of Russia, disposal of Kazakstan's RW was stopped (mainly, work on disposal of spent sealed sources).

The special storehouse in Kurchatov is now ready for action for long-term storage of spent sealed sources from the enterprises of the Republic of Kazakstan (it will be the regional point of storage). The storehouse has surface concrete buildings with special technological capacities for storage of RW. It is suitable for low and medium activity RW, and has special chambers for transhipment and identification. It allows safe storage of all spent sealed sources from enterprises of the Republic for a long period of time (100 years).

Part of the work dealing with collecting and transportation of RW can be performed by special teams from various enterprises in Kazakstan. However, there are some needs, for example recharging of apparatuses in oncologic institutes, where sources with residual intensity to several thousands Curie. These needs will be addressed by personnel from Russia under contract to industries of Kazakstan.

MAINTENANCE OF SAFE HANDLING WITH RW

The main supervising bodies in the national system are:

1) Bodies of sanepidiomilogical control (BSC) at the Ministry of Health - these supervise the observance of rules and norms of radiation safety for the protection of life and health of population and staff.

2) Ministry of Ecology and Bioresources - this national body of management and control of the environment gives permission for RW disposal.

3) Atomic Energy Agency (AEA) - executes the supervision of nuclear safety and the licensing of all kinds of nuclear work, including work on handling with RW.

Safe handling of RW is provided by safety services of each enterprise. These services are obliged to guarantee the safety of staff at execution of work and completely exclude irradiation of the general population. Moreover, the enterprise executes tracking for radiation conditions of object and around it. For these purposes health physics devices are used which measure fields gamma-radiation. For supervision of the conditions of underground water, measurements are performed around the objects in drilled boreholes. The quality of this work is supervised by BSC, the Ministry of Ecology and Bioresources and the AEA pursuant to the authorities.

LEGAL BASE

There are nuclear and radiation safety rules and separate rules of handling for RW on the territory of the Republic. At the same time, the legislative base for the handling of RW is absent. Till acceptance of appropriate laws, activities proceed under the "Temporary regulation about use of nuclear energy, nuclear activity, handling with radioactive wastes and spent nuclear materials, maintenance of radiation safety of population of the Republic of Kazakstan". This temporary act was authorized on April 1994 by decree of Cabinet of Ministers of the Republic of Kazakstan.

At the present, a project of the law "About handling with radioactive wastes" has

also been created under temporary status. This project established priority of protection of person and environment and obligatory indemnification of damage caused by RW. The following principle is used -whoever made the pollution should pay restoration money. This project fixed procedures of control and established the creation of a new body of national management for the handling of RW. This provided for separate functions of state management and state supervision. This project requires the ecological examinations during the creation of disposal points of RW or at the creation of large waste producing enterprises. The state accounting of all RW of the Republic will be established, including final disposal. The current law provides for the refusal to process spent nuclear fuel.

APPROACHES TO CREATION OF THE NATIONAL SYSTEM OF RW ISOLATION

In the future the following groups of RW can be allocated:

1) RW from uranium and non uranium mining and processing plants;
2) RW of low and medium activities with shortliving radio nuclides (less than 30 years);
3) special class of RW with long living radio nuclides (more than 30 years) and high active RW.

For RW isolation of the first group we will rely on the experience of our uranium mining and processing industry. Earlier , we described the principles of storage and disposal of RW of this branch. Probably technologies of disposal of these wastes will be developed in the direction of optimum choice of material and thickness of bottom cover and surface cover and stabilization of surfaces. And the procedures of supervision and control will be amplified and extended. It is not decided yet how long land used for RW disposal of this group will be excluded for human activity. It is possible, that with development of novel technologies for RW disposal, additional measures of protection will be needed or on the contrary the total safety of these disposals will be recognized.

The creation of regional points of disposal on the territory of the Republic of Kazakstan is under discussion. These sites could meet requirements for the final disposal of RW of the second group. After RW disposal, these sites will be not used for human activity for approximately 300 years. The sites will be observed, and final disposal of RW of this group will not present significant danger. The creation of such waste storages requires in the design stage a combination of natural and engineering barriers capable to supply reliable isolation of radio nuclides during waste storage operation and its supervision after disposal. Till creation of regional waste storages, wastes of this group will be stored in RW special storehouses of enterprises. All spent sealed sources will be stored in a regional storehouse of spent sealed sources in Kurchatov.

The decisions about RW disposal of the third group of RW is not under discussion and planning at the present time. In the interim, they are in special storehouses. There is a task of designing safe storage for high active RW which will need resolution after 10-15 years.

There is a well known necessity for development of RW processing before disposal. Currently, hardening of liquid RW for disposal is prohibited. Technologies enabling facilities to lower the volumes of RW and the inclusion of radio nuclides in matrix for decrease of their mobility are of interest. There is no experience in these technologies in RW disposal in the Republic of Kazakstan yet.

CONCLUSION

The Republic of Kazakstan has conditions for the safe and secure storage of RW containing low and medium activity levels. The problem of utilization of spent sealed sources will be solved by entering into agreements about the operation of a regional storehouse of spent sealed sources in Kurchatov.

After 10-15 years a solution for high active RW will be needed. For now, no safe permanent storages are created.

The Republic does not have facilities for storage of spent nuclear fuel. After preliminary cooling it departs to Russia. The processing of spent nuclear fuel will be made by facilities in Russia. RW from processing will be returned to Kazakstan. Now the future development of power in the Republic of Kazakstan is under discussion. Hereinafter, an increase of nuclear technologies will appear to require independent storage of spent nuclear fuel and its processing. The project of the law of the Republic of Kazakstan "About handling with RW" provides the opportunity to refuse the reprocessing of spent nuclear fuel on a temporary basis. To discuss these problems is premature until other issues are resolved.

The development of disposal technologies is required for good safety practices. Appropriate modern requirements for the final disposal of RW will follow. The development of a scientific base which designs or introduces new disposal technologies for RW is necessary and a scientific evaluation of the safety of existing are required simultaneously. At the present time, the Republic of Kazakstan begins with control and documentation of all existing locations for the storage and disposal of RW.

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IDENTIFICATION AND MEASUREMENT OF PERMANENT GASES AND VOLATILE ORGANIC COMPOUNDS IN HANFORD WASTE TANK HEADSPACES

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ABSTRACT

One of the prominent environmental and safety issues facing the Hanford cleanup effort is the highly radioactive wastes stored in 177 underground tanks. Before waste retrieval and long-term isolation efforts can be initiated, safety issues, including the flammability and toxicity of tank-headspace vapors and gases, must be addressed. While other species can contribute to flammability and toxicity, this paper describes only the analysis of the organic vapors and permanent gases in the tanks. Gas and vapor concentrations exceeding allowed flammability and/or toxicity limits may potentially be caused by 1) evaporation of semivolatile organic wastes, 2) generation of organic vapors by radiolytic and chemical degradation of organic wastes, and 3) generation of inorganic gases and vapors, such as hydrogen and nitrous oxide, by the waste.

To support Hanford tank vapor characterization, Pacific Northwest Laboratory set up a dedicated laboratory to identify and quantify permanent gases and volatile organic compounds in tank headspaces. Whole air samples are collected in SUMMATM passivated stainless steel canisters using two described methods. The variety of polar and nonpolar organic compounds found in tank-headspace samples required that modifications be made to the U.S. Environmental Protection Agency compendium method TO-14. This modified TO-14 method is now routinely applied to the analysis of tank-headspace samples. Roughly 200 organic compounds and permanent gases have been identified and quantified in the tank-headspace samples analyzed to date. Some compounds found in tank headspaces, such as the semivol atile normal paraffin hydrocarbons and tributyl phosphate, were constituents of waste sent to the tanks. The majority of volatile compounds found in the tank headspaces, including the homologous series 3 to 10 carbon alkanes, alkenes, alcohols, aldehydes, ketones, and alkylnitriles, are thought to be chemical and/or radiolytic degradation products of the semivolatile normal paraffin hydrocarbons and tributyl phosphate. Nitrous oxide and hydrogen are typically the most abundant species found in SUMMATM canister analysis. Nitrated organic compounds have also been found at low concentrations.

INTRODUCTION

Bounded on two sides by the Columbia River, the Hanford Site covers 560 square miles of southeastern Washington State. Between 1944 and 1988, Hanford supplied weapons-grade plutonium to United States national security efforts. After five decades of service, the end of the Cold War brought the plutonium-production mission to a close, leaving a legacy of environmental problems.

One of the most prominent environmental and safety issues facing Hanford cleanup efforts is the highly radioactive wastes stored in underground tanks. About 60 million gallons of high-level waste and 260 million curies of radioactivity are stored in 177 tanks that hold between 55,000 and 1.1 million gallons each. The long-term strategy is to reduce the volume of tank wastes, remove organic constituents, and separate remaining wastes into low- and high-level components for permanent isolation.

Several safety issues of immediate concern must be addressed before waste retrieval and long-term isolation can begin. These issues include the potential flammability and toxicity of waste-generated hydrogen, nitrous oxide, and organic vapors in the tank headspaces. Tools to characterize these tanks are available, but are incomplete (1). We have evaluated available methods and chose to modify them from U.S. Environmental Protection Agency (EPA) TO-14 to meet the needs of this program.

Westinghouse Hanford Company (WHC) directs the Tank Characterization Program, which conducts the vapor sampling and contracts sample analysis. WHC is responsible for sampling-methods development and sample collection. Pacific Northwest Laboratory (PNL), Oak Ridge National Laboratory, Sandia National Laboratories, and Oregon Graduate Institute of Science and Technology have assisted the program in the past year by developing and applying analytical methods. In this effort, PNL set up a dedicated laboratory to identify and quantify permanent gases and volatile organic compounds in tank headspaces sampled with SUMMATM passivated stainless steel canisters (2). The goal of this laboratory is to provide timely identification and quantification of permanent gases and volatile organic compounds in tank headspaces.

While it would be desirable to use only established methods to analyze the tank gases and vapors, the broad range of analytes found in the tank headspaces precludes this. For example, U.S. Environmental Protection Agency (EPA) compendium method TO-14 was developed and validated to analyze 40 volatile organic compounds in ambient air (3). This method is based on the use of SUMMATM passivated stainless steel canisters to collect whole air samples. Samples are analyzed by using a NafionTM dryer to dewater the sample, followed by cryogenic preconcentration and gas chromatography/mass spectrometry (GC/MS) analysis. The following issues have limited the direct application of this method to the Hanford waste tanks headspace vapor analysis:

- the TO-14 method was originally designed for ambient and not source-level sample analysis;
- of the 40 compounds that have been validated by the TO-14 method, only a few have been found in the tanks sampled to date;
- the NafionTM dryer used for sample dewatering in the TO-14 method will also scavenge many of the polar oxygenated compounds known to be present in the tanks.

We have successfully applied new vapor-analysis methods for analyzing tank headspaces over the last year (4). This paper provides a review of the tank-headspace sampling and analysis methods developed under this program and a summary of data derived from the application of these methods to various Hanford waste tanks. Finally, the composition of headspace vapors is explained in terms of known liquid waste streams and radiolytic decomposition processes.

TANK-HEADSPACE SAMPLING

Two techniques are used at the Hanford site to collect vapor samples from the waste tanks. The first method, in situ sampling (ISS), is a relatively simple technique in which sorbent traps attached to plastic tubing are lowered into the tank headspace and a measured amount of headspace air is pulled through them. Permanent gas, organic gases, and volatile organic vapors samples are collected in SUMMATM passivated stainless steel canisters via open-ended teflon tubing from the same headspace location. The second method transports headspace air, gases, and vapors via heated tubing to a remote sampling manifold. The heated sampling probe inside the tank headspace, the transfer tubing, the sampling manifold, and all associated pressure, flow, and temperature monitoring instrumentation comprise the vapor sampling system (VSS). Huckaby et al. describe the application of both the ISS and VSS methods to resolve vapor issues with tank 241-C-103 at the Hanford Site (5)

In Situ Sampling System

The ISS system is a site-developed method used to collect tank-headspace vapors from high-level waste tanks. It consists of a weighted sampling head connected to unheated flexible tubes, rotometer flow meters, needle valves, and a vacuum pump. The sampling head, in which the sorbent traps and an open tube for the collection of

SUMMATM canisters reside, is lowered into the tank headspace, as depicted in Fig. 1. After the open tube has been well purged with tank air, the vacuum pump is isolated, and samples are collected into evacuated SUMMATM canisters at the surface. The ISS equipment is constructed from commercially available components and mounted on a 2-wheeled hand cart. The advantage of this system is its simplicity and cost, allowing several of these generally robust sampling systems to be operational at any one time.

Fig. 1. Illustration of Waste Tank Sampling Methods Showing the Mobile Vapor Sampling System (VSS) and the In Situ Sampling System (ISS). In the VSS, SUMMATM and sorbent tubes as well as mass flow controllers and vacuum pumps are located and/or controlled inside the tanks.

Vapor Sampling System

The VSS consists of a mobile laboratory, a hot-water-jacketed stainless steel probe that is inserted into the tank headspace, and heated transfer tubing that connects the mobile laboratory with the probe. Sample air is drawn from the tank headspace and through the transfer tubing and sampling manifold by an air pump. Air flow within the VSS is directed by electrically activated, pneumatically actuated valves. Instrumentation housed in the mobile laboratory provides system temperature monitoring and controlling, absolute and differential system pressure monitoring, mass flow metering and controlling, and hydrocarbon vapor monitoring. Mahon et al. describe the VSS, its performance, and its operation in detail (6).

A key feature of the VSS is its use of heated sample transfer tubing and a heated sampling manifold. Maintaining the system temperature at an electronically controlled, elevated temperature prevents vapor condensation and reduces vapor adsorption on surfaces exposed to sample air. Mahon et. al describe various tests and observations that indicate the VSS sample transfer efficiency is consistently high (6).

The VSS sampling manifold has two sampling stations for sample collection. One station is set up for collecting tank vapor samples using evacuated SUMMATM canisters. The second station is equipped for collecting vapor samples on a variety of sorbent traps. To prevent condensation and to minimize adsorption of vapors during SUMMATM canister sampling, the connection tubing between the sampling manifold and the canister itself, as well as the mechanical valve on the SUMMATM canister, are heated with electrical heat tape. Mass flow controllers are used to measure and control the sample flow rate for sorbent-trap sample collection. These are located downstream of the sorbent-trap station and downstream of in-line dryers that remove water vapor from the air before it is metered. The sorbent trap station allows collection of two samples simultaneously, and most samples are taken in pairs.

TANK-HEADSPACE ANALYSIS

The PNL Vapor Analysis Laboratory currently analyzes SUMMATM samples for permanent gases and volatile organic hydrocarbons. An additional method for the determination of total nonmethane organic compounds (NMOC) has been developed. The total NMOC method will be used in the future to screen samples and determine if additional speciation and quantification of the volatile organic constituents is necessary. Figure 2 provides the SUMMATM analysis scheme from sample media preparation through sample analysis.

Fig. 2. SUMMATM Analysis Scheme for Permanent Gases, Total NMOC, and Volatile/Semivolatile Organic Compounds.

Permanent Gas Analysis

Several permanent gases are quantified in tank samples. These include hydrogen, carbon monoxide, carbon dioxide, methane, and nitrous oxide. The analysis method, developed using a gas chromatograph-thermal conductivity detector instrument {Hewlett Packard (HP) 5890 series II GC-TCD} with gas sampling and column switching valves, is typically capable of detecting analyte gases to 80 ppmv. This instrument has been recently modified with the addition of a mass selective detector (HP 5972 MSD) that allows mass-spectral identification of analyte gases and reduces detection limits to about 10 ppmv for carbon monoxide, carbon dioxide, methane, and nitrous oxide. The hydrogen detection limit is reduced to about 40 ppmv. Gases are sampled directly from the SUMMATM canister using a syringe to withdraw a 5-mL aliquot. No preliminary sample preparation is necessary for this analysis. A 1-mL gas-sampling loop is subsequently filled from the syringe and injected directly into the GC-TCD. The individual gases are separated on a 25 m x 0.5 mm PoraPLOT Q coupled with a 5

MolSieve Plot column. A 10 m x 0.53 mm HP-1 is used as a blackflushed sample precolumn to eliminate potential interference and contamination from the volatile and semivolatile organic compounds also present in tank vapor samples.

Total Nonmethane Organic Compound Analysis

In addition to the permanent gas and speciated organic compound analysis methods, we have recently configured one of our cryogenic preconcentration systems to determine the total NMOC concentration in a tank-headspace vapor sample. The method is adapted from the EPA Compendium Method TO-12, (3) and was optimized for samples in which significant quantities of semivolatile organic compounds, such as normal paraffin hydrocarbons (NPH) are found. The use of total NMOC concentration data as a screening mechanism can substantially reduce the cost and complexity of laboratory analysis and reporting through eliminating the speciation and quantification stages used in our modified TO-14 analysis. This method employs an EnTech 7000 cryoconcentrator system coupled to an HP 5890 gas chromatograph equipped with a flame ionization detector. The cryoconcentrator is coupled with a 16 position SUMMATM canister autosampler for automated analysis. Canisters, transfer lines, and gas sampling valves are heated to 120C to ensure recovery of polar and semivolatile compounds. Compounds are quantified by using a multilevel calibration curve with a propane standard reference material from the National Institute for Standards and Technology (NIST) as recommended by EPA method TO-12.

Volatile/Semivolatile Organic Compound Analysis

The PNL Tank Headspace Analysis Laboratory uses methods derived from the EPA Compendium Method TO-14 (3). The TO-14 method was originally developed and validated by the EPA to analyze hazardous VOCs in SUMMATM-collected ambient air samples. The method has EPA approval for 40 organic compounds that have been successfully collected in pressurized canisters and subsequently quantified at the parts per billion by volume (ppbv) level. Several instrument and method modifications were required to apply this technique to Hanford waste tank-headspace analysis. These modifications have allowed us to overcome analytical problems associated with the high concentrations of semivolatile organic compounds typically found in tank vapor samples and allow us to analyze a large number of targeted polar constituents.

The PNL analysis method employs an EnTech 7000 cryoconcentrator and an HP 5890/5972 GC/MS system for tank-constituent speciation and quantification. Each preconcentration/analysis system is fitted with a 16-canister autosampler, allowing the system to operate overnight without intervention of the bench scientist. The autosamplers meter in analytical standards, internal standards, and samples, preflushing and purging sample lines and traps as required. During analysis, the SUMMATM canisters are heated to 120C in specially constructed heating mantles to ensure recovery of polar and semivolatile compounds. The cryogenic concentration/sample dewatering front end operates on the principal of a microscale purge-and-trap system. Using a cryogenically cooled glass bead trap, sequentially followed by a Tenax adsorbent trap, this technique successfully dewaters the sample while allowing relatively polar compounds such as ketones, alcohols, aldehydes, nitriles, and nitrates to be concentrated from the air volume and subsequently analyzed. All transfer lines and sample contact surfaces are heated to 120C and the SILCOSTEELTM (Restek Corp.) is treated to minimize sample sorption. After the preconcentration stage, the organic constituents are separated on a 60 m x 0.32 mm J and W DB-1 capillary column with a 3 m liquid-phase coating. Target analysis compounds include the 40 VOCs listed in the TO-14 method (3) and 15 additional compounds selected by a toxicology review panel as being representative of tank-related compounds. Standards are prepared by diluting certified gas standards and/or using a Kin-Tech permeation tube standard generation system.

Compounds detected in tank vapors that are not on the target analysis list are tentatively identified by standard GC/MS analysis. These are further quantified through comparison with the response factor of the nearest eluting internal standard. An experienced mass spectrometrist reviews the mass spectrum of the unknowns using conventional mass spectral interpretation techniques and visual/computer comparison with the NIST/EPA/NIH (65,000 spectra) and Wiley (120,000 spectra) mass spectral libraries. If an unknown cannot be tentatively identified by conventional 70 eV electron ionization GC/MS, chemical ionization is used to obtain additional molecular-weight information. In addition, the PNL Tank Safety and Characterization programs have a wide variety of analytical tools to further the identification process if more information is required. These include a GC/atomic

emission detector, a four-sector high-resolution mass spectrometer with numerous ionization and MS/MS capabilities, a Fourier transform infrared (FTIR), and a Laser-Raman FTIR.

Laboratory Data Integration

Analyzing SUMMATM samples for TO-14 list organic compounds, additional Hanford tank-headspace analytes, and permanent gases is a highly automated process. Using an HP-UX chemserver workstation has enhanced productivity and data quality in several ways. First, the use of Thru-Put System's Target3 and ReportWriter software permits a high throughput of complex data analysis, meeting contract laboratory program (CLP)-type reporting and quality assurance/quality control (QA/QC) requirements. Production control is maintained by evaluating calibrations and sample analysis immediately after the GC/MS data are available. Calibrated compound data are processed using software that was developed and validated for reporting TO-14 gas-analysis results. Final concentrations are based on the volumes analyzed, dilution factors used, and molecular weights of each of the target analytes. All calculations are made by the data system, minimizing transcriptions and the associated potential for transcription errors. Molecular weights for non-target compounds are taken directly from the NBS mass-spectral database and used in the final calculation of concentrations when expressed by volume. Calibrations are performed directly on-column, which permits direct evaluation of whether a sample analysis falls within the calibration range. Data are fully auditable from sample preparation and analysis to reporting. The system tabulates both target and non-target tentatively identified compound (TIC) data , eliminating the potential for errors due to compiling and transcribing multiple sample-results data. Data for TO-14, TO-12, and permanent-gas analysis are processed and archived in a consistent and systematic manner.

RESULTS AND DISCUSSION

The headspaces of over 30 tanks have been sampled in the past year with the two described methods. The advantages of the ISS method include 1) it is relatively simple, 2) it consists of low-cost hardware that can be moved quickly from site-to-site, and 3) transfer tubing wall adsorption of constituents is eliminated for sorbent trap samples. Sampling setup and tear-down time are relatively short. One disadvantage is that unheated sample lines are used to bring the sample from the tank dome up to the surface for collection into evacuated SUMMATM canisters. Depending on the difference between the tank-headspace temperature and the ambient temperature during sampling, cold trapping of higher molecular weight compounds can influence results. This is generally not as severe a problem during summer months, but it is a concern during the colder winter months. Another disadvantage is that in the currently used configuration, flow regulation and measurements are somewhat crude, and there is no automated data logging during sample collection. Current plans are to correct some of these deficiencies by adding field-stable mass-flow meters and flow totalizers.

The advantages of the VSS are that it provides high quality samples with tight control over hardware performance, sample manifold heating, sample-collection flow rates, and sample-collection time. Much of the sampling process can be automated and recorded through integrated process-control/data-logging technology. Potential cold trapping zones have been eliminated through carefully heating all areas of the sampling manifold and sample-transfer line. The system is less susceptible to outside temperature fluctuations as it is housed in a climate-controlled vehicle. It also permits the use of limited real-time analysis techniques. However, the sophistication of this system does lead to some disadvantages, such as relatively time consuming setup, tear down, and cleanup stages. Extended equilibration times are also required to ensure that all adsorptive surfaces are in equilibrium with the tank vapors, and thus, staff members must spend long hours manning the sampling stations. These sampling times are of special concern when staff are required to dress out in full sets of radiation-protection clothing.

The analysis methods described were applied to vapor samples collected from over 20 different tanks representing 4 different tank farms. The initial focus was to develop and apply the methods for the speciation of the organic constituents; we have recently added the permanent gas and total NMOC methods. Over 200 organic compounds and several permanent gases have been identified and quantified in the tank-headspace samples analyzed to date. While a complete evaluation of these results is outside the scope of this paper, we would like to summarize the results

from tanks analyzed. Table I presents a overview of the classes of compounds that have been identified and the observed concentration ranges.
The chemical composition of tank headspaces is directly related to both recent and historical usage. Tank farms contain groups of tanks that are regionally located in conjunction with various facilities associated with different stages of the plutonium production, extraction, and finishing processes. Tanks in a particular farm have similar histories and therefore often share similarities in the composition of stored wastes. This appears to be especially true in the cases where tanks are linked together through cascading overflows. Tank vapors are generally comprised of varying concentrations of a homologous series of alkanes, alkenes, alcohols, aldehydes, ketones, and alkylnitriles. We have observed that these series often extend from C3 to C10. It is possible that other homologues are also present but not observed under current procedures. Nitrated organic compounds are also found, but generally at sub parts per million by volume (ppmv) range. Nitrous oxide and hydrogen are consistently above ambient levels.
Compounds found in tank headspaces are present because they were constituents of waste from plutonium-extraction processes or because they are formed through the oxidative and reductive radiolytic decomposition of process materials (7,8). The concentration of many organic compounds, especially the n-alkanes, can range over 4 or more orders of magnitude when comparing one tank to another. This can be explained by differences in the degree of venting, the amount of radioactivity in tank wastes changing the rate of waste aging, and the presence or absence of floating organic liquid layers. The semivolatile NPHs, ranging from C11 to C15, are examples of common extraction-process parent materials. The NPHs were constituents of diluents used with tributyl phosphate (TBP) to extract plutonium from spent uranium fuel. DecalinTM, which is a mixture of decahydronaphthalene and substituted decahydronaphthalenes, was also used as a TBP diluent. Both NPH and DecalinTM, as well as many of their radiolytic decomposition products, have been found in several tanks. Butanol, the main volatile decomposition product of TBP under alkaline conditions, is usually also found at relatively high concentrations when present.
Our laboratory has successfully analyzed vapors from Hanford waste tanks. The approach described in this paper describes tools used to characterize these vapors. The methods used have allowed flammability and toxicity issues to be resolved. The data from these analyses have helped in classifying and evaluating the Hanford tanks. This is a small but significant step toward complete tank waste characterization. Vapor sampling and analysis have been relatively easy and inexpensive when compared to the sampling and analysis of liquid/solid waste samples from these tanks. This difference is largely due to the high radioactivity and complexity of the wastes.
The procedures followed in our laboratory are well suited to Hanford waste tank analysis. These waste tanks are typically warm (ambient to 100C) because of the radioactivity of the constituents. The warmth allows semivolatile components to become vaporized and available for release to the atmosphere. Thus, both volatile and semivolatile components can contribute to flammability and toxicity. Because many of the standard methods available for sampling and analyzing organic vapor components are tailored to volatile components, we have added appropriate modifications to include both semivolatile and polar compounds.

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17-54

THE DEVELOPMENT OF NUCLEAR WASTE VITRIFICATION IN FRANCE

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ABSTRACT
The conversion of an ultimate waste form into a stable, inert product is a beneficial operation, especially in the case of potentially toxic waste materials. One widely recognized approach today is vitrification, in which a glass or crystalline material is fabricated from a particular waste composition. This is a high-temperature process involving the use of selected additives - notably silica - to form a glass network. The waste is thus contained in a stable, inert and nontoxic material suitable for safe disposal; the process also generally results in a significant volume reduction that has a decisive effect on disposal costs.
The nuclear industry was a forerunner in this area. Research on containment of uranium fission products began in 1957, and has been extensively developed in France, where industrial vitrification facilities are now operating at Marcoule and La Hague. Glass is now a universally recognized containment medium.
Research is continuing in France, not only to enhance the quality of the matrix and increase production capacities, but also to extend the process to low and intermediate level radioactive waste produced in nuclear power plants. New melting equipment has been designed to implement a "cold crucible" melting technique, in which glass is heated by induction inside a cooler solidified layer of the same material. Recent work by SGN, an engineering subsidiary of the CEA and COGEMA, substantiates the technical and economic advantages of vitrifying this type of waste.

INTRODUCTION

Nuclear power today represents a significant fraction of the world's energy balance. It accounts for about 17% of the electricity generated worldwide, including 35% in the European Community and over 75% in France.
Nuclear power produces practically no gaseous wastes that would represent an environmental hazard, resulting in acid rain or affecting the climate. This does not mean that it produces no waste at all, however, but in fact it produces very little. Nuclear wastes are separated from fissionable material by fuel reprocessing, and may then be confined in a suitable containment matrix. In France, for example, 99% of the waste produced by a nuclear reactor consuming 35 tons of fuel each year can be contained in less than 4 m3 of glass, a stable, insoluble matrix. Fuel reprocessing and waste containment operations are performed in France in the UP2 and UP3 plants operated by COGEMA at La Hague, on the English Channel.

THE NUCLEAR WASTE PROBLEM

Most nuclear waste - the 99% mentioned above - are present in concentrated high-level solutions containing uranium fission products. The remaining 1% include low and intermediate-level liquid waste, ion exchange resins used for liquid decontamination, and incinerable waste.

Processes are now available to treat all types of waste. Fission product solutions are vitrified to form stable, compact and insoluble glass packages. Other conditioning processes are also used for lower-level waste, generally bituminization or cement encapsulation, although these processes involve a weight and volume penalty.

The Commissariat l'Energie Atomique (CEA), together with the Socit Gnrale pour les Techniques Nouvelles (SGN), is currently investigating new enhanced techniques, notably high-temperature processes: incineration combustible waste materials, followed by ash melting, for example, to achieve a significant volume reduction and improved containment quality; or melting of incombustible materials to obtain the most stable oxide form.

EXISTING TECHNIQUES

Vitrification

Vitrification produces a glass or vitrocrystalline material from a specific waste composition to obtain a stable, compact solid waste form that ensures radionuclide containment and mechanical strength, and meets applicable standards for handling, interim storage and final disposal. Since the first nuclear application in France in 1957, vitrification has been adopted world-wide for containment of high-level radioactive waste; today in France and other industrialized countries, the scope of this technique is being extended to other waste categories, including toxic industrial waste.

Vitrification is now implemented at industrial scale to solidify concentrated fission product solutions produced by reprocessing spent reactor fuel. A pressurized water reactor with an electric power rating of 1300 MW and a nominal burnup of 33,000 MWd.t-1 consumes 35 metric tons of fuel annually, producing 1200 kg of fission products in the form of concentrated nitric acid solutions containing precipitates. France and the United Kingdom are the only countries in the world where such solutions are now vitrified industrially. The plants operated by COGEMA at La Hague in France, and by BNFL at Sellafield in the UK both implement the French two-step continuous vitrification product to convert the feed solution into a stable material of significantly smaller volume.

The process has been widely discussed in the literature (1,2). A rotating kiln first evaporates the solution and calcines the dry residue; the calcinate is then heated to 1150C in an induction-heated metal melter together with glass frit and suitable additives to obtain a glass melt of the desired composition, which is cast at regular intervals into a metal canister. The calcining step also generates dust that is trapped and recycled in the process by a scrubbing column, making the process particularly clean and simplifying the off-gas treatment system (which includes simply a condenser, a nitrogen oxide absorption column and an absolute filter). All the process equipment is maintained under slight negative pressure.

By the end of 1994, the three French vitrification units had solidified over 4000 m3 of solutions in nearly 2000 metric tons of glass, filling more than 5000 canisters and isolating over 1.5 x 109 curies (Table I). Vitrification has thus largely proved its capacity to dispose of concentrated fission product solutions. In the near future, the process will probably also be applied to other types of low or intermediate-level waste.

Incineration

Another waste treatment and conditioning technique is incineration. Although currently implemented industrially to very limited extent in Europe and in the USA, it is increasingly seen as an advantageous alternative to encapsulation and other low-temperature conditioning processes.

The CEA has developed a process designed to incinerate highly contaminated combustible-bearing waste, i.e. containing large amounts of plutonium (3,4). This is a two-step process involving pyrolysis of previously ground waste material at 550C under argon to form pitch, which is then calcined into ashes at 900C in oxygen-enriched atmosphere. Both steps are carried out in resistance-heated rotating tubular kilns. A third step, which could be performed on-line, is vitrification of the ashes to reduce the waste volume ten-fold, by increasing the specific gravity of

the material from 0.25 (ashes) to 2.5 (vitrified waste form).
The off-gas treatment system naturally includes a preliminary afterburning step in a vast, resistance heated refractory chamber; the gas stream is then cooled, preferably by dilution in air, water quenching or in a simple heat exchanger, before filtration by bag filters or electrostatic filters specially designed for the low process flow rates, and finally by absolute filters. The off-gas treatment also comprises a dechlorination treatment and maintains the system under negative pressure.
The incineration process is suitable for highly -contaminated waste, which is generally produced in small quantities; the design capacity is therefore limited accordingly. A nonradioactive solid waste incineration research prototype (IRIS) has been operating for several years at Marcoule with a throughput of 4 kg.h-1. Two industrial facilities have already been planned: one, with a capacity of 7 kg.h-1, is nearing completion and is scheduled to begin operating in 1996 at the Valduc military site; the second, with the same capacity, should be commissioned after the turn of the century at Cadarache, where incinerable -bearing waste from all CEA facilities is consolidated for processing.
This incineration process could also be used for waste, and could be scaled up to handle several hundred kilograms of waste per hour with no serious difficulties. Compared with conventional burners, this process has a number of significant advantages for highly contaminated waste:

The very limited gas flow during pyrolysis, which releases only the gases generated, and during incineration in oxygen-enriched air, minimizes entrainment of radioactive particles in the off-gas stream: less than 1%, compared with 5% or more in conventional incineration processes. The total off-gas decontamination factor may therefore reach 1011!

The process is suitable for highly chlorinated waste. A typical feed stream contains 50% polyvinyl chloride; the chlorine is released during pyrolysis at 550C, a temperature at which its corrosive action is very limited.

The carbon content of the ashes can be minimized by adjusting the ash residence time in the incineration step: after 3 hours the residual carbon concentration in the final ashes is well below 1 wt%. As carbon is a chemical reducing agent, limiting the carbon content allows the use of oxidizing processes to recover plutonium from the ashes.

The modular process design allows the use of compact equipment items, and facilitates monitoring of plutonium retention, minimizing the criticality hazard.

INCINERATION AND COLD CRUCIBLE MELTING

For many years the CEA has been investigating the potential of induction heated cold crucible melters (CCM) for waste conditioning, and the processes developed are now reaching maturity, due in part to the experience acquired and to recent progress in electric power technology. The same technique is now being applied to glass melting in non-nuclear industries.
This is an ingenious and apparently simple technique, requiring only relatively simple equipment (Fig. 1). An electromagnetic field oscillating at high frequency within the glass is created in a sectorized water-cooled metal crucible. The electric currents induced in the molten glass dissipate considerable power by Joule effect. A solidified glass "skull" forms between the melt and the cold wall, protecting the crucible from damage by the molten glass. A layer of solidified material above the melt also lowers the surface temperature, and very effectively limits the release of volatile elements.
Several melters of this type have already been built, notably by the CEA, although they are only nonradioactive prototypes 50 to 60 cm in diameter with capacities of about 50 kg.h-1. The principal advantage of this type of melter, other than its extended service life due to the virtual absence of corrosion, is its ability to reach very high temperatures - well above 1500C if necessary. In this diameter range, the melters operate at a frequency of 200-300 kHz and routinely achieve 70% efficiency. They may be scaled up to larger dimensions and capacities with few problems, and mechanical stirring provisions would allow even higher capacities to be reached.
The CEA is currently investigating several approaches:

Permit operation of this type of melter in the existing La Hague vitrification facilities by nuclearizing the process equipment for use in hostile environments.

Scale up the current melter design to obtain higher capacities.

Adapt the process for application to other types of waste, such as low and intermediate-level radioactive solutions from reprocessing plants or nuclear power stations (6).

Adapt the process to vitrify fuel waste incineration residues immediately after the incineration phase, notably for ion exchange resins from nuclear power plants. If direct incorporation into molten glass proves feasible, this would allow waste processing and conditioning in a single operation, inside a single highly compact unit that is not subject to corrosion.

THE ECONOMICS OF LOW-LEVEL WASTE VITRIFICATION

Low-level waste vitrification can only be considered if the technology is cost-effective, i.e. if the overall waste treatment cost is no higher than with conventional concrete grouting or bituminization processes.

The improved containment integrity of glass is an undeniable advantage of vitrification. A containment matrix must meet the conditioned waste acceptability criteria specified for a final repository site; the enhanced quality of glass packages constitutes a qualitative advantage with regard to public acceptance or to more stringent regulatory requirements in the future, but it is impossible to quantify objectively and was therefore disregarded in this economic assessment.

Another hypothesis concerns the ultimate destination of the conditioned waste package, which should be compatible with near-surface disposal. This is a conservative hypothesis in that it limits the quantity of radionuclides incorporated in the glass, but it also allows comparison with conventional processes.

Finally, vitrification is applicable only if contact maintenance is feasible after source removal. This is the case for liquid waste containing up to about 108 Bq.l-1. The cold crucible itself requires no maintenance; relatively thin biological shielding around the melter allows direct maintenance operations on peripheral components.

Based on these considerations, the comparative cost of vitrification and of cement encapsulation was assessed, including not only treatment and conditioning costs (capital, operating and maintenance costs), but also transportation and final disposal costs. The study was conducted for concentrated borate solutions from VVER reactors in eastern Europe, for capacities ranging from 200 to 1200 m3 per year; high capacities requires a preconcentration step before the calciner/melter or the mixer. The main specifications taken into account are indicated in Table II.

The overall waste treatment cost is shown in Fig. 2 for the two extreme capacities considered in this assessment: a) 200 m3 and b) 1200 m3 per year, with the depreciation time plotted along the X-axis.

The depreciation period necessary to obtain identical costs for concrete encapsulation and vitrification range from 2 years for a capacity of 1200m3/year to 7 years for 200 m3/year. With low capacities, the cost difference remains relatively low for periods exceeding 7 years, while for higher capacities the differences becomes more significant after 2 years. Any increase in the disposal cost would constitute a further argument in favor of vitrification.

These findings were supplemented by a sensitivity study covering final disposal costs (+100%), the capital cost of the vitrification facility (+30%) and the cost evaluation basis (data corresponding to central and east European countries). The results did not change the preceding conclusions.

Economic assessments are still in progress, focusing on very low capacities. The vitrification process has been modified accordingly, notably including provisions for supplying waste directly to the cold crucible to vitrify incineration ashes, for example.

CONCLUSION

Nuclear waste disposal problems have been taken into account from the outset by the CEA, and research continues with the objective of ensuring waste treatment and containment under optimum technical and economic conditions. After developing and implementing high-level radioactive waste containment jointly with SGN at industrial scale in the reprocessing plants operated by COGEMA at Marcoule (UP1) and La Hague (UP2 and UP3), the CEA is now orienting its research effort toward increasingly effective and reliable processes that are also applicable to other waste forms which, although less radioactive, warrant enhanced conditioning as part of an optimized overall nuclear waste management strategy. The CEA, COGEMA and SGN are working together to mitigate the waste produced by an advanced nuclear industry.

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Session 18 -- Progress: HLW Tank Safety and Disposal
Co-chairs: Ron Izatt, USDOE;
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18-1
BUILDING THE COLOSSUS OF RHODES OVER THE STRAITS OF GIBRALTAR AND HANFORD'S TANK WASTE REMEDIATION SYSTEM-ARE THEY THE SAME ANIMAL?
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ABSTRACT
The Island of Rhodes was the jewel of the Dodecanes Islands. It rivaled Athens in 300 B.C. The people of Rhodes built the Colossus of Rhodes in 283 B.C. The Colossus spanned the harbor entrance. It was 105 feet tall, a statue of the sun god Helios, the deity of the island. The statue stood for 56 years and was considered to be "The Seventh Wonder of the Ancient World." It collapsed during an earthquake and was sold for scrap and carted off by 980 camels, an ignominious ending for a wonder of the ancient world. The statue was built for approximately $2 million in 1993 dollars. Now how does this compare to the "Hanford Tank Remediation"? Hanford may not be the "Seventh Wonder of the World," but it is probably the most underrated "First Wonder of the Engineering World." The cost of remediating the tank farms is currently estimated at $53 billion in 1995 dollars. The Hanford site will have existed for 56 years when tank remediation starts, and was toppled by the Peace Initiative earthquake that changed the economics of the Tri-Cities area. If one extrapolated the cost of the Colossus to the cost of Hanford remediation in 2025 dollars, one could build the Colossus over the Straits of Gibraltar at its 8 mile width. Hanford could be considered a monument to the sun god, if one could relate the explosion of the first hydrogen bomb to the energy of the sun. The Hanford production site produced the resources to provide almost 20,000 suns.
The Hanford tank waste remediation project is probably the most underrated engineering marvel of the 21st century. It exceeds the engineering required for the Manhattan Project, the Nuclear Navy, the Polaris Missile Program, even the Mercury Space Program and placing a man on the moon. One can visualize 30 Trident submarines, the total fleet of nuclear carriers and their weapons systems, the total cost of the Suez and Panama canals, the Golden Gate, George Washington, Brooklyn and Straits of Mackinaw Bridges, the Channel under the English Channel. The amount of engineering at Hanford boggles the technical mind but is being treated as just another DOE project even though it exceeds the dollars and engineering of all the previous programs. In 1989, DOE signed a Tri-Party Agreement with the state of Washington, Region 10 EPA and DOE. The agreement originally required operation of a high-level waste vitrification plant to treat the high-level waste in 28 double-shell tanks by 1999. DOE recognized in 1990 that: 1) the higher risk

vulnerabilities were the 149 single shell tanks and 2) the ability to retrieve and process the waste had not been evaluated sufficiently to achieve these dates. After significant negotiation, the date for some form of vitrification has been revised to 2009 and a major effort is being devoted to waste retrieval technology and aqueous processing.

In April 1994, DOE initiated a "Call for Interest and Qualifications for Lead Organizations for the High Level Waste Tank Remediation Area." In this document, the DOE identified its objectives as stakeholder involvement, expediting regulatory constraints on technology, insuring DOE cross complex technology demonstrations, validating pretreatment technology needs, and immobilization as the key areas of concern by the interested parties. In response to the inquiry, a lead team was identified composed of the Hanford operations contractor, Pacific Northwest Laboratory and the Oak Ridge, Idaho Engineering Laboratory and the Savannah River Laboratory. As part of an "integration meeting", over 200 areas of "Need" were identified in the areas of: Waste Characterization and Analysis, Retrieval Technology, Waste Pretreatment, Waste Processing, Regulatory Uncertainties, Waste Stabilization, and Site Closure.

In conjunction with the seven major areas identified, more than 500 technology priorities were tabulated. The experience with DOE existing vitrification projects were major inputs into the 500 technology areas. The West Valley Demonstration project is presently four years behind its original vitrification schedule of vitrifying 600,000 gallons of commercial high-level waste. Also factored into the technology needs were 62 areas still to be resolved at Savannah River where the Defense Waste Processing Facility is 5 years behind its original schedule and a factor of 2-3 above original cost estimates to start vitrification.

Figure 1 displays the scope of the problem in the site technology problems

Safety

Characterization

Retrieval

Pretreatment

Immobilization

Stabilization & Closure

The figure is only a listing of the major technology initiatives. Thus, there are the 200 individual technology needs identified by Savannah River, West Valley, Oak Ridge, INEL and Hanford. When this is combined with over 500 serial, parallel and conflicting priorities, the initial scope of the challenge starts to be visualized. Once you realize that the cost of only the Hanford segment of this program is greater than $50 billion in 1995 dollars, the emotional and economic issues start to surface. In addition to the Hanford costs, Savannah River is estimated at $300+ million per year, West Valley is $120 million per year and INEL will require a $1 billion capital expenditure and $100 million per year operating expense at some period in the post 2003 time frame. The magnitude of the project becomes mind-boggling when one considers the diversity of managing contractors, the demands of the stakeholders, the regulatory overlap and conflicting scheduling demands, combined with the conflicting reinvention and decreasing budget availability.

DOE has identified a conceptual organization alignment shown in Figs. 2 and Fig. 3. The organization consists of a lead team, two interface steering groups, a management team, an implementation team and an economic development board. An ancient technology warrior once said "If you can't find the individual in charge, nobody is in charge." Figure 4 is the conceptual communications model which includes one committee, three teams, two groups, one council and three related management organizations.

Figure 5, "the decision making process" consists of one committee, three deputy assistant secretaries, two teams, one group, three headquarters program managers, and five site inputs. In Ref. 1, N. Nohria and James D. Berkley wrote an interesting article, "Whatever Happened to the Take Charge Manager?" (1). If one looks back on major projects of a similar nature-i.e., the Manhattan Project, the Nuclear Navy, the Polaris Missile programs--each program had a specific "take charge" manager in control and initially the programs were not as complex as the Hanford Tanks. General Groves, Admiral Rickover, and Admiral Rayburn were take charge types and the programs achieved their goals because of take charge focused decision making.

As seen in Fig. 1, there are enormous technology demands, hundreds of technology uncertainties, multi-year financial uncertainties, diverse interest groups and a

changing regulatory environment that is more politically driven then ecologically-health risk justified. In order to achieve decision closure, a key road map is needed that parallels the:
1. West Valley Program Schedule
2. Savannah River DWPF
3. Oak Ridge TRU Tank Program
4. INEL High Level Waste Program
5. Hanford Tank Focus Area
This road map needs to address West Valley as a bench scale initial research demonstration to identify problems and solutions to high level waste tank retrieval and small scale vitrification. Unknowns and needs for Savannah River, Oak Ridge and Hanford should be clearly identified for resolution in certain West Valley operations.
Next Hanford's needs and unknowns must be clearly identified in the Savannah River DWPF program. And the tank analysis, characterization and retrieval at Savannah River must clearly be part of a prototype project for Hanford technology resolution. The ORNL TRU tank retrieval program is a parallel extraction chemistry project that is directly applicable to Hanford TRU extraction and separation chemistry. Figure 2 illustrates the tank focus area proposed organizational interaction.
In Fig. 3, when one examines the diversity of unknowns that are interrelated, the multiple parallel programs communicating at Savannah River, West Valley, and INEL that will directly effect 2025 dollars of $200 billion, one understands the then the need for a more efficient decision making organization. It is clear that paramount to the success of the Hanford program is a simple chain of command. Although stakeholder participation appears to be a primary issue at Hanford, the need to address the total technical issue so that it can be presented to stakeholders, their constituents and their independent technical specialists in a credible way appears to be the initial feature to any successful dialogue. Even more important to a program that will require $200 billion dollars of potential expenditure is a leadership that will be recognized in Congress as technically, institutionally, and economically credible and one that would be accountable for such an expenditure of government and taxpayer resources.
Figure 4 was the illustration used in the Tank Focus Area Proposal on how decisions would be made by a committee, two teams, side bar management, a group and five sites.
Figure 5 identifies a simplified organization structure that meets the direct line of decision, responsibility and accountability that is considered to achieve an initial programmatic credibility requirement for early program direction and decision. The organization requires accountability at the top and responsibility for the implementation, and limits the responsibility of decision making to a limited number of individuals and organizations while still providing external review and support.
Figure 6 minimizes the circular decision process, provides input to EM-1 who Congress believes holds the responsibility for direction to the Hanford site manager and ultimately the program manager.
Congress will demand headquarters responsibility and continuity of management for a program with the potential of $200 billion expenditures. EM-50 will be the focus of continuity to Congress even though a field laboratory is providing technical research. Congress needs a headquarters technology guru who can convince them that the process will achieve the programmatic goals for the dollars expended. The state of Washington will look to the Hanford Operations Manager for regulatory and safety responsibility. All of the interested stakeholders, Congress, HQ, state of Washington, EPA, etc., will be looking to the Program Implementation Manager as the focal point of field accountability. Figure 6 reinforces the simplicity of the decision making process that will insure decision authority and responsibility. The DWPF, West Valley experience will help to convince Congress and the stakeholders that resolution of unknowns, the cost benefit, safety and regulatory decisions, and the warm feeling of someone being in charge will achieve success.
The lack of a single chain of responsibility inferred in Fig. 2 through Fig. 4 would cause Congress, the Defense Nuclear Safety Board and regulatory agencies to have significant qualms in continuity of direction and decision. Again, as Admiral Rickover would say, "If you can't point to the person in charge, no one is in charge."

The Hanford program needs to develop the following:
1. Program road maps with prime and alternative paths.
2. Technology availability and needs analysis.
3. Priorities of needs and availability.
4. Research flow diagram and technology availability.
5. Scheduled compatibility analysis.
6. Uncertainty analysis on meeting technology, research, scheduling and economics.
These areas must be fully integrated with West Valley, Savannah River, Oak Ridge, INEL and Hanford to insure that the projects are visible and clearly under control of the Department of Energy.
The Hanford project is greater, far greater, than Gibraltar, the colossus of Rhodes and the Pyramids, and needs to be recognized as such in the engineering and science communities to achieve success. There are no $200 billion 2025 dollar projects in either the private sector or the federal sector that approach the complexity of the Hanford Tank Focus Area, and the support of Congress and stakeholders will only be achieved through credible, visible leadership, expeditious decision communication and financial control and schedular success that can only be achieved through performance by take-charge leadership.
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18-2
TANK WASTE REMEDIATION SYSTEM: AN UPDATE
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ABSTRACT
The U.S. Department of Energy's Hanford Site, located in southeastern Washington State, contains the largest amount and the most diverse collection of highly radioactive waste in the United States. High-level radioactive waste has been stored at the Hanford Site in large, underground tanks since 1944. Approximately 217,000 m3 (57 Mgal) of caustic liquids, slurries, saltcakes, and sludges have accumulated in 177 tanks. In addition, significant amounts of 90Sr and 137Cs were removed from the tank waste, converted to salts, doubly encapsulated in metal containers, and stored in water basins.
The Tank Waste Remediation System Program was established by the U.S. Department of Energy in 1991 to safely manage and immobilize these wastes in anticipation of permanent disposal of the high-level waste fraction in a geologic repository. Since 1991, significant progress has been made in resolving waste tank safety issues, upgrading Tank Farm facilities and operations, and developing a new strategy for retrieving, treating, and immobilizing the waste for disposal.
INTRODUCTION
The U.S. Department of Energy's (DOE) Hanford Site, located in southeastern Washington State, contains the largest amount and the most diverse collection of highly radioactive waste in the United States. The Tank Waste Remediation System (TWRS) program was established in 1991 to safely store, treat, and dispose of those wastes. This paper provides an update on the progress of the TWRS program.
BACKGROUND
High-level radioactive waste (HLW) has been stored at the Hanford Site in large underground storage tanks since 1944. Approximately 217,000 m3 (57 Mgal) of waste have accumulated in 177 tanks. These caustic wastes consist of different chemicals, including liquids, slurries, saltcakes, and sludges.
The radioactive waste stored in these tanks came from different sources, including: 1) three plutonium and uranium recovery processes from approximately 100,000 metric tons uranium (Mtu) of irradiated fuel; 2) three radionuclide recovery processes from waste; and 3) miscellaneous sources (e.g., laboratories and reactor decontamination solutions). These wastes were then concentrated and mixed in order to minimize the

number of storage tanks required. The neutralized wastes include sodium nitrate/nitrite, sodium hydroxide, sodium aluminate, sodium phosphate, the hydrous oxides of iron, chrome, and other transition metals, large amounts of organics, and approximately 250 MCi of radionuclides.

The wastes are stored in 149 single-shell tanks (SSTs) and 28 double-shell tanks (DSTs). The SSTs are made of reinforced concrete with a carbon-steel liner and can hold 208 m3 (55,000 gal) to 3,800 m3 (1 Mgal) of radioactive waste. The DSTs consist of a carbon steel tank within a steel-lined concrete tank, and have a nominal capacity of 3,800 m3 (1 Mgal), as shown in Fig. 1. Of the SSTs, 67 have leaked or are suspected to have leaked a total of approximately 3,800 m3 (1 Mgal).

No waste has been added to the SSTs since 1980. The pumpable liquids are being removed from the SSTs and are being sent to DSTs, so that the remaining waste is mostly sludge and saltcake. There is no evidence to suggest that any of the DSTs have leaked.

In addition to the waste stored in the tanks, significant amounts of 90Sr and 137Cs were removed from the tank waste, converted to salts, doubly encapsulated in metal containers, and stored in water basins. There are approximately 1,900 6.7-cm (2.6-in.)-diameter x 52-cm (20.5-in.)-long capsules containing approximately 160 MCi total.

TWRS PROGRAM DESCRIPTION AND STATUS

The Tank Waste Remediation System's mission is "to store, treat, and immobilize highly radioactive Hanford Site waste in an environmentally sound, safe, and cost-effective manner." Systems Engineering techniques are being applied to TWRS to establish functions and requirements, to identify all its relationships to other programs, and to evaluate alternatives for accomplishing the TWRS mission. This systematic, disciplined, and documented approach is an effective way to manage this large, complex Hanford Site waste management program, which will require many years to complete.

A technical strategy for storing, treating and disposing of the tank waste has been developed, and the strategy has been agreed upon through extensive interaction with the public, other public interest groups, and regulatory agencies. This strategy is shown in Fig. 2 and described in the following sections.

TANK FARM OPERATIONS

Newly generated wastes continue to be received in the double-shell waste tanks, and approximately 4 million gallons of liquid waste are yet to be pumped from the SSTs to the DSTs to reduce the risks of leaks. Continuous surveillance and monitoring of the 177 underground tanks is necessary to ensure the waste is safely stored. Significant progress has been made in the past year on improving tank farm operations to increase safety and cost effectiveness.

Safety statistics have improved in all categories, as shown in Fig. 3. The number of lost and restricted work case injuries decreased by more than 60% and the total number of days lost was reduced by 75%. During this same time period, the maintenance productivity increased 50% and the backlog of corrective maintenance work was at an all-time low.

The 242-A Evaporator was restarted, and recovered more than 19,000 m3 (5 Mgal) of DST space during two processing campaigns. The lessons learned and implemented from Campaign 94-1 made Campaign 94-2 even more successful. No safety incidents or personnel contaminations occurred during Campaign 94-2, and only one administrative-based occurrence report was issued. The Campaign 94-2 maintenance outage was completed 30% faster than scheduled, production targets were exceeded, and the overall campaign was completed well ahead of schedule.

The SST pumping program, for the first time since the early 1980's, was actively removing liquid waste from 6 SSTs at one time. More than 760 m3 (200 Kgal) of waste was transferred to DSTs during the year. This included pumping two tanks that were "assumed leakers."

The plan is to continue improving tank farm operations to increase productivity and keep the workplace safe.

TANK FARM UPGRADES

Many of the Hanford Site facilities and much of the equipment at the tank farms are 40 to 50 years old and have not been well maintained. Upgrading the tank farms is a high priority, because the tank farms will continue to provide interim waste storage for many years until disposal of the waste can be achieved. Upgrades include: new

safety analyses; replacing old instrumentation, equipment, and ventilation systems; constructing new waste transfer lines; removing obsolete contaminated equipment; and removing contaminated soil.

SAFETY ISSUE RESOLUTION

The highest TWRS program priority is resolution of waste tank safety issues. The safety issues of primary importance are summarized below.

Flammable gas concentrations in 25 tanks (SSTs and DSTs). The flammable gas tanks safety issue involves the potential release of flammable gases in concentrations above the lower flammability limit. The worst tank, 101-SY, has been successfully mitigated with a mixing pump. The pump is operated up to three times a week to mix the waste and release gas.

Hydrogen monitors are being installed on all 25 flammable gas tanks. A better understanding of the physical properties of the tanks will be gained using a retained gas sampler, a viscometer, and void fraction devices. The void fraction test instrument has been successfully demonstrated in tank 101-SY. This monitoring data will be used, along with knowledge acquired during the process, to determine if any tanks other than 101-SY require active mitigation for safe storage. Documentation to close the 101-SY Unreviewed Safety Question was submitted in 1994.

Potentially explosive mixtures of sodium nickel ferrocyanide and sodium nitrate and nitrite in 18 tanks. In certain concentrations, these chemicals are known to react exothermally at high temperatures. Extensive testing of waste samples and simulants, along with temperature measurements and modelling, suggests that these wastes are safe and can continue to be safely stored for many years. Six of the original Ferrocyanide Watch List tanks have been removed from the Watch List based on records that showed no ferrocyanide had been added to these tanks. (Watch List tanks are those that the Secretary of the DOE reports on to Congress because of potential safety concerns.) The understanding and evaluation of this safety issue have progressed to the point that the Ferrocyanide Unreviewed Safety Question was closed in 1994.

The potential for exothermic nitrate-nitrite organic chemical reactions in SSTs. The organic tanks safety issue involves the potential for uncontrolled exothermic reactions of organic chemicals and nitrates/nitrites and for vapors from semi-volatile organics entrained in waste to exceed the flammability limits (4). Recent laboratory tests showed that fuel concentrations and temperatures required to support propagating exothermic reactions are comparable to those for ferrocyanide. Ten tanks that received organic complexants were added to the Organic Tanks Watch List following a review of sampling data and waste transfer records (5,2). Vapor sampling and safety analyses were completed and formed the basis for closing the Unreviewed Safety Question concerning flammability of the floating organic layer in tank 103-C (3).

Mitigation of worker safety concerns resulting from the random release of fugitive noxious vapors from passively ventilated SSTs. These vapors have a strong odor and, on occasion, have made some workers ill. A health and safety plan that includes appropriate sampling/respiratory requirements has been implemented.

High-heat load in SST 241-C-106 requiring periodic addition of water to provide evaporative cooling. Because many of the SSTs have leaked, attempts are underway to remove as much liquid from them as possible. Currently, the liquid cannot be removed from tank 241-C-106 because the tank would overheat, potentially damaging its structural integrity. Therefore, plans are underway to remove the waste from this tank, and a project has begun to provide the facilities and equipment to start sluicing the waste from this tank in 1996.

WASTE CHARACTERIZATION

Waste characterization data must meet the needs of safety, waste treatment, and disposal program elements. The waste core sampling equipment is being improved and additional systems have been placed in service. Additional analytical laboratory hot cells are under construction, faster and more accurate analytical methods and instruments are being developed, and the laboratory data management system is being upgraded. In response to Defense Nuclear Facility Safety Board Recommendation 93-5, waste sampling and characterization for safety issue identification and resolution will be accelerated. This acceleration includes screening all single-shell tanks over the next three years to determine if any other tanks have safety concerns that may require resolution.

WASTE RETRIEVAL

Waste will be retrieved from all the tanks for treatment, immobilization, and disposal; the waste in some safety issue tanks may be retrieved in order to resolve the safety issues. Hydraulic sluicing has been the method used to retrieve waste from underground radioactive waste tanks at the Hanford Site and elsewhere. While sluicing is the preferred method of waste retrieval, it may not be acceptable in single-shell tanks that leak, and it may not remove some of the hard sludges. A robotic, long-reach arm with an assortment of tools is being developed to solve these problems. Design of the first SST sluicing system (for tank 241-C-106) is in progress. Subsurface containment barriers, which could be installed around and beneath the leaking tanks, are also being considered.

WASTE PRETREATMENT

The waste retrieved from the tanks will be separated into two fractions so that most of the radionuclides, and only a small part of the waste volume, are in the HLW fraction. The HLW will be vitrified and shipped offsite for disposal in a geologic repository. The bulk of the chemicals, and only a small amount of radionuclides, would be in the low-level waste stream that will be vitrified and disposed of near the surface onsite. The strategy is to use proven separations technology, to the fullest extent possible. More advanced separations technologies will be developed, but will only be implemented if needed to achieve the required level of radionuclide removal or an acceptably small volume of HLW. The pretreatment processes will include:

Solid-liquid separation, and sludge washing and leaching, with the soluble liquid fraction destined to be the low-level waste stream. A sludge settling test in tank 241-AZ-101 was initiated.

Radionuclide removal from the soluble liquid fraction to assure the waste can be categorized as low-level waste. Ion exchange processes are planned to remove 137Cs and possibly 90Sr, which will be routed to the HLW stream. Removal processes for long-lived mobile radionuclides (e.g., 99Tc) will be developed as a contingency.

Enhanced sludge washing, leaching, and blending to minimize the amount of high-level waste.

Technology development for selective sludge dissolution and advanced radionuclides separation processes, continued as a contingency. These technologies may be needed if the amount of vitrified HLW to be produced is not acceptable to the repository or if there is an economic reason to reduce the amount of glass.

Organic destruction process development, continued as a contingency for resolution of waste tank safety issues or, if needed, to achieve radionuclide separation.

LOW-LEVEL WASTE IMMOBILIZATION

The low-level waste will be vitrified and disposed of onsite near the surface, in a retrievable form. The existing grout immobilization process and the disposal system will be maintained as a contingency for freeing up DST space if required. The low-level waste vitrification facility will have a capacity of approximately 100 tons of glass per day. Standard glass industry melter technology may be adapted for this application, and the radiation level of the waste stream should be low enough to allow minimal shielding. Some melter tests have been conducted, and an aggressive program of waste form development and vendor melter tests must be carried out to meet the Tri-Party Agreement (1). Neither the final vitrified waste form (large monoliths or small pieces) nor the disposal container have been selected.

HIGH-LEVEL WASTE VITRIFICATION

Vitrification as borosilicate glass is generally accepted as the method that will be used to immobilize high-level waste. The TWRS strategy is to provide a high-level vitrification capacity that will be able to vitrify all of the high-level waste in 20 years. This will require a melter capacity of approximately 15 tons per day. A high-capacity melter development program is being conducted. A stirred melter and a high-temperature melter are being acquired for testing. The number and size of melters that will be installed in the HLW vitrification facility will depend upon the results of the development program.

The waste container capacity and configuration will be optimized considering such factors as the vitrification plant, interim storage, and the geologic repository. This may include a larger HLW "package" (e.g., 10-m3 container) to reduce the cost of disposal at the repository.

INTERIM HLW STORAGE

HLW containers will require onsite storage for many years until a geologic repository is ready to accept them.

137Cs AND 90Sr CAPSULES

The 137Cs and 90Sr capsules will be stored until they can be packaged and shipped to the geologic repository for disposal. Overpacking multiple capsules in a canister is the reference plan. If the overpacked capsules do not meet repository acceptance criteria, the 137Cs and 90Sr capsules will have to be processed and vitrified with other HLW.

SCHEDULE

The schedule for carrying out the TWRS program includes: 1) completing retrieval of all SST waste by 2018; 2) closing all SST farms by 2024; and 3) completing all waste vitrification by 2028. The major milestones embodied in a recently approved amendment to the Tri-Party Agreement (1) are listed in Table I.

CONCLUSION

The Hanford Site TWRS program is a large, complex program that will require many years to complete. Acquiring the funding and stakeholder commitment to conduct this program will require a national consensus that this work is necessary and is being done in a cost-effective manner. In order to significantly decrease the budget requirements, an ongoing process is in place for review of options from single processing lines to commercialization. Privatization of selected processing units could reduce the funding needs even more. We must continue to review each option's level of risk while achieving the program objectives with significantly fewer dollars. It is imperative to work safely, protect the public, seek the best technology, and use national expertise in planning and conducting the TWRS program. The DOE is committed to an open, responsive policy and encourages public participation in Hanford Site cleanup discussions. The Hanford Site has been selected as a place to test the Clinton administration's thrust to "reinvent government." This designation is particularly appropriate because cleaning up the Hanford Site is estimated to cost tens of billions of dollars.

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PROGRESS IN RESOLVING HANFORD SITE HIGH-LEVEL WASTE TANK SAFETY ISSUES

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ABSTRACT

This paper summarizes recent progress toward resolving Hanford Site high-level radioactive waste tank safety issues, including modeling and analyses, laboratory experiments, monitoring upgrades, mitigation equipment, and developing a strategy to screen tanks for safety issues.

BACKGROUND

Interim storage of alkaline, high-level radioactive waste, from two generations of spent fuel reprocessing and waste management activities, has resulted in the accumulation of 216 million liters (57 million gallons) of waste in Hanford Site single- and double-shell tanks. Before the 1990's, the stored waste was believed to be: 1) chemically unreactive under its existing storage conditions and plausible accident scenarios; and 2) chemically stable. This paradigm was proven incorrect when detailed evaluation of tank contents and behavior revealed a number of safety issues and that the waste was generating flammable and noxious gases.

In 1990, the Waste Tank Safety Program was formed to focus on identifying safety issues and resolving the ferrocyanide, flammable gas, organic, high heat, noxious vapor, and criticality issues. The tanks of concern were placed on Watch Lists by safety issue.

To ensure continued safety until needed data were collected and interpreted, the tanks associated with safety concerns were placed under stringent operating controls.

Data from extensive work done in the last few years (2) have reduced uncertainties and allowed closure of several unreviewed safety questions. Technical findings, which resulted from laboratory studies on waste simulants and actual waste, accomplished the following: 1) identified and bounded the energetics of the fuel-rich materials added to the tanks; and 2) determined that waste aging processes over the last 30 to 40 years due to waste chemistry and tank physics reduced the potential risk. Waste aging processes have led to dispersion or degradation of these fuel-rich species (3).

RECENT PROGRESS AND PLANS TO RESOLVE SAFETY ISSUES

Ferrocyanide. The ferrocyanide safety issue involves the potential for uncontrolled exothermic reactions of ferrocyanide and nitrate/nitrite mixtures (19). Laboratory studies show that temperatures must exceed 250C for a reaction to propagate. The hottest tank's temperature is currently 54C and decreasing.

Moisture levels above 20% will prevent reactions from propagating regardless of fuel concentrations. Current plans were to take two core samples from each of 18 tanks to determine ferrocyanide levels and moisture content. However, recent laboratory work with simulants confirmed that ferrocyanide fuel content decreases over time in intense radiation fields (16,17). Core sampling will be used to confirm aging in the tanks projected to have the highest ferrocyanide concentrations and exposed to the lowest radiation field and/or alkali concentrations. The sampling results are expected to show that the ferrocyanide has aged to levels low enough to resolve the safety issue. Six of the original Ferrocyanide Watch List tanks have been removed from the Watch List based on records that showed no ferrocyanide had been added to these tanks.

Organic Tanks. The organic tanks safety issue involves the potential for uncontrolled exothermic reactions of organic chemicals and nitrates/nitrites and for vapors from semi-volatile organics entrained in waste to exceed the flammability limits (21). Recent laboratory tests showed that fuel concentrations and temperatures required to support propagating exothermic reactions are comparable to those for ferrocyanide. In addition, moisture levels above 20% will prevent reactions from propagating regardless of fuel concentrations.

In prior years, work controls were implemented to prevent introduction of ignition sources to these tanks. Vapor sampling and safety analyses were completed and formed the basis for closing the Unreviewed Safety Question concerning flammability of the floating organic layer in tank 103-C (20). Ten tanks that received organic complexants were added to the Organic Tanks Watch List following a review of sampling data and waste transfer records (22,12).

Aging processes have destroyed or significantly lowered the energy content of the organic tanks. Recent laboratory work (1) demonstrates that in the presence of alkaline aluminum salts and radiation, complexants such as hydroxyethylethylenediamine triacetic acid (HEDTA), ethylenediamine triacetic acid, and glycolic acid decompose to form less reactive fragments, ultimately creating a mixture of sodium formate and sodium oxalate. During this aging process, hydrogen, nitrogen, nitrous oxide and ammonia are evolved. This finding has been confirmed by studies on tank 241-SY-101 waste (9) that found the HEDTA had almost disappeared. Other complexants had partially degraded to sodium oxalate and formate and a complex mixture of degradation products such as the sodium salts of ethylenediamine diacetic

acid and nitrilodiacetic acid.
Recent laboratory findings (5) indicate that the more energetic complexants and the primary degradation products of tributyl phosphate (TBP) are water soluble in saturated nitrate-nitrite salt solutions and would a large percentage have been removed from the single-shell tanks when they were saltwell pumped.
In conclusion, the combination of waste aging, waste dissolution and dispersion appears to bound the inherent risk in fuel-rich nitrate-nitrite systems. When work on waste surrogates and simulants is completed, limited core sampling in bounding tanks will be used to verify the conclusions derived from these conceptual models of waste behavior.
Flammable Gas Tanks. The flammable gas tanks safety issue involves the potential release of flammable gases in concentrations above the lower flammability limit. In prior years, work controls were instituted to prevent introduction of spark sources, and evaluations were completed to ensure that attached equipment was intrinsically safe.
The worst tank, 241-SY-101, has been successfully mitigated with a mixing pump. The pump is operated up to three times a week to mix the waste and release gas.
Hydrogen monitors are being installed on all 25 flammable gas tanks. The Standard Hydrogen Monitoring System consists of a cabinet equipped with piping and instrumentation that support an on-line hydrogen detector and a "grab sampler." The cabinet isolates the equipment from the environment. The hydrogen detector is presently a Whittaker electrochemical cell, but other detectors can also be installed into these cabinets. The grab sampler allows gas samples, captured during a gas release event, to be removed from the tank for highly detailed analyses using state-of-the-art gas chromatograph and mass spectral analysis techniques. A better understanding of the physical properties of the tanks will be gained using a retained gas sampler, a viscometer, and void fraction devices. The void fraction test instrument has been successfully demonstrated in tank 241-SY-101. This monitoring data will be used, along with knowledge acquired during the process, to determine if any tanks other than 241-SY-101 require active mitigation for safe storage. Documentation to close the Unreviewed Safety Question in SY tank farm was submitted earlier this year.
High-Heat Tanks. The high-heat tanks safety issue concerns tank 241-C-106, a single-shell tank that requires water additions for evaporative cooling. Without the water additions, which would be discontinued in the event of a tank leak, the tank could exceed structural temperature limits, resulting in potential tank collapse. Tank 106-C is on an accelerated program for early retrieval to a double-shell tank. Double-shell tanks handle heat-bearing materials better than single-shell tanks, thus lowering the potential hazard from a structural failure if, in the event of a tank leak, cooling water additions to the tank are discontinued. Tank 106-C retrieval is scheduled to begin in late 1996. A process test and considerable thermal analyses were completed on tank 106-C to evaluate alternate cooling approaches. The studies concluded that the tank could be adequately cooled using an air chiller.
Criticality. The criticality safety issue involves the potential for criticality in tanks. Tank criticality prevention controls have been strengthened by improved administrative procedures and training, and the Unreviewed Safety Question was closed (6) using analyses that showed criticality during storage was highly unlikely. All the single- and double-shell tanks at the Hanford Site contain sufficient neutron absorbers to ensure safe storage; however, additional sampling or controls will be required for retrieval and pretreatment-related activities.
Noxious Vapor Concerns. The noxious vapor safety issue involves potential health and safety issues related to toxic vapors that may be present in some of the high-level waste tanks. The issues stem from an insufficient understanding of the causes of reported exposures of tank farm personnel to unacceptable levels of noxious vapors, and the concern that until the vapors in the waste tanks are well characterized, the risks to worker health and safety cannot be determined or controlled (18,13). In prior years, worker protection controls were instituted to prevent worker exposures, and a program was implemented for routine workspace air monitoring and periodic personnel dosimetry.
In-tank vapor sampling equipment has been developed and tested. Two methods are used to collect vapor samples from the waste tanks (14). The primary method employs: heated transfer tubing; a heated sampling manifold; temperature, flow control, and

valving technology; and an air pump to draw air, gases, and vapors out of the waste tanks. This method currently requires that a customized vapor sampling probe be installed by crane into a tank riser. This integrated equipment is referred to as the vapor sampling system (VSS). The VSS was specifically designed to collect representative samples from warm, moist tanks, even if a fog exists in the tank headspace.

A second method for collecting vapor samples from the waste tanks is referred to as in situ sampling (ISS). Rather than transferring the air, gases, and vapors to be sampled to a remote location, the sampling devices themselves (specifically sorbent traps) are lowered into the tank headspace. This minimizes the loss of constituents by adsorption on transfer tubing walls, and circumvents the need for heated probes, heated transfer tubing and a heated sampling manifold. Currently, the ISS equipment consists of a simple manifold and air pump mounted on a two-wheeled hand cart. Small bundles of about eight sorbent traps are lowered into the tank headspace, each sorbent trap having its own 0.64-cm (0.25-in.) plastic tube connection to the sampling manifold. The current sampling manifold is capable of collecting four samples simultaneously, and uses needle valves to control flow rates and a rotameter to monitor flow rates. In the ISS method, constituents that are not amenable to sorbent trap sampling (e.g., gases such as H2, N2O, and CH4), are sampled using an unheated plastic tube that transfers sample from the headspace to whole-air sampling devices (e.g., SUMMA1 canisters) located outside the tank. By the end of 1994, 18 high-level waste tanks had been vapor sampled.

UPGRADED APPROACH TO WASTE TANK SAFETY SCREENING

In 1990, all Hanford Site high-level waste tanks were segregated into four different categories based primarily on historical and monitoring information. These tanks were placed on "Watch Lists" to ensure increased attention and monitoring. In mid-1993, a program was initiated to screen all single- and double-shell tanks using core sample data. This program is being updated to take advantage of recent improvements in vapor sampling capability and an improved understanding of conditions necessary for selected chemical reactions.

Key changes in the revised safety screening strategy will include placing flammability controls on all tanks until associated sampling is complete, and using vapor sampling to help determine tank contents. Rather than initially sampling the waste to determine fuel content, as was the past strategy, evaluation and/or checking of moisture levels will be performed. If moisture levels are above 20%, fuel-rich propagating reactions will not occur.

Figure 1a, b shows the logic chart for the proposed safety screening. Key steps include:

Perform tank-by-tank evaluations of historical records (including past sampling results, laboratory waste testing, tank-specific modeling, and monitoring records), assessment of the adequacy of present monitoring systems, and tank-specific safety bases.

Vapor samples will be used to check for flammable or explosive gases. Headspace samples will be evaluated for hydrogen, organic gases and vapors, including ammonia, and nitrous oxide levels. If concentrations exceed pre-determined requirements, additional monitoring, waste or vapor sampling, or mitigation using enhanced ventilation or inserting, may be required.

Vapor samples will also be used to help identify the presence of volatile reactive materials and might be used to identify changes in waste chemistry resulting in significant changes in overhead gas compositions.

Vapor samples will be further used to check for noxious vapor sources. Treatment systems may be required if vapors are detected in the tank headspace above levels immediately dangerous to life or health.

Visual inspections will be used as an initial screening for moisture. For tanks with continuous aqueous layers (all double- shell and about five single-shell tanks), no additional sampling will be required. Other tanks will require either evaluation or near-surface samples to determine moisture content. If moisture levels are 20% or higher, no further safety screening sampling will be required.

If moisture levels are not adequate, fuel levels will be determined by near-surface sampling, or a low-cost mitigation process will be implemented to increase moisture levels.

This sampling program is expected to greatly accelerate tank safety screening, with considerable cost savings.

This new approach will require vapor sampling of all passively ventilated tanks (26 have already been completed) and near-surface moisture sampling of up to 70 tanks. Either auger samples or an in situ device will be used to determine moisture. An auger sampling device is like a drill bit that is rotated into the waste to obtain the sample. Auger samples are much easier to obtain than core samples; however, sample recovery may not be adequate. Commercially available devices such as an electromagnetic induction moisture monitor could be modified and used in a penetrometer. Some limited core sampling will be required to support the technical basis for the new strategy.

TECHNICAL BASIS FOR UPGRADED SCREENING STRATEGY

To provide greater assurance that all tanks with safety issues have been identified, a safety screening will be conducted to determine whether any other tanks generate flammable gases or noxious vapors above limits, or if conditions exist that would support exothermic propagating reactions due the fuel and oxidizers mixtures, or if surface combustion could occur due to entrained or pooled organic solvents.

Previous testing with vapor sampling has demonstrated that waste tank headspaces are essentially homogeneous (15). In addition, near-surface moisture levels are believed generally high enough to reduce the number of waste samples required for adequate moisture assessment to reasonably attainable levels, particularly for sludge-containing tanks.

Ferrocyanide and organic chemicals act as a fuel when combined with an oxidizer, and nitrate salts (an oxidizer) have been precipitated in the tanks. These compounds have the potential for propagating exothermic reactions between fuel and salts of sodium nitrate and sodium nitrite. Two types of propagating exothermic reactions are possible in fuel-rich wastes: 1) bulk runaway reactions, in which an entire tank's contents reach a critical temperature and then self-heat; and 2) propagating reactions in which the reactive zone spreads at a measurable rate as the result of a localized initiator (11).

For a bulk runaway reaction to occur, a major fraction of the waste would have to be heated to above 250C (Fauske 1994). Heat loads in most of the Hanford Site tanks are low, and temperatures are declining with time. In addition, most single-shell tanks are passively cooled (10). Bulk heating of waste to 250C is not credible.

For a propagating reaction to occur, a mixture of fuel-oxidizer must be heated to high temperatures or energized by applying an ignition source (7). Because specific conditions of fuel, moisture, and temperature are all required to support a propagating reaction, screening criteria have been set based on extensive energetics studies on waste stimulants. For the ferrocyanide safety issue, these results have been confirmed on actual waste. Safety screening criteria for the propagating exothermic reactions in the condensed phases are listed in Table I.

All double-shell tanks contain adequate moisture levels to prevent propagation. Ferrocyanide fuels are found only in sludges in single-shell tanks. Organic fuels may be in sludges or saltcakes and in double- or single-shell tanks. Most sludges, except for those in shallow tanks or high-heat tanks, contain adequate moisture to prevent propagating exothermic reactions and will remain adequately moist indefinitely unless temperatures increase above 90C. Considerable research work has shown that formation of dry regions due to local hot spots in sludges is not credible. In addition, no credible mechanisms exist to raise temperatures inside wastes to reaction levels. Therefore, only external initiators are credible. Several potential external initiators have been evaluated, including instrumentation, cameras, pumps, core samplers, vehicles, and lightning strikes.

The maximum waste temperature criterion of 90 C reflects the need to preserve existing waste moisture contents. It is important to note that this is a bulk waste temperature, and that point source temperatures (e.g., from rotary core sampling) are allowed to be only as high as 150C, well below the critical ignition temperature for a propagating reaction (Fauske 1994)(23). Experimental work at Fauske and Associates (Fauske 1994) measured the actual energies released for a variety of bounding organic-nitrate-nitrite-containing surrogates, and bounded the actual propagating energy ranges and conditions for a variety of mixtures containing organic complexants, TBP and related materials. Fauske and Associates is completing documentation of the quenching effects of moisture on such systems.

Evaporation of saturated salt solutions in the passively ventilated tanks is slow; therefore, extensive dryout of the tanks is an improbable event.

Operational upsets were considered by using a "what if" approach that focused on

upset conditions that would significantly increase the amount of energy deposited in the tank or waste. Heating a small portion of dry, reactive waste to fuel-nitrate ignition temperatures could initiate a sustainable rapid exothermic fuel-nitrate reaction accident if the waste were reactive. The potential for tank farm equipment and operations to heat a portion of waste to ignition temperatures has been evaluated in (4). Energy from external initiators and natural events such as lightning strikes were also considered. Lightning was the most energetic, and therefore was used to bound the maximum depth that an external initiator would penetrate the waste surface.

Given that credible initiators transfer their energy to the waste surface, sufficient fuel and oxidizer must exist near the surface in order for a propagating chemical reaction to be induced. Conversely, if insufficient fuel and oxidizer exist in a surface layer that could be heated by an initiator, then it is physically impossible to initiate a propagating reaction.

In addition, preliminary work by Camaioni and Samuels (8) has confirmed the general findings on complexant decomposition and also demonstrated the facile destruction of TBP and its primary hydrolysis product sodium dibutyl phosphate. Only normal paraffinic hydrocarbon, which would evaporate or boil out of hot waste, was resistant to gamma-radiation-induced aging under initial test conditions.

Further insight into the distribution of organic material was gained from a series of solubility studies by Barney (5). Accurate data have been obtained for solubilities of selected organic compounds in tank supernate solutions. Initial research resulted in development of a model for fuel distribution in saltcake that shows the more energetic species to be soluble in the aqueous-saturated waste solution and thus susceptible to removal when drainable liquid is removed from the single-shell tanks by saltwell pumping. Work is continuing to determine whether the salts of calcium, aluminum and/or iron form insoluble precipitates of complexants leading to their deposition in saltcake (creating localized concentrations of fuel). In the absence of localized insoluble salts, a case can be made that organic complexants are dispersed in the aqueous phase.

In summary, the presence of separable organic liquids can be quantified from concentration data obtained from headspace air. Because the energy required to ignite an entrained solvent is likely to be small, it is likely that mitigation efforts such as inerting the tank headspace (e.g., with nitrogen replacing air) may be necessary. Meanwhile, as mentioned earlier, flammability controls will be applied, not just to known Watch List tanks, but to all tanks.

CONCLUSION

Westinghouse Hanford Company and supporting organizations and laboratories have made remarkable progress toward resolving waste tank safety issues. Three unreviewed safety questions were closed in 1994, the worst tank was successfully mitigated, understanding chemical reactions limits has improved dramatically, instrumentation to enhance understanding of tank conditions is being installed on an accelerated schedule, and an innovative cost saving approach has been developed to screen all of the tanks.

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18-4

OPERATIONAL EXPERIENCE IN MITIGATING FLAMMABLE GAS RELEASES FROM HANFORD TANK 241-SY-101

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ABSTRACT

Flammable gases consisting of hydrogen, nitrous oxide, ammonia, and methane are periodically released from Hanford Site waste tank 241-SY-101 at concentrations above the flammable limit. A large mixer pump installed in the tank in 1993 has effectively mitigated this problem by continuously releasing small amounts of the flammable gases at the rate they are generated.

BACKGROUND

Since its original fill in 1980, Hanford Site high-level underground waste tank 241-SY-101 has periodically released flammable gases as a result of a buoyant rollover of the lower viscous layer of waste. At intervals of approximately 100 to 150 days, the flammable gases hydrogen, nitrous oxide, ammonia, and methane evolve episodically from the tank at concentrations above the flammable limit. Because of the potential consequences of ignition, this problem has received a high national priority for mitigation.

In July 1993, a large mixer pump was installed in tank 241-SY-101 to test whether mixing could mitigate the episodic hydrogen releases. The results of initial testing were reported in February/March at Waste Management '94 by Lentsch et al. (1). In April 1994, a series of full-scale tests were completed, and the pump was placed into routine operation.

TANK AND MIXER PUMP

Hanford Site tank 241-SY-101 and the mitigation mixer pump are shown schematically in Fig. 1. The tank is 23 m (75 ft) in diameter and contains 4,163,953 L (1.1 million gal) of caustic sodium nitrate/nitrite/aluminate waste with 3% to 5% organic complexants. Approximately 3 million curies of radioactive Cs137 in the tank generates 40,946 Btu/hr (12 kW) of decay heat, resulting in an average waste temperature of approximately 49C (120F). Overall waste depth is 1,016 cm (400 in.); the lower 635 cm (250 in.) consists of non-convective settled solids with a specific gravity of 1.7, covered by approximately 254 cm (100 in.) of convective saturated solution with a specific gravity of 1.4. A 102 to 127-cm (40 to 50-in.) layer of solid crust floats on top of the waste.

Prior to mixer pump installation, gases formed by radiolysis were retained in the lower (non-convective) waste layer. Every 100 to 150 days, the gas inventory built up to the point where the lower waste layer was less dense than the upper layer. The waste would then "roll over" or "burp," releasing the gas inventory to the dome of the tank, over a period of a few minutes, at concentrations above the lower flammability limit.

The mixer pump shown in Fig. 1 is a 150-hp, submersible, centrifugal unit that draws waste from the middle of the tank at 660 cm (260 in.) and discharges it through opposing 7-cm (2.6-in.)-diameter converging nozzles at 71 cm (28 in.) off the bottom at a maximum velocity of approximately 20 m/sec (65 ft/sec) and flow rate of 10,599 L/min (2,800 gal/min). The pump nozzles can be rotated through a 360-degree sweep. The mixer pump is located in a tank riser located 3 ft off center in the tank. Two multi-function instrument trees (MITs) are located at distances of 8.5 and 9 m (28 and 30 ft) from the pump to measure the vertical temperature profile of the waste at .30 to .61-m (1 to 2-ft) intervals. These two MITs provide the best measure of the degree of mixing taking place in the tank.

Tank 241-SY-101 is also equipped with multiple high-sensitivity gas monitoring systems and level detection systems to measure the quantity of gas that is retained in and released from the waste.

TEST AND OPERATIONAL RESULTS

The best measure of the effects of pump mixing on gas retention in the waste is waste level. A graph of the waste level before and after mixer pump operation is shown in Fig. 2. Prior to pump installation, the waste level grew at a rate of about 0.25 cm (0.1 in.)/day as gas accumulated. Following the rollovers, or burps, the waste level dropped 0.30 m (1 ft) or more after the gases were released. Then the cycle took place again. This behavior has not occurred since the pump was installed in July 1993. The first 4 months after the pump was installed, it was only operated for 5 to 10 minutes per day maximum. This operation was adequate to keep the pump nozzles from clogging with the viscous waste at the bottom of the tank. During this period, the waste level grew 5 to 8 cm (2 to 3 in.), but at a reduced speed. As more aggressive mixing was performed (up to 3 hours per day with 360-degree sweeps of the tank), the tank level was knocked down to a historical low value of 400 in.

Fig. 2. Tank SY-101 Surface Level.

Following demonstration testing in 1993, gas was allowed to regrow into the waste

during early 1994 to show that mixing had not altered the waste or created any safety issues. During February-April of 1994, full-scale testing was performed. Daily test sequences of 5-minute, 1-hour, and 3-hour runs and 360-degree directional sweeps were performed to determine the minimum pump operation necessary to control gas releases, and to assess how deep the pump jets were penetrating into the waste. From the full-scale tests and experience gained since then, it has been determined that the mixer pump should be operated at a speed of 1000 rpm for 25 minutes three times a week. Figure 2 shows that since the summer of 1994, the waste level has been closely controlled at 1,013 to 1,016 cm (399 to 400 in.) by this pump operating frequency. This frequency has also been shown to effectively prevent clogging of the pump nozzles.

Figure 3 shows a plot of hydrogen release data during a typical month of pump operation. Hydrogen is measured continuously with electrochemical cells and with gas chromatographs. Each 25-minute operation of the pump typically causes an increase in hydrogen concentration of 50 to 200 ppm. These values are not only controllable, but are far lower than the concentrations of up to 50,000 ppm that were seen during the naturally occurring, uncontrollable burps prior to pump operation.

Fig. 3. Hydrogen Releases from Tank SY-101 During Mixer Pump Operation.

Since September 1994, a steady state has been maintained using mixer pump operation. An average of approximately 3 m3 (100 ft3) of total gas is released each day. This amount corresponds with the approximately 100 ft3 of total gas that is estimated to be generated by radiolysis in the tank.

Concentrations of other gases in the tank dome exhaust are measured continuously with a Fourier transform infrared analyzer. Concentrations of the oxidizer gas nitrous oxide show close correlation with hydrogen at approximately 1.5 times higher values than hydrogen. Ammonia, on the other hand, shows a more steady release of roughly 35 to 50 ppm with only small increases of 10 to 15 ppm during pump operation. Much higher concentrations of ammonia (to 2,000 ppm) were seen during burps prior to pump installation). Only trace amounts of methane have been detected at concentrations of 1 to 2% of the nitrous oxide concentration.

JET PENETRATION

Waste temperature profiles were used to measure how deep the pump jets are penetrating into the waste. Figure 4 shows graphs of waste vertical temperature profiles from the thermocouples of one of the MITs (this tree is located 8.5 m [28 ft] from the pump). Profiles are shown for a date prior to pump installation, and for a recent date during steady state pump operation. The parabolic temperature profile shows that prior to pump installation, the lower 635 cm (250 in.) of waste are unmixed (non-convective). With pumping, only the lower 4 ft of waste is non-convective. Further, when the pump is running, the temperature reading on the very lowest thermocouple on this tree (10 cm [4 in.] from the tank bottom) clearly increases when the warmer pump jet passes by.

Fig. 4. Tank 241-SY-101 Multi-Instrument Tree Riser 17B Temperature Profile.

The other MIT, located 30 ft from the pump in the opposite direction, shows less than 41 cm (16 in.) of unmixed waste. However, each successive operation of the pump further reduces the thickness of the unmixed waste.

Recent operation of a newly-developed instrument for measuring the fraction of gas voids in the waste has confirmed the mixing depth inferred from the MITs. Also, infrared photogrammetry of the primary tank wall was performed from the tank annulus. The tank wall temperature was shown to be vertically uniform before and during pump mixing. From this and thermal calculations it has been inferred that the waste is vertically well mixed all the way out to the cylindrical tank walls, at a distance of 10 to 12 m (34 to 40 ft) from the pump.

The mixer pump cannot be operated continuously due to thermal operating limits. However, the continued systematic operation at 3 days per week is expected to eventually mix all of the tank waste all the way to the bottom in all directions.

PUMP REPLACEMENT

Because of the proven success of the mixer pump for mitigating tank 241-SY-101 flammable gases, keeping a pump operating in this tank at all times has become essential. Thus, a spare pump has been fabricated and is ready for installation in case the current pump fails. The estimated lifetime of the current pump is approximately eight years, based on its current duty factor, operating temperature, and radiation exposure.

Removal of the current pump, in case of a failure, is expected to be a difficult

task involving radiation levels up to 50 R/hr. The pump is nearly 21 m (70 ft) long and weighs approximately 9,072 kg (20,000 lb). Remote equipment to decontaminate, monitor, and contain the pump during withdrawal is currently being tested, and a special shielded container and hydraulic trailer assembly have been built to contain and transport the pump after removal. Detailed plans and training are being developed to assure that a failed pump can be removed and a new pump installed within 30 to 60 days of a failure, before hydrogen builds back up to unacceptable quantities.

Testing is also underway to determine what amount of dilution of tank 241-SY-101 would be required to "passively" mitigate the tank (thinning the waste to the point where it no longer retains gas). A 1:1 dilution (diluent: waste) might be needed. At this time, there is insufficient tank capacity in the West area at Hanford to accommodate the additional waste volume that would result from dilution.

CONCLUSIONS

The mixer pump in Hanford Site tank 241-SY-101 has been in operation for 20 months. It has effectively eliminated the episodic release of flammable quantities of hydrogen and other gases. Although the pump is only operated about 1% of the time, it has been shown to mix most of the viscous settled solids in this tank. A steady state has been reached where the pump releases gases at the same rate as they are generated.

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HANFORD TANK WASTE REMEDIATION SYSTEM: BASELINE SIMULATION MODEL

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ABSTRACT

A dynamic event simulation model for the Hanford Tank Waste Remediation System (TWRS) top-level functions has been developed for the in-tank enhanced sludge wash option. The model simulates the expected activity occurring between the 177 underground waste storage tanks during the waste remediation campaign period of about 30 years for the current TWRS baseline case. An animation is also developed which shows the material flow of tank farm waste from storage tank through processing to treated waste form. Preliminary results are reported.

INTRODUCTION

The primary mission of the Hanford site is waste clean-up and site remediation. A major aspect of the remediation is long-term protection of the environment from hazardous waste stored in 177 underground tanks at Hanford's 200 Area plateau. The hazardous portion primarily consists of longer-lived fission products from reactor fuel processed during various separations campaigns over the past 40 years. Remediation involves retrieving, treating and immobilizing the waste in a highly stable glass material for safe disposal in an isolated underground location. The activities and functions necessary to accomplish tank waste remediation in a timely/cost-contained manner are being studied to support the Tank Waste Remediation System (TWRS) Program at Hanford (1).

To assist with the study, a dynamic simulation has been developed to help predict how process and logistic assumptions influence behavior of the system. The

simulation tracks tank waste volume from storage tank to final treated waste form. Tracking involves numerically simulating the intermediate processes which change and transfer waste. Therefore, the primary dynamic variables are the waste volumes contained in the 177 underground storage tanks. The primary processes which are modeled and thereby control the dynamics are the following:

1) Concentration of dilute liquid waste in the double shell tanks (DST) to provide space for storage and processing,
2) Retrieval of the waste from remaining DSTs and from single shell tanks (SST),
3) Separation of liquid and solid portions of retrieved waste,
4) Pretreat liquid portion to remove higher activity aqueous components,
5) Vitrification of the solids and separated aqueous components, and
6) Concentration and vitrification of the pretreated liquid portion in LLW glass facility.

The interrelationships of the various processing activities are shown schematically in Fig. 1. The simulation model estimates flow rates, capacities, and interdependencies associated with the boxes in Fig. 1 to estimate the time dependence of waste moving through the system (2).

The model is organized such that storage capacities are fixed and preassigned rather than floating. Therefore, the delays caused by throughput bottle-necks are computed by the model and can be displayed in the animation. There is no strong advantage to this organization for performing sensitivity studies; but, it is necessary to drive an animated representation of the simulation.

Fig. 1. Top-level processing activities.

General tank farm information used by the model is contained in three text files which are read during program execution. Initially, all waste volumes are set by two of these data files to the current double and single shell tank waste inventory reported by Hanlon (3).For the SSTs, the third data file contains a retrieval schedule which directs the removal of waste from the tanks to designated transfer annexes. The schedule is based on a modification of the retrieval sequence outlined by Williams (4) to reflect the recent Triparty Agreement (5) (TPA) goals. Retrieval is one of the first activities to occur and continues well into the simulation when other activities are concurrently being performed. Even though a relatively fast retrieval rate is conceivable, transfers are constrained by the available volume in designated target tanks and by other processing activities involving those tanks. Since the model accounts for this type of intertank dependency, the retrieval schedule is best understood as tentative - the actual retrieval date is tentative, yet the retrieval sequence is preserved. As already mentioned continued retrieval of the SST wastes is postponed until a certain number of DSTs become available for storage and for performing the in-tank wash. Additionally, dependence on the operation of new (to-be-built) facilities coming on-line will also affect the actual time of retrieval during the simulation. Presently, the simulation assumes the existence of transfer annexes, cross-site transfer line and 2 new 1 Mgal double shell tanks. The assumptions are consistent with the TPA milestones (5), since the simulation starts in the year 2003. The time at which particular resources and facilities become available can be easily varied in a sensitivity study.

On the 200 East side, waste received from the B and C-Farm annexes is stored directly in the A-complex DSTs which also serving as feed for the in-tank wash. On the 200 West side, waste received from T, U, and S-Farm annexes is stored in the 3 SY-Farm DSTs, the 2 new 1 Mgal capacity DSTs. When one or more of the 1 Mgal DSTs are full, a cross-site transfer to 200 East Area is attempted. Cross-site transfer is allowed, if space in AN or AW-Farm is available. As waste slurry is transferred from the transfer annexes to the DST system, in-tank washing of dispersed solids begins becomes the rate limiting process in the simulation.

The most complex intertank dependence in the model occurs within the in-tank sludge wash process. This process allows the possibility of requests for transfers from multiple tanks to a single tank and/or partial transfer from a single tank to multiple tanks.

SIMULATION MODEL ASSUMPTIONS AND SELECTED RESULTS

LIQUID-SOLID SEPARATION/SLUDGE WASHING

To reduce the fraction of waste going to HLW vitrification a multitank settle/decant of retrieved waste slurry and sludge washing of dispersed solids is considered. The first wash effectively occurs during retrieval as insoluble solids are mobilized and dispersed through the retrieved waste slurry. Retrieved waste is stored in 101-102AN

and 103AW tanks which supply feed to settle/decant tanks. An exception is made for retrieved waste with less than 0.2 wt.% solids in this case the waste is sent directly to feed for low level pretreatment. For example, retrieved TX farm saltcake will completely by-pass sludge washing. Since only tanks with small amounts of sludge are retrieved before 2009, no significant accumulation of solids occurs in settle decant tanks before 2009. Figure 2 shows the (4 tank case) volume of total waste and of sludge in a first stage settle decant tank over the entire processing time.

For this study, it is assumed that settle/decant and washes occur within the same tank. Other schemes involving waste transfers could more effectively accomplish mixing and chemical additions, but overall timings are not expected to change significantly. Notice that the first cycle of Fig. 2 is extended because of the small amount of solids introduced for early times. Similar behavior occurs for other settle/decant tanks. Subsequent cycles show a mild variation based on the retrieval sequence. The timing and number of cycles depends on the batch size as well as the fraction of solids assumed for the settled sludge.

Fig. 2. Volume of waste in a 1st stage S/D tank.

The details of a typical cycle can be seen in Fig. 3. The first full cycle of Fig. 3 shows available waste added to a settle/decant tank in the beginning of 2012. At the end of the 30 day settling period, 250 kgal of sludge has settled to 20 wt.% within the tank. The supernatant liquid is decanted and additional waste slurry is added to obtain the specified 10 ft (330 kgal) of settled sludge batch for washing. Caustic is added (1:1 by liquid volume) to the sludge with a 30 day mixing period followed by a 30 day settling period. It is assumed that 30 % of the solids are dissolved (6,7). The wash is decanted and three additional water washes (2:1 by volume) are performed to remove interstitial sodium from the sludge. Since the density of the interstitial liquid is reduced by removal of sodium, the volume of the sludge increases slightly during the water washes. Finally, water is added to transfer the sludge to high level waste accumulation tanks (101-102AZ,101-102AY, & 107AN).

Fig. 3. Typical volume cycles for a 1st stage S/D tank.

Table I summarizes the processing steps and timings assumed for the simulation. While 20 wt.% solids in settled sludge is taken as the base case, the precise value will depend on specific waste characteristics, settling time, and chemical additions. A fixed settling time of 30 days is assumed in the model (of course longer settling occurs if space limitations restrict decanting). Table II shows the end dates for sludge washing for various solids/sludge fractions for two different batch sizes.

For 4 tank case at 100% TOE, Table II shows that the sludge wash operation completes in 2020. This is almost two years after the completion of SST retrieval, since the final years are spent processing waste within the DST system. The actual retrieval completion dates for the calculation of Table II occur about 2 years earlier. In following sections, additional calculations are made for retrieval progress which apply an overall 60% TOE to the timings presented in Table I.

INITIAL CONDITIONS AND DST ASSIGNMENTS

The start time of the simulation is assumed to be the beginning of CY 2003. While numerous tank transfers and use of 242-A evaporator occurs prior to 2003, the net effect of those activities is approximated. This net effect includes the presence of 8 Mgal of 7 M Na waste stored in AP farm as feed for low level pretreatment and 4 Mgal in 200 West area resulting from 101SY and 103SY retrieval. As low level pretreatment starts-up late in 2004 (processing feed at 30 gpm), AP farm tanks become available. The seven tanks 102-108AP continue to serve as feed (lag storage) to low level pretreatment. Once emptied, 101AP along with 104-106AW are assigned as second stage settle decant (S/D) tanks. For first stage S/D tanks, 103-106AN and 101-102AW are assigned, yet Table II assumes only four first stage S/D tanks are used. If the use of 242-A evaporator continues beyond 2004, 102AW and 106AW may not be available for these assignments.

The DST sludges remain in 101-102AZ, 102AN, 107AN, and 101AY; additionally, it is assumed that retrieved solids from 106C are contained in 102AY. Retrieval of the six DST`s with sludges begins in 2009 with the availability of high level vitrification. After their retrieval, 101-102AY, 101-102AZ, and 107AN are used for sludge accumulation, while 102AN becomes a receipt tank for retrieved SST waste. Tanks 101AN and 103AW are also assigned for receipt of retrieved SST waste. When available, cross-site transfer can be made to any of these three receipt tanks, but

it is assumed that 200 East area retrieval of SST`s is restricted to 103AW and 102AN. Effectively, 101AN is exclusively reserved for cross-site transfer.
Two new DST`s are assumed for 200 West area for retrieval of 101SY and 103SY; they continue to serve as storage of retrieved SST waste prior to cross-site transfer. No new DST`s are assumed for 200 East area from in-tank processing considerations.

SST RETRIEVAL

The retrieval of each of 149 single shell tanks containing distinct wastes is considered. While a large range of chemical constituents are contained in the tanks, only sodium, unidentified anions (mostly nitrates), water and solids are represented in the simulation. Hydraulic retrieval (or sluicing) is assumed in each case. In the simulation, this is accomplished with 5 waste receiver facility (WRF) annexes (each containing four 75 kgal tanks) distributed among the 12 single shell tank farms (9). Table III shows the grouping of SST farms with the five WRF annexes.
A given WRF may services more than one SST at a time from it`s assigned farms. The historic retrieval rate has an average instantaneous value of 14.4 m3/day of sludge or saltcake (6). Assuming this rate for four sluicers with a 70% equipment availability along with an overall 70% efficiency, each waste receiver facility has the retrieval rate capability of (0.49)14.4 m 3/day or 28.2 m3/day of sludge or saltcake. It should be mentioned that these assumed rates may be overly optimistic, since (without bottlenecks) a single WRF could retrieve all SST waste in 14 years. This is purposely imposed to insure overall process timing is controlled by sludge washing rather than by retrieval. Water addition is calculated on a tank-by-tank basis for a dilution to 5 M Na or 10 wt.% solids. Overall, the 36 Mgal of SST waste is diluted to 143 Mgal.
In 200 West Area, waste is stored prior to cross site transfer in two new 1 Mgal DSTs and in previously retrieved SY Farm tanks. When one million or more gallons of retrieved waste is accumulated, a cross site transfer to 200 East area is sought. Transfer is made at 200 gpm when one or more targeted AN Farm DSTs become available. For 200 East Area, waste is transferred from the WRF directly to 102AN or to 103AW. The choice of the retrieval sequence will effect process timing. Low level pretreatment is available to process supernatant liquid by 2005, yet space for additional sludge accumulation dose not appear until after 2009. Therefore, prior to 2009 the following tanks containing primarily saltcake are specified for retrieval: 110TX, 111TX, 112TX, 113TX, 114TX, 115TX, 116TX, 117TX, 118TX, 102TX, 104TX 105TX, 106TX, 107TX, 108TX, 109TX, 102TY, 109SX, 102BY, 103BY, 112BY, 101A, 101AX, 102S, 105S, 108S, 112S, 102U, 103U, 105U, 106U, 107U, 108U, 109U 111U.
Processing saltcake early in the retrieval sequence offers an additional advantage---the reduced dilution of sludge for later retrieval times will reduce the cycle time for sludge accumulation in a settle decant tank. Any further study attempting to optimize process times for sludge washing will need to consider retrieval sequence variability.
With the retrieval assumptions outlined, Fig. 4 show retrieval progress for 100% and 60% TOE for three cases in which 4, 6, and 8 DSTs are assigned as first stage settle decant tanks.
Fig. 4. SST Retrieval with 60 % TOE for Sludge Wash
Notice that significant differences in retrieval progress do not occur until after 2009 when full cycles for sludge washing become routine. Assuming the overall 60% TOE applied to the timings of Table I, Fig. 4 shows that 6 first stage settle/decant tanks just miss the SST retrieval TPA milestone. At the 60% TOE, 7 DSTs are sufficient to meet the SST retrieval milestone.

SUMMARY OF RESULTS

A dynamic event simulation model for the Hanford Tank Waste Remediation System has offered valuable insight for process timing and system performance behavior. System performance studies with respect to a few parameters are compared with milestone expectations of the Hanford Tri-Party Agreement. Assuming a 20wt% fraction of solids in sludge settled for 30 days, 7 existing DSTs are sufficient to perform the in-tank settle-decant/enhanced sludge washing and meet the TPA requirements for SST retrieval. Naturally, more tanks are required if smaller weight fractions or longer settling times are assumed.

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18-6

PERFORMANCE AND RISK ASSESSMENT OF SUBSURFACE BARRIERS FOR SINGLE-SHELL TANK WASTE RETRIEVAL

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ABSTRACT

Subsurface barriers are among various alternatives under evaluation to mitigate the threat of leakage from the Hanford Site's 149 single-shell high-level radioactive waste tanks. The Tank Waste Remediation System (TWRS) division of Westinghouse Hanford Company is conducting this evaluation of subsurface barriers and other alternatives, focusing on risk and cost as performance measures.

A number of alternative retrieval/closure approaches were evaluated in terms of risks (carcinogenic and toxicological) to a postulated maximally exposed individual. In addition, worker and accident risks were evaluated and factors developed for each alternative on a relative basis. The work performed to date indicates the use of subsurface barriers may potentially reduce public risk by limiting contamination of groundwater below the Hanford Site; however, the cost in terms of actual funding and in elevated worker risk is significant. The analyses also assume certain performance levels for technologies that have not been demonstrated in field conditions similar to Hanford Site tank farms.

The evaluations summarized herein are being used to support a decision by representatives of the U.S. Department of Energy, Richland Operations Office, the Washington State Department of Ecology (Ecology), and the U.S. Environmental Protection Agency (EPA) regarding potential further development of subsurface barrier technology.

BACKGROUND

The Hanford Site's 149 single-shell tanks (SSTs) currently store approximately 140 million liters (37 million gallons) of high-level radioactive liquid waste. Objectives for environmental cleanup of the Hanford Site include retrieving, treating, and disposing of the SST waste in an acceptable manner. The TWRS program is responsible for achieving these objectives in accordance with the Tri-Party Agreement, U.S. Department of Energy (DOE) orders, and state and federal regulations applicable to environmental clean-up.

Historically, tank wastes have been released to the ground from leaks from SSTs, and associated transfer lines, and other miscellaneous spills. Sixty-seven SSTs are assumed to have leaked a total volume of approximately 2,271 to 3,407 m3 (600,000 to 900,000 gal). Several studies have indicated that the contaminants present in this

volume of leakage could cause violation of groundwater quality standards, once the contaminants migrate through the vadose zone to the groundwater.
The Hanford SST tank farms were constructed from 1943 to 1964. Retrieval of waste from the SSTs is currently scheduled for a 15-year processing campaign beginning in 2003 and ending in 2018. One tank, C-106, is planned for retrieval in 1997 to provide a demonstration of retrieval technology. Accordingly, the SSTs will be 60 to 75 years old at the initiation of retrieval, depending on the sequence for individual tanks.
Hydraulic sluicing is the primary approach currently envisioned for retrieval of the SST waste. The sluicing operations will typically add some level of working liquid to the tank to mobilize the solidified salt cakes and sludges. The design life of the tanks has expired, and the integrity of the tank containment boundaries is in question. The age of the tanks and the fact that leakage paths exist in a number of tanks already strongly indicates that sluicing retrieval operations will likely cause additional leakage of tank waste to the surrounding soil unless preventive actions can be taken.
The concerns of environmental impact due to migration of the contaminants that have already leaked from the tanks and the potential for additional leakage during retrieval have driven the initiative to evaluate alternatives for leakage mitigation. The TWRS program is investigating a number of options to mitigate past and potential future leakage from the SSTs. The evaluation and decision processes are reflected in the following Tri-Party Agreement milestones:
M-45-07A Complete Evaluation of Subsurface Barrier Feasibility (September 1994)
Complete a feasibility study of barriers to accomplish the following:
1. Estimate the potential environmental impact of waste storage and retrieval activities without the application of barriers.
2. Establish functional requirements of barriers to minimize the impact associated with the waste storage and retrieval activities.
3. Evaluate the application of existing subsurface barrier technologies to meet functional requirements of barriers and the potential reduction in environmental impacts from the application of barriers to SST waste storage and retrieval activities.
M-45-07B Reach Decision on Whether to Proceed with Demonstration (January 1995)
Based on the results of the subsurface barrier feasibility study, Ecology, EPA, and DOE will make a decision on whether to proceed with a sub-scale demonstration. If the decision is negative, then interim milestone M-45-07 will be considered complete.
EMERGING TECHNOLOGIES
A number of subsurface barrier development projects are active in the DOE Office of Technology Development (EM-50) Program under the Crosscutting Technology, Plumes, and Landfills Focus Areas. Industry has made a number of proposals to DOE-Headquarters and the field in technologies such as a heated, continuous air barrier and a cryogenic soil freezing system. The continuous air barrier concept is designed to stop/prevent leaks by crystallizing salt in the soil pores by evaporation, thereby sealing the soil from further leaks. The soil freezing system is designed to create a basin of ice beneath the tank, thereby providing a mechanism for capturing any leaked waste. The Underground Storage Tank - Integrated Demonstration tasks conducted in fiscal years 1992 and 1993 (TTPs RL421210 and CH321203) began work on an instrumented test facility and evaluated several barrier systems. The proponents of a number of these subsurface barrier systems, including the air barrier, frozen soil basin, and a technology that provides a physical barrier adjacent to and beneath the tank, claim that they are "ready for demonstration."
The Plumes Focus Area is supporting ongoing research and development of materials that may serve as effective solid barriers around tanks and other waste sites. These materials include organic polymers, cementitious grouts, and paraffins. While several of these materials appear promising, methods of injecting the materials into the soil to effect a seal around tanks are largely untested. Several fiscal year 1995 projects will focus on deployment of subsurface barriers below waste sites. One

example is called the horizontal soil saw, a device that is pulled below the waste site by cables previously placed in directionally drilled holes. As it moves through the soil, a rotating "grouting bar" both cuts the soil and injects barrier-forming grout or other material.

The Hanford Site's TWRS Program has conducted several studies and workshops to evaluate the potential application of subsurface barriers at the tank farms. Based on this work, three subsurface barrier concepts have emerged for consideration: 1) injected or infused material barriers, 2) cryogenic barriers, and 3) desiccant barriers. These barrier types may be installed in two configurations: close-coupled (against the tank structure) and stand-off (with a soil layer between the tank(s) and barrier).

PERFORMANCE AND RISK ASSESSMENT

The overall objective of the feasibility evaluation was to analyze a logical set of subsurface barrier technologies in terms of overall system performance, as measured primarily by public risk reduction, worker and accident risks, and cost.

Integrated Alternatives

To compare the various alternatives on an even basis, integrated alternatives were constructed by assuming a retrieval method, the use or nonuse of subsurface barriers for leakage mitigation, and several closure methods. A total of fourteen alternative retrieval/leakage mitigation/closure approaches were evaluated. Figure 1 provides a flow chart that depicts the integrated alternatives.

The evaluation process required that a number of assumptions be made regarding performance of the alternative concepts. Key assumptions included the performance of the retrieval system in terms of the percentage of waste removed from the tanks, and the effective recharge rate of the tank farm and vadose zone soil system (with or without a surface barrier) following retrieval.

Public Risk

The public risk assessment was performed as a two-step process: the first step estimated the inventories of contaminants of concern in the various sources, then estimated the rates and durations of the releases to the vadose zone, and the second step modeled the transport through the vadose zone and aquifer to obtain potential exposure to a hypothetical maximally exposed individual (MEI).

The analyses assumed a hypothetical tank farm consisting of 12 tanks with a capacity of 3,785 m3 (1 million gallons) each. The waste composition was assumed equal to the average of the 149 SSTs. Soil chemistry and lithology representative of overall Hanford Site properties were assumed. The SST waste inventory includes more than 150 chemicals and radionuclides; however, a relatively small subset of the constituents are sufficiently toxic and mobile that they can produce a significant public risk impact. Eight constituents of concern were retained in the analyses as the dominant contributors to public risk.

The dominant sources in terms of contribution to the public (MEI) risk values were residual waste in the tanks following retrieval and the contamination resulting from old and new leaks. However, other sources resulted in substantive contributions; other sources analyzed included: waste trapped between steel and concrete, residual waste that diffused or advected into the concrete, residual waste that diffused or advected into barrier material (if applicable).

The Multimedia Environmental Pollutant Assessment System code was used to model the contaminant transport through the vadose zone to the aquifer and to compute risk factors to the MEI resulting from use of the groundwater. Cancer risk factors were calculated for exposure to carcinogens, and hazard index (HI) factors were calculated for exposure to noncarcinogens. The receptor was assumed to have a 70-year lifetime. Exposure pathways included drinking water ingestion, crop ingestion, and animal product ingestion.

The transport and risk calculations were run over a 30,000-year postoperational period. The duration of this model was needed to show the magnitude and time of the peaks in risk to the MEI. Figures 2 and 3 show the carcinogenic and HI curves for all alternatives.

Worker and Accident Risks

Potential risks to worker health and safety were evaluated by further defining the alternatives in terms of individual technology options. The work force needed to support each option was estimated on a life-cycle basis and included technology readiness, capital, operating and maintenance, and decontamination and decommissioning. The risk factors were formulated to address exposure to radioactive

and hazardous materials and physical hazards associated with the activities. Table I shows the results of the evaluation.

Cost/Cost Benefit

Cost estimates were developed for each of the alternatives based on recent engineering studies and other data. The cost estimates were based on the life cycle elements as described in the worker safety analysis. The cost benefit for each alternative was based on risk reduction versus cost. Table II presents costs as total net present worth (TNPW). The relative risk value is the peak carcinogenic risk calculated for the postoperational phase (Fig. 2). Cost benefit (based on carcinogenic risk) is calculated using Eq. 1.

Eq. (1)

The cost-benefit factors calculated based on hazard index are numerically different; however, the relative changes between alternatives would lead to the same conclusions as the values based on carcinogenic risk shown above. To provide clearer resolution among the various technologies, an incremental risk reduction value was calculated using Eq. 2:

Eq. (2)

Table III presents the incremental risk values based on implementation of individual technologies.

Sensitivity of Results

A five-phase analysis was performed to examine the sensitivity of the results to the following: variability in performance of subsurface barriers, variability in geology and waste composition, variability in release rate of constituents, the affects of high leakage and variation in performance of the surface barrier and soil flushing, and variability in vadose zone water potential due to leakage. The analyses concluded the results are highly sensitive to the assumptions regarding performance of the protective surface barrier to limit the affects of recharge to the tanks and vadose zone soil system. The results are also highly sensitive to the assumption that the tanks will be reasonably effective in containing the waste inventory during sluicing operations. Other variations had little affect on results.

CONCLUSIONS

The results of the feasibility evaluation support the following conclusions:

Taking no action would result in risks approximately three orders of magnitude higher than the assumed upper limit (10-4) of the target risk range.

Taking no action other than capping a tank farm with a surface barrier capable of limiting recharge to 0.05 cm/year (0.02 in./year) may result in acceptable risks for some tanks, but only if collapse of the tank domes could be prevented.

The use of either traditional sluicing (assumed capable of achieving 99% tank waste retrieval), robotic sluicing (99.9% retrieval), or mechanical retrieval (95% retrieval), in combination with stabilizing the structure of emptied tanks and using a surface barrier, appears potentially capable of attaining the target risk range for most tanks.

The use of any of the subsurface barrier concepts (chemical, freeze wall, and circulating air in close-coupled and standoff configurations) in general applications to tank farms would result in a relatively small incremental reduction in the risk level achievable using baseline technologies. (Baseline technologies include traditional sluicing, emptied-tank stabilization, and surface barriers.)

Uncertainty in the performance of subsurface barriers is high, but because the impact of subsurface barriers on risk and cost-effectiveness is very low, even best-case assumptions of subsurface barrier performance have a relatively small effect on improving overall risk and cost-effectiveness of SST disposal options.

Stand-off barriers offer no benefit in terms of reducing public risk unless soil flushing or excavation is used in conjunction. The requirement for soil flushing in the stand-off barrier alternatives results in a large cost impact.

The use of a close-coupled barrier to support clean-closure activities may be cost-effective in comparison to the clean-closure alternative without a barrier because it would limit the volume and reduce the cost of contaminated soil requiring excavation and treatment, while reducing risk.

Except for the clean-closure application, cost-effectiveness of subsurface barrier technologies is essentially equal and relatively low. The cost-effectiveness of the subsurface barriers, calculated by the method most favorable to subsurface barriers, is about 0.0001 times that of surface barriers, and 0.01 times that of the set of baseline technologies.

The retrieval of all tank waste, including tank structures and contaminated soil to effect clean-closure, would likely result in bettering the public risk range. The landfill created to contain washed, retrieved soil and debris from the tank farm would represent a new, but relatively small source of risk.

The clean-closure alternatives would be about as cost-effective as other tank waste retrieval alternatives assuming that all recovered contaminants of environmental concern would be destroyed or treated and disposed offsite in a federal repository, and assuming that benefit can be represented as a ratio of initial risk to achieved risk. If benefit is represented by the difference in these risks, the cost-benefit is two to eight times lower than for the other retrieval alternatives.

Worker risks of the baseline alternative are 2 to 5 times lower than subsurface barrier alternatives and about 10 to 15 times lower than the clean-closure alternatives.

Functional requirements have been established in Functions and Requirements for Single-Shell Tank Leakage Mitigation (1). All functional requirements potentially can be satisfied using any of the subsurface barrier options evaluated.

These conclusions are based on the ability of subsurface barriers to reduce risk and improve cost-effectiveness in general-use applications to tank farms. A broader set of values beside risk and cost-effectiveness should be considered. Conclusions presented here may be modified as a result.

The risk and cost factors and the methods of their calculation used in the evaluation process were developed using the most recent information available, given the available resources and schedule. The evaluation team considers the approach taken to be correct and defensible. However, there is no set standard for this kind of evaluation and decision process. This paper is a very cursory summary of the work performed. Individuals with an interest in reviewing the detailed information may contact the authors to obtain the supporting documentation (2,3).

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STATUS AND PROGRESS IN SLUDGE WASHING: A PIVOTAL PRETREATMENT METHOD

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ABSTRACT

Separation of the bulk soluble chemical salts from the insoluble metal hydroxides and radionuclides is central to the strategy of disposing Hanford tank waste. Sludge washing and caustic leaching have been selected as the primary methods for processing the 230 million L (61,000,000 gal) of Hanford tank waste. These processes are very similar to those selected for processing waste at the West Valley site in New York and the Savannah River Site in South Carolina. The purpose of sludge washing is to dissolve and remove the soluble salts in the waste. Leaching of the insoluble solids with caustic will be used to dissolve aluminum hydroxide and chromium hydroxide, and convert insoluble bismuth phosphate to soluble phosphate. The waste will be separated into a high-level solids fraction and a liquid fraction that can be disposed of as low-level waste after cesium removal. The washing and leaching operations involve batchwise mixing, settling, and decanting within the existing underground storage tanks.

PROCESSING PLANS

The waste will be retrieved from the single-shell underground storage tanks with as much blending as possible. The retrieved waste will be accumulated in million-gallon underground storage tanks to a volume of approximately 1 million L (300,000 gal) of

settled solids. The waste sludge will be processed through a series of leaches using 3-molar caustic, followed by inhibited water washing. Because methods for retrieving single-shell waste are still being developed, early sludge washing will be performed on selected double-shell tank waste with limited blending.

The first sludge to be washed is the neutralized current acid waste (NCAW) that was generated during the first solvent extraction cycle in the plutonium-uranium extraction (PUREX) process. The radionuclide inventory consists of fission products (primarily Cs-137 and Sr-90) with significant amounts of transuranic elements. With the exception of Cs-137, the fission products have low solubility in the alkaline solution and are present in the solid phase. Transuranic compounds as well as aluminum, iron, and zirconium compounds are also present as solids. Soluble components consist of nitrate, nitrite, hydroxide, carbonate, aluminate, and sodium ions.

The NCAW is stored in two 3.8 million-L (1 million-gal) tanks. Both tanks will be decanted using a floating suction pump down to a level of 30 cm (12 in.) above the sludge, or until the suspended solids concentration in the pump discharge exceeds 100 ppm. The first wash water will consist of 3 million L (800,000 gal) of dilute liquid waste, which will be mixed for 7 to 10 days using 300-HP mixer pumps. The contents of both tanks will be combined, mixed with mixer pumps for 7 to 10 days, and allowed to settle for 1 month. The solids will be washed once more with 2.3 million L (600,000 gal) of water and decanted.

It is estimated that washing the two tanks together will remove 97% of the soluble salts while only 0.4% of the insoluble solids will be carried over with the supernatant. A total of 10.2 million L (2.7 million gal) of supernatant solutions are generated, but these will be concentrated by evaporation to 3.8 million L (1 million gal). Thus, the two tanks containing NCAW will become one tank containing washed sludge ready for vitrification and one tank filled with solution ready for cesium removal.

The above outline plan was initiated as the "AZ-101 Sludge Washing Process Test," began when the airlift circulator operation in the tank was terminated on August 4, 1993. The next phase is to install a decanting pump, a control system, and instrumentation, then decant the supernatant to another storage tank. The final phase is to install and test two, 300-HP vertical centrifugal mixing pumps.

Airlift Circulator Operation

Every tank at the Hanford Site designed to contain PUREX high-level waste contains 22 airlift circulators. These circulators mix the tank contents to prevent insoluble solids from settling. If the solids settle and compact, the possibility exists that the high-heat-generating solids can cause steam to form in the sludge layer. Trapped steam might accumulate until it is released suddenly, resulting in a burp or steam bump. Such an event has the potential of adverse consequences due to overpressurizing the tank, thus causing potential airborne release of small quantities of radionuclides. In the case of a large burp, the physical damage may influence the integrity of the tank structure.

In-tank washing or leaching of insoluble solids requires that the solids be allowed to settle so that the solutions can be separated by decantation. This requires that airlift circulator operation be discontinued during the settling period. There was concern that a steam bump could occur during the settling period. Therefore, a computer model was developed to simulate the tank dynamics (1), and a process test was carried out to determine if a bump could occur in the tank with the highest heat-generating rate (2).

The test consisted of turning off the airflow to all the circulators in tank AZ-101. The temperature readings from the many thermocouples permanently installed in and around the tank were monitored and recorded. It was found that the average temperature of the solution increased about 11F, from 136 to 147F. The average sludge temperature increased about 13F, from 141 to 154F, with a maximum temperature of 190F near the tank center when the annulus ventilation system was not operating. With the bubble point of the solution being about 220F at the waste surface and 260F at the bottom, the formation of steam was not possible. The results of the test agreed well with the predictions of the computer model discussed later in this report.

Decantation of Waste Tank AZ-101

To decant tank AZ-101, a decanting pump, instrumentation to measure suspended solids concentrations and the sludge interface level, and a control system were needed.

These have been designed, procured, and fabricated for installation. The total system has been tested at the Sulzer Bingham Pump Co. facility in Portland, Oregon. A drawing of the tank with equipment and instrumentation and including the mixing pumps is shown in Fig. 1.

The decanting pump is a modified vertical turbine pump with a floating suction intake. A flexible metal hose is attached to the intake at one end, and the stainless steel float is attached to the other end. A cable attached to the float controls the position of the float, and allows the float to be pulled up out of the waste. A load cell is used to measure and control the tension on the cable, and an encoder or resolver measures the unreeled cable length.

The pump operation is controlled by a Siemens Power Corporation programmable logic controller, with a remote operator station located outside the tank farm fence line. There is instrumentation to measure the pump discharge flow rate, the discharge liquid turbidity (calibrated in ppm suspended solids with a waste simulant), a tank turbidity profiler, and a buoyancy liquid level monitor. This instrumentation will be wired into the controller so that the pump can be automatically switched between recycle back to the tank and transfer out of the tank or turned off. The position of the diverter valve or on-off status of the pump will be based on discharge flow and turbidity as well as tank waste liquid level and sludge/liquid interface level.

When the decant pump system is installed, tank AZ-101 will be decanted to about 0.3 m (1 ft) above the sludge layer. Then the tank will be refilled with a very dilute wastewater from another tank to reduce radiation exposure for the installation of two vertical centrifugal mixing pumps. This dilute waste will also be used for the first wash of the waste after the pumps become operational. The mixing pumps will be the first installed at the Hanford Site for the purpose of mobilizing and washing compacted waste sludge.

Future Test Plans

Following the testing of the mixing pumps, it is planned to decant a neighboring PUREX high-level waste tank (AZ-102) and pump the contents of tank AZ-101 to it. The combined sludge from the two tanks will then be washed and stored as feed for the future high-level waste vitrification plant. The washed sludge will also be available for laboratory or pilot-scale vitrification tests.

Also planned is a test of in-tank leaching and washing of single-shell tank waste that is scheduled for removal and transfer to a double-shell tank. This waste sludge contains a substantial amount of strontium-90 and large amounts of iron, aluminum, silicon, and phosphorus. It is an ideal waste to demonstrate the caustic dissolution of aluminum and methasis of phosphate to hydroxide. The waste will be sluiced from the single-shell tank C-106 into double-shell tank AY-102, which is in close proximity to tanks AZ-101 and AZ-102.

LABORATORY STUDIES

Laboratory studies have been conducted with both simulated waste and real tank waste. These studies were designed to:

1. Evaluate the efficiency of washing and caustic leaching the waste to separate alkaline water soluble nonradioactive species from radioactive species
2. Determine the settling characteristics of simulated waste, including the effects of high-heat generation in the sludge and pumping the waste through mixing pumps.

These studies were conducted by the Pacific Northwest Laboratory for Westinghouse Hanford Company.

Leaching and Washing of Actual Waste

Samples of waste taken from several single-shell tanks were subjected to a series of water washes and alkaline leaches with sodium hydroxide solutions (3). This was done to determine the solubility of aluminum, chromium, phosphate, and other species in 3 M OH^- solution and water.

Tables I, II, and III data indicate that much of the aluminum and phosphorus, and some of the chromium, can be leached out of tank waste with hot caustic soda solutions. The amount varies with each tank waste because the chemical composition varies from tank to tank. Overall, the majority of the aluminum and phosphorus compounds, and some of the trivalent chromium compounds, can be leached from the sludge with 3 M NaOH solution. Computer simulation of these processing experiments is discussed in a following section.

Settling and Washing of PUREX Waste Simulant

Experiments were conducted to determine the unhindered and hindered settling rates and supernatant clarity of a simulated PUREX waste (4). The experiments were carried

out in a 26-L (7-gal) tank, using 19 L (5 gal) of slurry. The experiments were also designed to measure the effect on settling of natural conduction caused by heat generation in the settled sludge, and to determine if a steam bump could occur. This was done by placing a spiral heating coil on the bottom of the tank. The effect on settling of pumping the slurry through a centrifugal pump was also studied. Preparations were made to test various flocculants as settling aides, but the tests were not carried out because very clear settled solutions were obtained during the experiments without flocculants. Initial settling rates were found to be around 1 cm/h, and supernatant clarities were less than 10 nephelometric turbidity units. Heating the sludge layer increased the settling rate, presumably because the temperature of the solution was increased, thereby decreasing the viscosity and density. Washing the sludge with alkaline, inhibited water also resulted in higher settling rates, possibly again because of a decrease in density. The slurry was pumped through a centrifugal pump with the same impeller tip speed as planned mobilization mixing pumps. No effect was observed on the settling characteristics of the slurry.

The conclusions from the experiments with the PUREX simulant is that flocculants will not be needed to gravity separate tank waste insoluble solids and solutions. Neither radiological heating in the sludge layer nor water washing the sludges will adversely affect the separation. The effects of shearing the slurry through mixing pumps is not expected to affect the settling properties.

THERMODYNAMIC SIMULATION

To predict the results of full-scale leaching and washing of various mixtures or blends of tank waste, a method to simulate processing with computers is necessary. A licensed process simulator can calculate activity coefficients and, thus equilibrium concentrations, for mixtures of aqueous and nonaqueous solutions, solids, and vapors. This simulator, produced by OLI Systems and called Environmental Simulation Program (ESP), includes extensive data banks for aqueous, organic, solid, and vapor phase species. It is being used at the Hanford Site to simulate a wide variety of processing. Two examples are discussed in the following sections.

Leaching and Washing of Actual Waste

Waste samples are taken from Hanford Site storage tanks by coring the tanks in a manner similar to geological coring. These samples are analyzed in a variety of ways. As discussed earlier, they are also processed in the laboratory by leaching and washing to obtain data on the solubility and leachability of the various constituents. A large amount of very valuable data comes from these laboratory leaching and washing experiments. However, because the results are for a single tank waste and are experimental, it is not easy to extrapolate the results to waste mixtures.

The ESP is being used to simulate the leaching and washing experiments. In ESP, the extensive data banks are searched for all species that possibly exist in equilibrium with the input species. Semi-theoretical methods are then used to calculate activities and concentrations for all the species. If the results of the laboratory tests can be predicted with reasonable accuracy, then the results from processing mixtures of tank waste can likely be predicted.

Simulations have been completed of the first three tank waste leaching and washing experiments (5). These simulations were for tanks B-110, C-109, and C-112. Comparisons between the laboratory results and ESP simulations are shown in Tables I, II, and III. The ESP simulations, in general, agree well with the laboratory results.

Sludge Washing Process Test

As discussed earlier, a full-scale process test of sludge washing is in progress in Hanford Site tank AZ-101. In preparation for this test, an ESP simulation was run of the separation and combining of waste that will occur as part of the decanting and refilling of the tank. This simulation was valuable for assessing whether any safety problems could exist, and for determining phase material balances during processing.

COMPUTATIONAL FLUID DYNAMICS SIMULATION

To evaluate the effects on safety of performing various operations in Hanford Site double-shell waste storage tanks, a tool has been needed to simulate the transport phenomena in the tank systems. Of particular concern has been the effects of heat-generating radionuclides, which can cause steam generation and possible "bumping" of the tanks. This need has lead to the development of computational fluid dynamics (CFD) models to simulate waste tank dynamics and heat transfer. The GOTH

computer code has been adapted to simulate the dynamics in a PUREX waste storage tank and ventilation system (1).

Fluid Dynamics in Tank AZ-101

As discussed earlier, the operation of the airlift circulators in tank AZ-101 was discontinued as the initial step in the sludge washing process test. Before shutoff of the circulators, a CFD simulation was made of the tank waste and vapor space dynamics to determine if steam could accumulate in the sludge and then be released suddenly. For this to occur, the transfer of heat from the sludge to the ventilated air space above the tank would have to be slower if the airlift circulators were not operating. This, in turn, would mean that the rate of heat transfer from the sludge layer to the liquid surface is faster when the circulators are operating.

The simulation was designed to simulate the heat, mass, and momentum transfer in the liquid and vapor phases as a function of time. The initial conditions used were those measured in tank AZ-101 with the circulators on. The simulation was then run until near steady-state conditions were reached. GOTH solves for the local fluid velocity, pressure, and temperature in the liquid and vapor space of the tank.

The results indicated that the supernatant solution temperature would rise about 10F, but that the bubble point would not be reached anywhere in the tank after the circulators are shut off. The temperatures in the sludge layer varied, but again, the bubble point of the solution in the sludge was never reached. When the actual test was performed and the circulator operation was stopped, the temperature changes corresponded well to what was predicted by the CFD simulations.

Decantation of Tank AZ-101

There was concern that when tank AZ-101 was decanted, the temperature of the remaining sludge and solution might rise and possibly boil. This decanting process was simulated with the GOTH model to determine what temperatures are predicted to be reached. The results were that as long as some liquid covered the sludge there will be very little change in temperature (6). This is because the force-ventilated vapor space remains well mixed, even when the tank is nearly empty. Thus, the rate of transfer of water vapor, and therefore latent heat, from the liquid surface to the vapor space remains relatively constant during decanting. The CFD model showed that most of the heat removal is through evaporation.

ENGINEERING EVALUATIONS

Although separation of metal hydroxide sludges traditionally has been performed by sedimentation because of the very small particle size, examination of other methods to accomplish the sludge washing process has recently been initiated. If a large radioactive processing facility is required it might make either economic or operational sense to perform this processing on a semi-continuous basis using centrifugal or filtration methods. A study has shown that it may be possible to use a crossflow filter for washing and separating the insoluble solids. This would be accomplished by continuously circulating waste slurry through the filter while adding water to it to maintain a constant suspended solids concentration.

The feasibility and efficiency of both the in-tank and crossflow filter processes needs to be demonstrated.

CONCLUSIONS

Sludge washing and leaching is central to the Hanford Site's approach to disposal of the tank waste. All experimental and analytical work done to date supports the feasibility of washing and leaching the tank waste currently stored at the Hanford Site using an in-tank process. An approach using computer simulation, physical simulant testing, and in-tank testing is being pursued and is providing validation of planning assumptions. By continuing to look at alternative methods it is expected that Hanford Site tank waste will be processed in the most cost-effective manner to allow disposal in accordance with legal commitments.

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THE SORT ON RADIOACTIVE WASTE TYPE MODEL: A METHOD TO SORT SINGLE-SHELL TANKS INTO CHARACTERISTIC GROUPS

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ABSTRACT

The Sort on Radioactive Waste Type (SORWT) Model is a method to categorize Hanford Site single-shell tanks (SSTs) into groups of tanks expected to exhibit similar chemical and physical characteristics based on their major waste types and processing histories. The model has identified 24 different waste-type groups encompassing 133 of the 149 SSTs and 93% of the total waste volume in SSTs. The remaining 16 SSTs and associated wastes could not be grouped according to the established criteria and were placed in an ungrouped category. A detailed statistical verification study has been conducted that employs analysis of variance (ANOVA) and the core sample analysis data collected since 1989. These data cover eight tanks and five SORWT groups. The verification study showed that these five SORWT groups are highly statistically significant; they represent approximately 10% of the total waste volume and 26% of the total sludge volume in SSTs. Future sampling recommendations based on the SORWT Model results include 32 core samples from 16 tanks and 18 auger samples from six tanks. Combining these data with the existing body of information will form the basis for characterizing 104 SSTs (70%). These 104 SSTs represent 79% of the total waste volume, 63% of the total sludge volume, and 88% of the salt cake volume.

INTRODUCTION

Between 1943 and 1964, 149 SSTs were constructed to store liquid and solid radioactive wastes generated during the production of plutonium at the Hanford Site. Over 36 million gallons of wastes are currently stored in SSTs. Before the last tanks were removed from active service in November 1980, various waste volume reduction programs were undertaken to minimize the amount of occupied tank volume. These programs involved intertank transfers, evaporation, and chemical alterations of the waste. These actions, combined with the ongoing chemical and radiolytic in-tank processes, have changed the character of the waste in the SSTs over time. Characterization of these wastes is currently a top priority to alleviate safety concerns and to support the development of retrieval, pretreatment, and disposal systems for the tank wastes.

The wastes in the SSTs originated from a limited number of chemical processes and waste solidification schemes. However, because of the complex physical and chemical history of the tank waste, especially when several different waste types were mixed or processed together, the model does not attempt to predict the precise composition of a tank. Instead, the sorting method concentrates on the different types of waste introduced into each SST, each waste's distinct contribution to the known properties, the individual significance of each waste type, and the process history of each tank. Although the actual chemical reactions and phase equilibria may be unknown when two waste types are combined in an SST, it can be assumed that similar reactions and equilibria occur in other SSTs when the same two waste types are mixed. Therefore, tanks that received the same waste types in the same approximate proportion and had a similar processing history will be more similar to one another than SSTs that received several different waste types in varying amounts and had a relatively unique process history. This forms the basis of the SORWT Model.

Validation of the SORWT Model indicates that a limited number of tanks (representatives of their corresponding SORWT groups) could provide sufficient information to begin developing pretreatment and disposal systems, assuming the selected tanks provide an accurate representation of the conditions within the SSTs.

DATA SOURCES FOR THE SORWT MODEL

The principal source of information used by this model is A History of the 200 Area Tank Farms (1). This document contains much of the available processing history for each of the 149 SSTs from 1944 until 1980. However, the historical records used to generate A History of the 200 Area Tank Farms were often inaccurate and/or incomplete. The methods used to measure accumulated solid and liquid volumes during the early history of the Hanford Site produced inconsistent estimates of inventories. Indeed, solids inventories were not routinely taken until the mid-1950s, and tank transfer information was often missing. Despite these inconsistencies, it is still one of the best sources of SST historical information, and it is believed a qualitative assessment of the principle solids-forming waste types contained in each SST can be accurately determined from this information.

The volume of waste contained in each SST was obtained from the Tank Farm Surveillance and Waste Status Summary Report (2). These values include, on a per-tank-basis, total waste volume, volume of salt cake, volume of sludge, and volume of supernatant liquid. It is assumed that these values are more accurate than those final values found in A History of the 200 Area Tank Farms because they were obtained more recently; however, it is understood that these values have deficiencies because of the limited access to the tanks.

SORWT MODEL ASSUMPTIONS

The underlying assumptions used by the SORWT Model are as follows:

The information contained within A History of the 200 Area Tank Farms is sufficient to qualitatively identify and rank, relative to one another, the waste types that contributed to the accumulated solids in each SST.

Primary solids-forming and secondary solids-forming waste types were responsible for the majority of the physical characteristics and chemical compositions of the waste remaining in each SST.

Supernatant wastes that were not allowed to remain in a tank for a long period of time and were later pumped out of the SST had less influence on the physical and chemical character of the waste than did the insoluble solid waste types that remained in the tank.

Tanks were often sluiced at some time during their processing history. Sluicing involves removing solids from waste tanks using high-pressure water jets. Waste types present in the tank prior to the most recent sluicing were not considered relevant by this model.

Using a broad-ranging, less descriptive waste type such as noncomplexed waste (NCPLX), complex concentrate (CCPLX), evaporator feed (EVAP), and/or double-shell slurry feed (DSSF) was avoided whenever possible. Process-specific nomenclature was preferred, if available; however, a broad category identifying the tank waste as either noncomplexed, complexed, or ferrocyanide-scavenged waste has been included in the SORWT Model to aid in evaluating the results of the model.

SUMMARY OF SORWT MODEL RESULTS

The SORWT Model has predicted the existence of 24 groups ranging from a high of 22 tanks per group to a low of two tanks per group. These 24 groups encompass 133 tanks and 93% of the total waste volume. An additional group contains the 16 SSTs which could not be grouped with any other tanks based on their primary and secondary waste types. Table I presents a summary of the SST groups predicted by the SORWT Model. A review of Table I quickly reveals that Group I is by far the most significant group. This group includes 22 tanks, 37% of the total salt cake volume, and over one quarter of the total waste in all 149 SSTs. The first three groups represent over one-half of the total waste volume in all 149 SSTs. This categorization demonstrates the potential usefulness of the SORWT Model in making management decisions. Table I also identifies groups that have relatively no significance, such as Groups XIV and XVIII, which contain almost no waste. This information can be used in allocating time and resources for characterization activities, pretreatment, and immobilization development.

Larger families of related tank groups may exist. Examples of potential families are Group I (R, EB) and Group IX (EB, R). The relative differences between these two groups are due to their respective designation of which of the two waste types is

primary and secondary. These differences may be small when compared with the overall group variability. Identifying larger families of tanks will reduce the overall number of different groups being evaluated and the corresponding number of sampling and analysis events.

STATISTICAL VERIFICATION OF THE SORWT MODEL

Approach to Verification of the SORWT Model

First, the analytical results were arranged into groups as predicted by the SORWT Model, then an ANOVA was performed on the grouped data for a selected number of analytes. An ANOVA is a quantitative method to test the significance of the effect a particular treatment has on the response or dependent variable. In the SORWT Model verification study, the treatment being studied is SORWT groups, and the dependent variable is analyte concentration. The ANOVA method was used to test whether the mean concentration of a particular SORWT group is statistically significantly different from the mean concentration of other SORWT groups. The null hypothesis tested by this statistical model was as follows:

The deviations between the means of the different groups were due only to random variation within the entire data set.

If the null hypothesis was proved valid, then no group effects were present, and the SORWT Model would be discredited. However, if the null hypothesis was proved incorrect, then the converse would be true (i.e., group effects are present and the SORWT Model methodology is supported by the data). If significant group effects were observed, a Tukey pairwise comparison was conducted to investigate the groups that differed significantly from one another.

Analytical Data Sources for the Verification Study

The analytical results data used in the SORWT Model verification study were obtained from the official core sample data packages produced by the Hanford analytical laboratories in support of the Westinghouse Hanford Company Tank Waste Characterization Program. The SSTs and SORWT groups that were used in the verification study are presented in Table II.

The core sample data packages contain a great deal of analytical data measured using several alternative digestion methods and analytical instrumentation. These measurements were often taken both on segment level aliquots and on core composites, which represent the nominal or average composition of an entire core. Because the SORWT Model verification study compares the differences between the mean nominal compositions of one group and the mean nominal composition of other groups, only core composite data for the analytes that significantly contribute to the overall character of the waste were considered. The analytes included in the verification study, along with the sample preparation method and analytical instrumentation, are presented in Table III.

Graphical Description of the Verification Data Set

The data set used in the SORWT Model verification study consists of 109 separate cases with 22 total measurements per case for a total of 2,398 pieces of information. This is a rather large amount of information to comprehend and only a small subset of the total data available. A useful tool for summarizing and understanding large data sets is a box plot, which is a graphical representation of the spread or variance in a given data set. Figure 1 is an example box plot for sodium (Na).

The example box plot shows the spread in the Na data for the five different SORWT groups to be tested in the verification study. The vertical axis is Na concentration presented in units of mg/g. The horizontal axis represents the five different SORWT groups. The spread in the data is depicted by a box and whiskers plot. The median of a set of data is marked by a horizontal line in the box. The lower and upper hinges are the edges of the central box. The median splits the ordered set of data in half such that 50% of the values are above the median and 50% of the numbers are below. The hinges split the remaining halves in half again such that the interior of the box represents 50% of the data. If we define the hinge spread as the absolute value of the difference between the two values of the upper and lower hinges, the whiskers show the range in values that fall within 1.5 hinge spreads of the hinges. Any data farther than 1.5 hinge spreads from the hinges are outliers and plotted as asterisks (*). Values that are more than three hinge spreads away from the hinges are considered far outliers and plotted as open circles. Examples of both of these outliers can be seen in Fig. 1.

As can be clearly seen in the figure, the median value and range of values for Na in

some of the SORWT groups are substantially different from other SORWT groups. It is also clear that not all groups are necessarily different from one another. It appears that Groups XII and XVI show comparable Na concentrations and that Groups VII and XV are indistinguishable from one another. However, the spread of values from Groups XII and XVI does not approach the spread of values in Groups VII and XV. Group IV appears to be different from all the other groups presented.
Similar box plots were generated for each analyte included in the SORWT Model verification study and are included in The Sort On Radioactive Waste Type Model: A Method to Sort Single-Shell Tanks into Characteristic Groups (3).
Analysis of Variance (ANOVA) of SORWT Groups
The ANOVA performed for each analyte included in the SORWT Model verification study used the general linear model of the SYSTAT for Windows(a) statistical data analysis software package. If a significant grouping effect was observed, then a Tukey pairwise comparison was also conducted for each analyte to determine which groups were significantly different from the others. The output reports generated by the statistical software for each analyte are presented in The Sort On Radioactive Waste Type Model: A Method to Sort Single-Shell Tanks into Characteristic Groups (3).
The ANOVA provides two estimates for the variance, one between groups and one within groups. If the null hypothesis (i.e., no differences among SORWT groups) is accurate, then the estimate for the between-group variance should be similar in magnitude to the within-group estimate of the variance. Conversely, if the between-group estimate of the variance is significantly greater than the within-group estimate, the null hypothesis would be untenable, and some of the between-group variation must be caused by real differences between treatment groups. A summary of the ANOVA results for each of the analytes tested is presented in Table IV. The F-Ratio is defined as the ratio of the between-treatment variance (mean sum of the squares) and the within-treatment variance. This ratio should follow an F distribution for the appropriate numbers of degrees of freedom. The significance of the F-Ratio is called a P-value and can be determined from the relevant F distribution. The significance is the fractional probability of the F-Test ratio occurring only by random chance. The benchmark probabilities typically used to test the significance of differences between means are 5% and 1%, which correspond to significances of 0.05 and 0.01. For the purposes of the SORWT Model verification study, the 5% benchmark was selected. If the significance is greater than the benchmarks, then the differences between treatment means can be explained by random chance. If the significances are below the benchmarks, then the discrepancies between treatment means cannot be explained by random chance, and real differences exist between the subject groups.
As shown in Table IV, all 22 analytes and measurements listed have a significance well below the benchmark 5% level. In fact, all but two analytes have a significance below 0.1%. Table IV indicates there is virtually no probability that the differences between the means of the SORWT groups are due only to random chance. Therefore, the null hypothesis is invalid, and the data strongly support the premise that SORWT groups provide meaningful information about waste characteristics.
Because a significant grouping effect was observed, a Tukey pairwise comparison was performed to identify the groups that were significantly different from one another. The Tukey pairwise comparison first generates a matrix of pairwise mean differences. These are the differences between the mean concentration of a pair of groups. The routine then compares this difference to the mean square error for the analyte calculated from the ANOVA table and calculates a P-value (probability) that the difference between the mean concentration of any two groups is due to random chance. Table V presents a summary of the Tukey pairwise comparisons. The analytes that vary significantly between groups and the total number of significantly different analytes are listed. As shown in Table V, 18 out of 20 analytes were significantly different in Group IV than in Group VII. The smallest number of analyte differences between groups was eight between Groups XVI and XII. More than half the analytes considered in this study were significantly different for 7 of the 10 pairwise comparisons. This is another strong indication that the grouping methodology used by the SORWT Model predicts real differences between the characteristics of tank groups. Density and pH were not included in these Tukey summary tables.
RECOMMENDED TANK WASTE SAMPLING
Tanks recommended for sampling based on the results of the SORWT Model are listed in Tables VI and VII. The list takes advantage of the SORWT Model groups to establish a

substantial amount of characterization information from a relatively small number of core and auger samples. Thirty-two core samples and 18 auger samples are recommended. If this new sampling and analysis information is combined with the existing data, nominal compositions of 104 tanks (70% of the SSTs) could be established. This would represent approximately 79% of the total waste volume, 63% of the total sludge volume, and 88% of the salt cake volume.

ADDITIONAL INFORMATION

Information included in The Sort On Radioactive Waste Type Model: A Method to Sort Single-Shell Tanks into Characteristic Groups (3) but not presented in this report includes the following:

Descriptions of waste types used in the SORWT Model. The waste types are general categories based on process history rather than chemical or physical properties.

Descriptions of the 24 groups predicted by the SORWT Model. A brief history of the tanks included in each group and general comments about similarities between them.

Nominal compositions of the five SORWT groups included in the verification study. Mean concentrations of each analyte were calculated for each SORWT group included in the verification study. These mean concentrations, along with the current waste volume inventories, were used to project an inventory of each chemical analyte over the entire group.

Results of a pairwise comparison of expected and observed analyte concentrations between groups. A semi-quantitative comparison of analyte concentrations between groups was made based on estimated analyte concentrations in various waste types. A similar comparison was made based on the projected nominal compositions of the groups.

RELATED STUDIES

Two additional studies are currently underway to investigate the SORWT Model's effectiveness in grouping SSTs with similar waste types. One study focuses on the physical and rheological properties of the tank waste. It is assumed that tank waste with similar process histories will have similar physical and rheological properties and may therefore be grouped together. The other study is based on chemical analytes; however, the data set used for the verification study will be much more extensive than the one used in The Sort On Radioactive Waste Type Model: A Method to Sort Single-Shell Tanks into Characteristic Groups (3).

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19-1

THE NEW GERMAN APPROACH TO WASTE MANAGEMENT AND FINAL DISPOSAL

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ABSTRACT

By enacting an amendment to the Atomic Energy Act, the German Parliament has recently added a new alternative of waste management to the nuclear fuel cycle back end. The former Atomic Energy Act wording, with its implicit commandment to reprocess all spent fuel, is now complemented to allow long term interim storage and direct disposal of spent fuel as alternatives. This new development is expected to have a substantial impact on all future activities of the fuel cycle back end. During the last 15 years, direct spent fuel final disposal was the subject of an important long term R&D programme. Since final disposal technology is meanwhile

virtually fully developed, real options are available. Policy decisions on waste management can be made now and in the future on a sound, objective basis.

INTRODUCTION

In June 1994 a new waste management option was legalized in the Federal Republic of Germany. By an amended to the valid version of the Atomic Energy Act of July 15, 1985, direct disposal of spent nuclear fuel is now permitted. Research and development work in this field started already in 1979, and it took 15 years from the initial steps to the legal effectiveness.

At the end of the seventies, the International Fuel Cycle Evaluation (INFCE) task force was set up by an initiative of US President James Carter with experts from 60 countries and five international organizations. This task force (INFCE) considered basically the question how the peaceful use of nuclear energy could be made available to all countries to cover their electrical energy needs and at the same time, how the proliferation of nuclear weapons could be reliable precluded. Among the variants analyzed for the back end of the fuel cycle, both reprocessing and direct disposal were considered. The misuse of facilities for peaceful application of nuclear energy to produce or to divert weapon-grade nuclear materials is in principle possible, but it can be however prevented by technical means, safety inspections, and administrative measures.

In May 1979 the Federal Government expressed again that it will adhere to the disposal concept with spent fuel being reprocessed. Nevertheless, it concluded to support investigations of waste management alternatives, like direct disposal of spent fuel, and to evaluate all available options on the basis of INFCE's experience while also considering the national context. This policy was also strongly supported by the Heads of the Federal States Governments. A long term research and development program was set up and launched, with the purpose of developing direct disposal in suitable steps up to technical maturity and evaluating possible advantages regarding safety and fuel cycle economics. At that time first evaluations were expected by the middle eighties.

REVIEW OF DIRECT DISPOSAL DEVELOPMENT

A first study called "Comparison of Different Disposal Alternatives and Evaluation of their Feasibility" was commissioned in 1979 by the Federal Minister for Research and Technology (BMFT) to the Karlsruhe Nuclear Research Center (KfK). Broad participation of other research institutes and the industry was recommended. The study compared final disposal after reprocessing with direct disposal of spent fuel considering technological, ecological, safety and energy policy aspects. A possible proliferation of weapon-grade materials had also to be considered. At the end of this first phase a concluding overall evaluation was found to be not sufficiently feasible, since several important issues remained open.

To resolve these particular remaining questions further research and development concentrated on "Other Disposal Technologies" during the period 1980 to 1985. The main objective was to find out whether direct disposal of spent fuel yields significant safety advantages not to satisfy political requirements.

During the development of a conceptual design for direct disposal of spent fuel a significant amount of technical documents was prepared. In many cases these documents already had a degree of details as required for the safety analyses to license a final repository. With regard to the comparison of both options for the back end of the nuclear fuel cycle a series of evaluation criteria was set considering main criteria as e.g. safety, technical feasibility, nuclear material safeguards and additional aspects, as economics and resource consumption.

Taking into account that such a comparison of complex and different alternatives should support political decisions reaching far into the future, it was necessary to consider not only existing facilities and fully developed technologies.

Consequently, the likely evolution of the state-of-the-art was considered as far as possible and it was assumed that future commercial-scale facilities with a yearly processing capacity of 700 metric tons of heavy metal are applicable. This seemed to be sufficient at that time for the NPP installed capacity forecasted for Germany by the end of the century, of some 25 to 30 GW.

In addition, a conceptual design for a second generation, an advanced reprocessing facility was developed, which should be built at the beginning of the next century. The experience gained during operation of a smaller plant, should be transferred to the final solution. Furthermore, it seemed not to be adequate just to compare different options for the fuel cycle back end standing alone, i.e. reprocessing and

direct disposal. Because of possible changes at the front end, of the fuel cycle such as the use of mixed-oxide fuel, additionally needed front end facilities were considered.
The analyses of several possible variants, led to a reference concept which was selected and developed in detail by companies of the nuclear industry. This concept of direct disposal was compared with the reprocessing path. Evaluation criteria were basically operational and long term safety, nuclear material safeguards and in addition also fuel cycle economics and energy policy issues.
With regard to operational and long term safety both options resulted to be nearly equivalent. The detailed analysis at that time clearly demonstrated a cost advantage for direct disposal, and that at present and for the foreseeable future. No macro-economical consequences were identified, but in the long term and from the view point of technology and energy policy reprocessing was still regarded very important. The final recommendation of the study was to stick to the reprocessing, and to continue developing direct disposal to technical maturity.
At the beginning of 1985, the results were evaluated by the Federal Government. Further development of the direct disposal technology was initiated, to achieve a technical level that fulfills all licensing requirements. Simultaneously, the Federal Government gave to understand that essential impulse for further development and application of these technologies should come from the industry, and especially from the utilities. According to the Atomic Energy Act, the Federal Government would limit its financial support to the field of final disposal.
In addition to the realization of the reprocessing path, direct disposal was further developed up to technical maturity by means of an extensive R&D programme aimed at providing an objective decision taking basis.
Until some months ago reprocessing of spent fuel has been considered as the only legal concept for the back end of the fuel cycle in Germany. The high level waste (HLW) resulting from reprocessing is to be vitrified and, after an interim storage period lasting some decades to allow a first reduction of the decay heat, disposed of in boreholes in a final repository. With the present legal option for final disposal of spent fuel, this fuel must be adequately conditioned and packed prior to final disposal.
A first reference concept consists of the following alternatives: either the fuel rods of disassembled fuel elements or the intact fuel assemblies are packed into self-shielding casks of the POLLUX type. These casks are transported to the repository site, via the shaft hoisted to the underground repository mine, transferred to the disposal position on a railbound carrier, and emplaced in a drift by a remote operated handling system (ELVIS). The remaining void space around the cask is subsequently backfilled with crushed salt.
Disposal of nuclear waste in deep geological formations has been chosen in Germany as the best solution to protect man kind and the environment from the radioactivity present in spent fuel and high level waste. The repository design assures that after closing and abandoning the repository no surveillance or any type of corrective or protective measures are needed. Different concepts have been proposed in several countries, to achieve this, all of them following the basic idea of a multiple barrier system. The multibarrier system is conceived and engineered in such a way that even after failure of some of the barriers the safety is still assured by the remaining barriers.
The barriers are either engineered ore natural. The engineered barriers are tailor made to fit into the site geology to enhance and complement the overall safety. The consideration of natural and man made barriers differs substantially due to the geological conditions of repository sites in various countries. In some concepts the repository safety relies more upon the engineered barriers, in others their role is of minor importance, because of the entrusted isolation function to the natural features of the site geology.
Fortunately, in Northern Germany exists a large number of salt domes with huge dimensions, many of them in principle adequate to host a repository. Due to the unique geohydrologic, thermal, and geomechanical properties of rock salt, as a self-healing impermeable rock, rock salt was very early selected as the preferred host rock for a repository for heat generating wastes. In the German concept the natural barriers are expected to assure by their own the long-term safety of the repository for most of the post-closure period. Engineered barriers enhance operational safety prior to repository closure and provide additionally required

redundancy during some time after the repository decommissioning.
It is important to mention that the advantages of rock salt as a host rock remain fully present for direct disposal of spent fuel. The benefits of salt compared to other host rocks are:

The spent fuel cooling time in interim storage can be minimized since rock salt can sustain a considerably higher thermal load than other host rocks

The viscoplasticity of the salt, enhanced by the waste decay heat, effectively closes all void space remaining in the backfilled drifts, compacting the crushed salt as backfill material after a period of some tens to a few hundreds of years. The backfill has then similar permeability as the original rock salt.

In order to develop the spent fuel direct disposal technology to technical maturity it was necessary to carry out a comprehensive series of demonstrations tests. The target of these tests was to prove the feasibility and safety of the proposed technical solutions. In addition, the results of these tests should provide all information required for licensing this new back end technology, satisfying the legal requirements for a German final repository. It is essential to mention that the required systems and components for the final repository may have to be tested by inactive and active demonstration tests.

In the framework of the mentioned long-term R&D effort DBE was contracted, besides the repository planning, with carrying out tests to prove the technical feasibility of systems and components under realistic conditions. The program which was set up had to supply comprehensive solutions for all relevant open questions with regard to the licensing of components and systems. It covered waste package handling and emplacement technology for drift and borehole disposal. The main objectives were:

construction and test of newly developed mechanical and mining components and systems for disposal technologies;

demonstration of safe handling for waste packages under realistic conditions;

ALARA-aspects, as calculation of radiation exposure to the operating personnel;

verification of temperature field and thermo-mechanical calculations for drift disposal by in situ tests.

One major aspect with regard to a new set of requirements was the handling of the POLLUX cask which will have a weight of 65 metric tons fully loaded with spent fuel. All necessary handling steps for drift disposal were demonstrated in separate test facilities. These are partly finished others are still under investigation:

Thermal Demonstration of Drift Disposal

This test started in 1990, and has now been running for more than four years. The drift disposal test is being carried out in the Asse salt mine at a depth of 870 m by using electrically heated waste POLLUX mock-ups. Among many other parameters the data sampling equipment records the temperature transient at the cask surface, drift convergency, backfill compaction, backfill porosity and permeability, displacements around the drifts, gas generation and release. Before starting the simulation, preliminary tests for selecting a suitable backfilling technique and a comprehensive geotechnical data sampling program were carried out. The test is planned to last until end of 1998. The results at present demonstrate clearly, that specifically the temperature is well within the required limits.

Handling Tests for Drift Disposal

These tests were accomplished to demonstrate safe underground transport, handling, and final disposal of the 65 tons self-shielding waste package of POLLUX type under realistic conditions. It allows to draw conclusions on operation, handling, equipment wear, reliability and availability. The tests were finished by the end of 1994. More than 2000 operating cycles have been performed under realistic conditions. The result was a clear confirmation of the newly designed equipment under normal and faulted conditions.

This test was a major step towards the future realization of the underground emplacement techniques.

Shaft Hoisting Test

Shaft hoisting systems for the handling of radioactive wastes are very sensitive pieces of equipment in a final repository. They are important to safety since a failure can eventually lead to detriment to personnel and to a possible release of activity into the environment. All shaft equipment must therefore be designed to satisfy highest safety standards.

The shaft hoisting demonstration test was run to demonstrate the technical feasibility of a shaft hoisting system for payloads up to 85 metric tons,

including all essential components. Radiological aspects where an essential issue during these tests. The mentioned payload corresponds to the weight of a POLLUX cask and of the flat cask carrier used for the internal transport in the repository. The tests provided conclusive evidence that the designed facility complies with all requirements of conventional and nuclear regulations. The results today are that such a system can now be licensed for a repository.

A shaft hoisting system was designed for a shaft diameter of 7.5 m, a maximum depth of 1000 m, and 85 metric tons payload taking into consideration the relevant boundary conditions, the applicable technical rules, and the operational and safety requirements. The hoisting facility design, and the design of the shaft cage loading and unloading systems on surface and underground as well, were performed with the view to safely prevent all accidents which may lead to a release of radioactivity or to cause severe detriment to the personnel.

Among many other details worth to be mentioned is the airlock door at the surface shaft station, which was designed as a redundant shaft safety gate. It is not only electronically monitored and interlocked, but also equipped with an additional mechanical system to prevent the accidental drop of a loaded cask carrier into the shaft. The shaft cage itself has a unique design, equipped with a movable floor which rests on the cage frame during hoisting and which can be fixed to the loading and unloading stations on surface and underground during loading and unloading. This system allows a very slow and smooth transfer of the payload to the cage ropes, which are tensioned by slowly rising the shaft cage with the hoisting machine.

The single tests carried out comprised a considerable number of loading and unloading cycles for obtaining statistically significant reliability data and the testing of all equipment and devices important to safety. Separate tests were run to simulate operational disturbances. The design of all components with regard to their mechanical stability was verified under normal and faulted operations. All tests have been completed and the final report is close to completion.

Active Handling Experiment

By using active neutron sources the neutron radiation dose around a POLLUX cask mock-up was measured. Special attention was paid to scattering effects in the underground environment. The test, was started with an above ground measurement campaign and continued during the fall of 1994 with underground measurements in the Asse mine. The main target is the validation of the existing calculations for cask shielding and radiation exposure of personnel during operation. These tests are scheduled to end during the current year.

CONCLUSIONS

Successful realization of the above mentioned tests has proven the technical feasibility and licensability of all key system and components of the German spent fuel direct disposal concept. The shaft hoisting test series was a key element since a payload of 85 metric tons had never been hoisted before. Meanwhile, the German Commission for Reactor Safety (RSK) confirmed the feasibility of direct disposal in a recommendation to the Federal Authorities issued on December 8, 1993, and underlined that the test results so far obtained and those of the still running tests enable industry and government to license and construct safe systems for spent fuel conditioning and final disposal.

The mentioned test results were technical preconditions for the authorities to accept direct disposal as a valid waste management option. After the law had passed the German Parliament, the Bundestag, it became effective on July 20, 1994. Hence, now there are in Germany two alternatives available for the fuel cycle back end, spent fuel reprocessing and direct disposal of spent fuel.

19-3

STORAGE/DISPOSAL STRATEGIES IN UN MEMBER STATES

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ABSTRACT

As of the end of 1993, 32 UN Member States had electricity-generating nuclear power reactors in operation or under construction. There are currently 429 operating units worldwide, with a total capacity of about 337 gigawatt-electric (GWe). In addition,

there are 67 units under construction, which would bring the total electrical generation capacity to about 391 GWe.
Generating electrical power, whether the energy source is coal, oil or nuclear, results in a by-product. In the case of nuclear power, the by-product of concern is radioactive waste.
The management of radioactive waste is not a new concept. It has been safely practiced for low- and intermediate-level wastes (LLW/ILW) for almost 40 years. For high-level waste (HLW), including spent nuclear fuel, many countries with nuclear power are developing strategies, approaches and technologies for its disposal. Disposal facilities for low- and intermediate-level waste and other short-lived wastes are in operation. Currently, there are three common disposal options for these types of waste, ranging from direct land disposal to near-surface facilities and geologic repositories.
There is agreement that interim storage will be an important component of the waste management system - most countries plan to store spent fuel and/or vitrified HLW for at least 20-50 years prior to disposal Presently, spent fuel is being stored by some Member States, at the reactor or at a separate storage facility, with the intent for its direct disposal; others are reprocessing the spent fuel and storing the HLW from the reprocessing. The reprocessed waste product will be placed in canisters in a borosilicate glass form and placed in interim storage, with eventual disposal in a geologic repository. To date, about half the countries with nuclear power are planning for direct disposal of conditioned spent fuel.

INTRODUCTION

The use of the atom has come a long way since that famous day in December 1942 in Enrico Fermi's lab. Since then, the atom has been applied to many uses, including the eradication of worms and flies, shelf-life extension of fruits and vegetables, medical research and therapeutic applications, and most notably, generation of electricity. Nuclear power in 1993 supplied over 23% of the total electricity in OECD countries and about 17% worldwide.
However, using the atom creates a by-product of concern: radioactive waste. Although all activities generate some waste, most comes from nuclear fuel cycle activities that support nuclear power plant operations. These activities include four main categories, which are uranium or thorium mining and milling; fuel conversion, enrichment and fabrication; nuclear power plant operations; and decontamination and decommissioning. The types of waste generated are diverse in terms of levels of radioactivity, physical characteristics and volumes.

QUANTITIES OF RADIOACTIVE WASTE

As mentioned earlier, the majority of the radioactive waste arises from the generation of nuclear power. Thus, if nuclear energy is to achieve its potential, countries with or planning nuclear power must develop effective waste management programmes that will safely immobilize and isolate radioactive wastes from human beings and the environment.
While the actual volumes of waste generated from nuclear energy are small relative to other energy industries and alternatives such as coal-burning power stations, one of the major concerns of nuclear energy is the long period of time that radioactive wastes can pose radiological risks to man and the environment. Although the larger portion of the radioactive waste generated has a relatively short half-life (20 to 30 years), there are some elements of the waste product that have half-lives of hundreds to thousands of years.

DISPOSAL STRATEGIES FOR LOW AND INTERMEDIATE LEVEL WASTE

The management and disposal of low- and intermediate-level waste (LLW/ILW) in Member States with nuclear power plants has been established and proven during the past 40 years. In the early days, LLW/ILW were usually disposed of in near-surface disposal facilities, sometimes referred to as shallow-ground burial sites, with little or no treatment and conditioning. However, as national programmes developed a better understanding of the effects and impact of LLW/ILW disposal, many countries have redefined their disposal strategies and practices.
In most Member States, the disposal strategy for LLW/ILW includes near-surface disposal facilities and geologic repositories. In some countries, the present strategy is to dispose of these wastes in deep rock formations. The four major options currently used or planned by countries for waste disposal are: 1) near-surface disposal facilities, 2) rock cavities, 3) sea coastal facilities, and/or 4) geologic repositories.

Status of Low- and Intermediate-Level Disposal Programmes
By the end of 1992, significant progress in a number of UN Member States had been achieved, and LLW/ILW disposal facilities began operations in Finland, France, and Spain.
A disposal facility (Centre de L'Aube) near Soulaines, France, located approximately 200 km southeast of Paris, received its first shipment of waste in January 1992 (operating authorization was received in late 1991). The disposal facility at (L'Aube) will have a disposal capacity of at least 1.2 million cubic meters of packaged short-lived LLW/ILW. The Centre de L'Aube will replace the Centre de La Manche, which has been in operation since 1969. Closure of the La Manche disposal facility will not occur for a number of years. The waste volume at the La Manche facility is expected to be about 500,000 cubic meters of short-lived LLW/ILW.
In early 1992, the Finnish utility TVO completed construction of a final LLW/ILW repository at Olkiluoto, the location of the first Nuclear Power Plant site in Finland. A cave was excavated in the bedrock to a depth of about 70 meters. Waste from TVO's Olkiluoto Nuclear Power Plants will be stored in 200 liter-drums and 1000 liter boxes in the two rock silos. The repository's total capacity is 15,000 drums and 1000 boxes. A second repository of similar design, which is to be located near Loviisa the second Nuclear Power Plan site, started construction in February 1993, and is expected to initiate operations in 1996. The overall capacity of the new repository will be about 113,000 cubic meters. This facility is being built on the Finnish Island of Hastholmen.
Spain's final storage/disposal facility for LLW/ILW at El Cabril in the Andalucia region began operating in 1992. Spain's disposal model is very similar to that being used by ANDRA in France, which is the near-surface disposal concept with engineered barriers for LLW/ILW. The first phase of El Cabril has a capacity of 60,000 cubic meters, sufficient to last until the year 2000. In Spain, ENRESA is responsible for radioactive waste management, including the storage and disposal of radioactive wastes.
The LLW generated by Belgium's seven nuclear power plants is treated and temporarily stored onsite. Solid LLW/ILW is temporarily stored in a storage facility at Mol until the repository is available in the mid-1990s.
The LLW generated by the 18 nuclear power plants in Canada is temporarily stored onsite. A surface land disposal facility is in operation at Chalk River, Ontario, and is used by AECL and Ontario Hydro for LLW disposal. A facility at the Bruce Nuclear Complex in Tiverton, Ontario, is used for the storage of ILW. The storage units are either in-ground concrete-tiled trenches or above-ground concrete quadricell units. The Canadian government is currently dealing with the political and financial issues of implementing the IRUS (Intrusion Resistant Underground Structures) system. The system stores LLW and ILW in resistant concrete vaults for 500 years, when the radioactivity in the waste has been reduced to acceptable levels.
Site investigations are underway in China to locate and build 10 waste storage and disposal facilities in the Sichuan, Gansu and Zhejiang Provinces. Other storage locations for LLW and ILW will be identified on an as needed basis. The current concept, includes disposal in near surface facilities and rock cavities.
In accordance with the Czech Republic Atomic Energy Commission Waste Management Act, all radioactive waste from nuclear power plants must be converted to a solid form prior to disposal at one of the two regional near-surface disposal sites, which are located at Dukovany and Mochovce. Short-lived wastes that are low-active and low-leachable are disposed of in landfills.
In Germany, notification of intent to develop the Konrad iron mine as a site for a LLW/ILW repository has been filed; site studies are continuing, however, construction of underground facilities as part of the repository has not started. The current schedule calls for a full license review to be completed in early 1995, and operations of the repository are expected to be initiated within 3 years after the approval of this action. The capacity is expected to be about 650,000 cubic meters. LLW/ILW was disposed of in the Asse salt mine from 1967 until 1976, and more than 42,000 cubic meters of LLW was placed in the mine. In addition, the Mosleben disposal facility, located in Eastern Germany, initiated operations in 1993 after a 2 year shut down to evaluate safety of the facility. It is expected to operate until about the year 2000.
A shallow land disposal site for industrial and institutional LLW has been in

operation in Hungary for about 12 years. A number of locations are being considered for siting a geologic repository including a clay formation in the southwestern part of Hungray, an old abandoned uranium mine, the PAKS Nuclear Power Plant site, and others. Research programmes will be conducted at these locations to determine site suitability. Decisions covering site selection are considered long term.
India's strategy is to dispose of short lived LLW/ILW in near-surface disposal facilities, and long-lived ILW (such as alpha-bearing wastes) in deep geological formations. LLW disposal facilities are engineered to meet site specific conditions and are located at BARC and at the nuclear power stations near Tarapur, Rajasthan and Madras, and the reprocessing plant at Kalpakkam.
Japan Nuclear Fuel Limited (JNFL) was authorized in late 1992 to begin operation of its disposal facility for low level waste from the country's nuclear power plants. The disposal facility is located at Rokkasho in Aomori Prefecture. Construction of a repository for 50,000 drums (200 liters each) was completed in November 1992. The facility is to be enlarged gradually to hold up to one million drums of waste - with an ultimate target of three million drums.
Korea's strategy to dispose of LLW includes evaluation of shallow-land disposal, engineering-trench burial and disposal in a mined-out cavity in granite. From the analysis performed, the underground "engineered cave" concept was selected, and engineered barriers will be used to minimize waste migration. Of the 25 sites that were considered, three were selected for further evaluation. As a result of public reaction to considering the disposal site on the uninhabited island of Anmyon, continued search for a disposal site has been delayed. The LLW/ILW disposal facility will have an initial capacity of 500,000 drums of waste, with a final placed volume estimated to be at least one million units.
The storage facility for LLW in the Netherlands, which was built near the village of Borssele, initiated operations in 1992. This is also the location of the 450 MWe PWR-1 reactor. The municipality was one of two in the country that agreed to store nuclear waste in its territory. LLW/ILW will be stored for at least 50 years prior to disposal.
The Swedish Final Repository (SFR) for LLW and ILW at Forsmark began operating in 1988 and is considered by some to be one of the industry's most successful operating facilities. The solid LLW from power plants is segregated, and the combustible materials are incinerated in an excess air-shaft-type furnace. The ash is then cemented in 200-liter concrete shielded drums. Arisings at Oskarshamm and Ringhals Power Plants are solidified using cement in concrete boxes with an internal volume of 900 liters. These wastes are then transported to Forsmark for disposal. The current capacity of Forsmark is 90,000 cubic meters; however, space can be made available to dispose of all reactor wastes from D&D, which is estimated to add about 100,000 cubic meters of LLW.
In 1988, the National Cooperative for the Disposal of Radioactive Waste (NAGRA) in Switzerland, received approval for Project Gewahr from the Government. The approval authorized NAGRA to select and construct a facility for LLW/ILW. The current design concept envisions a disposal facility inside a mountain with horizontal access. The disposal area will be several hundred meters below the surface of the mountain, its design capacity is expected to be about 100,000 cubic meters of waste, which corresponds to 40 years of operation. Successive and independent multiple barriers will be used for long term protection and isolation of the radioactive waste from man's environment.
A LLW/ILW disposal site has been in operation for over 30 years at Radon, which is near Zagorsk, Russia, about 30 kilometers northeast of Moscow. About 5000 cubic meters of waste is received at Radon for disposal each year.
Since 1959, the U.K. has disposed of more than 700,000 cubic meters of LLW in eight trenches at the Drigg Low-Level Waste Disposal Site near the Sellafield Nuclear Complex. Seven of the eight trenches are direct land disposal units; and number eight is an engineered concrete barrier unit, constructed as three separate bunkers, whose overall size is about 220 meters long, 60 meters wide and 5 meters deep. The facility is expected to operate beyond 2020. The Drigg Disposal Site occupies about 107 hectares, of which about 40 hectares are used for disposal of waste.
The radioactive waste management company Nirex is considering Sellafield in Cumbria as a site for a LLW/ILW geologic repository. Test drilling at Sellafield has been under way for several years. Based upon data collected and current studies being performed, indications are that the site is suitable for further investigations and

characterization.
The U.S. Low-Level Waste Policy Act (LLWPA) of 1980, as amended in 1985, mandated that each state, either individually or in compact agreements with other state(s), must have a plan for a LLW disposal facility by January 1993 and in operation by January 1996. In June 1992, the U.S. Supreme Court ruled that the LLPA of 1980 was not valid or binding; therefore, LLW disposal site investigations were not necessary, and individual states were free to develop their own programmes and determine methods for disposing of the waste. There are two commercial LLW disposal sites in operation in the U.S.: they are located at Hanford, Washington and Barnwell, South Carolina. A third site located at Beatty, Nevada receives only hazardous wastes, which began this mode of operation in January 1993. However, due to the political and institutional concerns raised within each of these states, waste receipts from other states may be restricted or stopped at any of these sites.
DISPOSAL STRATEGIES FOR HLW AND SPENT FUEL
Radioactive waste disposal systems are designed to isolate the waste from humans and the environment for the necessary time frames to ensure that no potential future releases of radioactive substances would constitute an unacceptable risk. Such systems have been built at or near the surface for low-level and short-lived wastes, and are widely envisaged to be built deep underground in geological formations for high-level and long-lived wastes. Disposal of HLW in deep geologic formation has received strong support from the scientific and technical experts for more than 35 years. In 1957, the U.S. National Academy of Sciences issued a report that stated that HLW should be disposed of in a deep salt formation. The position, that HLW can be safety disposed of in a deep geology, has been supported by many organizations. In 1991, the NEA Radioactive Waste Management Committee and the IAEA International Radioactive Waste Management Advisory Committee issued the report "An International Collective Opinion on the Disposal of Radioactive Waste: Can Long-Term Safety be Evaluated?" which states that the methods, techniques and analytical tools are in place to determine the long-term safety for disposing of HLW in a geologic repository, and which also notes that this action should be pursued.
Research and Development Activities
In conjunction with the Commission of European Communities (CEC), the Hades underground research laboratory (URL) near Mol, Belgium, was constructed in clay in 1984. Since 1984, the Nuclear Research Center (CEN/SCK) has conducted various tests and investigations to develop a technology for disposal of ILW, HLW and TRU. One of the main objectives of the research center is to evaluate the geology for long-term disposal of HLW in clay. Galleries (drifts) have been excavated at a depth of 225 meters. Various tests are under way (some have been completed), including thermomechanical heating, demonstration of excavation techniques and waste-handling methods.
Investigations are under way on the island of ASPO, with the drilling and mining of a shaft for access to the Swedish URL, to evaluate the granite for a possible geologic HLW repository. The URL is expected to be operational in 1995.
The Waste Isolation Pilot Plant (WIPP) in Carlsbad, New Mexico, suffered a series of setbacks in 1988 - awaiting the official transfer of land from the Bureau of Land Management to the DOE, certification of the TRU-PACT canister, and a hazardous waste permit from the Environmental Protection Agency, thus delaying its experimental schedule for a number of years. The original plan called for waste acceptance in October 1988. It is expected that delivery of waste to the WIPP will not occur until at least 1995-96.
Status of HLW Disposal Programs
To date, most countries with nuclear power are developing and implementing programmes that will safely and effectively manage HLW and/or spent fuel.
China's national programme for HLW disposal has four discrete phases: 1) technical preparations for site screening, 2) geologic research and site characterization, 3) in-situ testing and URL experiments, and 4) repository construction. China is currently in the first phase.
In accordance with a HLW management bill passed by the French Senate and the National Assembly in 1991, ANDRA (Agence Nationale pour la Gestion des Dechets Radioactifs) is continuing detailed geographical investigations at four candidate sites (granite, clay, salt and schist). Upon completion of the site surface work, one or more sites will be selected for construction of an underground research laboratory. Studies and investigations will aim at characterizing the local geology,

radionuclide migration and developing waste emplacement and backfilling techniques, to determine if the site(s) is suitable. Construction of a repository would begin in 1995 for ILW, with scheduled operation around 2000. Emplacement of HLW would start around 2010 if studies confirm the site's suitability.
Out of 100 sites screened, Finland has narrowed the number to three for conduct of site studies to determine site suitability for disposal of HLW and/or spent fuel. A programme to conduct site investigations and confirmation studies until 1997 is planned, and should lead to the identification of a suitable geologic media by the year 2000. A design specific to the selected geology and completion of confirmation studies is expected by the year 2020. Operation of the repository is targeted for the year 2050.
Germany's construction of the Ahaus interim storage facility for HLW was completed in 1991. In June 1992, the facility received its first shipment of fuel from the shutdown THTR-300 high temperature gas reactor. Construction of shaft one at the Gorleben site resumed in March 1989. A final evaluation to examine the legal, technical and organizational impacts of the May 1987 mining accident was completed. Both shafts have been drilled and lined to a depth of 450 meters, which penetrates the salt dome about 100 meters. As currently envisaged, the Gorleben repository is scheduled to begin operations in the year 2008.
For HLW, a geological survey has been conducted at Horonobe in Hokkaido, on the northern tip of Japan. However, due to many siting problems with locating a HLW repository, interim storage facilities (dry storage cask units) may be used. Japan is continuing investigations as to the feasibility of sub-seabed disposal for radioactive waste.
Geological disposal of radioactive waste in the Netherlands may be in salt domes. The Netherlands and the Energy Research Foundation (ECN) are conducting preliminary research evaluating potential suitable sites and repository concepts; however, there are no firm plans at the present to dispose of the radioactive wastes by geologic disposal. A national storage facility at Borssele became operational in 1992. The facility has a capacity of 110,000 cubic meters for LLW and ILW, and 24,000 cubic meters for vitrified HLW and 5000 tonnes of spent fuel.
Spain's current HLW strategy includes: increasing spent fuel storage capacity until a disposal facility is available; selecting a site for disposal in a geologic formation (clay, salt or granite); and, developing an URL pilot facility (IPES) in granite in collaboration with the CEC.
In Sweden some of the spent fuel will be reprocessed, resulting in a small amount of HLW to be stored-disposed of as vitrified glass. The remaining spent fuel will be disposed of in multiple encapsulated copper-clad waste packages. Spent fuel is currently being stored at the CLAB facility. Investigations are under way at ASPO, which is about 20 km north of Oskarshamn, with the drilling and mining of a shaft for access to the URL to evaluate the granite for a possible geologic HLW repository. The URL is expected to be operational in 1995, with about 15 years of testing to follow before a final decision is made as to suitability of the site for a repository.
In Switzerland, NAGRA is now proceeding with a site selection programme, taking into account four candidate sites for a HLW repository. A site in crystalline rock in northern Switzerland is being considered in addition to sites in the central part of the country, and investigations in the Jura Mountains area are also being conducted.
In Russia, spent fuel is to be returned by foreign customers; however, it is to be stored by the customer for at least 5-10 years prior to its return. Due to experience obtained in reprocessing WWER440 fuel, a large-scale plant for WWER-1000 fuel is being built. Industrial-scale reprocessing plants with annual capacities of about 1500 tonnes uranium will be built on an as-needed basis. The resulting HLW will then be immobilized in a glass matrix and stored until a permanent disposal facility is available.
In the United Kingdom, at the present time, there are no plans for HLW disposal in a deep geological repository. Vitrified HLW will be stored for at least 50 years.
The Nuclear Waste Policy Amendments Act (NWPAA) of 1987 designated Yucca Mountain in Nevada as the sole candidate site for detailed characterization as the United States first HLW and spent fuel repository. The NWPAA directed DOE to prepare a site characterization plan (SCP) to determine suitability of the Yucca Mountain Site. DOE published the voluminous SCP for Nuclear Regulatory Commission (NRC) review in December 1988. The SCP describes all the site work completed to date, but more

importantly, all of the geotechnical and engineering test programmes required to determine the site's suitability. The SCP also includes a repository conceptual design, waste package performance criteria, exploratory shaft test programme, levels of quality assurance to be applied for the test programme, repository engineering and performance needs, and plans for engineering, construction, operation and closure of the repository. Exploratory Studies Facility Startor Tunnel Construction was started in 1993. This activity continues underground site suitability studies in the proposed repository block area. The tunnel boring machine was delivered in April 1994 for on-site erection and follow-on operations, which is part of the site characterization activities.

SUMMARY

The disposal of low and intermediate level waste has been practiced in a number of countries for about 40 years. At present, countries are continuing to rely on a mix of near surface and subsurface disposal facilities for disposal of LLW/ILW, but increasing their reliance on the use of engineered barriers to isolate the radioactive materials. Some countries such as Finland and Sweden have constructed disposal facilities at greater depths, and a number of other Member States are leaning towards deep geologic disposal for all radioactive wastes. Even though there are no deep geologic disposal facilities on-line for disposal of HLW, most Member States with nuclear power programmes are evaluating potential repository sites for HLW. In continuing the development of radioactive waste disposal facilities, the IAEA, in response to requests by Member States, is developing standards, guides, practices, etc., which should lead to the harmonization of approaches to radioactive waste management at the international level. In addition, safe and effective waste management practices can be implemented by means of open coordinated national programmes, and exchange of ideas and experiences with the technical community, politicians, media and the public, leading to closer collaboration and cooperation with these groups at all levels.

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19-4

RETHINKING THE STRATEGY FOR MANAGING LOW-LEVEL RADIOACTIVE WASTE

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ABSTRACT

While it remains the Department of Energy policy to transition as promptly as possible from waste acceptance to disposal, a variety of circumstances in the last decade have resulted in extensive delays in establishing new disposal and treatment capabilities for managing significant portions of the Department's radioactive

waste. As a result, waste management programs are experiencing extended and unexpected storage requirements. Moreover, this situation has delayed the development of disposal waste acceptance criteria and specifications. The combination of contending with expanded long-term storage and the uncertainty about the waste form and packaging to use for such storage in order to avoid further processing and repackaging for disposal has resulted in the need for consideration of more robust components for the waste management system, e.g., waste packages that exhibit enhanced waste containment characteristics and greater flexibility to perform well in all reasonable storage and disposal environments.

This experience is not unique to the Federal waste management program; similar delays and uncertainties exist with efforts in the private sector to establish new waste management capabilities. These circumstances have inspired several efforts to demonstrate and advocate technologies that would provide more conservative and robust systems for managing low-level and low-level mixed radioactive waste. However, there has been little overall system life cycle cost-benefit analysis of deploying the more robust pre-disposal components of the system pending the availability of a disposal capability. To adopt a more conservative and robust strategy introduces certain policy issues, such as the merits of Federal and private sector sharing of selected waste treatment capabilities as a means of reducing capital costs for new or improved facilities and the possible use of lower risk, long-term storage to facilitate a more conservative and structured approach to establishing a disposal capability with approvals by all appropriate stakeholders.

INTRODUCTION

The Department of Energy's Office of Waste Management has a program vision statement that reads:

The Waste Management Program permanently protects people and the environment from the hazards of DOE wastes by providing an effective and efficient system that minimizes, treats, stores, and disposes of DOE waste as soon as possible.

For purposes of the arguments presented herein the operative words of the vision statement are "as soon as possible." Disposal is defined as the least health and safety risk, least cost isolation of the waste from the human environment that can be realized with the resources, technologies, regulations and schedules provided for the waste management programs. An obvious advantage for disposing any hazardous waste as promptly as possible is the reduced risk for human exposures and environmental impacts by isolating the materials from the biosphere. Recognition of this fact is reflected in the regulatory principle of discouraging long-term storage of radioactive waste in favor of early disposal.

Transitioning from waste acceptance to waste disposal is not occurring as expeditiously as expected for large segments of the Department of Energy's radioactive waste inventory. Such delays, which are occurring for a variety of reasons, are having an impact on the strategy for managing all phases of the waste management system. This situation is not unique to the Department's program for managing its waste; the "commercial" low-level waste management programs are experiencing similar difficulties.

The strategy for managing any class of waste with its inherent radiological, physical, and chemical characteristics is predicated on the functional objectives to be achieved by the elements of the management system and the technological, legal, regulatory, and fiscal constraints with which the system must comply. The strategy must also consider the views of legitimate stakeholders as a prerequisite to receiving public acceptance.

For purposes of a common perspective, the key functional elements of a waste management system include:

- Waste acceptancecharacterization, packaging and certification for transfer by the generator;
- Storagean interim function pending treatment and/or disposal;
- Transportationshipment usually for treatment, storage or disposal;
- Treatmentfor storage, transport and/or disposal; and
- Disposal.

TRADITIONAL MANAGEMENT STRATEGY

Once a waste stream is characterized and classified, the traditional approach to defining and implementing a strategy for managing the waste is to identify the method for disposal of the waste, define the criteria and specifications for acceptance of the waste for disposal (e.g., the waste form and packaging

requirements), and to use these considerations as the principal "drivers" for determining critical requirements applicable to each pre-disposal phase in the management system, i.e., acceptance, treatment, storage, and transportation.
This approach to managing the waste has been generally acceptable since it was felt one could identify preferred generic methods of disposal for the principal classes of waste. For example, the bulk of the waste characterized and classified as low-level radioactive waste qualifies for disposal in near-surface land disposal systems that have been demonstrated in practice with both commercial and defense-generated waste. Long-lived and/or highly radioactive high-level and transuranic waste generally require isolation in deep geologic repositories. Since the regulatory performance requirements have been established for these preferred generic disposal systems, and since it was felt one could identify technically acceptable sites for these disposal systems, then one could with a reasonable degree of confidence plan and project schedules for the availability of the necessary disposal capabilities. Armed with the source term inventory for disposal and the characteristics of the host site for the disposal facility, an assessment of the long-term performance of the disposal system could establish the criteria and specifications for acceptance of the waste for disposal. With this information, one could develop the design and performance requirements for the remainder of the waste management system, i.e., the pre-disposal phases.

EXPERIENCE IN IMPLEMENTING THIS STRATEGY

The employment of this strategy for the management of both "commercial" and DOE low-level waste enjoyed reasonable success for over two decades. Disposal capabilities were established on relatively prompt schedules, thereby avoiding prolonged storage of the waste packages. There have been six LLW disposal facilities sited, developed, licensed, and operated by the private sector over the past three decades. Over the same period, DOE operated six major sites with low-level waste disposal capability.
Experience in operating and monitoring the performance of these disposal facilities added greatly to knowledge about the use of enhanced waste forms and packages as well as the potential advantages of engineered barriers to migration of hazardous components. There have also been improvements in modeling the performance of LLW disposal systems and in the codes used for quantitative assessments of the disposed waste in interacting over the long-term with the man-made and natural occurring environment in which the waste is emplaced.

RECENT DIFFICULTIES IN IMPLEMENTING STRATEGY

It would appear, therefore, that with the emergence in the early 1980's of the need for establishing a new generation of LLW disposal facilities in the U.S., the responsible parties could build on this extensive experience base in a reasonably expeditious and straightforward manner in establishing the necessary new LLW disposal capacity. Unfortunately, this has not been the experience of the past decade. Even though 11 States are currently identified to host new "commercial" LLW disposal facilities pursuant to the terms specified in the Low-Level Radioactive Waste Policy Act in 1980, there has not been a new disposal facility brought into operation to date. In fact, in spite of extensive efforts and expenditures of resources, construction has not been initiated on a single new disposal facility to serve non-DOE LLW generators, nor has final approval been obtained for a site to host one of these facilities. Over the same period, there have been only 3 new DOE LLW disposal facilities brought into operation and two of those were technology demonstration projects of limited duration (i.e., the Savannah River Site concrete vaults, the tumulus facility at the Oak Ridge Reservation, and augured boreholes at the Nevada Test Site).
In an effort to promote progress in the establishment of new "commercial" LLW disposal capacity by the States, Congress amended the Low-Level Radioactive Waste Policy Act in 1985 to provide incentives for the States to comply with Federal policy. One of the more compelling incentives was authorization for the three operating disposal facilities in the States of Washington, Nevada, and South Carolina to deny access to their facilities after 1992 by generators outside their specific compact regions as ratified by the Congress. As a result of this provision and the continued delays in opening new disposal facilities, only generators in the 8 States of the Southeast regional compact and in the 11 States in the Northwest and Rocky Mountain compacts served by the disposal facilities in Barnwell, South Carolina, and Richland, Washington, respectively, have access today to disposal

facilities. Approximately 47% of the LLW being generated currently by NRC and Agreement State licensees is being stored, pending the availability of a licensed disposal capability.

As noted above, DOE also has had only modest success in recent years in providing improved or new LLW disposal capacity. There are some common factors that have limited progress in both the "private" and DOE programs. They both have experienced difficulties and associated delays in siting new facilities. In most cases, there have been no unusual problems associated with the process of selecting a preferred site by the responsible organization in its application of such technical criteria as hydrogeologic conditions, avoidance of wetlands, and access to appropriate transport routes.

The difficulties and delays usually occur after selection of a preferred site. Three reasons appear to be the cause of such delays. First, sufficient technical information is not obtained by site developers. Second, technical information, including risks and benefits, are not adequately conveyed to State officials and regulators. Third, local publics will not accept a disposal facility regardless of the adequacy of the technical and regulatory validity of the information. This could be due to a general lack of trust and confidence by the public and the lay decision-makers in the developers and promoters of radioactive waste disposal facilities.

Changing regulations also play a role in opening new disposal facilities. Recently two regulatory uncertainties have emerged that could have significant implications for siting and development of design requirements for new LLW disposal facilities. The NRC is planning to formally propose a revised Branch Technical Position for conducting performance assessments of LLW disposal systems. It will employ probabilistic risk assessment methodology with increased emphasis on uncertainty analyses. The net result will reflect a more rigorous and conservative approach to assessing the performance of the LLW disposal system which is also likely to result in more stringent acceptance criteria and specifications for the disposal facility. Concern has also been expressed by some private sector facility developers that application of the NRC-proposed methodology will make risk communication more difficult to convey and thereby jeopardize the schedule of the licensing process.

The second new development emerging in the regulatory field with potential adverse implications for existing and planned LLW disposal facilities is EPA's draft LLW standards (40 CFR 193) planned for publication in April of this year. These standards include a provision for protection of underground sources of drinking water which is keyed to the maximum allowable contamination limits for drinking water. Options under consideration for this standard may significantly limit operations at some existing disposal facilities and limit siting options for new facilities.

A direct consequence of these delays in realizing adequate disposal capability for low-level waste is either to invoke extended storage of waste or to curtail operations that generate the waste. This assumes that access to an alternate disposal capability is not available, which is the prevailing situation for many private sector waste generators in the country.

Off-site disposal options may or may not be available for DOE generators with waste that is not acceptable to current on-site disposal facilities. If DOE's degrees of freedom for transferring waste among its sites are significantly limited, the Department may have to rely on a strategy of complete waste management self sufficiency for LLW at each major site. Given the wide range of characteristics associated with the low-level class of waste and the quite different site-specific characteristics of the Department's arid and humid sites, cradle-to-grave management of LLW at the site of generation would impose significant economic penalties on DOE environmental restoration and waste management programs. It would also result in extended periods of storage for portions of the LLW inventory at most of the large DOE sites pending the availability of necessary on-site treatment and/or disposal capabilities.

In any case, both licensed and DOE LLW generators that do not currently have access to a disposal facility are confronted with uncertainties about how long this situation will persist during which storage of the waste will be necessary. This leads to further uncertainties and questions about the adequacy of existing storage capacity, what waste forms and containers should be employed for storage in view of the uncertainties about the term of storage, and what will be the resource

requirements for these additional functions. If the wastes are generated as a result of services rendered, one must also consider how best to pass these additional costs to the beneficiaries of the services. Concern has been expressed that the additional costs resulting from the inaccessibility of a waste disposal capability could force the discontinuance of some services using radioactive materials, such as industrial and medical research, diagnosis and therapy, which are deemed important to society. Those organizations confronted with indefinite storage of their waste must ensure that the waste form and container provide the necessary containment integrity over the term of storage. Moreover, they would like the storage package to qualify for acceptance by the disposal facility without further processing or repackaging, thereby avoiding additional costs as well as additional exposure risks to workers and the public.

Preoccupation with this concern about selecting appropriate waste form and packaging for storage has been apparent in workshops on temporary storage of radioactive waste provided to the States and regional compacts by the Department's National Low-Level Waste Management Program over the past couple of years. Among the first to reflect interest in this subject were generators in the State of Michigan who lost access to a "commercial" LLW disposal facility in 1990. Additional workshops on this subject are scheduled in 1995 at the request of States and their waste generators. It is a concern for both the generators and the regulators.

The most obvious response to these concerns would be to store the waste in stabilized forms and containers that comply with the current acceptance criteria at the operating disposal facilities. This may or may not be adequate for the planned new generation of LLW disposal facilities. These facilities with their new designs may require waste packages of new configurations to be compatible with the waste emplacement cells and modes of closure. More stringent waste form/package performance standards may result from planned changes in the regulatory standards governing release rates of radionuclides from disposal facilities. Moreover, the long-term storage facilities may place certain special demands on the waste packages for handling and maintenance of long-term containment integrity.

These considerations could be particularly relevant for managers of DOE LLW. They would prefer to minimize storage while avoiding expenditures of resources on improving or establishing new waste management capabilities that may not comply over the longer term with resolution of their current set of uncertainties:

The degree of flexibility DOE will have in transferring waste between sites for treatment or disposal;

The extent to which different and possibly more stringent regulatory requirements will govern the management of DOE's LLW as a result of the following current regulatory-type initiatives:

- EPA's proposed LLW management standards;
- Revision of the Department's Order 5820.2A that applies to management of radioactive waste;
- The Department's programmatic environmental impact statement that pertains to management of radioactive and mixed waste; and
- The on-going debate as to whether or not external regulatory authority should be imposed upon DOE's nuclear programs, including waste management.

Possible changes in acceptance criteria and specifications for several of DOE's LLW disposal facilities as a result of performance assessments for these facilities having been conducted recently;

Potential policy changes with implications for the long-term strategy for managing waste that could result from periodic changes in the senior Department managers; Resource availability beyond the current fiscal year; and

The uncertain nature and volume of waste to be generated by the environmental restoration program in cleaning up and decommissioning a multiplicity of sites and facilities contaminated with radioactivity materials.

In this latter connection, the cleanup standards, which are still somewhat uncertain, could have a significant impact on the amount of waste to be generated and, therefore, on the capacity requirements for the elements of the LLW management system.

NEED FOR MODIFIED STRATEGY

In addition to the uncertainties noted above, several other known factors contribute to the need for reexamination of the practice of predicating the LLW management strategy on prompt transition from waste generation to waste disposal.

Whatever the result of the debates over the public policy issues of:

The continued use of nuclear energy for generating electrical power;

The continued generation and use of radioactive materials for various industrial, research, and medical applications; and

The continued use of nuclear materials in the U.S. national defense systems;

there remains the need to safely and effectively manage the very large amounts of radioactive wastes already generated by these programs, as well as the waste from the cleanup and decommissioning of sites and facilities that have possessed radioactive materials. However, the experience of the past decade indicates that under today's circumstances the establishment of new LLW disposal sites involve very long lead times. In addition, recent performance assessments of some DOE LLW disposal facilities have resulted in more stringent waste acceptance criteria. As a consequence, some of the waste streams expected to qualify for disposal on-site no longer have an identified place for disposal. An example are waste streams containing transuranic nuclides ranging in concentrations from 10 to 100 nanocuries per gram of waste that are no longer acceptable for disposal in current facilities at certain DOE sites. These wastes will be stored until a disposal facility with suitable containment is established at the site of generation or they can be transferred to another DOE site with a LLW disposal facility that can accept such waste.

Given these facts, one can only conclude that the strategy of transitioning on a reasonably timely schedule from low-level waste generation through any necessary treatment and storage to disposal is no longer viable. This has been a long-recognized fact in the management of DOE's transuranic and high-level radioactive waste. Extended and uncertain periods of storage will likely be the rule rather than the exception for the foreseeable future.

In light of this situation, are there initiatives that can and should be taken by DOE waste managers that provide an enhanced level of confidence in:

a) The integrity of the system to contain the radioactive waste during the lengthy pre-disposal phases of management; and

b) The stored waste package qualifying without further processing for acceptance at the disposal facility when one becomes available.

Such initiatives to modify the strategy for managing the waste must pass the test of providing net overall benefits in terms of public health and safety and life cycle costs.

The most obvious initiative to ensure integrity of a waste storage system for containing the radioactive constituents over whatever time period is necessary while awaiting the availability of a disposal capability is to provide very robust components for each pre-disposal phase of the waste management process. The components in question are primarily the waste form and the waste container. The term robust is intended to mean the application of conservative requirements for the waste form and for the waste container thereby exhibiting a high degree of containment in any likely storage and disposal environment. The bounding case would be design requirements for containment over the 500-year lifetime of the waste package as contemplated by 10 CFR 61.

An additional characteristic of the robust system would be a high degree of flexibility exhibited by the system components to be compatible with and to perform well in a broad spectrum of storage and disposal environments. Just how robust the waste form and how high the integrity of the container may have to be determined at each DOE site in terms of the circumstances that prevail in the candidate options for managing its LLW. For example, the characteristics of the waste streams and engineering judgements about the disposal acceptance criteria and specifications for a specific DOE site (humid or arid) should provide a basis for selecting conservative performance and design requirements to provide a sufficiently robust waste form and container.

To varying degrees a recognition for the need of a more conservative approach is beginning to occur at the major DOE sites as they are confronted with new challenges, constraints, and delays in managing their LLW. This trend should be examined in terms of the need for a strategic principle(s) that would be applied across the entire DOE complex with respect to harmonizing and optimizing the LLW forms and containers for all components in any candidate waste management system. The intent would be to recognize and take into account any uncertainties associated with the disposal and treatment technologies, the schedules on which they can be

deployed, and the associated spectrum of fiscal, regulatory, and institutional constraints when deciding the waste form and container requirements of the system. This process is essentially the general application of system engineering principles.

Several initiatives have been taken in recent years by the Department's Office of Waste Management that will contribute to the more disciplined and fully integrated approach embodied in the application of system's engineering methodology to the management of waste across the DOE complex.

A DOE-Headquarters coordinating manager has been designated for each class of waste, with each chairing a Steering Committee having representation from DOE Operations Offices. Together they maintain a current snapshot of the state of the Department's waste management systems, compile projected waste volumes and the capabilities required to manage these volumes in a manner consistent with the goals to be achieved with time and in compliance with the multitude of regulatory, institutional, and fiscal requirements and constraints on the system. The major system issues requiring resolution should be identified in these studies. A target product of these efforts is the development of a long-term strategic plan for management and disposal of each class of waste.

The DOE regulations covering the management of the Department's various classes of waste products are set forth in DOE Order 5820.2A, "Radioactive Waste Management." This Order is currently being revised to reflect the experience and changes since it was initially promulgated in 1988.

A separate but related study has been recently initiated by the Department's Office of Waste Management pursuant to a recommendation by the Defense Nuclear Facilities Safety Board. It constitutes a survey of all facets of the management of low-level waste across the DOE complex, including the identification of issues and options for their resolution. Elements of the Department's practices will be compared to those of the private sector.

TRENDS IN COMMERCIAL PROGRAMS

A similar situation exists in the management of low-level waste in the private sector. Given the delays in access to a disposal facility and uncertainties about what waste forms and containers will be acceptable when a disposal capability is available, each generator is attempting to ensure access to adequate interim storage under affordable terms and conditions. If generators are providing an in-house storage capability, they are also facing the question of what waste form and container to use and the availability of any necessary treatment capability to produce the preferred waste form.

Some private entrepreneurs have apparently concluded that licensed generators, facing these uncertainties, will opt for more robust waste forms than generally employed in the past. According to recent reports in the trade journals, several projects have been initiated that are designed to provide
more conservative and robust waste forms. These efforts have included the initiatives listed in Table I.

TRENDS IN DOE PROGRAMS

We have seen a similar trend toward conservative waste forms emerging in the management of DOE LLW. As various DOE sites have been confronted with increasing external influences on aspects of managing their LLW, such as tripartite agreements among DOE, the host State, and EPA, they have encountered significant uncertainties about the schedules and technologies for treating and disposing of their LLW. As a result of these uncertainties, there have been case-by-case conservative decisions or considerations given to the adoption of more robust waste forms for certain waste streams. For example, a vitrified waste form has been a prime candidate for stabilizing the low-activity waste stream that will result from processing the HLW at one of the Department's production sites.

Other developments or issues currently under consideration that are conducive to adoption of a DOE strategy to deploy a more robust LLW management system include:

Savannah River Site is eliminating use of engineered earthen trenches for disposal of low-level waste in favor of using concrete vaults. Specially formulated concrete is used to resist cracking and erosion. A waste/grout slurry will be introduced into a vault for in-place solidification (1).

Low-level waste generated in processing Hanford high-level waste is expected to be vitrified rather than stabilized with grout for disposal (2).

- DOE has contracted to demonstrate for vitrification of liquid low-level waste at

Hanford.

Vitrification of residual waste from operation of the proposed Idaho Waste Processing Facility is planned when treating mixed alpha-low level waste (3).

Use of sulfur polymer cement has been proposed for solidifying and stabilizing radioactive and mixed low-level waste, particularly those with high concentrations of toxic metals and salts resulting from high-temperature treatment processes (4).

Pilot-scale tests have been conducted for the use of an existing electric arc melter to evaluate vitrification processes for low-level and transuranic mixed waste similar to those occurring at the INEL (5).

Oak Ridge National Laboratory plans to demonstrate in 1995 a ceramic waste form for nitrate salts (2).

Proof of concept tests were completed in 1993 at INEL for a fixed hearth plasma thermal treatment unit, thereby demonstrating the ability of this process to turn compacted waste drums and buried waste into a vitrified product that complies with Federal and State laws for disposal (2).

Based on past experience and current considerations, one can identify advantages of extended duration and uncertain disposal criteria:

Provides an additional margin of safety and confidence in the containment integrity of the waste package if the term of storage is extended;

Results in a storage waste form acceptable for disposal over a broader range of acceptance criteria, thereby minimizing adverse schedule and cost impacts due to possible changes in regulatory performance requirements and more stringent waste acceptance criteria;

Likely reduced total life-cycle costs, reduced worker exposures by avoiding further processing and packaging operations, and increased safety margins for potential public exposures;

Provide enhanced confidence in and acceptability of proposed waste management system by the public and host State officials; and

Reduction in potential adverse implications from changes in the planned configuration of TSD facilities across the DOE complex resulting from host State equity decisions, PEIS results, or changes in DOE policies.

On the other hand, there may be disadvantages associated with the more robust system to be considered:

There would be increased near-term capital outlays for the more expensive treatment, storage, and/or disposal components of the system; and

There could be higher unit costs than originally projected for management of the waste.

Both of these potential disadvantages may be more than offset by use of the more robust system, thereby avoiding additional costs and personnel exposures associated with a future need to further treat and repackage the stored waste to comply with more stringent requirements.

If the Department adopts a strategy of reliance on more conservative and robust systems for management of its low-level radioactive and mixed waste, it would also be useful to consider other policy issues including:

1. Should there be some degree of accessibility for low-level waste generators to treatment and perhaps even to disposal facilities between the Federal and the private sector waste management systems? Such an approach could be conducive to harmonizing waste management practices and associated risks on a national scale and could be cost effective for both sectors in terms of minimizing the number of capital projects, optimizing land use, and economies of scale.

2. Should DOE rely on "privatization" as a means of establishing selected low-level waste treatment and disposal capabilities, thereby taking advantage of the experience of private industry with the more conservative and advanced engineered disposal technologies, use of high-integrity containers, and stabilized waste forms customized to specific waste streams, which are required by some State waste disposal requirements.

3. Should external regulatory authority be applied to the Department's waste management operations? An affirmative resolution of the two aforementioned policy issues would certainly be conducive to a similar resolution of this issue.

4. Should the Department's regulations include prescriptive performance requirements for low-level waste forms and container integrity in terms of radionuclide confinement (release rates)? Given the progress in RD&D for the long-term performance of concrete structures for low-level waste disposal applications (1) and

the advances in high-integrity waste forms and containers, should consideration be given to adopting a system performance objective of zero release for the first 300 years. Such a requirement, if realistically achievable, would minimize concerns about siting disposal facilities on DOE reservations, even if the more stringent low-level waste standards proposed by EPA in 40CFR193 become a reality. It could also result in portions of the GTCC LLW qualifying for disposal in the near-surface low-level waste disposal facility.

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HUNGARIAN STRATEGY FOR MANAGEMENT OF RADIOACTIVE WASTE OF NUCLEAR POWER PLANT ORIGIN

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ABSTRACT

In Hungary, no long term strategy has been worked out yet for the waste coming from the nuclear power plant operation and decommissioning - a strategy that covers all important questions related to the problem.
The original Soviet concept for low- and medium level radwaste storage - i.e. the storage of the radwaste in the plant site -till the time of decommissioning was refused by Hungarian authorities as early as in 1977. The proposed new concept - to dispose of the cemented waste into trench - type facility at a distance of less than 100 km from the NPP - remained only an idea after the 1990 January ministerial refusal concerning the planned site and repository.
The recognition that to solve the safe, socially acceptable final disposal of the radwaste generated in the NPP - which provides almost one half of the Hungarian electrical energy production -needs a complex (scientific, economical/technical social, legal and financial) activity went into operational program at the beginning of 1992. The Hungarian Atomic Energy Commission in his resolution related to this topics deemed it important to launch an inter-departmental Task Project aiming at finding solution for the final disposal of the NPP radwaste. Within this Project - called National Project - a comprehensive strategy had to be established covering all the low-and medium level waste produced during the whole life of the plant and the back-end of the fuel cycle, handling and disposal of the waste coming from the plant decommissioning, and implies technical, licensing, regulatory activities, the financial solutions and public relation issues.

INTRODUCTION

The concept laid down in the late 1960s for the management of wastes at WWERs was to store the arising wastes on site and to postpone the decisions on conditioning and disposal until the decommissioning stage so that wastes from operation and dismantling can be handled together. Waste collection and storage systems were developed to accommodate ten years' treated operational wastes with possible extension of storage capacities. The only exception were very low level solid wastes where on-site disposal was proposed.
A truly remarkable feature of the concept was the return of the spent fuel to the USSR where it would be reprocessed and the arising wastes would be disposed of. No reprocessing wastes were to be sent back to the operators of the reactors. However, after the political and economic changes in Russia, this practice may not continue and probably new arrangements will have to be made.

Due to differences in design and operating philosophy from those practiced in Western Europe, the waste management systems at power plants with WWER reactors have to deal with relatively large volumes of waste. This resulted in the accumulation of large amounts of radioactive wastes stored at the sites of the NPPs and has increased the risks of radiological incidents and contamination of the environment. Now, following the political changes in Eastern Europe, there is an intensive re-evaluation of the safety of WWER reactors under way. This program also deals with the waste management practices at these power plants. During the IAEA's Technical Assistance Regional Project on Advice on Waste Management at WWER type reactors, the lack of effective waste management strategies has been identified as the most significant common problem apart from the need for further technical improvements. Nuclear activities in Eastern European countries are carried out on the basis of nuclear energy acts which only define general principles. Application of these principles would require a more detailed system of regulation which is not available yet. In most cases only the period of the NPP's operation is covered and the problems connected with decommissioning and final disposal of all types of wastes remain to be resolved. Furthermore, in the centrally planned economies the state was responsible for all decisions concerning the financing of these activities and now there is an urgent need to reallocate the responsibilities between the state and the operators according to the needs of the new economic system. As the IAEA experts put it, the safe and reliable management of radioactive wastes is only possible within the framework of a clearly defined national waste management strategy which covers all of the long term aspects of waste management.

The main requirements for such a strategy should include:

- a legal framework;
- a clear structure and division of responsibilities between operators and national authorities at all levels;
- national regulations for the releases of radionuclides into the environment and exemption limits;
- key decisions on the type of disposal for low and intermediate level wastes arising from operation and decommissioning;
- evaluation of solutions for the back end of the fuel cycle (duration of and technologies for intermediate storage, preparation for geological disposal of spent fuel or high level reprocessing wastes);
- timing and extent of decommissioning activities;
- long-term liabilities, financial arrangements and insurance for all waste-related activities.

However, the development of the individual countries' waste management systems were also influenced by their own regulatory authorities, the performance of the plant, and previous waste management experience. Many of the power plants have changed the original system arrangements or some of its components.

NATIONAL PROJECT

On 28 February, 1992, the Hungarian Atomic Energy Commission (HAEC) declared in a resolution that the safe and socially acceptable disposal of radioactive waste required the co-ordination of complex scientific, economic, technological, social, legal, financial and international activities.

It was found that the most appropriate means to achieve this was the establishment of an interdepartmental project in which all the organizations concerned would be involved. Basically, this resolution represented a realization that since the reliable operation of the nuclear power plant was a national interest, the company could not be left on its own in its attempt to find a solution to the radioactive waste disposal problem.

To implement this resolution, a proposal was prepared by the representatives of several ministries and Paks NPP.

The starting point taken in the proposal was that the project should be realized as a part of the new tenders program of the National Committee of Technical Development and thus the tasks of the project must essentially be commissioned through the invitation of tenders.

According to the proposal, the Project was to be funded primarily by the Paks NPP Ltd. although substantial contribution would come from central resources as well. The setting up of an appropriate fund or other financial arrangement to cover the costs of the waste disposal was also scheduled to be done in the framework of the Project.

When discussing the proposal, the Hungarian Atomic Energy Commission considered it very important that a complex strategy be elaborated that would cover all the issues related to the management and final disposal of the low and intermediate level radioactive wastes arising during the operation of the plant, the wastes resulting from the closing of the nuclear fuel cycle (spent fuel and perhaps high level reprocessing waste) and finally the wastes coming from the decommissioning of the plant.
It was emphasized that the strategy would have to deal with the different options of the technical realization of the disposal facility (near surface or geological disposal), the economic conditions of the disposal, the licensing and regulating activities of the authorities, and the question of public acceptability.
On 26 February, 1993, the Ministry of Industry and Trade, the Ministry of Public Welfare (the successor of the Ministry of Health and Social Affairs), the Ministry of Environmental Protection, the National Committee of Technical Development, the State Holding Company (which is responsible for the proper management of state owned enterprises as a trustee), the Hungarian Electricity Works Ltd., and the Paks Nuclear Power Plant Ltd. signed a Co-operation Agreement to solve the handling and final disposal of radioactive wastes of nuclear power plant origin.
According to the agreement, the National Project, which is aimed at the establishment of the repository for low and intermediate level radioactive wastes, will be carried out in two phases. The complex strategy will be worked out in the first phase (1993-94). This phase will provide the basis for the decision on the siting of the facility and the invitation of tenders for its realization. The second phase is the actual realization of the facility.
The costs of the project will be covered by the fulfillment of the economic conditions that are to be worked out in Phase 1.
Informing the public about the National Project, the results of its implementation and other issues that might arise during the process was considered as a high priority and a decision was made to carry out these activities in a co-ordinated manner.
The responsible organization for the realization of the Project is the Paks Nuclear Power Plant Ltd. The first phase of the Project is jointly funded by the company and the Central Technical Development Fund.
The following tasks of the Project are listed in the supplement of the agreement:
  elaboration of a complex strategy for the management and final disposal of radioactive waste;
  siting of radioactive waste repositories;
  having the site and the facility accepted by the public;
  selection and implementation of waste handling and volume reduction technologies;
  drawing up the conceptual design of the facility for the disposal of low and intermediate level radioactive waste;
  preparation of the engineering report, licensing and construction of the facility.

COMPLEX STRATEGY

The complex strategy was worked out on the basis on the following principles:
  Radioactive wastes generated should not hamper the operation of the NPP during its whole life.
  At the plant site radioactive wastes shall be stored in the possible lowest amount and volume.
  Pre-disposal conditioning of the radwaste and the final disposal itself has to be solved within the shortest time possible.
  By examining the alternatives of the technical solution, a complex evaluation shall provide a framework for the further tasks that later would be submitted to tenders for design and construction of the facilities.
  By designing the final disposal of the radwaste, not only the waste from the plant in operation, but that from a contingent further one shall be considered.
  For selecting the technical solution, the experiences of the countries that are advance of us have to be used and provide the proper references.
In Hungary - as previously mentioned - no long term strategy has been worked out yet for the waste arising from the nuclear power plant operation and decommissioning - a strategy that covers all important questions related to the problem.
The original Soviet concept for low- and medium level radwaste storage - i.e. the storage of the radwaste in the plant site -till the time of decommissioning was refused by Hungarian authorities as early as in 1977. The proposed new concept - to

dispose of the cemented waste into trench - type facility at a distance of less than 100 km from the NPP - remained only an idea after the 1990 January ministerial refusal concerning the planned site and repository.
Since 1983 the Paks NPP has been sending low activity solid radwaste to Central Radioactive Waste Processing and Storage Facility at Pspkszilgy - a disposal facility built for receiving radioactive waste of non-nuclear fuel cycle - with the consent of its constructor, the Hungarian Atomic Energy Commission and with permission of the health authorities.
Long term concept of the Paks NPP fuel back-end - to ship back the spent fuel to the Soviet Union as made previously - has became rather uncertain because of the radical changes in foreign policy and economical environment in the ex-SU.
The complex strategy was worked out on the basis on the following principles:
Radioactive wastes generated should not hamper the operation of the NPP during its whole life.
At the plant site radioactive wastes shall be stored in the possible lowest amount and volume
Pre-disposal conditioning of the radwaste and the final disposal itself has to be solved within the shortest time possible.
By examining the alternatives of the technical solution, a complex evaluation shall provide a framework for the further tasks that later would be submitted to tenders for design and construction of the facilities.
By designing the final disposal of the radwaste, not only the waste from the plant in operation, but that from a contingent further one shall be considered.
For selecting the technical solution, the experiences of the countries that are advance of Hungary shall be used.
The first version of the complex strategy outlines the work to be performed by considering the knowledge available at the time of compilation and the previously declared and approved ideas, giving the priorities for the participants in the Project and for the experts getting in touch with it (Authorities and decision makers etc.). At the end of the first stage of the Project, on the basis of the decisions and the knowledge acquired in the meantime, the first version of the strategy is to be reviewed and finalized focusing on long term tasks for each decision-making and executive level.
The structure of the complex strategy goes along with the logical sequence radwaste treatment and disposal and assigns the tasks required from the origin through the treatment, packing, transport up to the final disposal according to this sequence.
Basic premises of the approach applied for solving the problems are as follows:
Elaboration of a coherent international recommendation system related to waste management as a whole is currently in progress. Principles and the requirements are laid down, they can be applied in Hungary, too.
Operation and economical electricity production of the Paks NPP is of a national interest. So it is not solely the Paks NPP's interest to solve the waste management problems.
Treatment of the nuclear power plant wastes means a very complex system. Arrangement of some single elements (solving the problem of partial areas) results not necessarily the optimum of the whole system. In decisions, the "system view" and the avoidance of "irreversible" alteration shall dominate.
As a part of the "safety culture" a change in the way of thinking should be reached, where the success were not measured by solving the known and urgent problems in short range but - by seeing the waste management as a process - favors the forming and implementation of such procedures that provide adequate handling for both the known and the - for the time being - unknown problems.
Having the priorities of the partial areas scrutinized, the following statements can be made:
Source - side reduction of the waste produced is a general requirement.
A further general requirement is to avoid the long living radioactive isotopes getting into the waste stream, or at least to minimize this process.
Without the availability of waste disposal facilities, the waste management is unsolved. Disposal for the low - and intermediate level radioactive waste has to be constructed even in the operational period, and it should have the capacity also for housing the decommissioning waste. Therefore design work shall account both with the operational and decommissioning waste.
It should be set as a requirement to carry out developments programs of the Paks

NPP technological system in such a way (the availability of the intermittent storage capacities included) that results a minimal amount but conditioned waste suitable for disposal.
By taking into account the above principles and objectives a hierarchically arranged proposal for the tasks to be performed are as follows:
The following questions should be discussed in a government-level submittal
a) Acceptance of areas proposed for detailed site survey and that of the disposal concept
b) Proposal for establishing a national waste management agency
c) Settling the long term financing and liability problems
d) Ideas for the nuclear power plant decommissioning:
- alternatives of the total decommissioning or further utilization as a nuclear site;
- the time scheduling of decommissioning.
e) Principles of the high activity waste disposal
- Reprocessing or direct disposal
- Intermittent storage, the program scheme of the final disposal with decision making, time schedule and cost estimation
Problems requiring authority regulation or further development of the regulation:
a) Relation between waste classification and solutions for disposal (adaptation of the IAEA proposals is suggested)
b) Determination of the limits of exemptions
c) Determination of the waste acceptance criteria
d) Settling the waste qualification - and requalification procedure
e) The problems of waste disposal to be regulated include the site selection requirements, settling the safety requirements on dose - or risk basis, establishing
the licensing scheme and system of terms up to the end of institutional surveillance,
as well as integration of the aspects of the public needs and concerns.
Questions belonging basically to the Paks NPP authorization area:
a) Source side minimization
b) Expand the storage capacities or improvement of their maneuverability
c) Apply the waste volume reducing technologies
d) Liquid waste conditioning
e) Preparations for decommissioning waste treatment

SUMMARY

In 1993 a National RW Management Project was launched to solve handling and disposal of LLW/ILW of the Paks Nuclear Power Plant and to elaborate a complex strategy for the management of radwaste from the NPP, including HLW, spent fuel and wastes from the decommissioning. It was intended to realize the project so as to have selected the possible site (or sites) for the LLW/ILW waste repository. The nation-wide screening for suitable areas and the comparative evaluation process to select potential sites for a surface or geological LLW/ILW disposal facility is under progress. International tenders were issued to find the most appropriate technology to reduce the quantity of liquid and solid radwaste in the Nuclear Power Plant. Their results will provide a better basis for planning the characteristics and quantity of radwaste. The applications revealed that supercompacting can be ordered as a service, when need arises thus it was possible to re-allocate the funding foreseen for the equipment to treat solid wastes. Great importance is attached in the Project to public acceptance and PR activity. An expert organization was selected in a two-phase bidding process, and it was decided that detailed exploration of a potential site will take place only if public acceptance is assured. The original program of the Hungarian RW Management Project was extended in 1994 to make on-site underground investigations with Canadian support in a silt-stone formation. The first results confirm that this is a potential site for deep geological disposal of HLW. The financial and legal framework of the RW management is also to be solved. The new law on nuclear energy now in preparation will deal with that problem in one of its most important chapters, defining the responsibilities for RW management and the sources of funding.

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19-7

EXPERIENCES, CONCEPTS AND STRATEGY WITH THE MANAGEMENT OF RADIOACTIVE WASTE MATERIALS IN FR YUGOSLAVIA

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ABSTRACT

From the early fifties, when the first research reactor started with operation inside the "Vinca" institute site, parallel started generation of the radioactive waste materials. These materials had been generated in the industry and nuclear medicine, as well, where the radionuclides found their useful application. All these materials were transferred to the covered interim storage on the Institute site, that was built in early seventies, whether they were in liquid or solid form. Compressible solid waste materials were introduced to the 50 tone hydraulic press, for the volume reduction purposes. Capacity of the facility is five thousand drums. Main radionuclide presented in the most of the drums were 14C and 3H. Dose rate in the storage does not exceed 5 mSv/h. This storage facility was fulfilled, and in the early eighties, second interim storage facility was built. Liquid radioactive waste materials, that were generated in the research reactor, were transferred, using specially manufactured pipeline, to the four stainless steel reservoirs, each of 300 m3 capacity. As concepts for immobilization processes of the generated and stored radioactive waste materials, cementation and bitumenization processes are adopted. These two processes are suitable for immobilization of different types of the radioactive waste materials, depending on the radwaste material origin and structure. Pilot bitumenization plant was built in the "Vinca" institute, and the experiments were carried out with three types of the Yugoslav produced bitumens, used as an inactive matrices. Optimization was performed on the basis of the lowest level of the radionuclides leaching from the matrix-radwaste form, as well as on the other physico-chemical and mechanical characteristics. Cementation facility is under construction. In the technological procedure of the radwaste materials chemical pretreatment and treatment before the cementation, several steps were taken into account, such as: concentration, chemical precipitation, phase separation with or without evaporation. As a concept for the final disposal system of such prepared solidified radwaste-mixture forms, engineer trench system is adopted. This system consists of four barriers to the influences that might occur to and from the trench environment, and they are: immobilized matrix-radwaste mixture form, solidified inside 200 l metal drums or in concrete made containers, depending on the applied immobilization technique, drainage materials and trench walls, made of the concrete of the high mechanical strength characteristics. Engineer trench system will be divided into section, each representing independent trench system, with the control possibility of the system potential damages, expressed as radionuclide presence in the drainage network system, that goes under the trenches and collects liquids that might penetrated through facility. All R&D work that is performing in the "Vinca" institute, dealing with the treatment and management of the radioactive waste materials has its final goal in constructing the Yugoslav radwaste materials final disposal centre.

INTRODUCTION

Since early fifties, when the "Vinca" nuclear research reactor of 5 MW power started with operation, as a product in the nuclear fuel cycle, radioactive waste materials of different levels of specific activities, were generated. Simultaneously with research rector operation, laboratories for fuel reprocessing, radiation chemistry

and application of radioisotopes in nuclear medicine, industry and agriculture, started with work. In the same time, in the "Vinca" Institute, the Radiation and Environmental Protection Department was established. Its primary task was collecting, conditioning and storing of the generated and accumulated radwaste materials, on the site of the "Vinca" Institute,in absence of a final storing centre,being in project stage.

Quantities and nature of radioisotopes presented in collected radwaste materials until nowadays, have established a task in front of the "Vinca" Institute investigators, to solve the problems that are arising from the collected radwaste materials in their safety handling, treatment, conditioning, storing and final disposal.

All the accumulated radioactive waste materials should be treated due to the international regulations and recommendations of IAEA and ICRP (1,2). Quantities of the accumulated radwaste and a level of bonded activities, ordered necessity of safety handling of these materials, as well as development of appropriate technologies, which have to provide final products of the immobilization processes to be safely stored and disposed.

Numerous procedures that are adopted in developed countries, having a great deal of experience in the radwaste materials management, were adopted in the earliest stages of our investigations. First of all, we had to established types and quantities of the radwaste already stored on the "Vinca" institute site, their generating frequency and to adopt safety procedures related to their handling in transport, immobilization, storing and final disposal. Due to these safety procedures, we were obligated to confirm safety of an each step in the radwaste materials management. This task was related to an investigations of the radwaste materials chemical and mechanical treatment, before their introduction to an immobilization processes and treatment of the mixtures of radwaste materials and matrix materials, as well as materials that are anticipated for the final disposal facility construction on prospecting their physico-chemical and mechanical characteristics. Processes of radwaste materials pretreatment and treatment of the radwaste materials are depending on their form and phase, as well as on possibility of recovering the valuables components, that are presented in the bulk radwaste form, using one of the appropriate methods (1).

SOURCES, TYPES AND QUANTITIES OF RADWASTE MATERIALS AND MANNER OF THEIR STORING

Radioactive waste materials that arised as off-products in the work of research reactor, as well as in industry and medicine, were interim stored on the Institute site. All these materials were transferred to the covered interim storages on the Institute site, that was built in early seventies, whether they were in liquid or solid form.

Liquid Waste Materials

For the purpose of collecting of the liquid waste materials, generated in the research reactor, four stainless steel reservoirs, each volume of 300 m3, were built. These underground basins were connected with reactor building through a special net work system. The main radioisotopes in these reservoirs are: 137Cs, 60Co and 3H, where the specific activity of the liquid is about 4 KBq/ml. The other source of the liquid radwaste materials is radioisotopes production laboratory. These radwaste materials have different radionuclide composition from the materials of the first group, and the preconcentration of liquid is usually done before its introduction to the underground reservoirs. This group of radwaste is divided into two subgroups, due to half-life of the radioisotopes present. The third source of the liquid radwaste materials arising from the isotopes application in research, medicine and industry.

Radioactive waste materials arising from radionuclide application in medicine and industry, are temporary stored inside the storing facilities on the site of the "Vina" Institute. Liquid radwaste materials are placed mainly in the plastic made barrels, V=50 l. Liquid radwaste materials collected in the underground reservoirs are periodically inspected on pH value and specific activity. Building of the facility for the liquid radioactive waste materials treatment, is in progress.

Solid Waste Materials

Solid radwaste materials are generated from the sources already discussed, as well as from collection of the spent sealed and unsealed sources, fire alarms, radioactive lightning rods, radium needles and discarded filters from the ventilation systems. These materials are divided into two groups: compressible and

incompressible materials. If the materials from the both groups are valuable, they could be decontaminated in the first treatment step. These materials are stored in the 200 litres steel barrels, with or without interior concrete shield. The annual production of these materials till 1990 was about 70 m3, depending on intensity of work in the nuclear program. For the volume reduction of the compressible waste in drums, 50 tone press was used. At the beginning, drums were stored on the uncovered storage, where they were under the destructing influence of corrosion.
In early seventies, first covered interim storage was built, with the capacity of 5000 drums. The main radionuclides in almost 50% of all drums are 14C and 3H. Dose-rate in this storage does not exceed 5mSv/h. In 1984., the new interim storage shed was built. In this facility, 1500 drums are stored and the great deal of the stored radwaste materials are spent sealed sources from medicine and industry, dominantly containing 60Co, 137Cs and 153,154Eu.

EXPERIENCES

When the row radwaste material is conditioned mechanically and chemically in aim to decrease the volume of the waste form, such a treated material is immobilized in the inactive matrix. In our investigations, as a matrix, domestic cements and bitumens were used. After solidification of the radwaste-matrix mixture form, hardened monoliths are obtained. These solidified radwaste mixtures are investigated on their physico-chemical and mechanical properties. Good experimental results of the investigations, performed in aim to confirm good embedding properties of the matrix-radwaste mixture forms, which could guaranty stability of the solidified waste forms during the operations of their transport, storing and final disposal, even in the elapsed periods of time.

Research and Development Work (R&D)

Even the regulatory bodies of the Yugoslav government imposed temporary prohibition of nuclear power plants building, objects for treatment, storage and disposal of radioactive waste materials were excluded and we were in position to continue with our R&D work, in attempts to start with the preliminary works of the first Yugoslav radwaste disposal facility and all activities, that are related to this goal. This work is related to the procedure steps in treatment of solid and liquid radwaste materials, before they are introduced into the immobilization processes, developed until nowadays in the "Vinca" Institute: cementation and bitumenization (3,4,5,6). Immobilization processes in which two component resins were applied as matrix materials were neglected in the early stages of the matrix material selection, mainly because of expensive immobilization process, as well as some weak physico-chemical characteristics. Developed immobilization processes have, as a final goal, production of the solidified radwaste-matrix mixture form that is easy for handling, and that satisfy safety requirements for interim storage and final disposal of such materials, on the appropriate sites. Shallow land burial engineer trench system, as a concept of the final disposal system for low level radioactive waste materials, is adopted. Following this strategy, materials that the engineer trench system consists of: mortar or bitumen as matrices, concrete for containers and trenches, back fill and drainage materials, were investigated. Research was done in aim to define the most important physico-chemical properties, that characterized implemented materials as barriers to the radionuclide migration from the trench system to an environment. Optimization of the matrix materials, either mortar and bitumen, were performed, as well optimization of the concrete formulations for containers and trenches in aim to promote the best physico-chemical characteristics of the materials applied in three stage barriers concept for radwaste materials final disposal system.

Experimental Work Related to the Radwaste Management

Testing methods, which are applied mainly in radwaste-mortar mixture form properties examinations, are:

- leaching test in static conditions,
- leaching test in static semi-real conditions,
- leaching test in dynamic conditions,
- leakage test,
- test of radwaste-mortar mixture form accelerated ageing,
- testing of mechanical characteristics,
- permeability measurements.

All the experiments were ruled up to the IAEA recommendations and standardized procedures (1,2,7).

The most important characteristics that were investigated for each material in the final disposal system, are: main radionuclides leach-rates and adequate diffusion coefficients, permeability, porosity, compressive strength, resistance on freezing and open flame. One of the specific investigation methods that was developed in the "Vinca" Institute was "leakage"-test, that should simulate radionuclides migration from the encapsulated solidified radwaste-mortar mixture form through the well defined inactive barriers of the final disposal system, into the inactive environment, e.g., distilled water in these experiments. One of the aims of the R&D work is, by using the results of the performed experiments, to be in the position to predict behaviour of each segment of the future disposal system, especially solidified waste-matrix mixture, in the prolonged periods of storing time, in normal and accidental conditions on the disposal site. In this manner, experimental results for prospected radionuclides leach-rates, as well as their efficient coefficient of diffusion, obtained in leaching-test, using Standard Hespe`s and modified, accelerated and Soxlet`s methods, were used as input parameters in the mathematical modelling of the phenomena of radionuclide leaching from the matrix materials to the environment (3,4,7,8,9). For this purpose, modified program packages are used. Results of the experiments, performed in a period of 3 to 5 years shown good accuracy of the applied mathematical models to the experimental results, when simulating the radionuclide leaching from the inactive matrices in the prolonged time periods (5,6,8,9,10).

FINAL DISPOSAL SITE CONCEPT

Radioactive waste materials generating in FR Yugoslavia established a task in front of the organizations responsible for handling with the hazardous and radioactive waste materials, starting with development of an idea project in such hazardous materials safety treatment in collection to a final disposal.

As a concept for the final disposal system of such prepared solidified radwaste-mixture forms, engineer trench system is adopted. This system consists of four barriers to the influences that might occur to and from the trench environment, and they are: immobilized matrix-radwaste mixture form, solidified inside 200 l metal drums or in concrete made containers, depending on the applied immobilization technique, drainage materials and trench walls, made of the concrete of the high mechanical strength characteristics. Engineer trench system will be divided into the sections, each representing independent trench system, with the control possibility of the system potential damages, expressed as radionuclide presence in the drainage network system, that goes under the trenches and collects liquids that might penetrated through facility. All R&D work that is performing in the "Vinca" institute, dealing with the treatment and management of the radioactive waste materials has its final goal in constructing the Yugoslav radwaste materials final disposal centre.

STRATEGY

In our attempts to prepare all the pre-work that is necessary for a future Yugoslav final disposal centre for the low and intermediate level radioactive waste materials, we decided to established National Agency for Radioactive Waste Management (NAROM). This body of the national importance will have its main task in planning and leading all the actions that are related to the radioactive waste management, such as: making a decision on the final national disposal radwaste materials disposal centre, concerning the disposal site and techno-economical aspects of the facility, and it will have advisory role for all aspects of the radioactive waste management: appropriate pretreatment, treatment and immobilization techniques for the certain waste materials, manners of solidified waste materials transport, storing and disposal, etc.

Figure 1 graphically represents concept of the organization of the Yugoslav agency for the radioactive waste management.

CONCLUSION

Preliminary investigations and research work related to the final disposal site for the future Yugoslav radwaste disposal site is in progress. Until the final decision concerning facility site is made, all the other, necessary previous step should be finished, having a common goal in being positive about quality of the all applied processes and techniques in radwaste management. Optimization of the chosen techniques in treatment, conditioning, immobilization and storing the radwaste materials is nearly completed and great deal of the work is almost done. Investigations are still carrying out on materials that are adopted as components of

the engineer trench system, in aim to improve their physico-chemical properties, mainly retention the radionuclides release from the disposal facility to environment, as well as their mechanical characteristics. Parallel, collection of the new generated waste materials and their interim storing at the site of the Institute "Vinca" is continuing.

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SPENT FUEL RERACKING EXPERIENCES AND PROSPECTS IN KOREA

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ABSTRACT

High density storage racks (HDSRs) have been, and will be installed to alleviate the saturated condition of spent fuel pool storage capacity in some operating nuclear power plants in Korea. Korea Power Electric Company (KOPEC), the A/E, has been performing the technical support in the reracking projects for Korea Power Electric Corporation (KEPCO), the utility. This paper presents the experiences and some issues encountered from previous and current reracking projects. The prospects for future reracking projects and the design of spent fuel pool under consideration are also discussed focusing on the work to be performed by KOPEC.

INTRODUCTION

In Korea, nuclear power plants have been an important role of the national energy system since the first commercial operation of KORI-1 in 1978. Since then, the nation's nuclear power program has continuously expanded. At present, Korea has nine nuclear power plants (8 PWRs and 1 CANDU) in operation with generating capacity of 7,616 MWe. The nuclear share of total installed electrical capacity is about 36%

which requires approximately 180 MTU of PWR fuels and 100 MTU of CANDU fuels every year. Seven(7) nuclear power plants, with capacity of 6,100 MWe, are currently under construction with an additional 7 more nuclear power plants, with capacity of 6,700 MWe, planned by the year 2006.

There are several long-term issues to be resolved in order to continue Korea's nuclear energy policy as planned. The major issue is the radioactive waste management which includes interim spent fuel storage, and low- and intermediate-level radioactive waste disposal. The spent fuel storage capacity of most of the nuclear power plants operating in Korea will be nearly saturated by 2000. To resolve this issue, the Korean Atomic Energy Commission, the governmental regulating authority, decided in 1988 that a central away-from-reactor interim storage facility for spent fuel will be constructed by December 1997. Several studies on the interim storage of spent fuel have been performed. However, the construction of an interim storage facility has been delayed due to the difficulty of site selection. Therefore, reracking projects to increase the at-reactor storage capacity of the spent fuel are under consideration since the utility is still responsible for the at-site storage of spent fuel until the time the interim storage facility is constructed.

In this paper, the experiences and technical problems encountered from past reracking projects, will be reviewed and future plans will be discussed.

RERACKING PROJECTS FOR OPERATING PLANTS IN KOREA

Spent fuel storage racks ensure a safe storage of spent fuelunder normal and postulated accident conditions. The fuel storage racks are designed to withstand operating loads (dead loads of fuel assemblies) as well as safe shutdown earthquake loads, as required by Category I seismic requirements of Regulatory Guide 1.29.

As of November 1994, the number of discharged spent fuel assemblies is 3,155 for 8 PWRs and 57,196 fuel bundles for the single CANDU unit. A dry storage facility was installed to increase the storage capacity for the CANDU spent fuel in 1992. However, there are no dry storage facilities for the PWRs, therefore several reracking projects have been completed to increase the at-site storage capacity of PWR spent fuel since UCN-2 (Ulchin Unit 2) reracking in 1990. KOPEC, the A/E, has been performing the engineering work, including technical specification preparation, bid evaluation, reviews of manufacturer's documents, and licensing support, for the reracking projects of KORI-3, UCN-1, UCN-2, KORI-4, YGN (Younggwang)-1 and YGN-2. In the reracking project of UCN-2, KOPEC only prepared technical specifications and performed the bid evaluation. In Korea, to rerack existing spent fuel racks and install High Density Storage Racks (HDSRs) requires approval of regulatory body, KINS (Korea Institute of Nuclear Safety). The utility must amend the related section of FSAR, submit all licensing analysis and resolve all questionnaires raised by KINS. KOPEC and manufacturer also resolve the problems encountered in licensing procedure to support the utility. Companies which participated in reracking projects and their respective work scope, are summarized in Table I.

The outline of reracking projects performed or under consideration is described below.

UCN-2 (ULCHIN UNIT 2) RERACKING

UCN-2 is a 3-loop pressurized water reactor designed by Framatome, and has been in commercial operation since 1989. The core is sized for 157, 17x17 fuel assemblies. The original storage capacity of 472 fuel assemblies was expanded to 893 fuel assemblies by completely replacing the old austenitic stainless steel racks with newer boraflex HDSRs of region I type in 1990.

As of November 1994, UCN-2 has stored 252 spent fuel assemblies in its spent fuel pool. It is expected that the storage capacity will be saturated in 2005. However, the plant is expected to be reracked again by replacing partially (about one and a half core) of existing boraflex HDSRs with the boral HDSRs of region I type to accomodate fuels of high enrichment. After the second reracking, the spent fuel pool will consist of two regions, boraflex HDSR and boral HDSR. The remaining boraflex HDSRs of region I type will be used as region II racks. The two region concept permit increased spent fuel storage by the use of denser fuel storage array for full burnup fuel assemblies. Region I has no restriction on burnup history of stored fuel assemblies. However, region II is restricted for use with fuel having a minimum cumulative burnup, which is dependent on the initial enrichment for each fuel assembly. Design parameters for each spent fuel pool for respective plants are summarized in Table II.

KORI-3 RERACKING

KORI-3 is a 3-loop pressurized water reactor designed by Westinghouse and has been in commercial operation since 1985. The core is sized for 157, 17xl7 fuel assemblies. As shown in Tables I and II, the spent fuel storage capacity was expanded from 746 fuel assemblies to 1201 fuel assemblies by installing an additional 455 borated stainless steel HDSRs of region II type in empty space of spent fuel pool. The old austenitic stainless steel racks is used as region I rack which can only store no
burnup fuel of 4.2 w/o initial enrichment.

As of November 1994, KORI-3 has stored 444 spent fuel assemblies from its own core and 156 spent fuel assemblies discharged and transported from KORI-1 in 1991 because the storage capacity of
KORI-l was nearly saturated at that time. KORI-3's storage capacity is expected to reach saturation in 2001.

UCN-1 RERACKING

UCN-1 is a 3-loop pressurized water reactor designed by Framatome and has been in commercial operation since 1988. The design parameters of the plant are the same as those for UCN-2. As of November 1994, UCN-1 has stored 265 spent fuel assemblies. As shown in Tables I and II, the storage capacity of UCN-l's spent fuel pool will be expanded from 472 to 1114 fuel assemblies by completely replacing the existing spent fuel racks with boral HDSRs in 1995. Without reracking in 1995, the plant will lose full core reserve capacity. The new boral HDSRs consist of 264 racks of region I and 850 racks of region II. Both region I and II storage racks are designed to accommodate fuel assemblies with initial enrichment up to 5 w/o U-235.

In UCN-1 reracking, two divers will remove seismic bracings and cut off the rack interlock connections in spent fuel pool. After reracking, the storage capacity will ensure unit operation until 2005.

KORI-4 AND YGN-1 & 2 (YOUNGGWANG UNITS 1 & 2) RERACKING

Design parameters of KORI-4 and YGN-1 & 2 plants are same as those of KORI-3. Both KORI-4 and YGN-l have been in commercial operation since 1986 and YGN-2 has been in operation since 1987. As of November 1994, KORI-4, YGN-l and YGN-2 have stored 388, 400, and 360 spent fuel assemblies, respectively. It is expected that the storage capacity of KORI-4 and YGN-l will be saturated in 1997 and that of YGN-2 will be saturated in 1999. Therefore, to increase the storage capacity, the installation of HDSRs is is scheduled for 1996 through 1997 for all three units. The HDSRs will be installed successively in non-racked (empty) space of spent fuel pools for the three plants. Since the spent fuel pool of three plants are identical, a single contractor will be selected, by competitive bid, to complete the work on all 3 units. It is anticipated that project cost and licensing uncertainty will be reduced since it is a blanket contract and licensing application.

For these 3 units, the utility and the A/E are planning to remove some of the spent fuel pool cooling piping and supports which would interfere with the HDSR installation to maximize the space in which HDSRs are to be installed. Generally, in Korea, such piping and supports are maintained and the contractor is required to work around the interference. The piping and supports will be removed by underwater divers after the range of pipe to be cut is determined and the maintaining of structural integrity and sufficient cooling capability is proved by all analysis needed. The removal of cooling pipe and supports is a common practice in countries other than Korea. Particular emphasis will be placed on thermal-hydraulic, criticality, structural/seismic, radiological, and mechanical accident analysis.

There are many alternatives in HDSR design and arrangement in spent fuel pool, but according to KEPCO's requirements, the new HDSRs to be installed in empty space of pool will be designed as region I racks. Existing racks will be designated as region II. When the reracking is completed, at least 400 spaces for each plant will be available for the spent fuel storage. Design parameters for the spent fuel pools using HDSR are summarized in Table II.

In Table III, differences between poisoned-rack (HDSR) and non- poisoned rack are summarized.

DESIGN IMPROVEMENTS FOR PLANTS UNDER CONSTRUCTION AND CONSIDERATION

As described above, four PWRs (YGN-3 & 4, UCN-3 & 4) and three CANDUs (Wolsung-2, 3 and 4) are under construction. Four PWRs are planned to be commercially operated in 1995, 1996, 1998 and 1999, respectively and three CANDUs in 1997 through 1999. YGN-3 & 4 were designed by Combustion Engineering, while UCN-3 & 4 were designed using the

Korean Standardization Plant design, which incorporates the results of standardization study and experiences gained from YGN-3 & 4. Nuclear power plants under construction were designed to ensure the spent fuel storage for at least 10 years.

The spent fuel storage racks of YGN-3 & 4 and UCN-3 & 4 consist of nine 10x12 modules. The spent fuel, which is CE standard fuel, will be stored in two regions of the pool. Region I provides core off-load capability for 252 spent fuel assemblies (equivalent to one core and maximum refueling batch plus spares). This is achieved with 50% density storage in checkerboard array using "L" inserts in the usable cells. The "L" insert is a non-poisoned stainless steel insert which provides the needed flux trap water gap. It also provides additional metal thickness for neutron absorption and limits the displacement between the fuel assemblies and the rack wall, which minimizes loads due to seismic acceleration. New fuel assemblies placed in the spent fuel pool in preparation for a refueling outage are located in region I. Region I also includes storage space for failed fuel assemblies. Region II provides 75% density storage for 426 spent fuel assemblies. "L" inserts are required in the region II cells to provide a flux trap water gap. The cells that are not used are blocked to prevent improper storage. A total of 678 storage locations are thus provided. This capacity allows storage for approximately 11 years of unit operation (380% of a full core). The structural design of future spent fuel racks includes provisions for accepting spent fuel with 100% storage using neutron poison inserts for future expansion potential (approximately 600% of a full core).

Nuclear power plants which are under consideration, including YGN-5 & 6 will be constructed incorporating the latest high density storage techniques. To this end, the responsibility for design of spent fuel storage rack was transferred from NSSS scope to A/E scope. Planned storage capacity will allow storage for at least 20 years of unit operation. Therefore, in the plants, the partial storage concept of spent fuel, designed by CE, will be changed to the full storage concept using HDSRs. Spent fuel storage racks will be designed for extended burnup fuel of 5 w/o initial enrichment in nuclear power plants under construction and consideration.

DESIGN REQUIREMENTS FOR NEXT GENERATION REACTOR

Korea is studying the design of an advanced reactor, often called the Korean Next Generation Reactor, having increased safety and improved economics. The basic design requirement of the spent fuel pool storage capacity for the Korean NGR is that it will only ensure the storage of spent fuel discharged for 10 years plus one core reserve as described by K-SRED (Korea Standard Requirements Document). However, if we consider the future improvement of storage technology, including boron credit, poison rack, and/or rod consolidation, the spent fuel pool storage capacity of the Korean NGR will have sufficient capacity for the lifetime of the unit.

CONCLUDING REMARKS

The issues of radioactive waste management should be resolved to drive actively the national nuclear energy policy in Korea. Specifically, the major issue to be solved immediately is that of spent fuel storage since the spent fuel storage capacities of several operating plants are nearly saturated and construction of away-from-reactor interim storage facility has been delayed. The utility, KEPCO, reviewed the alternatives for spent fuel storage since the utility is responsible for the storage of spent fuel until an interim storage facility is constructed.

Their review concluded that the reracking was the first option to increase the at-reactor spent fuel storage capacity. Thus the spent fuel pools of UCN-1 & 2 and KORI-3 have been reracked and reracking of KORI-4, YGN-1, and YGN-2 is under consideration with bid evaluation process in progress. CANDU spent fuel is stored in dry storage facility after being cooled for 10 years in wet storage pool. Future nuclear power plants will be designed to guarantee sufficient spent fuel storage capacity (more than 20 years).

If an interim storage facility for spent fuel is not constructed by the early 2000's, additional reracking or at-reactor dry storage facility is inevitable. KOPEC is expected to play an important role in this facet of nuclear power.

Session 20 -- Public Participation, Education and Training
Co-chairs: Chris West, SAIC;
Margery Olson, SAIC
20-1
THE INSTITUTIONAL BARRIERS TO PUBLIC PARTICIPATION

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ABSTRACT
In the past twenty years, the mission of the Department of Energy (DOE) has evolved from national defense to environmental clean-up. The Cold Was nearing an end, the Department had begun shifting its priorities from nuclear materials production for nuclear weapons to management of the resulting waste and contamination. The culture of the Department of Energy has evolved along with its mission. Once a bastion where secrecy prevailed, the Department is now changing, opening up to allow citizens to impact DOE's decisions. The cultural change - still underway - has come slowly, due to institutional barriers at the Department. This paper examines some of these barriers and the methods employed to overcome them.
INTRODUCTION
The primary barrier to institutionalizing public participation at DOE has been the prevalence of the old culture. Chipping away at the old culture has involved two major initiatives. The first is the publication of the DOE Public Participation Policy, which defines and communicates the new culture and direction of the Department. The second - and probably, the most important - is ensuring that commitment to public participation comes from the top down. A recent study of DOE and DOE contract employees revealed that the only factor which predicts an employee's predisposition toward public participation was the stance of his/her manager on public participation. Senior managers are the decision-makers and role models at federal agencies; as such, their behavior should reflect the new direction of the agency.
So how to ensure that senior managers will be committed to practicing public participation? Communication. Explain to managers what effective public participation really means. It does not mean relinquishing the Department's decision-making authority to the public; DOE is still the ultimate decisionmaker. Public participation is reaching out and educating the public so they can provide guidance and input to the Department on strategic decisions. It is then taking this input, incorporating it (or not), and letting the public know why.
Two realities about public participation must be communicated to senior managers and agency officials. The process of public participation is one which will likely increase controversy, not decrease it. If the Department is opening up the decision-making process, it will become apparent that many of DOE's decisions are based on value judgements. This will outrage and impassion people - and the Department will hear about it. But this does not mean that a decision will never be made. An informed, involved public is much less likely to challenge a decision than an excluded public. The Bonneville Power Administration case clearly illustrates this premise. If an agency can document that it went the extra mile to promote education, outreach, and involvement efforts, and made the best decision it could based on these efforts, then people - and judges - although upset with the decision, are more likely to accept it.
Another important tool for institutionalizing public participation - not just at the senior manager level, but throughout the Department - is an effective training program. The Department offers informal lecture series on public participation to all DOE employees and contractors. DOE has also organized a half-day training program on public participation for senior managers and a two-day training program for program managers and below. The training programs are crucial to the ultimate success of DOE's public participation efforts. Most of the people at DOE were not recruited to do public participation. Many of them were recruited due to their backgrounds and disciplines: geography, water resource management, nuclear engineering...To expect these people to go out to the public and deal with outraged community groups, or Native American tribes, or environmental groups, is unfair. Education and training is necessary to prepare DOE representatives to effectively deal with and involve the public.
Finally, the Department is working to generate some incentives for public participation efforts. Performance standards have been develop which evaluate employees on their accomplishments and initiative in the public participation arena.

A public participation awards program has also been instituted, which recognizes outstanding efforts in the public participation realm.
Changing the culture of an agency is not an easy task, especially when the culture is imbedded in the system, as is the case with the Department of Energy. It will be a long process, but the culture will change - and continue to evolve in response to the changing times.

20-2
STAKEHOLDER PARTICIPATION PROGRAMS THAT WORK
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ABSTRACT
The U.S. Department of Energy's (DOE's) Environmental Management Program is committed to creating a working relationship with its stakeholders by sharing information about its activities and inviting public input. Successful stakeholder involvement results in decisions that are technically and economically feasible and environmentally sound, that protect health and safety, and that address public concerns and values. Public involvement is a legal requirement in many cases, but DOE is committed to moving beyond required involvement to provide a wide range of opportunities that are tailored to DOE's individual stakeholder groups. The purpose of this paper is to describe the efforts of DOE's Environmental Restoration (ER) Program to characterize the types of information and participation techniques that are used at its more than 130 sites, and to begin developing qualitative measures of success for its stakeholder involvement programs.
INTRODUCTION
DOE's activities directly affect public health and safety and the environment-for which DOE must exercise stewardship and be responsive to the public interest. Citizens have the right to influence decisions about matters that effect them, and public participation rights have been codified in many of the environmental laws with which DOE activities must comply. As a result, DOE needs broad-based support and participation in order to implement its environmental programs effectively. DOE plans and implements the ER Program, and serves as the lead agency in making decisions at most ER sites. However, it does so within a complex web of organizations that have roles in authorizing, overseeing, regulating, funding, reviewing, and participating in ER activities. These organizations, which include the U.S. Congress, the U.S. Environmental Protection Agency (EPA) and State regulatory agencies, and Site-Specific Advisory Boards (SSABs), are key ER stakeholders.
The ER Program has several objectives for its public and stakeholder involvement efforts:
Solicit the public's help in identifying ER site problems and issues;
Solicit the public's involvement in identifying the full range of approaches for addressing those problems and issues, and working towards a broad-based consensus early in the cleanup process;
Increase public understanding of the complex regulatory, political, technical and funding environment in which ER decisions are made, as well as the need to balance a variety of often competing interests and considerations;
Coordinate, integrate, and communicate information about ER program activities and participation opportunities;
Provide a range of participation opportunities that are tailored to meet the needs and interests of ER site-specific communities;
Provide timely feedback on how input was considered in the ER decision making process; and
Meet the letter and spirit of all laws, regulations, and negotiated agreements relating to public and stakeholder participation.
In meeting these objectives, the ER Program works closely with DOE's Office of Public Accountability, which serves as the central coordination point for public participation activities among all Environmental Management (EM) program offices, with other DOE offices, and between EM Headquarters and field offices.
The Need To Define Terms

DOE often uses the terms "customer," "stakeholder," and "public" interchangeably when describing its participation programs. Similarly, the terms "participation" and "information" are often confused. The ER Program uses the following definitions to guide its public and stakeholder involvement programs:

Customer: one who directly benefits (or should benefit) from an ER product or service; may be external or internal to the ER Program.

Stakeholder: any group or individual who is affected by or who can affect the future of the ER Program-as a subset of "customer" includes customers, employees, suppliers, owners, other agencies, Congress, the public at large, supporters, and critics.

Public Participation: meaningful opportunities to take part or share in the ER program decision making process; a two-way exchange of ideas, concerns, and issues with the goal of impacting outcomes. Example activities: public meetings and hearings, advisory boards, and comments on technical and programmatic documents.

Public Information: communication and receipt of data, facts, messages, and knowledge regarding the ER Program; materials prepared by the ER Program that inform stakeholders and enable them to participate effectively in ER decision-making. Example tools: fact sheets, newsletters, exhibits, and videos.

The ER Program believes there are clear differences among these terms, and that different approaches should be used depending on the audience and the intended outcome of the activity.

Conflicting Stakeholder Needs and Desires

In one sense, the ER Program's stakeholders include not only citizens who live near sites or in the same State, but every taxpayer in the Nation. Trying to reach and satisfy a group as large and diverse as this is difficult; compromises must be made and site decisions must be carefully negotiated and clearly communicated.

Each of the more than 130 ER sites has a different set of interested publics. For example, stakeholders at one of ER's largest sites in Washington State include two States, three separate counties, several Indian Tribal Nations, as well as regulators, Members of Congress, environmental groups, agricultural interest groups, labor groups, and a variety of community action groups. While some of the concerns of these different groups are similar, each has a different approach for resolving its concerns. For example, some groups are looking to DOE for money to fund their participation efforts. Other groups want information so they can provide more meaningful input into ER decisions. While others want to scrutinize Departmental actions with an eye towards impacting how DOE spends its ER dollars.

In addition, each stakeholder group has a preference for how it wants to participate in DOE's decision making process. In response, the ER Program provides a wide variety of information and participation opportunities, including numerous publications describing the Program in general as well as site-specific activities; site tours, public meetings and workshops; traveling exhibits describing ER accomplishments and priorities; Site-Specific Advisory Boards; information repositories; and many other mechanisms.

ER's challenge is to develop public information and participation programs and tools that are tailored to these varied groups. If ER is successful, the Program can help facilitate a broad-based consensus on the ER Program's objectives and how to achieve those objectives. With a successful program, DOE/EM may overcome the public skepticism that has resulted from decades of secrecy and public mistrust, and may be able to create new relationships with its stakeholders that are built on common trust and a dedication to solving environmental problems.

ER's Customer Focus Analytical Team

In the Spring of 1994, DOE's Office of Environmental Restoration developed six functional Analytical Teams for the purpose of conducting program-wide, crosscutting analysis. The purpose of these teams is to enable the ER Program to better focus its efforts on meeting EM's six goals. The analyses conducted by these teams will enhance the overall communication and programmatic understanding among individual ER projects and between the ER Program and other EM programs. The six Teams parallel EM's six major goals, as follows: 1) Technical Scope, 2) Program Management, 3) Technology Applications, 4) Risk Management, 5) Health and Safety, and 6) Customer Focus. Each Team has broad representation across the Office of Environmental Restoration, and has coordinated extensively with other EM offices, other DOE organizations, other Federal agencies, and private sector companies.

The Customer Focus Analytical Team (CFAT) is aligned with EM's Strategic Goal 6:

Develop Strong Partnerships Between the DOE and Its Stakeholders. CFAT is chartered to analyze and recommend to the ER Program a process to ensure that ER hears its customers; supports its customers with accurate, timely, and helpful information; and provides opportunities for their involvement. To fulfill this charter, CFAT has embarked on a series of analyses:

Development of a baseline of information and participation techniques and tools used by the ER Program and an analysis of trends in their application;

Benchmarking of current ER customer support and stakeholder participation efforts against other Federal agency and private sector practices;

Coordination with the Office of Public Accountability and other DOE offices in assessing customer and stakeholder satisfaction with ER Program activities;

Coordination with the Office of Public Accountability in developing a performance measurement and evaluation system to ensure that ER Program customer support and stakeholder participation efforts continuously improve; and

Development of specific recommendations to improve ER Program information and participation processes and activities, when necessary.

To date, CFAT has completed an inventory or baseline of information and participation activities across ER sites. In addition, the Team has begun to identify the attributes and techniques of "Best in Class" public participation programs and compare how these attributes and techniques have been adopted by the ER program.

ATTRIBUTES AND TECHNIQUES OF SUCCESSFUL STAKEHOLDER PARTICIPATION PROGRAMS

The first analysis conducted by CFAT was to identify the specific activities that DOE uses to inform the public and other stakeholders and gain their participation in the ER Program. To gather this information, CFAT interviewed program managers from 24 ER sites and reviewed numerous public participation/community relations plans. CFAT determined that the ER Program supports at least 12 unique participation or information programs, encompassing more than 145 activities. Eight of these programs, comprising 82 activities (57%), are considered information programs. These information efforts include routine newsletters and fact sheets that help the public acquire the understanding they need to become more informed participants. Four programs, comprising 63 activities (43%), provide participation opportunities. These participation opportunities include SSABs, tours, workshops, and meetings, all of which enable stakeholders to engage in a dialogue with DOE on issues that concern them. Fifteen sites reported that they either have or are in various stages of establishing an SSAB. Three sites (Kansas City, Mound, and Pinellas) concluded with the help of their stakeholders that a formal SSAB was not required at this time. For example, instead of an SSAB, the Mound site is using "Mound Action Consortia," small focus groups set up for limited periods to address specific issues. Figure 1 summarizes the types of activities that are being conducted across the ER Program.

Attributes of "Best in Class Programs"

As a follow up to this inventory of ER Program stakeholder involvement activities, CFAT investigated and developed a preliminary list of attributes and techniques most widely cited as characteristics of "Best in Class" public participation programs. The Team then evaluated whether these attributes and techniques are characteristics of ER Program stakeholder participation efforts. The Team has not yet assessed the quality of individual information or participation techniques nor evaluated the impact each technique has had on the overall level of customer satisfaction.

CFAT's analysis determined that the ER Program exhibits many of the attributes cited for "Best in Class" organizations, with less emphasis, however, on evaluation activities. This analysis also shows that the ER Program is utilizing all of the top ten successful public participation and information techniques that are being used by the external organizations that were investigated. In fact, the ER Program utilizes a variety of innovative public participation and information techniques that are not widely utilized by other organizations.

The methodology used to conduct this analysis involved three activities: 1) an extensive literature search of more than 16,000 titles and the review of information on effective public participation programs and practices and improvement methods of "Best in Class" and award winning private sector companies; 2) interviews with professional and trade associations, private industry, and other Federal and local government agencies; and 3) interviews with EM Public Participation Coordinators. The information collected by CFAT is considered representative of the types of approaches and techniques being used by the various organizations. The data are not

comprehensive of all outreach and participation techniques that are being used in the public and private sector.

CFAT identified several common attributes or characteristics of successful outreach and participation programs. These attributes are broad in nature and provide generic guidelines for the development and/or assessment of information and participation programs. The Team organized the attributes logically within the chronological life cycle of a program: research, planning, execution, and evaluation. Figure 2 summarizes the key attributes.

Information and Participation Techniques That Work

In addition to identifying the attributes of successful public participation programs, CFAT has identified and begun to assess specific information and participation techniques that are used by the "Best in Class" programs. As shown in Tables I and II, there are few techniques which are unique to any one organization. The private sector companies that were contacted reported fewer techniques, with an emphasis on print and broadcast publicity campaigns and citizen advisory boards. Private industry also does not appear to consider hearings/meetings/workshops as key to the success of their programs. This type of participation is viewed as a regulatory requirement rather than a tailored outreach approach. One notable exception is the use of public debates, through which industry provides a public forum for open discussion of a particular technology or other area of citizen concern.

Insert Tables I and II

Industry, particularly the chemical industry, has been employing citizen advisory boards for more than five years. The boards are viewed as a key connection to the community. Some companies that have had boards in place for several years reported that the current challenge is keeping the board members interested and involved, since much of the mystery and secrecy of plant operations has been erased. Print and broadcast campaigns are used to acknowledge the efforts of the citizen advisory boards and increase general awareness of plant operations. Private industry also uses exhibits at trade shows, public libraries, and other public venues to encourage partnerships with the community in conducting plant activities.

Cable information shows have been used at only one EM facility, with limited duration and mixed results. One of the private companies contacted by the Team produced a cable television show for a six-month period. While viewer ratings were high, the company could not continue to justify the high production costs and canceled the effort. Alternatively, one local government that was contacted reported using cable access as a way to announce and describe special programs and new opportunities for citizen involvement. ER was unique in its development and application of interactive simulation games like the "Priority" game and "Cleanopoly," which may represent an innovative tool that could be adopted by other government agencies, as well as the private sector.

Three Innovative Techniques

In many ways, DOE has been more aggressive, innovative, and responsive in developing and implementing tailored public information and participation tools when compared to many of the external organizations that CFAT reviewed. For a program that is only 5 1/2 years old, EM/ER has accomplished a great deal in providing a variety of options for citizen involvement. The following three examples highlight the creative thinking of EM/ER public involvement professionals.

The "Priority Game"

This interactive board game was invented by the DOE/ER Program's Site Manager at Mound. The game was introduced at a December 2, 1993, Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) Quarterly Public Meeting to inform stakeholders about the impact of budget reductions on the ER Program and about ER activities and their costs, and to gather stakeholder input on which activities to prioritize with the available funding. The game board itself resembles the well-known Monopoly Game. Each square on the game board represents a planned Fiscal Year (FY) 1994 operable unit activity and its cost. To start the game, each stakeholder is allocated $28 million in non-negotiable million-dollar bills. Twenty-eight million dollars represents the total funding needed to complete all of the planned FY 1994 activities.

The first two blocks on the board are mandatory "stops." The first is labeled "Budget Reduction, Lose $11 Million." Each stakeholder has to place $11 million in a basket labeled "budget reductions." The second stop, "Program Operating Costs,"

reduces available funding by another $6 million. These costs are defined as employee salaries, utilities, and public participation activities. As with the "budget reductions," the "operating costs" are collected in a basket that is passed around the room. Each stakeholder is now left with $11 million and a very clear understanding of how quickly funds are reduced or spent.

"Playing" the game then focuses on setting priorities given the reduced amount of funding available. Stakeholders place their money on the game board blocks representing their "Priority" activities. The game ends after each stakeholder "purchases" their "Priority" activities. The money placed on each activity is then counted as "used" and represents a preliminary result of stakeholder priorities. Within a week of this meeting, DOE sent a return postage-paid survey to all stakeholders asking for their formal input on priority selection. DOE received 100 completed surveys.

The "Priority Game" enlivened the meeting and provided a strong visual presentation of an otherwise abstract concept. The stakeholders could clearly see the impact of supporting their particular priority in light of the priorities of others. The formal surveys were returned quickly, suggesting that interest in the topic was retained after the meeting concluded. The "Priority Game" allows stakeholders to choose activities to be conducted while becoming aware of what activities won't get done. There isn't a "winner" in the "Priority Game." The stakeholders choose from an open set of possibilities, not a DOE selected option.

The "Roadshow" Traveling Exhibit

The EM Roadshow is an exhibit display accompanied by various publications designed to communicate technical and programmatic information about the EM Program to the public. The exhibit also highlights some of the successful applications of modern cleanup technology that are being used nationwide. There are five primary goals to the Roadshow: 1) inform and educate EM Program stakeholders and the general public; 2) increase public awareness of DOE's overall environmental mission; 3) encourage and improve public participation in EM programs and decision making processes; 4) provide an opportunity for a positive interaction with a DOE/EM representative; and 5) solicit feedback on EM Program directions and public information needs.

The Roadshow has been traveling to technical conferences and workshops, educator conventions, Community Days, and other public and professional venues since 1989. In that time, DOE has reached more than one million people who have attended more than 200 events. The exhibit staff is selected from a cadre of 75 EM Federal and contractor employees, with more than 50 percent of those individuals being Federal technical staff working in the EM Program. Roadshow staff have developed and distributed more 250,000 EM Program publications since 1989. The most popular publications are a set of EM Fact Sheets, the Nuclear Age Timeline Poster and Resource Guide (which is in its fourth printing), the EM "Primer (both the standard and Student edition)," a Q&A booklet on ER activities, and a series of classroom experiments and other teaching aids.

There are two key benefits to the Roadshow activity. First, by proactively reaching out to the public and providing information that is tailored to a broad spectrum of interest and understanding levels, the Roadshow has been able to increase DOE's public credibility. There is always a technical staff person at each event, who is familiar with EM sites and activities and can discuss cleanup progress. Attendees at Roadshow events are hungry for information, and take as many publications as they can carry. Roadshow staff are consistently greeted with the comment: "We are glad to see DOE here," which indicates a positive trend in abandoning the secrecy of the past. The second benefit is the opportunity for Roadshow staff to work across EM organizational lines, meet the public, and learn firsthand the concerns of various stakeholder groups. EM technical staff and contractors have become more sensitized to the challenges of addressing public information and participation needs.

The Fernald Envoy Program

The Fernald Envoy Program was established in February of 1994 by DOE and the Fernald Environmental Restoration Management Corporation (FERMCO) to promote one-on-one communication between Fernald personnel and local community groups. More than 75 FERMCO and DOE employees have been trained as Fernald envoys. These employees serve a wide range of stakeholder groups such as adjacent property owners, business leaders, schools, environmental groups and agencies, and public officials. Envoys provide detailed information about Fernald objectives, policy, mission, roles, constraints, and requirements, and then listen to stakeholder opinions, suggestions,

values, questions, and other input about Fernald issues. The input from stakeholders is then submitted to the decision makers at the site. Relationship-building through home visits and small group meetings is the foundation of the Envoy Program.

The Envoy Program has provided employees an opportunity to develop personal, face-to-face relationships with more than 100 stakeholder groups, many of which had previously not been included in the decision making process at Fernald. Through Envoy Program tours of the site and one-on-one relationships that have been established, stakeholder trust and confidence has increased at Fernald. Stakeholders have realized that some previously held perceptions were not accurate, that a cleanup program is underway at the site, and that regulatory milestones are being accomplished. In some cases, the feedback that Envoys have received from stakeholders has changed decisions made by management. For example, the final Strategic Plan for the site was revised based upon stakeholder comments received through the Envoy Program. The personal attention Envoys have given stakeholders has helped diffuse some potential controversial situations. In fact, several stakeholders now help Fernald employees educate the public and correct misperceptions about Fernald site cleanup efforts. Based upon the success of the Fernald Envoy Program, several other ER sites are considering developing a similar program.

CONCLUSION

ER/EM public information and participation efforts exhibit many of the attributes of "Best in Class" programs, with variations across ER sites and field offices. The ER Program is thinking creatively about what, how, and when it communicates. CFAT is continuing to examine other public and private sector practices and will recommend to ER Program managers new procedures, techniques, and tools that may be adopted. The ER Program is also continuing to work closely with the EM Office of Public Accountability to develop public involvement program measures of success.

About The Authors

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20-4

BETTER DECISION MAKING: PUBLIC INPUT IS ONLY PART OF THE EQUATION*

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ABSTRACT

According to Secretary of Energy Hazel O'Leary, "Public participation must be a fundamental component of the Department's program operations, planning activities,

and decision-making." The Department of Energy's (DOE) current focus on improving the form and format for public input is clearly a step in the right direction. But simply changing the way input is gathered from the public fails to help agencies make more informed and collaborative decisions. A paradigm shift is needed to achieve the purpose and goals of DOE's public participation policy--mutual understanding and trust between the Department and the public it serves. Managers and communicators must forge mutual understanding and trust between themselves to incorporate function, as well as form and format, into the public participation equation. This paper presents three case studies that exemplify the failure to optimize stakeholder involvement and input in the Department's public participation efforts and offers observations and suggestions for a new paradigm that focuses on the internal process that creates decision outputs.

INTRODUCTION

Activities by government agencies are increasingly performed under legally mandated public oversight and involvement. Numerous programs at the Department of Energy's Idaho National Engineering Laboratory (INEL) are no exception, and managers with mostly technical backgrounds are often faced with implementing public participation efforts in the course of making technical and program decisions. Conversely, communicators with mostly social science backgrounds are responsible for gathering public input for the technically oriented decision maker's use. Involving stakeholders up front at the points where these disparate but interdependent disciplines first meet may offer the greatest opportunity for optimizing public participation programs.

For the purposes of this paper, the authors have used James L. Creighton's definition for public participation: "public participation is the process by which public concerns, needs, and values are incorporated into governmental decision making. Public participation is two-way communication, with the overall goal of better decisions, supported by the public." (1) The three case studies below are used to illustrate some problems common in public participation activities: stakeholder involvement designed to provide input early in the decision process made "useless" by circumstance or poor planning, efficiency and potential public support lost by failing to incorporate public participation early in the process, and failure to deliver on implied commitments and a public mandate.

CASE STUDY 1, WENDY GREEN: THE SITE TREATMENT PLAN FOCUS GROUP PROCESS

Compliance with the Resource Conservation and Recovery Act's (RCRA) Land Disposal Restrictions (LDR) has presented difficulties for the Department of Energy (DOE). Previously, DOE could generate and store mixed wastes (wastes that are both radioactive and hazardous) in compliance with RCRA. In May 1992, however, the Environmental Protection Agency's (EPA) national capacity variance expired, and any future generation and storage of mixed wastes by DOE would violate RCRA unless an approved treatment technology existed for such wastes.

Because no approved treatment technologies existed for most of these wastes, Congress passed the Federal Facility Compliance Act (FFC Act) in October 1992. The FFC Act waived DOE's sovereign immunity for penalties under RCRA and provided a three-year window before penalties could be assessed for storage prohibition violations. In addition, the FFC Act required each DOE site to develop a site treatment plan (STP) to identify treatment plans for mixed wastes generated or stored at the site. Additionally, the Act mandated public participation during the development of the STP; the regulator for each DOE site was given the responsibility for conducting the mandated public participation. DOE published its strategy for complying with the FFC Act in the April 6, 1993, Federal Register--including a commitment to seek its own public participation prior to the publication of the Draft STPs.

As part of that commitment to public participation, DOE's Idaho Operations Office developed a focus group process to solicit input from Idaho residents. The process was designed to build a better understanding of how Idaho residents evaluate projects proposed for the INEL. At the time, the DOE thought it would be considering multiple options for dealing with each mixed waste at the INEL and planned to use focus group input to help provide a publicly defensible set of criteria for determining which of the technical options should be proposed as elements of the STP.

In December 1993, nearly 200 people throughout the State of Idaho were invited to attend a series of two focus group meetings to be held in four regions of the State.

An open invitation was extended to include other interested parties as well. At the first meeting, a neutral process-facilitator solicited issues of concern through a "brainstorming" session, resulting in a total of 196 identified issues. Afterwards, the facilitator categorized and reformulated the list of issues, combining related issues to the extent possible to arrive at a list of 45 potential evaluation criteria. At the second session, the participants ranked the 45 evaluation criteria using two social science tools, dot polling and a simplified nominal group process. The end product of the groups' work was a set of overall rankings of evaluation criteria that the focus group participants use to evaluate proposed projects; all were presented in the final report on the focus group process. The report was sent to the focus group participants and to the DOE decision makers working on the STP. According to the overall rankings, the most important conclusion that can be made is that "Water Resources Impacts" are considered to be very important; the criterion ranked first in the five overall rankings. Four other criteria ranked in the top 10 in all rankings: Health Impacts, Air Impacts, Biological Impacts, and Accidents and Accident Impacts. The five can therefore be assumed to be very important to Idaho residents in evaluating projects proposed for the INEL.

It was planned that the criteria ranked in the report would prove useful to DOE in selecting technical options for inclusion in the INEL Draft STP when multiple options exist for treating specific mixed-waste streams. For example, if two options were considered technically feasible by DOE-ID decision makers, the one which offers more protection for water resources might be selected in response to predicted public opinion. Similarly, if the results of this study can be assumed to be representative of opinion among Idaho residents, such selections should be supported even if the non-preferred option is more costly because participants ranked water resource impacts much higher than "responsible use of tax dollars." Even when only one treatment option exists, however, the facilitator felt the results would provide insight into how a project design might be modified to meet the concerns of Idaho residents.

Unfortunately, the people responsible for selecting treatment options to be included in the INEL STP were unable to use the focus group input because, as the selection process proceeded, no real options existed. As previously stated, it was originally believed that multiple treatment options would be available for each mixed waste generated or stored at the INEL. Many of the options were eliminated on technical grounds, however, after a close reading of the LDR. As a result, there was only one option each still in consideration (after complying with the LDR) for most of the mixed wastes. The LDR treatment standards essentially left no real choices open to the DOE. The original mission of the focus group was lost, and the public input could not be used to help select treatment options.

In general, a wise strategy for public participation includes the collection of public input early enough in the decision process to be of use to decision makers. Conforming with that general principle, the schedule for conducting the focus groups was driven by a desire to gather input before DOE had gotten too far along in its decision process. The strategy backfired in this case, however, as the parameters of the decision making process were not well understood up front. The focus group component of the public participation process was selected too early in the decision making process to be designed to collect input that could be of use to those making technology selections. It should be noted that this did not result from a lack of effort--DOE fully intended to use the input to select from a range of options. Perhaps the first mistake was made by DOE-Headquarters in making the commitment to conduct public participation prior to the release of the Draft STP; the aggressive schedule did not make the effort any easier and sound public participation strategies require adequate time and budget.

Although the focus group process is discussed at length in the chapter of the STP dedicated to the public participation process, the document fails to explain why the focus groups' input could not be used in making technology selections. In addition, the Draft STP was not mailed to the focus group participants. After telling the focus group participants why their input was being sought, DOE should have (in the author's opinion) provided an explanation as to why that input could not be used as planned. Additionally, DOE should commit to using the results to modify project designs in an effort to address the concerns of Idaho residents.

DOE deserves commendation for the innovative, interactive manner in which it sought public input for the STP. Unfortunately, however, the agency did not know enough

about its internal decision making processes ahead of time to ask for input that could subsequently be used.

CASE STUDY 2, JOYCE POLE: The SNF AND INEL EIS

The Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement (SNF and INEL EIS) is as complicated as its lengthy name implies. It is both a complex-wide and a site-specific document, a first in the annals of environmental impact statements for the DOE. How this hybrid came to be is a complicated story by itself, but suffice it to say it is driven by a negotiated United States district court order that dictates its schedule; it involves the Department of Navy as a cooperating agency because the Navy and DOE jointly manage spent naval fuel; and ten different sites are considered under five management alternatives in the programmatic portion.

The EIS supports two sets of decisions and will have a preferred alternative for each of those. Volume 1 covers spent nuclear fuel management across the entire DOE complex over the next forty years until the department decides the ultimate fate of its spent nuclear fuel. Volume 2 is specific to environmental restoration and waste management at the INEL for the next decade. Several appendices provide detailed data for the main volumes to make a total draft document of more than 4,200 pages.

When the draft EIS was distributed to the public in June of 1994, it had a missing piece--no identified preferred alternatives for DOE. But there was a reason for this, and it didn't break any of the rules of the National Environmental Policy Act (NEPA) or circumvent regulations from the Council on Environmental Quality (CEQ). Preferred alternatives are to be called out in a draft environmental impact statement only if preferences exist. (CEQ does direct that the final document identify preferred alternatives.) The analysis of the environmental impacts of the five alternatives considered for spent nuclear fuel management and the four specific to the INEL Environmental Restoration and Waste Management Programs revealed no significant differences to warrant a preference based solely on environmental factors.

So DOE said it would use public input, along with technical and practical considerations of the impacts on DOE and Navy missions and projected costs for the various actions, to help derive preferred alternatives for the final EIS. The decision management process was structured to include three panels, one for each of the two volumes to develop technical criteria, and one for public issues that would provide input to the two other panels. The purpose of the public issues panel was to develop a sense of public concerns based on the 5,100 comments that had been received through the EIS comment process and then translate those concerns into useful factors for decision making. This panel's task was completely separate from the formal comment response process (prescribed by NEPA), which was focused solely on determining the adequacy of the document itself. In fact, many of the public concerns that were identified as "out of scope" in the formal comment response emerged as significant issues for the public issues panel.

A support team to the public issues panel, composed of EIS project office personnel, reviewed all 5,100 public comments and categorized them according to a pre-determined list of issues and concerns, statements of favor or disfavor for any of the alternatives, and the geographic location of the commentor. Following this, each team member independently drew up a list of what he or she thought the key public issues were. By continued refinement of the group list, the support team reached agreement on five concerns of most importance to the public.

Although specific social science methodology wasn't employed in the team's issues categorizations, that process was never intended to be--or portrayed as--a true random sampling of public opinion. Its intent was to advance an understanding of relative public sentiment rather than to attempt extrapolation of absolutes. Based on a limited review of available opinion polls related to environmental and nuclear issues, the issues panel found their roll-up of public concerns to be consistent with other published surveys.

Was the process successful? The decision panels for the programmatic and site-specific parts of the EIS did, indeed, take into serious consideration the work of the public issues panel. The concerns distilled from public opinion were folded into the two document-specific decision panels' structured and systematic process to arrive at recommendations for preferred alternatives. The public issues panel recommendations were subsequently approved by DOE high level managers who bought

into the three-part decision process and the role public opinion played in it. There is agreement among those involved that the preferred alternatives that evolved from the decision process represent the best balance of addressing public issues, DOE and Navy mission impacts, costs of implementation, and regulatory compliance. At the time of this writing, the preferred alternative recommendations were working their way to the Secretary of Energy for approval.

Will the public believe they have been heard and that what they have said has made a difference? That remains to be seen because the final document won't be available until later this spring. Also, the public's view may depend on how well DOE communicates the process that was used to reach the decision. Even more fundamental, however, is the fact that many of the public called into question the very purpose and need of the programmatic portion of the EIS. In their view, the scope of the document put the "cart before the horse". Many of the commentors wanted DOE first to identify the end of spent nuclear fuel, that is, how the department would dispose of it once and for all, before the department proposed the means for storing spent nuclear fuel until that end could be achieved. This issue was quickly identified by the public issues panel and evolved into one of the panel's decision factors.

But an EIS is not an end in itself; it's a tool to help an agency reach a decision. Public involvement has a rightful place in this decision process, too. Because NEPA and CEQ are woefully silent on this aspect of public involvement, federal agencies are left to their own devices. DOE's track record in this arena may be changing, if the decision process developed for this hybrid programmatic and site-specific environmental impact statement is used as a measure. Although DOE was treading into foreign territory without a roadmap as it developed this process, the end of the journey still represents one of DOE's best efforts yet at capturing and acting on public concerns.

CASE STUDY 3, AMY DIRKS STEVENS: THE SITE SPECIFIC PLAN

What would you do if you were producing a public communication product with a 94% positive response rating from its intended audience, which wants to see the document continued? If you were the U. S. Department of Energy, facing budget cutbacks and with no requirement to continue, you'd cancel it.

In this case, the product was a document--the annual Site-Specific Plan for the U.S. Department of Energy's Idaho National Engineering Laboratory. In late 1989, the DOE Headquarters issued the national Environmental Restoration and Waste Management Five-Year Plan, which outlined then Secretary Watkins' plans for cleaning up and managing the wastes at the Department's sites. The Five-Year Plan was updated each year to provide the public with a complex-wide progress report of the status of activities and projection for future actions and technology development.

A companion document for the public was required from each operations office to detail its site's specific Environmental Restoration and Waste Management conditions and plans. Content was both descriptive and prescriptive. Topics included an explanation of fiscal year budgetary proposals and distribution of funds by project; discussion of how the broad goals within the Five-Year Plan would be tailored and implemented at the site; an accounting for compliance with relevant laws and regulations; updates on project status and technology applications; and an overview of general site demographics (e.g., size, location, mission).

Public involvement associated with development of INEL's Site Specific Plan evolved and increased over time. The basic approach was to release a draft for public comment on the site's proposed activities for the coming year. Public meetings, surveys, and outreach office efforts were used to glean reaction and encourage face-to-face interaction on Department of Energy Idaho Operations Office plans for the Environmental Restoration and Waste Management Programs. The intent was to use public reaction to finalize the site's plans for the coming year and to better understand the concerns and level of support from its stakeholders. The final version of the Site Specific Plan reflected the effect of the public's input on the draft.

Involvement of the public changed the content, focus, and utility of the document. Comparison of the first version of the Site Specific Plan to the last issue shows visible improvements, such as a more approachable less technical format and language style; instructive graphics to enhance clarity of complex technical activities; educational sections on concepts such as radiation and toxicology; and highlights of key issues needing public evaluation, with accompanying explanations of how to take part in DOE decision processes. In the final year of production, the document was

split into two pieces: a concise summary, with a high-level overview of concepts, accomplishments, and proposed actions (8,000 copies printed) and a full plan with detailed data allowing for open scrutiny of the Environmental Restoration and Waste Management Programs (1,000 copies printed). This two-part approach satisfied both sides of a growing schism within the stakeholders' needs for and interest in salient information about the sites.

During the preparation of the Fiscal Year 1994 Draft Site Specific Plan, DOE Headquarters announced that it was considering elimination of the requirement and funding for both the overarching Five-Year Plan and the related Site Specific Plans. By production time for INEL's final version of the Fiscal Year 1994 Site Specific Plan, the document was known to be in its last required publication. However, the DOE Idaho Operations Office wondered if canceling the document would be an impediment to open communication with its stakeholders. If the Site Specific Plan was not produced, how would people get access to user-friendly, timely, and consolidated information on the site's two largest and most controversial programs? The answers to this question and others were captured through a mail-in survey form printed in the front of both the Site Specific Plan and Summary.

The response was unprecedented. The survey had the highest return rate ever received on any questionnaire or response method used in the history of INEL's public participation efforts. Intriguingly, the site's employees and contractors wanted the full document continued nearly as much as the public did. About 5% of each of the two target audiences were in favor of its cancellation. Eighty-seven percent of the employee respondents and 94% of the public (the intended primary audience) wanted the full Plan continued. It seems significant that all stakeholders in the public group responded to the question, but 8% of the employee respondents abstained from answering the query. The Summary, in its inaugural issue, received less favorable ratings, but not significantly so--84% of the public wanted it continued, as did 77% of the employees. The results were expected since the Summary had not benefitted from the evolution of several years of public input.

The survey also asked how the reader would respond to the absence of the documents. Only 3% responded that they would never even realize the full Plan was missing; 70% of both employees and public stakeholders said they would have to find the information themselves from some other source; and 19% said they would wonder what was happening at the site but not enough to do their own research (80% of these self-identified apathetics were employees, only 20% were the public). The fledgling Summary would have 14% never missing it, 53% of the respondents out on their own looking for the information (also equally split between employees and the public), and 27% wondering what was going on but not caring enough to follow up (a daunting research task for non-site workers).

Respondents were asked if they had ever sent in a DOE Idaho Operations Office comment form before. For the full Plan, 39% had not, 61% had; for the Summary, 64% had never sent in a comment, 36% had. This survey clearly brought in many fresh voices for the DOE to hear. And these new commentors wrote telling sentiments such as, "I had never been asked to comment until now." and "I never had a reason to comment before."

In spite of several years of public participation in the development of the document, which resulted in inputs to decisions on the Environmental Restoration and Waste Management Programs, and in the significant mandate evidenced in the survey results, the DOE Idaho Operations Office canceled the Site Specific Plan due to the lack of a DOE Headquarter's requirement and related funding. The public had been asked about the document's importance, but its unequivocal response was seemingly unimportant. However, the DOE Idaho Operations Office decision was in step with its guidance from the DOE Headquarters. By appearing to disregard the input it sought, the process called into question how much value the DOE had really attached to the public's contributions then and in the past.

Having invested in a comment process that yielded a 94% positive reaction, it may have been a more responsible approach for the DOE Idaho Operations Office to have at least responded to the mandate received when it asked for help in making a decision. The DOE Idaho Operations Office does take its public seriously--the fact that it asked for public input on a document that was no longer required or funded attests to that. DOE is trying to find other ways, within its financial constraints, to give stakeholders the information they have asked to receive. Stakeholder input on how to fill the information gap would contribute to the public confidence that the DOE

Idaho Operations Office still wants to share the same level of data and to solicit stakeholder input.

OBSERVATIONS AND SUGGESTIONS

The three case studies all typify DOE's growing commitment to involving the public in its activities at the INEL. While the intent of genuine stakeholder involvement appears to be present in each case study, each also illustrates that there is a missing component in the public participation equation: an awareness of function, or of how to optimize the investment in public participation efforts. The following lessons learned from the case studies offer insights into ways to improve the often uneasy alliance between the technical/managerial and communication disciplines. Although each of the points below has relevance for both communication professionals and managers within the DOE system, subheads highlight the party most likely in a position to authorize or implement, not just influence, the suggestions.

Be Careful Not to Promise More than can be Delivered

Management: The case studies show a disconnect between the Department's ultimate intention of what and how decisions will be made. Although it is impossible to exactly predict program processes and constraints, a reasonable attempt should be made to forecast how public involvement could be used. If there is no real ability or desire by staff or management to use input to cooperatively develop plans and make decisions, then the public should be told the Department is only gathering information on reactions to a proposal being made within rigid boundaries. When unforeseen changes occur that negate or effectively reduce the utility of public input already received or in progress, the Department should immediately explain the evolving situation and let people know why their input is no longer relevant or useful in the context it was solicited. An effort also should be made to capture the general value that was gained from the input or interaction (e.g., include findings in a database on general public attitudes and preferences).

Communicators: Practitioners also need to be honest with management about what can realistically be expected from communication activities. Changing the minds of individuals with strongly held opinions is unlikely. Control of the press is hardly realistic, or advisable. But what communication professionals can do is help level the playing field on which the Department's ideas are presented. Communicators can work to make accurate information easily accessible to the public and the media. Mechanisms can be offered to reduce the public's sense of alienation or exclusion (and the resulting distrust and anger) from the information exchanges and decision processes. The communication practitioner's roles are most productively and realistically those of negotiator, facilitator, and mitigator, not savior, placater, or protector. To behave otherwise is to be an acquiescent partner to a closed internal decision process.

Public Involvement Projects are Conducted in a Fluid Environment

Management: Managers must learn to live with their discomfort related to public involvement. The current trend is towards more public participation, not less. And poorly conducted public participation erodes DOE credibility. Technical managers, while unaccustomed to the highly dynamic, often uncontrollable nature of public involvement, must learn to relinquish control and let communicators do their jobs. As managers give communication practitioners the opportunity to run involvement programs using assumptions (such as "planned unknowns" and rolling schedules) and techniques (such as Communication and Stakeholder Involvement Plans) unfamiliar to those trained in more rigid technical project management, they will gain confidence in the ability of the communication staff to effectively manage their element of the project.

Communicators: A basic tenet of communication planning is to expect the unexpected. The specific tactics will often need to be responsive to the project's changing conditions (e.g., revised regulations or guidance, funding cuts, political shifts, schedule requirements). Practitioners are aided by planning allowances into their overarching strategies. Many communication professionals specialize in strategy (not just tactics) and can contribute significantly if they are allowed to stay closely apprised of the technical and management aspects of the project. They need to be alert that the planned form of the communication element does not dominate the opportunity for the internal decision process to function dynamically.

Management and Communicators Must Invest Time Initially to Fully Understand the Technical Decision Process and the Role that Public Participation will Play Within that Process

Management and Communicators: An investment of time must be made at the outset of any project requiring public participation to understand the full scope of the technical program and the communication process implications for implementing the technical program, as was evidenced in the Site Treatment Plan case study. Time must also be spent on creating a public participation process that will work for the technical basis of the project, before activities begin.
Timing has a tremendous impact on the benefits an involvement program can yield. Going out to stakeholders prematurely or with too short a time line will inherently limit the viability of the interaction. And, it may appear to outsiders as proof that involvement was not really desired since its implementation was handled poorly.
The Department must teach the public about internal decision processes.
Management and Communicators: Decision makers cannot criticize the quality and usefulness of input or involvement if it is not adequately explained what type of contributions will be welcomed by DOE's internal system and deemed to add value when decisions are contemplated. For example, in the EIS case study the public was told how to make a comment from a process and logistical perspective, but was not instructed on how to craft the comment itself. Telling the public "we want to know what you think" can be interpreted as an open invitation for those sweeping referendum-type comments on DOE policy that are generally relegated to the "out of scope" pile. What's the result? DOE complains that the public didn't provide substantive comments specific to the document while the public counters that, once again, DOE didn't listen to what they had to say. DOE owes it to stakeholders to help them understand the prescribed processes and the public's role in it. Tools for collecting consistent data from stakeholders could be considered, such as surveys of value rankings and alternative acceptance scales in addition to open-ended, non-directed comment (the latter of which is the current approach for collecting comment).
It will be to the Department's benefit in the future to explain the larger decision making system in which the Department operates (e.g., cooperating agencies, regulatory bodies, legislative directives, political expediences). If, for instance, a stakeholder wants to challenge a decision that was justified on the basis of DOE's mission or support DOE efforts to avoid funding cuts, that interested citizen needs to know alternative and perhaps more appropriate paths to register concerns or support. There is great fear associated with teaching stakeholders how to "work the system," despite the fact that many of them are more astute at it than the Department is. But not telling stakeholders only builds dissatisfaction and encourages belief that DOE must be hiding something.
The Department Should Reward Stakeholder Involvement Champions Within Decision-maker Circles
Management: Those with the authority to make decisions need also to be people who appreciate and know how to use public input to enhance internal decisions. Managing communication programs should be deemed as important as managing technical projects--because good science alone can no longer provide an adequate justification for the merit of a program. Reward structures should encourage--if not require--evidence of effective inclusion of public issues in program management.
Communicators: Practitioners need to lead this shift by providing the impetus and demonstrate the value of designing stakeholder sensitivities, ideas, and partnerships into the process.
Departmental decisions should be grounded in results from public involvement activities relying on solid social science methods.
Management: Decision makers can benefit from the use of communication objectives and criteria in grappling with decisions. In the case of a NEPA public comment process, for example, being able to state that public comments were analyzed with proven content analysis techniques could have bolstered the legitimacy of decisions. Use of sound social science methods also reduces the traditional rift between the often questioned competence and appropriateness of including communication professionals in key roles within technical projects--a situation which has often caused management to overlook the value of the communication field in arriving at well-rounded decisions.
Communicators: Public participation and associated communication specialties are theoretically grounded disciplines, not unlike the engineering and physical science professions. Too often practitioners neglect to present communication strategies and tactics with the academic and field research data to support the soundness of their

proposals. Like technical verification and validation, communication methodologies are available for the communicator's defense and offense. In the EIS case study for example, comments could have been coded using content analysis techniques from the social sciences. Surveys, such as those used in the Site Specific Plan case study, can be constructed to provide valid, reliable quantitative and qualitative findings. Preoccupation with opposition groups can be put into perspective when simple principles are foundational to strategies--e.g., persuasion efforts focused on either end of an attitude spectrum will produce negligible results.

The Department should make better use of public input across projects and programs. Management: Amongst all the projects with communication components is a cumulative wealth of untapped understanding. Absent a common mechanism for gleaning and compiling separate data sets and lessons learned on communication elements of projects, mistakes are repeated, inefficiencies continue, and individuals, instead of a central repository, retain the hard-won knowledge. It would be beneficial to develop baselines and databases, analogous to technical benchmarks and codified models, that decision makers and practitioners alike could turn to for factual evidence of public attitudes and concerns. Such efforts could provide trend analyses, historical perspectives, and inter-/intra-agency case studies for use in contemplating decision options. Field operations seem no more equipped to capture and share evolving internal knowledge about public interactions than do headquarters functions. Toll-free lines, museums, and science, visitor, and information centers have been established for the purposes of fielding questions, distributing program materials, and educating the public (such as those for the Office of Civilian Radioactive Waste Management, the Office of Environmental Management, and the Yucca Mountain Project). The DOE should consider a similar resource for internal consolidation and dissemination of communication "technical" data--if not because it will aid project management, provide consistency within the Department, and increase professional knowledge, then at least for the tremendous cost savings it could afford.

Public input on the design of the decision process should be as critical as public input on public involvement process design.

Management: The Department has made laudable advances in working together with stakeholders to develop functional mechanisms for soliciting better input into decisions. The same path should be extended into unveiling the "black box" into which the input is dumped. The perspective an outsider brings to redesign of internal systems could yield innovative, fresh approaches with the potential to satisfy more parties and conduct negotiations in parallel rather than serially. And it would go far to educate stakeholders on the facets and constraints operating in the internal system.

DOE as a federal agency has yet to truly engage stakeholders at the front end of any contemplated action so that they can help frame the process and, therefore, have some ownership for its success. If the public understands the process and believes it is credible, they may be more accepting of its outcome even if they don't particularly agree with the decision itself.

CONCLUSION

It is the authors' supposition that the Department needs to shift its emphasis on the input variable of the stakeholder involvement process and focus instead on the internal process that creates the decision outputs. Neither greater involvement from a broader array of perspectives nor increased interactive, personalized exchange will significantly advance the credibility and soundness of DOE's decisions. No level of improvement in input can substitute for the ultimate misuse or disuse of those inputs. The case studies suggest a particular set of potential options for improving the output side of the equation, as described above. These represent a small portion of the opportunities DOE has for strengthening and balancing stakeholder relations by reshaping its paradigm and looking inward for solutions.

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COMMUNICATING WITH THE PUBLIC ON NUCLEAR MATERIALS DISPOSITION ISSUES

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ABSTRACT

The storage and disposition of weapons grade fissile materials involves complex technical issues. The goal of reaching consensus on the best method or methods for storing and ultimately disposing of these materials can only be achieved after an open public discussion of all reasonable alternatives. The Department of Energy considered a number of factors in developing a program for public involvement. These factors included defining the affected public, public knowledge about the proposed action, Government decisions to be made, availability of pertinent information, current public perceptions, and responding to public input.

Based upon an analysis of these factors, three primary techniques were selected for communicating with the public. The techniques are distribution of a quarterly newsletter, establishment of an electronic bulletin board, and holding periodic conference calls and public meetings. The newsletter is designed to reach the widest segment of the public and to be meaningful regardless of the reader's level of familiarity with the project. Periodic conference calls and meetings are used to reach the public that have a real but periodic interest in the issues. The electronic bulletin board is used to communicate with a core group of stakeholders who have a continuing high interest in the total storage and disposition program.

Initial indications are that these techniques are effective in reaching a wide segment of the public on highly technical and complex questions related to weapons grade nuclear materials. The effectiveness is not the result of these being new techniques but rather using each technique, to the maximum extent possible, to provide open, on-going, two way communications.

INTRODUCTION

The storage and disposition of weapons grade fissile materials, primarily plutonium and highly enriched uranium, involves complex technical issues. There are aspects of every storage and disposition alternative that are at the frontiers of our knowledge. Therefore, the challenge for the Department of Energy is to effectively communicate important environmental impact information on the best method or methods for storing and ultimately disposing of these materials.

In January 1993, the Secretary of Energy created a project to focus on the control and disposition of these materials. Congress then authorized the creation of a Office within the Department with the responsibility for all activities relating to management, storage and disposition of fissile materials from weapons and weapons systems that are excess to the national security needs of the United States. Congress also authorized funds for the accomplishment of these responsibilities. These actions resulted in the creation of the Office of Fissile Materials Disposition.

One of the major activities being undertaken by the new office is to prepare a Programmatic Environmental Impact Statement on the Storage and Disposition of Weapons-Usable Fissile Materials as required by the National Environmental Policy Act. This act requires a certain level of public participation in the preparation of the document. However, the Secretary of Energy, recognizing the considerable benefits from full public participation, has issued policies that go well beyond the minimums of the law. The Secretary's policy defines public participation as an open, ongoing, two-way communication, both formal and informal, between the Department and those interested in or affected by its actions.

This paper presents the methods being used by the Office of Fissile Materials Disposition to communicate important information about potential environmental impacts and other technical issues in an open, ongoing and two way manner.

COMMUNICATIONS PLANNING

The first step taken by the Office was to obtain answers to several questions about the public with whom communication was needed. Then, based on those answers, a public participation plan, was developed. The questions asked were:

Who are the individuals and groups (stakeholders) that are interested in or could be affected by the proposed action?

What decisions are contemplated that would benefit from stakeholder input?

What is the stakeholder's level of knowledge about the proposed action? What additional information does the stakeholder need? What additional information do they think they need? What is the technical complexity of this information? What are the classification restraints, if any, on this information?

What is the current stakeholder perception about the proposed action? What are their primary areas of concern?

How will the Office use stakeholder input and show its response to that input?

The individuals and groups affected by the proposed action could, in the broadest sense, include every citizen of the United States because the proposed action could affect national security. In a narrower sense, the stakeholders can be divided into national and local levels of interest. At the national level, examples of stakeholders are arms control advocates, environmentalists, unions, and members of Congress. At the local level, examples of stakeholders are environmentalists, state officials, city mayors, chambers of commerce, contractors to the Department, employees, and union locals. Based upon this answer, communications have been established and maintained with a very wide spectrum of the public across the United States. The answer to this question was also used to determine the location of the PEIS scoping meetings. The meetings were held near the potentially affected sites and three regional locations.

The decisions to be made by the Department of Energy on the storage and disposition of fissile materials must be communicated to the public in a clear, concise and effective manner. Furthermore, the process for reaching these decisions must be clearly defined and allow for public participation. The decisions center around which storage and disposition alternatives will be selected for detailed technical, cost and environmental impact analysis plus which of these alternatives will finally be selected for implementation. The technical merits and costs of the alternatives, as well as their relative ability to meet the goals of United States nonproliferation policy, are being evaluated in other studies. All of these studies will be incorporated into the decision making process.

The Public Participation Program is the process by which the Department also factors diverse public values and perceptions into the decision process. For example, as part of the review of the Draft Programmatic Environmental Impact Statement, the public will have the opportunity to let the Office of Fissile Materials Disposition know whether they prefer leaving the material in its present location or whether to transport it to a new consolidated storage facility. Obviously, there are technical, economic and environmental aspects in deciding between these alternatives. But, there are also value judgements relating to whether the public would rather accept the risks associated with moving the material or those associated with leaving it in place.

The evaluation the public's existing knowledge about nuclear weapons showed that the level of knowledge varied widely among the stakeholders. On one end of the spectrum, are people who have little or no knowledge about nuclear issues and how the proposed action might affect them or the environment. This group of stakeholders is composed primarily of members of the general public. At the other end of the spectrum, are people very knowledgeable about nuclear weapons, nuclear reactors, and the proposed action. This group of stakeholders is composed of external special interest groups (arms control advocates; environmentalists) and internal special interest groups (unions, contractors, employees). Therefore, all communications need to be intelligible to the general public while not insulting the sensibilities of nuclear professionals.

The information the public needed or wanted was discovered to include some very basic information about nuclear radiation and weapons, the proposed action, how the proposed action could affect other environmental actions, public participation opportunities, and the Department's decision making process. To meet the public need for basic nuclear and process information, a series of short, one to three page, fact sheets were prepared. Each fact sheet covered one specific topic such a radiation, plutonium, uranium, and the National Environmental Policy Act. Every attempt was made to have each fact sheet easy to read by the use of high school level language and the use of simple graphics. Further, the content of each fact sheet was kept as generic as possible so that it could be used though out the life of the program.

The most technically complex information needed by the public relates to plutonium disposition. The National Academy of Sciences, at the request of the U.S. National Security Advisor, conducted a detailed study on the disposition of plutonium. The study was requested by Brent Scowcroft, the National Security Advisor to President Bush, and sponsored by the Department of Energy. The National Academy of Sciences, as a private, non-profit society of distinguished scholars, produced a report that

provided information that was helpful to both the general public as well as the nuclear professionals. In fact, the disposition recommendations made in the report were used by the Office of Fissile Materials Disposition as the starting point to develop options which would be studied in the Programmatic Environmental Impact Statement. The use of these options helped to establish the fact that the Department had not already decided on a course of action. The proposed action was, therefore, not a DAD situation, decide-announce-defend.

In order to answer the question about the public's perceptions on the storage and disposition of fissile materials, a public meeting was held before formally announcing the proposed action in the United States Federal Register. This meeting was conducted in an interactive format using small discussion groups and facilitators. The meeting format, agenda, and facilitators were used to encourage each person to actively participate. During the discussion groups, the public provided the Office of Fissile Materials Disposition information on the areas of concern to them. For example, there were differing views regarding the timing of disposition actions. Some participants wanted the Office of Fissile Materials Disposition to move as quickly as possible to preclude the potential threat of proliferation and to put the materials into a form that would prevent it from being reintroduced into the nuclear stockpile. Others wanted the Office of Fissile Materials Disposition to move slowly to ensure that potential environmental health and safety concerns would be thoroughly examined before any disposition decisions were implemented.

The discussions also focused on the methods that would be most useful in establishing open, on going and two-way communications with the public. The techniques finally selected to aid public participation in the Programmatic Environmental Impact Statement process were greatly influenced by the discussions during this meeting and comments submitted in writing as part of the participant's meeting evaluation.

The policy answer to the question about how the Office would respond to public input was easy and straight forward. All input would be documented, seriously considered and a response provided to the submitter. The process of implementing this policy was much more difficult because it involved all of the Office staff, Department personnel outside the Office, and contractors. It was decided to integrate this policy one way or another into each communication technique.

OPEN COMMUNICATIONS

One of the primary considerations in open communications was to overcome public perceptions about the Department's desire to keep information classified, the accuracy of its information, the completeness of the information and the currency of the information. This perception was significantly reduced by the Secretary of Energy's "Openness Initiative". This project resulted in thousands of documents being declassified and made available to the public. Additional classified information that would be useful to the public was identified and requests have been made to remove the classification so that the information could be included in program fact sheets, and presentations.

Open communications also require information to be accurate and honest and thus credible. Therefore, the initial information to be provided on the program was prepared and reviewed with meticulous care. It was important that the Office of Fissile Materials Dispositions's motives not be impugned by inaccurate information from the very start. Initial inaccurate or incomplete information would have made it very difficult for the public to accept the information put out at a later date. Early credibility will help the program recover from mistakes or inaccuracies that will inevitably creep into documents at one time or another.

ONGOING COMMUNICATIONS

Planning considerations lead to the realization that the public would have varying degrees of interest in the proposed actions and that this level of interest would vary during the process of preparing the Programmatic Environmental Impact Statement. Therefore, the Office's communication techniques were devised to recognize the public's level of interest and allow them to change their level of involvement in the decision making process. This approach can be explaining by using an "onion" analogy.

The outer layer of the onion consists of members of the public who are only slightly aware of or interested in this program. To establish an ongoing communication with them, a quarterly newsletter is being published and public meetings are being held

on topics of concern to the Office of Fissile Materials Disposition and the public. The newsletter is designed to provide a wide section of the public with a program overview, information on the direction, successes, failures and barriers to the program. The newsletter also provides information on opportunities for increased involvement in the program such as announcements of upcoming meetings. Newsletters can be expensive to prepare and distribute. In order to keep costs to a minimum, the newsletters have few photographs or graphics, and printed on recycled paper. Publication of the newsletter requires the development and maintenance of a mailing list. Keeping a mailing list up to date is not an insignificant cost. Our present mailing list contains over 2, 300 stakeholders. A mailing, at the present postal rate, costs over $2,000.

Another challenge to making a newsletter an effective tool is to attract and maintain reader interest on issues related to the subject of fissile materials storage and disposition. Sustaining public interest is difficult because the subject matter is technically complex, there are no dynamic actions or processes that would lend themselves to interesting visual images, and the time period for the completion of the proposed action may be decades. However, an attractive, periodic newsletter can be an effective mechanism to maintain ongoing communication with the general public. It is a comfortable, familiar format that does not require a heavy investment of an individual's time, but provides sufficient information for a reader to choose their level of involvement.

Public meetings should be designed to reach the widest cross section of the public possible. They are held to present information and discuss a topic of special concern to the public and the Department. Based upon the results of the initial meeting mentioned above, it was determined that a large segment of stakeholders support interactive format meetings. This format gave the participants a chance to ask questions and get immediate answers. It also gave the Office the chance to determine why certain issues were of concern. Interactive workshops were also used to obtain public comment about the scope of the proposed action. The public comments were summarized in a report that was made available to the public by various means including providing copies to DOE reading rooms. The Office of Fissile Material Disposition's responses to the comments have been provided to the public in a separate report. A critical element of each public meeting is the use of feedback forms and questionnaires. For example, the use of questionnaires was helpful in obtaining specific public comments on the screening criteria to be used in eliminating unreasonable storage and disposition alternatives. Approximately 15 percent of the public attending the scoping meetings completed the screening criteria questionnaires. The information obtained was analyzed and used to improve the screening process.

Public meetings are expensive and require a great deal of logistical support to make them effective. However, they are worth the expenditure of significant time and money because they are such an extremely effective means of ongoing communication. One note of caution about public meetings. Their use should be limited to those times when important information is to be discussed. Interactive public meetings require a significant commitment of time by all the participants. This commitment must be acknowledged. Meetings are more productive when substantive discussions are held, based on new or developing program information, and when it is clear how the results of the meeting will contribute to the decision making process.

The middle layer of the "onion" includes those members of the public who have a continuing interest and want to be quickly informed about program changes that affect them. To communicate with this group periodic conference calls are made with specific interest groups. The largest groups in this layer are state officials, local officials and members of the business communities near DOE sites. As stated before, these calls, for the most part, are established with different special interest groups. This has two effects. First, it allows the agenda of the call to be tailored to the needs and interests of that group and second, it keeps the number of participants to a manageable number. As a matter of policy, no one is excluded from joining us the call. For a variety of reasons, including economics, the length of each call is kept to between 30 and 45 minutes. A constraint of this technique is the inability or reluctance of some groups or individuals to spend their resources on long distance telephone charges.

In the inner layer of the "onion" are those members of the public who have an intense, almost daily interest in the program. This group of stakeholders is

primarily composed of people who are full time representatives of environmental, arms control, nuclear industry or other special interest groups. To communicate with them, an electronic bulletin board system has been established. This system allows the Office of Fissile Materials Disposition to maintain a dialogue with a core group of stakeholders on a daily basis. They can access the bulletin board in one of two ways. They can use a modem to dial into a toll free number, or through one of several methods on the Internet. The bulletin board contains five sections, four of which provide a different types of information. These sections are Announcements, Press Releases, Program Documentation, and Registration Forms. The Program Documentation section includes the fact sheets discussed above, briefing charts from public meetings, and technical reports. The fifth section is a public forum in which topics related to storage and disposition of fissile materials are debated. Users of the bulletin board can provide comments on information provided by the Office of Fissile Materials Disposition or comment on the comments made by other bulletin board users. The most significant down-side to this technique is the fact that not all members of the public have access to a computer and thus the technique does not reach everyone in this layer of the "onion".

TWO-WAY COMMUNICATIONS

Incorporated into each of the techniques listed above are very specific mechanisms for the public to provide the Office of Fissile Materials Disposition information about both the content of the program and the public participation process. Feedback forms are used to ensure two way during public meetings. Forms are prepared for all meetings. In order to maximize the use of these forms by the public, the meeting leaders and facilitators request the public participants to complete the form. The Department officials and the facilitators are also requested to stress the importance that is attached to public feedback. All completed forms are read, analyzed and appropriate changes made where possible. These changes are then communicated through each of the outreach techniques including the newsletter, the electronic bulletin board, and conference calls. The public has expressed an interest in feedback on meeting results. Therefore, attendance lists and meeting summary reports are always made available. The attendance lists provided to the public are restricted to names and organizations so that individuals do not become unwelcome targets for mailing lists of other organizations.

The method of obtaining two way communications in the newsletter is to include an explicit requests that readers write to the Office of Fissile Materials Disposition about anything in the newsletter. This section of the newsletter also requests that the reader inform the Office of Fissile Materials Disposition about changes of address, plus additions and deletions to the mailing list. Additional two-way communication mechanisms in the newsletter include "Letters to the Director" and guest articles by stakeholders.

By it's very nature, a conference call is a two way communication. It provides the Office with immediate feedback on new information from interested and active members of the public. This means of communication, more than the other two techniques allows the Office to ask "why" a comment or concern is being made. The answer to this question many times gives the opportunity to improve a response to that comment or concern. The primary two way communication mechanism in the electronic bulletin board is the Forum. The Registration Forms Section of the board can be used to request documents not available on the bulletin board because of their size or age. Users can also use the Registration Forms Section to provide comments on the program independent of the forum topic. As an indication of its commitment to two-way communication, the Office has pledged to provide an answer to questions on the bulletin board within five working days of the question being posted.

The critical part of two way communication is for the Office to react to the feedback and recommendations from the public. This may mean making immediate changes to information, and changing procedures. It also, and very importantly, means the Office must tell the public when the suggested changes are not being made and why.

CURRENT ASSESSMENT OF THESE TECHNIQUES

The Office of Fissile Materials has found that the techniques discussed in this paper have been effective in providing information to the public and obtaining feedback on that information. The communications program was instrumental in obtaining good comments and attendance at its PEIS scoping meetings. During the 12 meetings, over 1,000 individuals provided over 3,000 comments. The feedback forms

indicated that 4 out of 5 stakeholders who completed forms felt that the Department Officials had been very open to public comment The same ratio of stakeholders liked the interactive scoping meeting format instead of the traditional hearing format. The Office's communications program was also helpful in having stakeholders, for the most part, recognize the fact that plutonium has a greater potential for harming the environment and potentially affecting the health and safety of the public than highly enriched uranium. The stakeholders, again for the most part, have come to recognize that the disposition of plutonium will take longer, cost more and be technically more complex than the disposition of highly enriched uranium.
Management support for ongoing, two-way communication with the public has been a factor contributing the effectiveness of the techniques. The obvious willingness of key senior managers to engage in dialogue, even when the discussions were critical of the Department, has influenced the participants in two ways. First, more mid-level managers and technical staff have participated in the public involvement activities. As a result, the interactive and diverse communication methods listed in this paper have been successful in increasing the access of stakeholders to the technical staff and decision makers within the Department of Energy. These interactions appear to have both increased the Department's understanding of stakeholders' concerns and have increased the public's awareness and understanding of the technical, funding, and administrative constraints associated with program implementation.
However, the above techniques have not been effective in reaching the "general" public; that is the people who could be potentially affected by the proposed action but are not part of an organized special interest group. These techniques have also not been very effective in reaching and sustaining the involvement of state and local officials to the degree felt necessary.
SUMMARY
The use of newsletters, public meetings, conference calls and electronic bulletin boards systems are and will continue to give the Department of Energy ways of having an open, on-going, two way communication with the public on the storage and disposition of fissile materials. Further, we believe that communication with the public on the disposition of fissile materials, while technically complex, can be done effectively by providing information:
  on a continuing basis,
  by a number of communication techniques,
  in short, accurate, easy to read documents, and
  with easy to use feedback mechanisms.

Session 21 -- Federal Facility Compliance Act Year-2: Implementation, Status and Issues
Co-chairs: Lawrence Harmon, MACTEC;
Joseph Coleman, USDOE
21-1
DEPARTMENT OF ENERGY STRATEGY FOR MANAGING MIXED WASTE UNDER THE FEDERAL FACILITY COMPLIANCE ACT
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ABSTRACT
The Federal Facility Compliance Act (FFCAct) was enacted on October 6, 1992, amending the Solid Waste Disposal Act, which was previously amended by the Resource Conservation and Recovery Act (RCRA). The FFCAct set in place requirements for the Department of Energy (DOE) and other Federal agencies to bring their mixed (hazardous and radioactive) waste into full compliance with the requirements of RCRA and be treated in a manner similar to non-governmental operations. For DOE, this means its mixed waste treatment operations will be regulated by either the state in which the site is located or an Environmental Protection Agency (EPA) regional office. Over the past two years, the DOE, states, and EPA have developed a unique process for implementing the FFCAct.
The key feature of this process is DOE and its regulators working in an interactive, iterative fashion to meet the requirements of the FFCAct at both the local and national levels. The process is a challenging one involving 49 DOE sites and the requirement to negotiate compliance orders with the 22 affected states or EPA within

three years. The 49 DOE sites that are involved in this process are shown in Fig. 1. In order to be successful in complying with the FFCAct, the DOE needed to ensure the process met certain requirements. First, to meet the aggressive schedule required by the FFCAct, the process needed to be established quickly and needed to provide mechanisms for DOE Headquarters and its field offices to communicate and plan for meeting the FFCAct requirements. Second, the process needed to provide an opportunity for national coordination among the sites to blend the individual site treatment plans into a sensible national configuration of treatment systems to avoid redundant efforts, to provide comparable results, and to coordinate required funding and implementation schedules on a national level. And thirdly, the regulators who will approve the plans needed to be intimately involved throughout the process and the public, Indian tribes, and others needed to be provided with opportunities to participate within the process.

The DOE's interaction with the associated regulators to jointly discuss, analyze, and plan DOE's implementation is a unique aspect of the process. To accomplish this, the DOE established a cooperative agreement with the National Governors' Association (NGA) that provides for the NGA to act as coordinator and facilitator for the states, EPA, and Indian tribes throughout the development of the plans. This interaction has been advantageous to establish DOE policies and planning that are in tune with the regulators' interests. One early result of this communication was that DOE agreed to establish a "bottom-up" process, i.e., that the sites prepare their plans individually, rather than DOE Headquarters preparing a national plan that the sites must follow.

The process that emerged to meet the challenges of complying with the requirements of the FFCAct involved coordination at a number of levels, both within and external to the DOE.

INTRODUCTION

Aside from the technical challenges contained in the FFCAct, the DOE was faced with the challenge of coordinating a massive national planning effort during a time of significant change. Historically, the DOE mission of development and production of nuclear weapons in providing national security required that most operations be conducted in complete secrecy. To facilitate this high level of secrecy, the overall mission of the DOE sites were coordinated from Headquarters, but the details of the operations were conducted with a high-level autonomy at the sites. Very little information was transferred between the DOE sites with even less communication to the local stakeholders and regulators. The recent changes in the world order and the end of the cold war have resulted in a sea change to the DOE mission which have created a need to change the very culture of the institution itself. The DOE is no longer in the business of actively building nuclear weapons and supporting plutonium reprocessing operations have been shut down. Previously classified information is being released to the public and DOE is actively involved in one of the largest and most complex waste management and environmental remediation programs in the world. The program's goal is to properly manage waste produced as a result of past operations and clean up its facilities for future use. Attempts to change the culture of DOE from one of total secrecy needed to protect the national security to a more open way of doing business, with full participation of the public, stakeholders, and even within the Department has been steady, but uneven. One noteworthy success is the process that has emerged through efforts to comply with the requirements of the FFCAct. This process has laid the groundwork for a much higher degree of external communications and participation in planning a DOE program than ever before, and has resulted in the development of a cooperative working relationship with the state and EPA regulators and other stakeholders.

REQUIREMENTS OF THE FEDERAL FACILITY COMPLIANCE ACT

The FFCAct provided that federal agencies be treated as any other entity and waived the federal government's defense of sovereign immunity thereby allowing states to impose fines and penalties against federal agencies for RCRA violations. The DOE is not in compliance with the RCRA Land Disposal Restrictions (LDR) because it does not have sufficient capacity to treat its mixed radioactive and hazardous waste and is storing this waste in violation of the storage prohibition. Because of the complexity and enormity of the task of bringing almost 1,000 DOE mixed waste streams into compliance, Congress provided, in the FFCAct legislation, a three year postponement of the waiver of the defense of sovereign immunity for DOE for violations of the LDR storage prohibition if certain conditions are met. One of

these conditions is that DOE be in compliance with approved plans for developing treatment capacity and technologies for mixed waste generated or stored at the DOE sites. The FFCAct provides authorized states in which DOE sites generate and store mixed waste, or the EPA, authority to approve DOE's plans and issue orders requiring DOE to comply with those plans.

Additionally, the FFCAct required DOE to develop and provide to EPA and the affected states a comprehensive inventory of mixed waste that is generated, stored and projected to be generated by DOE over the five years following the enactment of the FFCAct. The DOE met this requirement through the publication of the Interim Mixed Waste Inventory Report in April of 1993. The DOE has since updated the original inventory and produced a new report in disk form in April of 1994 and anticipates a second update to be available in late 1995.

The most demanding requirement of the FFCAct is the development of plans to provide treatment capacity for each site at which DOE generates or stores mixed waste. The DOE utilized the Mixed Waste Inventory Report as a baseline for the development of these plans. These data have been continually updated and refined throughout the process. The DOE has adopted a "bottom-up," three-phased approach for the development of its Proposed Site Treatment Plans (Proposed Plans). The bottom-up approach requires the plans to be developed by each DOE site in consultation with other DOE sites. In the first phase, the Conceptual Site Treatment Plans were issued to the sites' regulatory agencies (State or EPA) in October of 1993. These Conceptual Plans identified for each site the broad range of options available to treat DOE's mixed waste. In the second phase, a narrow range of site preferred options were developed in coordination with the states and EPA. The Draft Site Treatment Plans were submitted in August of 1994. The third phase of the development of plans will culminate in the submittal of Proposed Plans by early April of this year.

As noted above, the Draft Site Treatment Plans presented each individual site's proposed treatment options for their mixed waste. As a result of this "bottom-up" approach, the states, EPA, and DOE recognized that the mixed waste treatment configuration that would result from implementation of the Draft Site Treatment Plans included inefficiencies and duplication in the number and type of treatment alternatives proposed. To help eliminate these redundancies and provide a bridge between the Draft and Proposed Site Treatment Plans (the second and third phases), DOE in concert with the states, EPA, and the NGA, established an Options Analysis Team (OAT). The Team analyzed accommodations necessary to blend the Draft Site Treatment Plan configuration into a sensible national configuration of treatment systems to achieve obvious efficiencies and economies over building multiple facilities at different locations to treat similar wastes. Because the strategies for the treatment of DOE's high-level waste and mixed-transuranic waste are already established, the focus of the Team's efforts was on mixed low-level waste. The states have been kept informed of these activities through the participation of NGA representatives in Team meetings. The Proposed Plans will represent a blended configuration: bounded on one side by the states' wishes for on-site treatment; and bounded on the other side by DOE's and the states recognition of the necessity to reduce inefficiencies. The process that is being used to develop these plans is shown on Fig. 2.

There is still much work to be done. The DOE will continue to work with its regulators, Indian Tribal governments, and other stakeholders in the coming months to finalize these Proposed Plans. The states or EPA, as appropriate, are on schedule for issuing compliance orders based on these plans by October 1995.

GROUPS ESTABLISHED BY DOE TO FACILITATE COMPLIANCE WITH THE FFCAct

To facilitate the communications that are required for DOE and the states/EPA to meet the schedules and milestones in the FFCAct, formal groups were established. Members of these groups communicate regularly through conference calls, informal discussions, and periodic workshops. The groups that have been formed by the DOE and the States/EPA are shown on Fig. 3. They include the following:

The FFCAct Task Force is a Headquarters team that includes members from various Waste Management offices who are assigned full-time to FFCAct-related activities. The mission of the Task Force is to lead, facilitate, and monitor FFCAct related efforts at the DOE Headquarters and Operation Offices, and to manage the coordination of documents for review and concurrence within DOE Headquarters. The Task Force also acts as the interface between the Operations Offices and DOE

Headquarters Offices, and others, such as the NGA, the Office of Management and Budget, the General Accounting Office, and the Congress.
The FFCAct Policy Coordination Group (PCG) is a management level Headquarters-Field team, chaired by the Task Force Director, established to provide overall coordination between DOE Headquarters and the Operations Offices, to facilitate the development of FFCAct guidance and strategies, and to resolve FFCAct implementation issues. The PCG consists of representatives from key participating DOE organizations: 1) the Headquarters FFCAct Task Force staff; 2) field Operations Office and facility staff who focus on FFCAct-related activities; and 3) other DOE Headquarters staff (in addition to the FFCAct Task Force) who focus on specific FFCAct-related issues, including representatives of Environmental Management's Office of Environmental Restoration and Office of Technology Development, the Office of Environment, Safety and Health, the Office of General Counsel, and the Office of the Chief Financial Officer. Regular communication is ensured through weekly conference calls; in addition, the PCG meets every few months among themselves and three or four times per year with the full NGA FFCAct Task Force.
The Site Treatment Plan Workgroup is a working level group established as a forum to discuss the development of each version of the Site Treatment Plans and other deliverables required by the FFCAct. This group consists of technical representatives from the sites who are responsible for the preparation of the STPs. Topics of discussion often overlap between the STP work group and the PCG; the STP workgroup discusses technical details of the Plans and FFCAct deliverables and, as needed, raises issues to the PCG for resolution.
Smaller focus groups were established to work on specific technical and policy issues identified in the process of developing the Site Treatment Plans. Each focus group developed its own procedures and processes to meet their needs in addressing their particular issue. Following are descriptions of three key groups that have been formed.
The Compliance Order Group was formed to develop guidance on compliance order administrative provisions that will implement the Site Treatment Plans, to develop sample language for large and small DOE sites, and to coordinate resolution of order-related issues. Each site is developing a negotiation strategy and draft language based on guidance provided by this group and will begin negotiations on the orders soon after submitting the Proposed Plans. In addition to Waste Management, the group's membership includes representatives from Environmental Management's Office of Environmental Activities and Office of Environmental Restoration, the Office of Environmental Compliance, the Office of Naval Reactors, the Office of General Counsel, and field representatives, usually field counsel, from the DOE Operations Offices.
The Options Analysis Team was formed to review the treatment options proposed by the sites in the Draft Plans and to blend those options into a sensible national configuration. The Team consists of technical representatives from the larger DOE sites and DOE Headquarters. The Team conducted their analysis based on the types of treatment required for the waste streams and has analyzed various treatment system configurations for cost and schedule impacts and other technical factors. They have analyzed a broad range of site treatment system configurations, including: a centralized configuration; a configuration based on the shortest possible treatment schedule; a configuration that maximizes the use of existing facilities; and others, to facilitate the development of a sensible national configuration of site waste treatment systems.
The Disposal Working Group was established to review potential sites for mixed waste disposal. Although the FFCAct does not specifically address disposal, both the DOE and the states recognize that disposal is an integral part of mixed waste management. This focus group developed a process, in consultation with the states/EPA and NGA, for evaluating options for the disposal of residuals resulting from the treatment of mixed low-level waste. The evaluation process includes only mixed low-level waste streams. DOE is relying on established processes implemented for studying, designing, constructing, and ultimately operating disposal facilities for high-level waste and mixed transuranic waste (e.g., geologic high-level waste repository and Waste Isolation Pilot Plant, respectively). The sites considered in this evaluation are the 49 sites that are currently generating or storing, or planning to generate or store mixed low-level waste in the next five years. Work has progressed steadily on the evaluation process. However, because of the complexity of

the issues and the work left to be done, proposed alternatives or final destinations for disposal of treatment residuals will not be known at the time the Proposed Plans are submitted to the states and EPA by April 1995. The interim results of this evaluation process are intended to be considered during the discussions of the Proposed Plans, both between the DOE and states and among states themselves, and to identify next steps. This process has been a vehicle for DOE and the states to discuss and to begin to resolve this important issue.

STATE TASK FORCE

The states proposed to DOE that the NGA coordinate the communications of the states that would be involved in the process of approving the Site Treatment Plans. A cooperative agreement was developed between the DOE and the NGA that established a three-year project with a total funding of approximately $2,600,000 and a first year budget of $833,000. NGA was tasked with the responsibility of coordinating the states and facilitating the exchange of information and technical, policy, and implementation issue discussions on mixed waste treatment and disposal. In addition to meeting jointly with the DOE, the arrangement also provides for the NGA and the states, EPA, and Indian Tribal government representatives to meet independently of the DOE to discuss these matters and to resolve equity issues. The NGA coordinated the formation of a State Task Force consisting of both technical and policy representative from the 22 states involved in the process. The State Task Force has established working groups to focus on public participation, technical, and legal issues. Communication between the DOE Task Force and the NGA is ensured through weekly conference calls. Meetings have been held quarterly with all of the state representatives and the PCG to keep all parties abreast of the latest developments and to resolve issues. The Options Analysis Team has provided regular briefings to the State Task Force at the quarterly meetings.

The NGA and the state representatives have been requested by the DOE to participate in some of the DOE focus group activities to ensure that their viewpoints are considered. The NGA has acted as the point-of-contact to review and provide comments on various documents and guidance for development of the Plans, and reports that are developed. This interaction with the state representatives has been advantageous to DOE in establishing policies and conducting planning that is in tune with the states' interests. This interaction has been advantageous to the states in that they have a better understanding of the Site Treatment Plan development process prior to DOE's submittals.

COORDINATION WITH OTHER ONGOING DOE ACTIVITIES

Programmatic Environmental Impact Statement Development

The Environmental Management Programmatic Environmental Impact Statement provides an analysis of the impacts of alternative strategies for conducting DOE's waste management program, including hazardous waste, high-level waste, transuranic waste, low-level waste, and mixed waste. The analysis of the mixed waste strategies is being coordinated with the discussions on, and preparation of, the Site Treatment Plans. As such, the analysis provides information on the impacts of a variety of mixed waste treatment system configurations.

Mixed Waste Technology Development Focus Area

DOE's Office of Environmental Management formulated a new approach to focus its environmental research and technology development activities on key program priorities. This approach is based on teaming the users and developers of technologies to identify, develop, and implement needed technologies. Mixed waste characterization, treatment and disposal is one of five key focus areas that have been established. The Mixed Waste Focus Area has provided key technical support to the development of the Plans in several ways: 1) reviewing, collating, and analyzing DOE complex-wide mixed waste technology needs; 2) providing direct technical assistance and recommendations to the sites to assist in the evaluation of treatment options; 3) reviewing Plans to ensure a consistent technical approach to the analysis of waste stream treatment requirements and treatment system selection throughout the DOE complex; 4) keeping the sites informed of emerging technologies being developed; and 5) aiding the Options Analysis Team by providing technical assistance and reviewing the analyses of treatment options for opportunities to optimize the treatment configuration through the application of innovative technologies. This coordination between the FFCAct process and the Mixed Waste Focus Area has provided valuable technical input to the development of the Plans and provided the Mixed Waste Focus Area hands-on understanding and knowledge of the

technology development needs, priorities, and issues facing the Waste Management program.

SUCCESSES

Since enactment of the FFCAct, the DOE has met every deadline under very aggressive schedules. Issues have been raised through early discussions with the state regulators, and changes to guidance documents and the Site Treatment Plan development process have occurred through these discussions. While there are many issues yet to be resolved, the process is on track to meet the October 1995 compliance date.

As this is an innovative process, there were concerns from everyone involved about whether it was achieving its intended goal. At the initial stages of the process, two years ago, the states requested that DOE allow the sites to develop the Site Treatment Plans themselves, with minimal guidance from DOE Headquarters. The DOE has complied with that request. Upon release of the Draft Plans, the regulators complimented DOE on the product, the process, and its adherence to the requested approach. The resulting trust and confidence in this process has allowed the process to move forward quickly and confidently.

The program is six months away from achieving compliance with the FFCAct. The Proposed Plans will be submitted to the states and EPA by early April 1995. The Proposed Plans will provide the framework for implementing the mixed waste management program for many years. Once negotiations are completed and compliance orders are in place, the DOE and states and EPA will have achieved the initial goal of the FFCAct--the creation of a blueprint for achieving compliance with RCRA. However, much remains to be done. Further discussions with Indian Tribal governments, the public, and others are needed. Moreover, DOE must seek authorization and appropriations from Congress in order to implement the compliance orders. The funding for FFCAct compliance will be a top priority for the DOE.

The success of this program is in the innovative approach that was taken to involve all affected parties throughout the planning process. The DOE has made great strides in changing its culture from one of total secrecy to protect national security, to a more open way of doing business with full participation of the public and stakeholders. The Site Treatment Plan development process is evidence of DOE's success in changing its culture.

21-4

FFCACT TECHNOLOGY SELECTION PROCESSES

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ABSTRACT

The Site Treatment Plans (STPs) required by the Federal Facility Compliance Act (FFCAct) for all sites that generate or store mixed waste will propose the use of existing facilities with existing technologies as well as new facilities. The construction of new facilities for waste management, as well as the use of existing facilities, implies the use of a technology selection process. This paper will present an overview of the impacts of technology selection, the process and some tools, and opportunities for the implementation of emerging technologies in the waste management activity.

INTRODUCTION

Technology selection for waste treatment has an impact on many aspects of the process and politics of waste management within the Department of Energy (DOE). The selection of a particular suite of technologies as the waste management system for a facility brings into consideration, in each case, an array of factors such as cost, risk, and treatment effectiveness and others. These factors associated with a technology or treatment system are influential in a number of areas. These interaction .s are shown in Table I. There are policy issues, both budgetary and statutory, as well as political sensitivities associated with the DOE Complex-wide

treatment configuration which will be influenced by technology selection. Stakeholders and regulators have a number of demands which need to be met in the choice of waste management technologies. The results of such decision support activities as the Programmatic Environmental Impact Statement (PEIS) and the Baseline Environmental Management Report (BEMR) are strongly dependent on technology selection assumptions. The analyses carried out and recommendations carried forward in the DOE FFCAct process by the sites, the Policy Coordination Group, the FFCAct Task Force, and the Options Analysis Team (OAT) are also based on assumptions influenced by technology selection.

TABLE I

The Mixed Waste Focus Area (MWFA) of the Office of Technology Development (OTD) is responsible for overseeing the development and implementation of new mixed waste management technologies for the Office of Environmental Management. The MWFA, in order to promote consistency in the approach to technology selection, has recorded the basics of the selection process and provided tools to assist the sites. In the following discussion the technology selection process, the tools, the Technology Resource Team (TRT), and the Alternative Technology Workshop will be discussed.

DISCUSSION

The three phase process by which DOE and its sites have approached the preparation of the STPs has provided a three stage structure within which technology selection has played an important role. The evolution of the STPs from the Conceptual Site Treatment Plans (CSTPs) through the Draft Site Treatment Plans (DSTPs) to the Proposed Site Treatment Plans (PSTPs) has involved refinements of the three elements of a treatment option. The three elements of a treatment option are location, facility, and technology. As the plans have evolved, the location of future facilities has become more certain, there is more detail concerning the facility requirements, and the range of applicable technologies is better defined.

The primary driver in technology selection are the treatment requirements for each waste. The technology selection process is straightforward and involves a small number of well defined steps. For convenience wastes of like characteristics are grouped according to treatability groups. These groups take into account the waste matrix (usually the majority constituent), the radionuclide and radiological characteristics, and the Resource Conservation and Recovery Act (RCRA) regulated contaminants. Radiological characteristics such as alpha or non-alpha contamination and 'contact handled" versus 'remote handled" are important factors in assessing treatment requirements.

The treatment requirements for the RCRA regulated materials in the waste are determined by the performance requirements set forth by the Land Disposal Restrictions in RCRA. The regulated organic materials need to be destroyed and the metals need to be stabilized. The radionuclides also require stabilization. Once the treatment requirements are determined, baseline treatment technologies are matched to the treatment requirements. The waste matrix is often the dominant factor in choosing the appropriate treatment technology. Incineration, for example, is not the best way to destroy low concentrations of organic materials in an aqueous waste. Once the core treatment technology is chosen, a flowsheet can be constructed which contains the unit operations and technologies that comprise the balance of the waste management system. When the baseline flowsheet is complete, opportunities to employ innovative or emerging technologies should be evaluated and appropriate flowsheets generated for each option. If there are several options for the primary treatment technology and the balance of the flowsheet, those options should be analyzed for their comparative value using the Treatment Selection Guides.

The Treatment Selection Guides (1) were provided by DOE Headquarters as part of the guidance for the preparation of the DSTPs. The seven elements of the Treatment Selection Guides summarize the important components of several technology evaluation criteria sets that have been developed by various DOE sites in recent years. These guides supply a tool that can provide a consistent framework within which technology or treatment system options can be evaluated and compared. Once these guides were defined they were provided to the DOE sites, OTD, and to stakeholders for review and comment. The current collection of guides reflects the comments obtained during this review cycle.

The Regulatory Compliance element gives high scores to technologies that have been previously permitted. Lower scores are given to technologies that require regulatory exemptions or demonstrations of equivalency that may pose additional permitting

difficulties. Environmental Health and Safety gives high marks to processes that decrease or provide no additional risk to the workers, the public, or the environment. The recognition of Stakeholder Concerns is important to the progress of DOE's waste management program. Stakeholders may include the local public, public near the intermediate and final destinations of the waste, state and local governments, Indian tribes, Congress, the Department of Defense, and industry. Technical performance issues are evaluated under the Treatment Effectiveness element. Process properties such as volume reduction, secondary waste generation, DRE (destruction or removal efficiency), flexibility, and final waste form performance are major considerations in this category. The Implementability guide assesses the ease and likelihood of bringing a treatment technology into operation within the proposed schedule and estimated cost. Technologies that can be designed, built, demonstrated, and put into production within specified schedules while exhibiting high levels of maturity, development, availability are given high scores. Life-Cycle Cost includes the total of the direct, indirect, recurring, nonrecurring, and other related costs estimated to be incurred in the design, development, production, operation, maintenance, support, and final disposition of a major system over its anticipated useful life. Evaluation of technology transfer opportunities and the value of the technology to the commercial sector is included under the Technology Development guide.

Another tool, the Technology Catalog, initiated at Los Alamos National Laboratory, has been developed by the MWFA to assist in the technology selection process. There are several versions of the catalog. The Waste Management Automated Technology Catalog (WMATC)(2) is in a portable electronic format supported on a Macintosh PC and operates under the 4D RuntimeTM software. DOS based versions are available either through Internet access to the LANL sewer using the MOSAIC utility or a local version that uses the MOSAIC utility on a DOS based PC.

Fig. 1. is an example of the way in which this catalog is organized. In the catalog's automated version there are five flowsheets of which the Aqueous Liquids Flowsheet is one example. The WMATC's utility is based on its ability to carry out searches for technologies that match specific waste management functions such as front-end handling, pretreatment, treatment, etc. A unique feature of the WMATC is that the technologies in the database have been tied to specific unit operations in top-level flowsheets. These flowsheets are suggestive of the flowsheets developed earlier by the Mixed Waste Treatment Project during the analysis of the Complex-wide mixed low level waste treatment system requirements. Similar flowsheets have also been used in the PEIS analysis.

The graphically oriented search strategy directs the development of flow sheets by providing technology choices from the catalog for five waste matrices - organic liquids, aqueous liquids, process solids, soils, and debris. The choice of a waste matrix such as aqueous liquid from the first menu screen leads to a flow sheet with the unit operations, e.g. solids separation, neutralization, organic removal, metal removal, etc., that are necessary to treat an aqueous liquid waste. Selection of a particular unit operation in a point and click graphical display will provide a selection of technologies relevant to that unit operation. Repeating this sequence for each recommended unit operation from the start to the finish of the flowsheet will provide several technology options for each unit operation. This, in turn, will lead to several differing configurations for a treatment system. These configurations can be evaluated and compared for local and technical efficacy by using tools such as the Treatment Selection Guides.

Figure 2 shows an example of the information available for a specific unit operation selection in the aqueous flowsheet. When organic destruction is chosen for the primary treatment unit operation, four options are displayed, i.e. chemical oxidation, photolytic/radiolytic destruction, wet air oxidation, and biological. In the example given in Fig. 2. photolytic/radiolytic destruction is selected. The WMTAC technology examples that are brought in the search are shown in the figure and range from lasers and electron beams to UV oxidation techniques. The catalog has narrative descriptions which describe more fully any technology that is selected.

In addition to the tools described above the Technology Resource Team (TRT)(3) is an organization that has been supported by the MWFA. The TRT is comprised of number of experienced process engineers and technologists. The membership of the team was structured in a way that mixes technology developers l with technology users. The function of this organization has been to provide technical assistance throughout

the FFCAct process. This assistance has taken the form of technology selection and options analysis workshops at various sites, technical reviews of the CSTPs/DSTPs/PSTPs participation in OAT activities, and evaluations of the opportunities to employ emerging or innovative technologies. One of the primary responsibilities of the TRT is improving information exchange with regard to technology issues. Information concerning technology needs must be discovered and details concerning the performance characteristic of technologies must be transmitted to the users to facilitate technology selection decisions.

The most recent activity completed by the TRT was the Alternative Technologies Workshop. One of the objectives for this workshop was to take advantage of the OAT's 12/95 configuration of waste streams and facilities in a treatment technology analysis. A crosswalk of the Mixed Waste Inventory Report, Phase II waste stream data with the OAT configuration created a database. The database represents the first opportunity to view treatment facility technologies and technology requirements against the framework of detailed radionuclide/contaminant/matrix data. In the case of OAT proposed new facilities, this technology analysis provided an opportunity to identify the proposed facilities where the implementation of emerging or innovative technologies would have the greatest impact. This analysis also provided information concerning areas of the waste management process where there are additional technology development requirements. This information is intended to supply the needs of various customers. The identification of implementation opportunities should help the operations people make technology selection. The needs information will help the MWFA understand what additional technology development activities need to be considered.

Figure 3 is a summary example of a product of the workshop. The Idaho Waste Processing Facility (IWPF) is one of the proposed facilities in the current OAT configuration. The waste streams shown in Fig. 3 are a small selection of the large number of wastes targeted for the IWPF. the core technologies chosen for this facility are thermal treatment (incineration is a prime candidate) and stabilization. These are shown at the middle of the figure. Alternate technologies have been suggested. An important consideration in the ongoing negotiations with stakeholders involved in the FFCAct is the comparison of thermal and non-thermal treatment technologies. It is increasingly important to develop data for non-thermal treatment options in order to be able to defend a thermal treatment choice based on performance data. Suggestions were made by the workshop for non-thermal options including sludge Washing, direct chemical oxidation, and acid digestion. Phosphate-bonded ceramics which are being studied in a program sponsored by OTD are a new material for use in stabilization applications may be applicable to these wastes. Technology needs related to mercury recovery, stabilization, and monitoring were identified. Concerns about processing mercury contaminated wastes are not confined to waste streams directed to IWPF. These processing problems exist at other sites in the Complex such as Savannah River and Oak Ridge.

SUMMARY

We have discussed the technology selection process, tools, and other assistance that is supported by the MWFA. The MWFA has provided support to the technology .selection activities associated with DOE's response to the FFCAct. -As the STPs evolve through negotiations with the states and stakeholders additional technology selection concerns will arise as facilities are planned or modified. There are areas in which the technology selection process can be improved. Information quality and dissemination are at the heart of these needed improvements. There needs to be a more precise definition of the treatment requirements for many of the wastes. Performance data for new technologies needs to be generated and distributed to the designers and operators of facilities where implementation opportunities exist. All of the important parties including regulators, stakeholders, and technology development staff need to be involved in the early stages of the development process. The planning of demonstrations should involve the regulators and stakeholders to insure that data that are important to these groups is generated. The issues associated with technology selection will continue to influence. e the process of planning a Complex-wide treatment capability for at least a decade. The technology selection process will continue to be important to those groups and activities discussed earlier including stakeholders and regulators, decision support activities, the DOE FFCAct process, the policy. making activities, and the MWFA.

REFERENCES

1. FFCA Task Force, Treatment Selection Guides, March 1,1994.
2. Mixed Waste Focus Area Fact Sheet, Waste Management Automated Technology Catalog, March, 1994.
3. Mixed Waste Focus Area Fact Sheet, Technology Resource Team, October, 1994.

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DISPOSAL OF TREATED MIXED WASTE - DOE'S RESPONSE TO STATES' CONCERNS

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ABSTRACT

In conjunction with the development of the Site Treatment Plans (STPs), required under the Federal Facility Compliance Act (FFCAct or Act), DOE and the States affected by the Act are working closely to resolve issues associated with the disposal of treated mixed low level waste (MLLW). This paper describes the process by which DOE has responded to concerns raised by States about disposal of the MLLW treatment residuals. The continuing dialogue between DOE and the States concerning the issues surrounding disposal and how to address them has fostered a better understanding for both parties of the options for managing MLLW treatment residuals. The process for addressing disposal of MLLW treatment residuals will continue to evolve and progress as such dialogue and exchange of ideas and values continues among DOE, the States, and other stakeholders and the public.

BACKGROUND

Prior to the enactment of the Federal Facilities Compliance Act (FFCAct or Act), the DOE planning baseline for mixed low-level waste (MLLW) disposal capacity included the development of disposal facilities at the six DOE sites currently disposing of low-level waste. These sites are: Hanford Site, Savannah River Site, Oak Ridge Reservation, Idaho National Engineering Laboratory, Nevada Test Site, and Los Alamos National Laboratory. However, this complex-wide planning effort was re-directed to align with the principles set forth in the FFCAct.

The FFCAct requires the Department of Energy (DOE) to develop a plan for the treatment and storage of mixed waste for each site which currently stores or expects to generate mixed waste, called a Site Treatment Plan (STP). The law does not impose any similar requirement for the disposal of mixed wastes; however, early discussions with the States that host DOE MLLW facilities identified the need to address disposal of MLLW along with treatment addressed in the STP. Recognizing the need to address this final phase of mixed waste management to make an informed decision on treatment options, DOE agreed to work with the National Governors' Association (NGA) on issues associated with the disposal of MLLW being considered under the FFCAct. In support of this goal, DOE established a Disposal Working Group (DWG) in June 1993 to work in consultation with the States.

Rather than describing in detail the technical aspects of the evaluation process that the DOE Disposal Workgroup and States are developing, this paper focuses on the concerns and values that lead to the development of the evaluation process and the way in which DOE is addressing and incorporating these concerns into the process. (For a detailed discussion of the technical aspects of the DOE process for evaluating disposal sites, see DOE'S Planning Process for Mixed Low-Level Waste Disposal, J.T. Case, et al, and DOE's Performance Evaluation Process for Mixed Low-Level Waste Disposal, R.D. Waters, et al).

FINAL DISPOSITION OF WASTE A PRECONDITION TO STATES' AGREEMENT WITH TREATMENT PLANS

During the initial DOE-State discussions concerning compliance with the FFCAct in December, 1992, a number of States indicated that final disposition of all treated mixed waste streams would be a precondition of their agreement with any plans for the treatment of mixed waste by October, 1995. A primary concern of the States with large DOE sites having existing waste management facilities, including low-level waste disposal facilities, was that such sites and facilities would bear a greater burden due to disposal activities than those associated with treatment only. These States felt that such impacts must be considered in order to ensure that each State is doing their fair share to support the DOE waste treatment plans. While the FFCAct does not require DOE to address the disposal of mixed wastes, DOE was sensitive to

the "equity" concerns of the States regarding final disposition of treated waste. However, given the lack of existing MLLW disposal facilities and the April, 1995, schedule for submitting the Site Treatment Plans, DOE felt that completion of MLLW disposal commitments by October, 1995, was not possible without relying on existing LLW disposal facilities. Both the States and DOE agreed that an alternative approach would be preferable.

DOE responded by proposing to develop a planning process, in consultation with the States, to evaluate potential disposal sites. The DOE Disposal Work Group (DWG) and disposal evaluation process were proposed to utilize a "bottom up" approach to resolve issues associated with MLLW disposal and emphasize the incorporation of the States' values concerning disposal into the process for evaluating potential disposal sites. As such, the disposal evaluation process is an inherently iterative process that relies on continued dialogue between DOE and the States concerning issues, values, and approaches for conducting the analyses.

DOE AND THE STATES ARTICULATED THEIR BASIC VALUES FOR THE EVALUATION PROCESS

The DOE Disposal Work Group initiated the development of the disposal site evaluation process by proposing that in order for the process to be useful to all parties, the goals of the process must be reasonable and attainable. The first of these goals that DOE articulated was that final disposition could not be known by October, 1995, and therefore the evaluation process should not be structured as a site selection process. Rather, DOE proposed that the goals of the process would be to bring to light information about each of the sites being evaluated that would help the States understand the extent to which each site may have a potential to conduct disposal activities and what options may exist for ultimate disposal of a site's treated MLLW.

A second simplifying assumption that DOE proposed was that the scope of the evaluation be limited to the universe of sites subject to the FFCAct. These 49 sites in 22 States are the only DOE sites that currently store or are expected to generate mixed waste, and therefore, provide a natural universe for analysis regarding potential MLLW disposal. While other opportunities could be possible, such as pursuing disposal at non-DOE military reservations being closed under the Base Re-Alignment and Closure Act, it was felt that such alternatives were inconsistent with the goals and timeframe to which the States and DOE were working. Other principles that the DOE proposed to be incorporated into the evaluation process included:

All MLLW disposal analyses would be worked cooperatively with the States where the facilities being analyzed are located;

Discussions and working sessions would be held directly with involved States and with a consortium of States through the National Governors' Association;

Disposal alternatives considered would ensure protection of public health, worker health and safety, and the environment;

Equity among stakeholders would be sought; and

All information and decisions would be documented and made available to the public and other interested stakeholders.

The States also articulated a set of basic principles that the disposal site evaluation process needed to be founded on. First and foremost, in order to ensure the broadest possible analysis of options, the States requested that a rebuttable presumption be established that every site would be considered for disposing of their own waste on site and that existing and historical inter-site transfers of waste for disposal would not be assumed. Second, the States suggested that any reduction in the number of sites being evaluated for disposal activities would only be based on good sound technical reasons and initially the only sites that might be excluded from further evaluation would be those for which radioactive waste disposal siting criteria could not be met. Finally, the States suggested that DOE use all applicable State and local regulations and federal statutes applicable to commercial waste disposal activities, as appropriate, during each stage of the analysis.

DOE AND STATES AGREED TO REMOVE 23 SITES FROM FURTHER EVALUATION

With the basic values of the disposal site evaluation process agreed to, DOE and the States proceeded to establish the first level of analysis. DOE proposed that while the existing universe of sites being evaluated was the 49 sites subject to the Act, a number of those sites may not be able to meet basic siting criteria. At a joint meeting on March 3-4, 1994, the States proposed that DOE could conduct a screening of 49 sites to eliminate sites from further consideration that could not meet siting

criteria. The States also proposed that the only criteria that should be used at this step of the analysis would be technical criteria related to the actual performance of a site, e.g., hydrology, physical conditions, and not criteria that were merely regulatory constructs and subject to change.

DOE responded by reviewing the pertinent State and Federal statutes and regulations regarding the siting of waste treatment, storage, and disposal facilities that were proposed by the States and identified three criteria that met the States' terms. These criteria were that a MLLW disposal facility:

Must not be located within a 100-year floodplain;

Must not be located within 200 feet of an active seismic fault; and

Must have adequate clear land to support a 100 meter buffer zone around the disposal facility.

The States and DOE agreed that these three criteria would be the basis of the first step of the analysis and used as exclusionary criteria for screening sites. Utilizing these criteria, 23 sites were eliminated from consideration, leaving 26 sites remaining for further evaluation. At a joint meeting on March 30-31, 1994, DOE and the States agreed to remove those sites not meeting the screening criteria from further evaluation.

DOE AND STATES AGREED TO EVALUATE REMAINING 26 SITES BASED ON SITE-SPECIFIC FACTORS

At the joint meeting on March 30-31, 1994, DOE and the States also agreed to collect additional information on the remaining 26 sites and identify additional strengths and weaknesses of each site. The States proposed that such site-specific analysis focus on a number of factors related to each sites' ability to perform disposal activities.

DOE responded by collecting the requested information for each of the remaining 26 sites and organizing the information in easily comparable tables and fact sheets for the States to review. The criteria that the States proposed were grouped in three main classifications; Technical Considerations, Potential Receptor Considerations, and Practical Considerations. Each of the remaining 26 sites were evaluated against these consideration groupings which included criteria such as: soil stability and topography; precipitation and evapotranspiration; population; proximity to sensitive environments; land acquisition; government presence at the site; and regulatory constraints.

DOE and the States met again on July 26-27, 1994, to discuss the site-specific data on the remaining 26 sites, and to identify sites suitable for further evaluation. Sites with marginal or no potential for disposal, based on the above criteria, were recommended to be removed or postponed from further evaluation. As a result of the meeting, DOE and the States agreed to eliminate five sites from further evaluation due to their limited potential for disposal, and agreed to merge the evaluation of two other sites due to their close geographic proximity.

While not eliminated from further evaluation, DOE and the States also agreed to lower the evaluation priority of an additional four sites. Issues such as the technical capabilities of the site, the volume of mixed waste which may be generated by the sites, and the acceptability of off-site waste made disposal at these sites particularly infeasible or undesirable. DOE and the States agreed to evaluate these sites in terms of their capability to dispose of their own mixed waste on-site if no other off-site disposal options could be identified. However, these sites were not to be considered further for disposal of wastes from other sites.

As a result of this meeting, DOE and the States agreed to carry forward 16 sites for further evaluation for the potential development of MLLW disposal activities.

DOE AND THE STATES AGREED TO EVALUATE PERFORMANCE OF REMAINING 16 SITES

At the joint meeting on July 26-27, 1994, the States also requested that DOE provide even more detailed information about the actual disposal capabilities of each of the 16 sites remaining under the evaluation process. DOE responded by proposing that the remaining sites undergo a performance evaluation to assess each site's disposal potential by estimating the concentration limits of waste that may be disposed at a given site. Based on the methodology and tools used to construct radiological performance assessments, DOE constructed a generic model which could be adapted to each site's particular conditions and used to estimate radionuclide specific limits at each site.

DOE presented the performance evaluation methodology and approach to the States in August, 1994, and after addressing comments and concerns from the States, began collecting site-specific data to perform the evaluations.

DOE AND THE STATES STILL NEED TO DEVELOP NEXT STEPS IN EVALUATION PROCESS
The results of the performance evaluations, expected to be completed in May, 1995, will be reported to the States for review, comment, and discussion. Most importantly, while the States and DOE have discussed general approaches for the next steps of the disposal site evaluation process, many details remain to be developed and agreed upon. In general, the next steps will include estimating waste volumes and expected radionuclide concentrations in treated residuals, comparing these estimates to the site-specific radionuclide limits from the performance evaluations, and analyzing the trade-offs and impacts associated with alternative MLLW disposal configurations. However, the issues to be addressed and values to be incorporated in these steps of the evaluation process will require a level of dialogue among DOE, the States, and the public similar to that which occurred throughout the process to date.
CONCLUSION: THE "BOTTOM-UP" ITERATIVE DIALOGUE BETWEEN DOE AND THE STATES IS WORKING
While the FFCAct does not require that DOE address the ultimate disposal of the treated mixed waste in its plans, DOE recognized the need to work with the States to address the disposal issue. DOE engaged the affected States early in the development of the disposal site evaluation process and committed to an iterative dialogue based on eliciting the values most important to both DOE and the States in addressing the disposal issue. While it is not assured that this process will result in the establishment of waste disposal strategy for treated MLLW, both DOE and the States recognize that the disposal site evaluation process has to a large extent been successful so far. DOE believes that the successes of this process can be attributed to:

- the establishment of reasonable and attainable goals;
- Identification and incorporation into the analysis of the values most important to both DOE and the States;
- An open and flexible process based on continual and frequent dialogue; and
- Use of appropriate and relevant levels of analysis.

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DEVELOPING MIXED WASTE TREATMENT OPTIONS AT THE SAVANNAH RIVER SITE - A LARGE SITE'S OVERVIEW
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ABSTRACT
By April 6, 1995, the U.S. Department of Energy (DOE) sites that generate and store DOE mixed waste will submit Proposed Site Treatment Plans to their states in compliance with the Federal Facility Compliance Act (FFCAct).
This paper describes the process that the Savannah River Site, a large DOE site, used to develop its Site Treatment Plan (STP). This paper also describes Savannah River's ongoing efforts to ensure that the regulator and stakeholders are fully informed and involved in the development of the STP.
INTRODUCTION
The DOE is required by Section 3021(b) of the Resource Conservation and Recovery Act (RCRA), as amended by the FFCAct, to prepare plans describing the development of treatment capacities and technologies for treating mixed waste. The Act requires STPs to be developed for each site where DOE generates or stores mixed waste. Mixed waste is defined by the FFCAct as waste containing both a hazardous waste subject to RCRA and a source, special nuclear, or byproduct material subject to the Atomic Energy Act of 1954. Upon submission of a plan to the appropriate regulatory agency, the FFCAct requires the recipient agency to solicit and consider public comments and approve, approve with modification, or disapprove the plan within six months. The agency is to consult with the U.S. Environmental Protection Agency and any state in which a facility affected by the plan is located. Upon approval of a plan, the agency shall issue an Order requiring compliance with the approved plan.
The STPs are being developed in three phases: 1) a Conceptual Site Treatment Plan (CSTP), 2) a Draft Site Treatment Plan (DSTP), and 3) a Proposed Site Treatment Plan (PSTP). The Savannah River CSTP, issued in October 1993, listed mixed waste streams

generated at Savannah River Site (SRS), and described various treatment options, including onsite treatment, offsite treatment at other DOE facilities, and vendor treatment either onsite or at the vendor's site.
For most waste streams discussed in the CSTP, the DSTP, issued in August 1994, selected a preferred treatment option. Waste streams proposed for shipment to SRS from other DOE facilities were addressed as well. The DSTP identified specific treatment facilities for treating a number of mixed wastes and, where feasible, proposed schedules for treatment or treatment technology development as required by the FFCAct. Treatment implementation schedules not identified in the DSTP will be proposed in the PSTP. If a schedule cannot be identified in the PSTP, a date for schedule development will be specified.
The PSTP will be issued by April 6, 1995. The PSTP will reflect the complex-wide needs and plans of the DOE Mixed Waste Management program. During the implementation of the PSTP, DOE will continue to evaluate and develop technologies that may offer advantages in the areas of risk, performance, and life cycle cost. These technologies will be incorporated into the STP process, as appropriate. Due to the complexity of defining a single preferred option, there may be more than one option given for some waste streams. If a waste stream is not sufficiently characterized to select a preferred option, the PSTP will propose a schedule for characterizing the waste and developing a treatment plan. In addition, if a waste stream does not have a developed technology for treatment, the PSTP will propose a strategy and a schedule for research and development of treatment technology. After the State of South Carolina accepts the SRS PSTP, it will become an approved STP and the basis for the Order.

DEVELOPMENT OF THE SITE TREATMENT PLAN

There were four basic steps in the development of the SRS STP. They included: 1) determination of the type and volume of mixed waste requiring treatment including current and five year projected volume, 2) identification of feasible options for treating the mixed wastes, 3) initial screening of feasible options to eliminate those with low technology risk and select the viable options to be further analyzed, and 4) in-depth options analysis to evaluate and score each option and selection of the best treatment option. Each of these is discussed further below.

Determination of Type and Volume of Mixed Wastes

As required by the FFCAct, SRS prepared a report providing an inventory of all mixed wastes in storage and to be generated in the next five years. This report provided a description of the type of waste, volume in storage, estimated volume expected to be generated in the next five years, the hazardous waste codes for each type of mixed waste and the RCRA Land Disposal Restrictions (LDR) treatment technology or technologies specified for the hazardous waste component of each type of mixed waste. This data was collected from the existing waste inventory records and from the SRS mixed waste generators. To facilitate analysis of the data, wastes with similar compositions were categorized into waste streams, and each stream was then assigned a unique identifying number. Annual updates to this report (referred to as the Mixed Waste Inventory Report, or MWIR) were completed by SRS to support the STP process. The updates facilitated inclusion of additional detail as each stream was analyzed by both the MWIR updates and the STP investigations. Waste stream experts evolved from this process and worked with developers of the STP to examine and select feasible, viable and preferred treatment options as discussed below.

Identification of Feasible Options

After identification of the mixed waste streams, each stream was evaluated to determine potential treatment capabilities.
A group of technical personnel familiar with the existing and planned site treatment facilities, along with personnel familiar with RCRA specified treatments, evaluated each waste stream to determine what type of treatment was required by RCRA and if an existing, modified existing, or planned onsite facility could perform that treatment. Additionally, technical personnel familiar with vendor treatment capabilities, and personnel familiar with RCRA specified treatments, evaluated each waste stream to determine if there was a vendor that could treat the waste either onsite or offsite. Lastly, the waste streams were evaluated against other known treatments available within the DOE complex to determine if the waste stream could be treated by another DOE site.
The potential treatment capabilities developed during this process were called feasible options.

Initial Screening

The first step in determining which feasible options merited further evaluation was to assess the technology risk of each of the options. The methodology used was based upon Risk Management Concepts and Guidance written by the Analytical Sciences Corporation for the Defense Systems Management College. The "risk" determined is not whether the system or process is safe but the "risk" that a process may be too new and too complex to perform as required. Some questions that were used to determine the technology risk indicator were:

Are state-of-the-art advances in technology being used in the design?
Is the equipment exposed to a harsh or unique environment?
Does the design require complex integration of control systems, or computer software?
Is the cost of recovering from system failure high?
Is the design evolving as construction is going on?
Is the design new, or an extension of a successful design?
Are standard components being used in new, non-standard ways?
Does the facility stand alone, or must it interface with other facilities or processes?

No process or facility can be simpler than its most complex part, or more mature than its newest part. Thus the technology risk assessment began with an examination of the whole process or facility to identify the part that has the most complex and the least mature technology. While the interaction of numerous parts and features may result in an overall process that is more complex and novel than its individual pieces, the identification of the crucial part is the first step in assessing the probability of a process or system failure.

Next, a magnitude was assigned to the maturity of the part, the Maturity Factor (Pm), and to the complexity of the part, the Complexity Factor (Pc). The magnitudes were assigned based on the guidance in Table I. When engineering assessment indicated the factor fell between the extremes noted, other magnitudes were assigned.

Next, a magnitude was assigned to the Consequence of Failure (Cf). Such consequences range from minor inconveniences from which recovery is quick and inexpensive, to technical catastrophes from which recovery, if possible at all, is prolonged and costly. Table II provides the guidance for assigning the magnitude.

A Cf equal to 0.7 was chosen for all assessments of the technology risk of the waste treatment options, because technical failure of a preferred treatment option was postulated to result in a costly and time-consuming redesign to develop another process to meet requirements. Until the redesign was complete and implemented, waste treatment performance would be significantly degraded.

The Maturity and Complexity Factors were then combined with the consequence factor to give the Risk Factor (RF):

First, the Maturity and Complexity Factors were averaged:

$$(Pm + Pc)/2 = Pf$$

Then, the RF was calculated:

$$RF = (Pf + Cf) - (Pf \times Cf)$$

The RF was a number between 0.19 and 0.99.

If Pf = 0.1 and Cf = 0.1, then RF = (0.1 + 0.1) - (0.1 x 0.1) = 0.19

If Pf = 0.9 and Cf = 0.9, then RF = (0.9 + 0.9) - (0.9 x 0.9) = 0.99

As can be seen from the above, the closer the RF is to 0.99 the greater the technology risk.

In the next step to screen and evaluate waste treatment options, numbers ranging from 0 to 100 were assigned to treatment option attributes, with high numbers representing more desirable features. To make technology risk assessment scores work the same way (high numbers indicating a low technology risk), the RF was converted arithmetically to a number between 0 and 100, called the Technology Success Factor (TSF). A TSF score near 100 indicates a high degree of simplicity and maturity for a treatment option.

In the initial screening of treatment options, those with TSF scores under 50 were discarded. At this time, such technologies remain unproven and would not be recommended as the preferred option in the STP. When these technologies mature, they will be reassessed for application to the site's waste treatment program.

In-Depth Options Analysis (IDOA)

After the elimination of treatment options with a low possibility for technological

success, most waste streams still had several viable treatment options. To determine the best option, all viable treatment options were subjected to an IDOA. Comparison among treatment options for a given waste stream is facilitated when each option can be assigned a number that reflects the degree to which the option satisfies a set of criteria or requirements. The method of developing a numerical ranking of treatment options is known as the IDOA model.
The IDOA process took several steps: 1) determination of the attributes by which all treatment processes would be analyzed, 2) determination of the relative importance of the attributes, 3) application of the IDOA model to each viable treatment option, and 4) determination of the Preferred Option to treat a given waste stream.
The categories and attributes analyzed were:
Process Parameters (total weight factor of 22%)
volume alteration (5%)
secondary waste generation (4%)
destruction, removal, and demobilization efficiency (2%)
flexibility (3%)
ability to be shipped (2%)
final waste form (6%)
Engineering Parameters (total weight factor of 19%)
system implementability (13%)
availability (3%)
scalability (1%)
remedial measures (1%)
schedule for treatment of waste (1%)
Personnel Parameters (total weight factor of 20%)
consequences of unmitigated accident scenarios (6%)
non-operational worker potential exposure (6%)
operational worker potential exposure (6%)
transportation potential exposure (2%)
Regulatory Parameters (total weight factor of 14%)
need for a variance (4%)
ability to obtain a permit (6%)
waste disposal (4%)
Public Acceptance (total weight factor of 9%)
public acceptance (9%)
Cost Considerations (total weight factor of 15%)
life cycle cost (14%)
funding availability (1%)
Industry Involvement (total weight factor of 1%)
market for technology (0.5%)
private sector involvement (0.5%)
"Enabling statements," clarifying the above attributes, assisted with the process experts' evaluation of treatment options.
To evaluate a viable treatment option, a team of waste treatment process experts applied the enabling statements to each option. The team assigned a number from 0 (low) to 100 (high) to each attribute. The score reflected the experts' assessment of how well the process satisfied the requirement posed by the attribute as described in the enabling statement.
For example, consider the attribute of "Secondary Waste Generation." If the process produced a small quantity, all of which could be handled by existing technologies, the enabling statement specified a "high" numerical rating (median 80). If the process produced as much as 10% additional waste that existing technologies could handle, the process was rated "medium" (median 50). If the process produced large amounts of secondary waste, or if existing technologies could not handle the secondary waste, the process rated "low" (median 20). If the experts felt a score other than the median better reflected conditions, they could assign another number, provided they gave an explanation for the variation (e.g., in the preceding case, if the process produced 20% additional secondary waste, the evaluation would include a statement such as "subtract 10 points because of additional waste generation").
For the cost attribute, a team of cost estimators determined the life cycle cost. The estimators developed:
  pre-operating cost to design and prepare initial documentation (including permits) for the facility

facility cost to build and equip a new treatment facility, or modify an existing one
operating and maintenance cost for the life of the facility
disposal cost of all final waste forms in compliance with the RCRA LDR
decontamination and decommissioning cost to return the facility to a safe and environmentally benign condition at the end of its useful life.

The process experts' evaluation resulted in a raw technical score for each attribute, and inclusion of the cost estimators' life cycle cost data resulted in a raw total score. Nevertheless, these raw scores did not reflect the relative importance of the attributes. Not applying a weighting factor to each attribute assigned the same weight to all. Therefore, each attribute received a weighting factor. The weighting factors were then reviewed and modified by independent reviewers, regulators, and a citizens' focus group. The final weight factors appear beside the attributes previously described.

Each option's weighted technical scores were summed. The total fell between 0 (least preferable) and 100 (most preferable). The sums enabled the treatment option to be ranked according to the technical weighted score. Then, the weighted life cycle cost data were added to the technical weighted score in a way that ensured that the cost of a treatment facility was equitably apportioned among the waste streams that would be processed using that facility. This resulted in a total weighted score. The IDOA model generated the technical and total weighted scores for each treatment option. These IDOA model scores were useful tools to narrow the entire population of options. The IDOA model:

ensured the same attributes were analyzed for every process or facility,
provided some guidance to help make analyses consistent among the facilities,
enhanced the engineering assessment by incorporating consistent structure and logic.

Application of the IDOA model ensured consistency and completeness in performing the in-depth analysis of the potential treatment options associated with each waste stream. The primary function of the model was to reduce the number of possible treatment options to a more manageable number for further analysis and review. The model was not developed to provide a clear preferred option (PO) winner. The application of the model results in a smaller set of POs that may have model scores within a 10 to 15% range of each other, that serve as the focus of engineering analysis. It was not expected, and in practice was not always the case, that the treatment with the best model score was the treatment option selected.

Sixteen of the waste streams also had treatment options proposed by commercial vendors. Many of these options, however, remain technologically unproven. The vendors have offered to perform studies to demonstrate that their technology can produce a waste form that will meet the LDR. Nonetheless, the technical viability of these technologies was assumed, and hypothetical vendor processes were projected, to allow application of the IDOA model for a comparison of the potential vendor processes with other treatment options.

The last step in the IDOA was to perform an engineering assessment, taking into account the score generated by the IDOA model. While application of the IDOA model analyzed the degree to which the treatment option satisfied the requirements of the prescribed attributes, engineering assessment took a broader perspective and considered other less tangible factors.

REGULATOR/STAKEHOLDER INVOLVEMENT

An aggressive program to inform the regulators and stakeholders in assessment of and selection of treatment options was implemented. Regular meetings were held with key state regulatory agency personnel to keep them informed of progress in developing the STP. Items directly affecting the selection of preferred treatment options such as assumptions, treatment selection methodology, and the IDOA, were reviewed with the agency, and their comments incorporated into the process. The agency was kept informed of the plans for treating each of the SRS mixed waste streams. The agency was also kept informed of the waste that other sites were targeting to send to the SRS for treatment, and the waste that SRS was targeting to send offsite for treatment.

Stakeholders have also been kept informed and their input obtained throughout the development of the STP. When the CSTP was issued, a fact sheet was mailed to stakeholders on the site's public involvement distribution list and a copy of the CSTP placed in the local Public Reading Room. In response to the fact sheet,

citizens volunteered to participate in a focus group to review the assumptions, treatment selection methodology, and In-depth Options Analysis Model. Comments from this group were incorporated into the process and used in the development of the DSTP.
When the DSTP was issued, copies of the DSTP and Executive Summary were placed in the local public Reading Room. The DSTP's availability and public workshops were announced through public service announcements, newspapers, television and radio advertisements and news releases, using the site's media list. SRS again issued a fact sheet summarizing the highlights of the DSTP to stakeholders on the sites public involvement list. Copies of the DSTP were also mailed to stakeholders upon request. Two public workshops were held in the local area to provide information on the treatments and facilities proposed to treat SRS mixed waste and on the wastes from other DOE sites proposed to be treated by SRS facilities. Stakeholders were invited to give comments during the workshop or provide them to SRS later. In addition, an edited videotape of the workshops was carried on cable channels in surrounding communities. During the airing of the videotape, SRS personnel were available via a toll-free number shown on the screen to answer questions about the DSTP and to solicit comments. Stakeholders attending the public workshops or calling the toll-free number during the videotape viewings were also invited to participate in focus group meetings to provide further comment on the DSTP. Although sparsely attended, some valuable input was obtained through these focus group meetings. SRS representatives also offered briefings on the highlights of the DSTP to interested community groups. Two such presentations have been provided. Both oral and written stakeholder comments gathered throughout this process have been incorporated into the development of the PSTP.
Once issued, copies of the SRS PSTP and the Executive Summary will be placed in the local Public Reading Room. The public will be made aware of the Plan's availability through public service announcements, newspapers, television and radio advertisements, and news releases, using the site's media list. A revised fact sheet will be developed and issued to stakeholders. Stakeholders will be informed that comments on the PSTP may be submitted to the state regulator.
SUMMARY
The SRS PSTP will be issued to the State of South Carolina by April 6, 1995. When issued, the PSTP will satisfy DOE's obligation under the FFCAct to develop and submit a treatment plan for SRS. Since the PSTP is intended to provide DOE's plans for achieving compliance with the LDR requirements of Section 3004(j) of RCRA at SRS, it is understood that no further civil enforcement action, administrative or judicial, will be initiated for violations of RCRA Section 3004(j) arising from storage of mixed waste covered by the approved STP for as long as DOE is in compliance with the requirements of the approved STP and its Order. This will include all mixed waste in storage at SRS and identified in the approved STP, as well as future generated mixed waste incorporated into the STP in accordance with the provisions of the STP, and any mixed waste received from offsite which is being accumulated to facilitate the treatment of such waste at SRS which is covered in another site's treatment plan approved by the appropriate regulatory agency after consultation with the State of South Carolina.

21-7
PUBLIC PARTICIPATION IN THE FEDERAL FACILITY COMPLIANCE ACT PROCESS
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ABSTRACT
The Department of Energy is continuing its efforts to open its decision making processes and to work with its state regulators, Tribal representatives, and other interested parties, or stakeholders, to resolve the issue of how to treat mixed hazardous and radioactive waste. The Federal Facility Compliance Act requires DOE to develop Site Treatment Plans that will identify, at each site, how mixed waste will be treated. Throughout the development of the Plans, DOE has worked closely with its regulators and has provided opportunities for interaction with its stakeholders. This paper discusses some of the key concepts being incorporated into the planning for stakeholder involvement and how they are working.

BACKGROUND

The Federal Facility Compliance Act (FFCAct) requires DOE to prepare plans for developing treatment capacity and technologies for any site at which DOE generates or stores mixed radioactive and hazardous waste. The plans are needed because DOE does not currently have adequate capacity for treating its mixed waste, generated by operations over the past 40 years, to standards required by the Resource Conservation and Recovery Act. The Site Treatment Plans must be submitted to the regulating state or the U.S Environmental Protection Agency (EPA) for approval. The regulatory agency will then ensure that DOE complies with the approved plan and schedules through a compliance order

DOE is developing the Site Treatment Plans in three stages: the Conceptual Plans, published in October 1993, which identify preliminary treatment options; the Draft Plans, published in August 1994, which identify the site's preferred options, and the Proposed Site Treatment Plans, due no later than April 6, 1995.

The Federal Facility Compliance Act (FFCAct) has provided an opportunity for DOE to work through an open process to develop a strategic approach to treating mixed hazardous and radioactive waste. This paper will discuss some of the keys concepts being incorporated into the planning for FFCAct stakeholder involvement and how they are working.

SUCCESSFUL INTERACTIONS TAKE TIME TO DEVELOP AND ARE BASED ON BUILDING TRUST

The Act requires DOE and its regulators, both at the State and Environmental Protection Agency, to work together in this planning process. Through this process, DOE and its regulators have developed a more cooperative working relationship and a greater understanding of the issues to be addressed in DOE's mixed waste treatment plans. At very early meetings, DOE and the state and EPA regulators agreed to use the National Governors' Association to facilitate quarterly coordination meetings between DOE and the regulators. This forum provided valuable opportunities for discussing the overall approach to developing the mixed waste treatment plans and related technical issues.

Despite the mutual goal of developing reasonable, feasible, and technically acceptable plans, DOE and the states both needed to invest time to get past the traditional regulator-to-regulatee relationship, and to begin building trust relationships. For example, in the initial meetings, DOE and the regulators were at odds regarding whether to proceed with DOE's proposed national planning effort or with the site specific planning effort proposed by the states. After significant discussion, DOE agreed to proceed with the state preferred site specific, or "bottom-up" approach. Although this approach proved to be difficult, DOE gained credibility with its regulators by agreeing to their preferred approach. When the Draft Site Treatment Plans were released, the sites recognized the inefficiencies and duplication of facilities prevalent in the plans due to use of the bottom-up approach, but supported DOE's use of that approach.

Although as a whole, DOE has focused on its FFCAct interactions on the efforts with its regulators, the DOE sites have been pursuing the philosophy of using an interactive approach with their local publics and other interested parties for some time, and have begun to see the evidence of success in their efforts. The FFCAct has been a part of the ongoing public involvement efforts at each site.

BE RESPONSIVE TO REQUESTS, BUT BE HONEST IN DEFINING LIMITATIONS

Any planning effort involving two or more parties has inherent limitations and requires mutual understanding of the issues and compromises. The keys to building successful partnerships is to listen to and understand the requests of other involved parties, to identify creative solutions, and to clearly define requests that cannot be implemented. Throughout the mixed waste treatment plan development process, DOE has shared information as it was collected and worked with the states to define the next steps in the process, but has stood firm when necessary. In fact, at several points, DOE and the States have "agreed to disagree" on issues that they were unable to resolve. In this way, the involved parties understood the issues and arguments of the other party, but the process did not stop. In most cases, disagreement was only temporary as DOE and the States identified creative approaches to work toward resolving the issues without stopping the entire process.

PUBLIC PARTICIPATION IS A PART OF EVERYONE'S JOB

In order to undertake a planning effort of this magnitude and implement the "bottom-up" approach, DOE was required to work on many facets of the mixed waste issue simultaneously. Data and technical analysis, policy evaluations, and

stakeholder involvement were ongoing at both the individual sites and DOE Headquarters. In addition to the interactions with the regulators, each DOE site was required to provide information and involvement opportunities for its local interested parties, and similar efforts were undertaken at Headquarters to reach national level interests. The consideration of effective communication and stakeholder involvement was present in the public affairs and outreach efforts and in the technical analysis. For example, initial evaluations of appropriate technologies for treating mixed waste included a consideration of stakeholder acceptability as a part of the technical criteria to be included in the evaluation. Also, development of public information was coordinated with the technical evaluation process to ensure accuracy and timeliness.

IT IS IMPORTANT TO RECOGNIZE THAT DIFFERENT LEVELS OF INVOLVEMENT MAY BE NECESSARY

Since the development of FFCAct plans is focused primarily at the individual site level, DOE has used a decentralized approach to public involvement at the site level. To ensure some consistency, each site was directed to provide a minimal level of activities, including notification and availability of the plans for review, informing the local interested parties, and providing direct interaction opportunities as appropriate.

However, DOE has recognized that it is important to provide opportunities for national level interests to be informed and involved in the FFCAct process. To address these national interests, DOE provided an information mailing to approximately 60 environmental, labor, political, and other individuals announcing the release of the Draft Site Treatment Plans. DOE held an open house prior to releasing the draft plans, in order to provide an opportunity for Washington, D.C. based interests to dialogue directly with the responsible DOE staff.

As the process of developing the plans has continued, DOE has broadened its national approach to include a national stakeholder meeting focused on identifying national issues and potential solutions. This forum provided DOE with direct feedback at the national level as the plans came together to define an emerging configuration of treatment systems.

By broadening its perspective, DOE has gained insight into the issues that are important to its stakeholders at both the local and national level. These issues are being considered in moving forward in the FFCAct process.

ACCURATE COMMUNICATION OF INFORMATION TO THE PUBLIC AFFECTS THE DOCUMENT DEVELOPMENT

The issue of treating DOE mixed waste is complex, involving 49 sites in 22 states throughout the United States and many different mixed waste with different characteristics. The analysis of how to treat these wastes involved identifying all of the types of waste, the various hazardous and radioactive characteristics of the waste, and the appropriate waste treatment based on these variables. Trying to effectively communicate the facts, the issues, and the DOE approach to resolving the issues required a focused effort to provide consistent information in various levels of detail. For example, in developing a national summary of all of the Draft Plans, DOE provided a variety of information to suit the interest level of the audience, including a one page information brief, a short fact sheet, a more detailed Executive Summary, and a document containing both a national and a summary for each site. Technical staff preparing these documents paid particular attention to trying to clearly and simply convey the information so that it would be understandable to the general public as well as the regulators.

In another effort to provide information even though it may be in draft form, each DOE site provided a brief fact sheet describing the contents of their Draft Plan and its possible treatment alternatives. These were compiled and made available to the public and Congress even before the plans had been evaluated from a national perspective. Early information of this type satisfied the level of interest at the stage of the draft plans.

Additionally, attempts were made to make information generally available and easy to obtain, with the exception of some very large documents. Documents and information were made available through the toll-free Center for Environmental Management Information (1-800-736-3282), at public reading rooms located near each site, and at an open house held to reach out to Washington, D.C. based interests.

DON'T LET AN INITIAL LACK OF INTEREST STOP YOUR EFFORTS

A crucial component of public involvement is to continue outreach and involvement opportunities throughout the process. In spite of an initially low level of interest in the topic of mixed waste treatment, DOE has continued to pursue an open process

at both the local site level and at the national level. As the FFCAct process has continued and a national configuration of treatment systems is emerging, DOE is discovering that the level of interest is increasing, and is trying to provide additional information and involvement at the sites, while pursuing avenues to provide national interaction opportunities. For example, in early December, DOE hosted a National stakeholder meeting that included members of environmental groups, labor interests, local government, American Indian tribes, and others to discuss national issues involved with the FFCAct and potential ways to resolve these issues. The forum provided DOE with insight to the issues that are important to a variety of interests, and as a result is defining further activities. Technical issues that were raised will be factored into the development of the Proposed Site Treatment Plans and the discussions with the State and EPA regulators. Through ongoing evaluation of its efforts, DOE is adjusting the public involvement opportunities to meet the level of interest and demand for information.

EVALUATION OF THE EFFECTIVENESS OF PUBLIC INVOLVEMENT EFFORTS IS CRITICAL TO SUCCESS

This is particularly true in the case of a national program such as the FFCAct, where the majority of the involvement issues are being addressed at the site level. Preliminary evaluation of the public involvement program revealed that there were groups interested in the national perspective as well as the site perspective. To address these groups, DOE sponsored a National stakeholder meeting in December, 1994. During this meeting, the participants indicated that they did not feel their involvement opportunities had been adequate, and that additional efforts were required during the decision making process between April and October 1995. As a result, DOE is working to increase the level of involvement and to focus on national issues. One of the challenges facing DOE during this time is the need to provide public involvement while initiating a negotiation process with the regulators to define the Consent Orders that will implement the mixed waste treatment plans. Throughout its public involvement activities, DOE has found that it is vital to be open, honest, and timely in its efforts, and to keep trying different approaches. It has been particularly important to involve the public throughout the process, as the state regulators will ultimately decide how the Site Treatment Plans are implemented through Consent Orders in October 1995. The FFCAct process will continue into the future with periodic updates to the Consent Orders, and the public will continue to be involved as the plans are implemented at the site level.

Session 22 -- Management of Institutional and Industrial Radioactive Wastes
Co-chairs: Ivan F. Vovk, IAEA;
Karen Yourish, Radioactive Exchange

22-1

MANAGEMENT OF WASTE IN RUSSIA

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ABSTRACT

A review of contemporary state is presented concerning the management of institutional and industrial radioactive wastes in the central part of Russian Federation. There are 16 specialized enterprises for management of radioactive waste of scientific and industrial centers in Russia. Almost 80% of institutional and industrial radioactive wastes from Russia are collected for treatment and disposal by Moscow Scientific and Industrial Association "Radon", which is responsible for the management of radioactive waste in the central European part of Russian Federation. SIA "Radon" provides collection of radioactive waste at enterprises -users of radioactive materials - and their transportation by means of special vehicles. After the treatment and conditioning at the Sergiev Posad division of SIA "Radon" radioactive waste is disposed into shallow ground repositories. The average volume of the waste to be treated at SIA "Radon" is 3000 m3 per year for solid waste and 350 m3 per year for liquid waste. Total radioactivity of processed waste is up to 4 PBq per year.

COLLECTION AND TRANSPORTATION

Solid and liquid radioactive wastes are collected for transportation at almost 2000

enterprises from Moscow, Moscow Region and other 9 neighboring regions in Central Russia (see Fig. 1). These enterprises use radioactive materials for scientific research, industrial and medical applications.
Transport section of SIA "Radon" deals with transportation of radioactive waste. Special vehicles are used. Solid radioactive waste is transported in vehicles OT-20 provided by transporting container with the volume of 3 m3. Liquid radioactive waste of specific activity up to 370 MBq/l is transported by vehicles OJ-2M in tight stainless steel vessels with the volume of 0,8 m3. Transportation of liquid radioactive waste with specific activity less than 1 MBq/l is performed by vehicles OJ-10 in tight stainless steel vessels with the volume of 3m3.
Transportation of high level solid radioactive waste is realized in the vehicle OT-31 with trailer-container made of stainless steel, designated for safe transportation of waste with total activity up to 150 GBq. For transportation of high level spent radiation sources shielded containers are used. They permit transportation of sealed radiation sources with the total activity up to 81 TBq.

TREATMENT

SIA "Radon" has the following basic methods for treatment of radioactive waste: incineration, compaction and water purification.

Incineration

Beginning 1983 SIA "Radon" uses two-chamber ceramic furnace for incineration of inflammable radioactive waste with specific activity up to 3.7 MBq/kg for b bearing waste. Capacity of furnace is 80 kg/h for solid waste and 65 kg/h for liquid inflammable waste. Temperature in burning zone is in the range between 800 and 1000C. Volume reduction factor of waste constitutes 50-60. Multi- step dry gas purification system, with sintered-metal filters as first main element, is used. Purification factor of the system is (1-5)104. Final product from combustion in the form of radioactive ash is included into cement matrix. Further high frequency inductive melting is provided for the ash conditioning.
Recently a new high temperature treatment method in shaft furnace was developed at SIA "Radon". It uses high efficient plasma burner, which allows slag melting directly in the bottom of the shaft and production of high stable materials for disposal (1).

Compaction

Non-inflammable solid radioactive waste is compacted in horizontal hydraulic press with the pressure up to 4,9 MPa. The waste is compacted in paper bags or inside of metallic casks. Volume reduction factor is from 2 to 10. Compacted waste is sent to the repository for the disposal of solid radioactive waste, where it is cemented.

Water Purification

For water solutions purification the system is used, which includes unit of mechanical purification on the base of cloth made of polycaproamid (SYPRON) and claydite filtering elements, electro-dialysis unit and ion-exchange purification unit with cation filters. Capacity of the system while purification of low level radioactive waste with specific activity 2 - 15 kBq/l and salt content 0,6 - 3 g/l constitutes 2,5 m3/h. Purified water is used for technical needs, related with vehicles, equipment and premises decontamination. The amount of waste arisen from the work of the system, and subjected to further conditioning is equal to 0,8 - 2,5 % of the purified water volume. Mobile module plant on the base of automobile with the capacity of 1m3/h is also used for water solution purification (2).

IMMOBILIZATION

Before disposal the radioactive waste is subjected to additional immobilisation to reduce possibilities of radionuclide migration. SIA "Radon" uses the following basic methods of radioactive waste immobilisation: cementation, bituminization, vitrification, inclusion into metal matrices.

Cementation

Low level liquid radioactive waste with specific activity up to 370 kBq/l and salt content not higher than 130 g/l is treated by cementation. Low level liquid radioactive waste is used for making cement mortar in the process of solid non inflammable waste cementation. Waste cementation is realised either directly in near-surface repositories or in metallic casks. Stationary system as well as mobile unit of high capacity on the base of automobile KRAZ-250 is used for cementation. The last unit is equipped with vacuum ejector-mixer, providing the capacity up to 20 m3 of liquid radioactive waste per hour.

Bituminization

In contrast to cementation bituminization allows to obtain hydrolytic resistant compound without growth of waste volume. Low level liquid radioactive waste bituminization is performed by the industrial system URB-8. Capacity of this system is 450 - 600 l of liquid radioactive waste per hour. Rotary film evaporator with the heat-exchange surface of 8 m2, operating at the temperature 130 - 140C, is the main part of this system. The URB-8 system provides bituminisation of low level liquid radioactive waste with salt content 300 -500 g/l. The bitumen-salt compound obtained is discharged into metal casks. It has from 40 to 50 mass% of salts, while its leaching rate is 10-4 - 10-5 g/sm2day.

Vitrification

Vitrification of intermediate and low level radioactive waste is a new perspective trend, providing the maximum waste volume reduction and maximum stable product obtaining. SIA "Radon" began experimental vitrification of radioactive waste in the early 70-th. Boron-silicate glass was selected as a matrix. Ceramic melter with direct joule heating and capacity up to 30 kg/h for glass mass was used for vitrification early. Capacity for treatment of liquid radioactive waste with salt content 200 g/l constitutes 50 kg/h. Specific glass rate is equal to 40 - 50 kg/m2 h and melting ratio 3,2 - 3,4 kWh/kg. Electrical power supply of the system-is 150 kW, temperature in melter 1250C. Many types of wastes with specific activity up to 37 MBq/l and among them wastes from atomic power stations with reactors WWR and RBMC were treated by vitrifying. Volume reduction factor in the process of vitrification is 4,2 - 4,5. Losses of radioactive Cs from the melter in the process of melting did not exceed 3,5 mass%. Leaching rate for Cs137 from final product is equal to 1,410-5 - 4,410-6 g/sm2day. Total amount of glass produced by ceramic melter constitutes more than 10 tons. This glass was disposed at the experimental site for observation and safety estimation (3).

Recently a new vitrification method was developed at SIA "Radon" (4). It is based on coreless induction melting of glass in cold crucible. Glass composite materials, which contain separately radioactive components in the form of disperse phase in the glass matrix, are considered for immobilisation of glass incompatible chemical components (such as sulphates, hard metal's oxides, etc.)

Inclusion in Metal

Being disposed in underground near-surface repositories, spent radiation sources are included into metallic matrix material. For this purpose mobile module plant assembled on the base of automobile chassis is applied. Lead or lead based alloy is used as matrix material (5).

RESEARCH PROGRAM

Research division of SIA "Radon" includes Applied Research Centre and other research departments. They provide for improving technologic processes of radioactive wastes treatment and environment remediation. Research programme includes many topics. Among them there are following items:

Improvement of technology and apparatus for vitrification of liquid radioactive waste (Tests of coreless induction melter, tests of gas purifying system, experiments with different glasses and glass composite materials, SYNROC and other ceramics, analysis of their structure and distribution of nuclides in matrix).

Development of safe disposal technology of spent radiation sources (Tests of mobile unit for spent radiation sources conditioning by using metal matrices, experiments with different metals as well as polymeric composite materials).

Long term tests of solidified radioactive waste (Tests on open testing area and tests in experimental shallow ground repositories by using real solidified waste).

Determination of economic efficiency of environment protection activities (Ecological and economical assessment of activities of radioactive waste handling enterprises in Russian Federation and development of database for financial information).

Development of solid radioactive waste treatment methods (Improvement of radioactive waste incineration technology, analysis of composition and structure of materials produced by melting ashes from incinerator, improvement of treatment technology of polymeric radioactive waste by high frequency heating, tests of shaft furnace for high temperature treatment of waste by using plasma heaters, development of gas purifying system for shaft furnace).

Development of systems of automated management of operation processes with radioactive waste (Development of database for sealed radiation sources -information system "SOURCE" - and development of an entrance radiation controlling

system for radioactive waste).
Development of treatment methods of liquid radioactive waste (Tests of mobile unit for liquid radioactive waste decontamination, investigations by using membranes and sorbents, radiation cleaning of water).
Ecological and geographical aspects of environmental protection while render harmless radioactive waste (Investigations of radiation burdens comparing with background natural reservation levels).

RADIATION MONITORING

Since 1974 SIA "Radon" carries out a complex programme of radiation monitoring of Moscow. Radio-Ecological Centre as a division of SIA "Radon" manages the activities on investigation and prognosis of radiation situation. It has a wide network, which includes more than 200 controlling surveillance elements. Meanwhile it was discovered a large number of radiation anomalies and lost spent radiation sources. In 1991 a special computer based system was introduced for the control of sealed radiation sources.

ECONOMICAL EVALUATION

Economical assessment of the efficiency of environment protection activities permits worthy decisions for selection of reliable treatment methods. Particularly for intermediate radioactive waste treatment it is important to evaluate possible treatment processes: vitrification, incineration in shaft furnace with ash melting, etc. A new method of economical assessment was developed, which take into account reliability of radionuclides retention by different matrix materials. For example, vitrification of liquid waste from nuclear power plants has an economical advantage, when specific radioactivity of waste is higher than 4 MBq/l.

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22-2

MANAGEMENT AND DISPOSAL OF RADIOACTIVE WASTE FROM OUTSIDE THE NUCLEAR FUEL CYCLE

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ABSTRACT

In the Federal Republic of Germany it is intended to dispose of all radioactive waste in deep geological formations. This includes radioactive waste originating from outside the nuclear fuel cycle. Detailed information on the various compositions of respective primary wastes, waste treatment and conditioning processes, waste forms, packagings and radionuclide inventories per waste package as well as on waste arisings have been compiled. Site-specific safety assessments covering a repository's operational and post-closure phase have been made. The results of these investigations served in particular to define the requirements to be met by the waste intended for disposal. Waste acceptance requirements have been established for the planned Konrad repository project and the Morsleben repository.

INTRODUCTION

At present, about 50 % of the radioactive waste arisings in the Federal Republic of Germany originates from outside the nuclear fuel cycle, e.g. from nuclear research establishments, other research centers, universities, industrial facilities, hospitals, pharmaceutical industry and the German Federal Armed Forces. The use of radioisotopes in these facilities results in a comparatively high volume of immobilized liquids and concentrates as well as of compressible and/or combustible

materials featuring a heterogeneous composition and a broad spectrum of radionuclides.

COMPETENCIES AND RESPONSIBILITIES

In Germany the Bundesministerium fr Umwelt, Naturschutz und Reaktorsicherheit (BMU, Federal Ministry for Environment, Nature Conservation and Nuclear Safety) is the competent authority for all aspects of spent fuel and radioactive waste management, and is the supervisory body for the licensing authorities in the federal states.

The peaceful use of nuclear energy in Germany is regulated by the Atomgesetz/Atomic Energy Law. On September 5, 1976, the Fourth Amendment to the Atomgesetz/Atomic Energy Law was enacted. It provided the legal basis for the disposal of radioactive waste. According to section 9a para. 3 of this law, the Federal Government has to establish installations for the engineered storage and disposal of radioactive waste, i.e. the disposal of radioactive waste is assigned to the Federal Government as a sovereign task. On November 1, 1989, this competency was assigned to the Bundesamt fr Strahlenschutz (BfS, Federal Agency for Radiation Protection). Accordingly, the BfS is responsible for the establishment and operation of those federal installations, acting on behalf of the Federal Government. In this respect, the BfS is supervised by the BMU.

The federal states are the licensing and supervising authorities for all nuclear installations, including the licensing of repositories, but excluding spent fuel interim storage facilities. In performing this task being legally regulated, the federal states act on behalf of the Federal Government. In addition, they have to operate collecting depots for radioactive waste originating especially from isotope application in industry, research and medicine, i.e. originating from small waste generators.

All other waste management procedures and steps, i.e. spent fuel storage, reprocessing, waste conditioning, transportation and waste interim storage are within the responsibility of the waste generators. In particular, the waste generators have to characterize and to specify their waste. They are responsible for the proper fulfillment of all conditions given within the licenses as well as of the requirements to be met when waste packages are shipped and transferred into an interim storage facility or a repository.

RADIOACTIVE WASTE FROM OUTSIDE THE NUCLEAR FUEL CYCLE

An essential prerequisite for the development of waste management and disposal strategies or the planning and construction of repositories is the provision of a realistic data base. Data on the origin, type and expected amount of radioactive waste from outside the nuclear fuel cycle are therefore necessary.

Origin and Type of Radioactive Waste

In the Federal Republic of Germany, radioactive waste from outside the nuclear fuel cycle in particular originates from

1. basic and applied investigations in the Karlsruhe and Jlich nuclear research establishments,
2. other research centers, universities, industrial facilities or medical application of radioisotopes (note: This waste is generally handed over to the collecting depots operated by the federal states), and
3. other waste generators, e.g. the German Federal Armed Forces and the pharmaceutical industry.

The term "radioactive waste" covers a wide range of various materials which are quite different according to their nature and quantity of the radioactivity associated with them. The major bulk of waste arisings comprise, among other things, the following waste types (primary waste):

Liquids, aqueous concentrates, sludges.
Scrap, filters, worn-out equipment, spent radiation sources.
Protective clothing, paper, plastics, medical waste.
Organic material, chemical residues, miscellaneous waste from laboratories.
Insulating material, debris, rubble, contaminated soil.
Indicating instruments and devices, pilot lights, compasses and sights containing luminous paints.

As the different waste types must be adequately separated to allow appropriate pretreatment and conditioning for interim storage and disposal, the development of a waste type catalogue seems to be of great advantage (1). Such a catalogue surveys the radioactive wastes from their arising to their disposal in a repository. It specifies the primary wastes, the pretreatment methods, the resulting intermediate

products, the conditioning methods and the resulting waste packages, thus describing the "flow" of radioactive waste. The waste type catalogue, therefore, forms the basis for the description and tracking of radioactive waste from the primary waste to a waste package suitable for disposal. It helps to provide the necessary transparency in waste management and, in particular, gives guidance to the waste conditioners (2).

To provide the necessary data base for disposal-related planning work comprehensive information on waste packages planned to be disposed of has been compiled by the waste generators and conditioners at the request of BfS. Details on the various compositions of primary waste, waste pretreatment and conditioning processes, waste forms and packagings, and the radionuclide inventories per waste package were listed. In addition, planning data on the expected number of waste packages per year has been gathered. According to this procedure, model data on each waste stream originating from outside the nuclear fuel cycle becomes accessible for radiological evaluation within the scope of a safety assessment.

Conditioning Techniques

Conditioning of radioactive waste includes processing and/or packing of the waste, eventually after a pretreatment or a sorting. Various strategies and techniques are applied. The selection of a conditioning process is dependent upon factors like the requirements for interim storage and disposal, acceptance of the process, and volume of the resulting waste packages. Therefore, it is not surprising that different conditioning techniques for the same type of waste may be applied. Furthermore, the necessity to minimize the volume of the conditioned waste because of in former times lacking repository capacity stimulated the development of new and advanced conditioning techniques.

Primary waste must be collected and pretreated in such a way that it is suitable for the selected conditioning process. Principal pretreatment methods are decontamination, crushing, compression, evaporation/distillation/rectification, decantation/dewatering/
filtration and incineration/pyrolysis.

Especially the incineration is attractive for all types of combustible waste. Solid or liquid waste and also alpha-bearing waste may be incinerated. The large volume reduction and, in particular, the inorganic and inert character of the intermediate product (e.g., ashes or slags) are reasons to recommend the incineration process from a repository-related point of view. Nevertheless, the off-gas treatment and the secondary waste must be taken into account.

The cementation of radioactive waste is the most well-known immobilization process being widely applied. It is used for the solidification of liquids, the embedding of solids as well as the grouting of voids in scrap, rubble or filters. Various cementation techniques are used and the equipment might be mobile or stationary. If necessary, special cement formulas and/or suitable additives are to be used. Reactions with the cement, e.g., gas generation by amphoteric metals in the ashes must be taken into account. Possible chemical reactions between the radioactive waste, the immobilization material and the packaging must be limited to permissible levels.

The high-pressure compaction with 1500 Mg to 2000 Mg compactors is an advantageous development to minimize waste amounts. Solid materials are compacted to a stable pellet. This technique is applied to, e.g., metallic materials, paper, plastic, rubble and even ashes from the incineration of organic radioactive waste. Due to possible gas generation occurring in compacted waste, a segregation before compaction is reasonable, i.e., to separate metallic and wet organic materials. Alternatively the compacted pellets may be dried.

Radioactive waste has to be packed for handling, transportation and storage. The necessary quality of a packaging is dependent on the type of waste and its radionuclide inventory. Sheet steel, reinforced concrete and cast iron are common as packaging material. Cylindrical and box-shaped packagings of different sizes and weights are being used. A standardization of the packagings has successfully been realized in order to harmonize the equipment as well as the repository-related handling and emplacement techniques.

Waste Amounts

On behalf of the Federal Ministry (BMU) the BfS carries out an annual inquiry into the amount of unconditioned and conditioned radioactive wastes generated in Germany. In the following, low and intermediate level wastes originating from outside the

nuclear fuel cycle (i.e., radioactive waste with negligible heat generation) will preferably be considered.
According to the latest inquiry (3), the total amount of unconditioned radioactive remnants and primary wastes was about 27,600 m3 on December 31, 1993. Of this, a portion of 15.5 % was produced by the nuclear research centers (4,271 m3) and 10.6 % by the collectings depots of the federal states, the pharmaceutical industry and the German Federal Armed Forces. In addition, an amount of about 200 m3 of short-lived waste was stored for decay in the collecting depots.
The total volume of conditioned radioactive waste amounted to about 63,900 m3 on December 31, 1993. Of this, waste originating outside the nuclear fuel cycle is compiled in Table I.
Based on the results of the annual waste inquiries from 1984 to 1993, a forecast of future waste arisings was carried out (3). This prognosis predicts a total amount of conditioned waste at the end of the year 2010 to about 238,900 m3. Major contributions are made by spent fuel reprocessing (73,140 m3), nuclear research centers (55,250 m3), operation of nuclear power plants (47,750 m3) as well as decommissioning and dismantling of nuclear facilities (50,000 m3). Further details are compiled in Table II.

Waste Disposal

According to the German disposal concept, all radioactive waste has to be emplaced in a repository constructed and operated in deep geological formations. As liquid and gaseous wastes are excluded from disposal in such a mine, only solid or solidified radioactive waste is accepted. In the Federal Republic of Germany, two sites are presently considered for disposal of low and intermediate level waste:
1. In the abandoned Konrad iron ore mine in Lower Saxony, it is planned to dispose of radioactive waste with negligible heat generation.
2. The emplacement of waste in the former Morsleben salt mine in Saxony-Anhalt which was operated as a repository for short-lived low and intermediate level waste with low alpha emitter concentrations has been resumed.
The planned Konrad repository is assigned to accept radioactive waste with negligible heat generation, i. e. waste packages which do not increase the host rock temperature by more than 3 K on an average. Iron ore, i. e. coral oolite, at a depth of 800 m to 1,300 m is the host rock for this repository. Waste packages will be disposed of in drifts with an excavated volume of about 1,100,000 m3 allowing an emplacement of about 650,000 m3 waste package volume. Operation of the repository is scheduled at least 40 years. A total activity in the order of 1018 Bq and an alpha emitter activity of about 1017 Bq are anticipated in this facility.
In the former German Democratic Republic an abandoned salt mine located near the village of Morsleben was reused for waste emplacement. From 1981 until 1991, radioactive waste with a total emplacement volume of approximately 14,500 m3 and about 6,700 spent sealed radiation sources were disposed of. Of this, the activity of alpha emitters amounts to 1.6 1011 Bq and that of beta/gamma emitters amounts to 4.8 1014 Bq. Subsequent to German unity the Morsleben facility has the status of a federal repository, the continuation of its former license being now limited by law until June 30, 2000. Until then, according to present planning, a radioactive waste volume of 40,000 m3 is envisaged to be disposed of. The estimated maximum activity of alpha emitters amounts to about 1013 Bq, that of beta/gamma emitters to about 1016 Bq.
Due to these planning data and marginal conditions the major bulk of low and intermediate level waste, i.e. radioactive waste with negligible heat generation, is intended to be disposed of in the planned Konrad repository. This is in particular relevant to alpha-bearing waste. The operation of the Morsleben repository will in the first instance contribute to the discharge of interim storage facilities.

WASTE ACCEPTANCE REQUIREMENTS

Pursuant to the Sicherheitskriterien fr die Endlagerung radioaktiver Abflle in einem Bergwerk/Safety Criteria for the Disposal of Radioactive Wastes in a Mine (4), the safety of a repository in the operational and post-closure phase must be proved within the scope of a site-specific safety assessment. Such an assessment comprises the undisturbed performance of the planned facility, assumed incidents, the thermal influence upon the host rock, the nuclear criticality safety and the radiological long-term effects in the post-closure phase.

Konrad and Morsleben Requirements

The results of the respective Konrad safety assessment have been converted into both

the design of the surface and underground facilities of this planned repository, and a system of waste acceptance requirements (5). They describe the general basic aspects and the general requirements to be fulfilled and then develop into more specific requirements on waste forms, packagings, radionuclide-specific activity limitations, documentation and delivery of waste packages to the repository. As the licensing procedure for the Konrad repository is still pending, the Konrad waste acceptance requirements are still in a preliminary form (2).

In analogy to the requirements derived for the Konrad repository project, corresponding requirements on the quality characteristics of a waste form, the waste containers respectively packagings of radioactive wastes, the limitations of the permissible radionuclide-specific activity concentrations, and the declaration of radionuclides have been compiled in the Morsleben waste acceptance requirements (6). The waste acceptance requirements were elaborated in such a way that a flexible system of requirements could be established, which is not only tailored to the radioactive waste presently generated but which also allows for improvements and future developments in waste conditioning techniques. Such a flexible system includes several alternatives and different options for the waste packages which ensure the required level of safety for the respective repository. The waste generators thus have the possibility of applying and fulfilling those requirements which are specifically applicable to the waste packages produced by them and to be disposed of.

Therefore, the Konrad and Morsleben waste acceptance requirements may be considered as an envelope or a frame covering all requirements to be met by the waste packages. Of course, such a flexible system of requirements may be considerably extended and more complicated but the advantages are immense if it is taken into account that a substantial modification of the waste acceptance requirements necessitates a new licensing procedure.

Acceptability of Waste Packages

In accordance with these requirements the radioactive waste originating from medical application of radioisotopes, industrial facilities or nuclear research and development laboratories was systematically checked for proper and complete evaluation. Thus, the acceptability for disposal of individual waste packages and of their annual amount was assessed. Special attention was given to the case where the activity-limiting values of the waste acceptance requirements are fully exhausted and to the necessary consequences. This assessment has shown that almost all waste packages from outside the nuclear fuel cycle can basically be disposed of in the Konrad repository (7).

Guidance for Waste Conditioning

Bearing these possibilities in mind, it is not excludable that the waste generators and conditioners will re-evaluate and optimize/rationalize present conditioning strategies and procedures. Up to now, those strategies and procedures have been determined by the available interim storage capacities, lacking repositories and the Konrad waste acceptance requirements. As a consequence, conditioning techniques are in particular aiming at volume reduction and observing the permissible activities per waste package due to the Konrad requirements. From now on, the operation of the Morsleben repository offers potential new developments or modifications of existing conditioning techniques. From a conditioner's point of view it is meaningful to analyze the Morsleben waste acceptance requirements (6) and to adopt conditioning strategies and techniques to these requirements. In addition to technical aspects, due to the Morsleben costs for disposal fixed at DM 12,500 per m3, i.e. DM 2,500 per 200 litre drum, it is henceforth possible to select appropriate conditioning procedures taking economic aspects into consideration, too. For example, as to combustible waste, it could be meaningful to use high-pressure compaction instead of incineration.

STATUS OF THE KONRAD AND MORSLEBEN REPOSITORIES

Konrad Repository Project

A final decision on the licensing procedure may possibly be expected in 1995. Thus, operation of the Konrad repository may be assumed to start towards the end of the nineties. However, the political will within the state government of Lower Saxony, i.e. the licensing authority, tries to prevent the Konrad repository project, so that the outcome of the process appears to remain open. Thus, in order to proceed, directives or instructions issued by the BMU can't be excluded.

Morsleben Repository

Since January 13, 1994 operation has smoothly been resumed. In 1994, a total waste package volume of about 1,400 m3 was disposed of. Nevertheless, due to the permissible rather low activity concentrations for U, Th, Pu and Cm difficulties concerning the emplacement of alpha-bearing waste originating from nuclear research establishments have arisen. Subject to results of additional safety assessments currently performed, final decisions on the acceptability of this waste are still to be made.

CONCLUDING REMARKS

The Konrad preliminary waste acceptance requirements are subject to the pending licensing decision. For planning and project implementation, the compulsory nature and reliability of these requirements are of great importance to the waste generators and conditioners. They have already started to adopt and to convert this guidance within their waste management system. According to a successful continuation and final realization of their efforts, the license for the Konrad repository is an important factor being a highly political and not a technical subject.

The operation of the Morsleben repository is an important step in the realization of a proper radioactive waste management system. Therefore, parallel to the Konrad work, present activities are in particular intended to increase the number of waste shipments to the repository according to the planned performance as well as to initiate further developments and improvements within the emplacement of radioactive waste in this facility.

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THE TEXAS/MAINE/VERMONT COMPACT: WASTE ACCEPTANCE CRITERIA PROBLEMS

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ABSTRACT

The disposal of low-level radioactive wastes for the State of Texas, as well as the proposed participating compact states of Maine and Vermont, will require the establishment of a stable classification scheme and a mutually acceptable series of waste related definitions. Additionally, the development of a precise waste acceptance criteria will be necessary for the orderly planning, development, emplacement, and closure of the proposed Texas low-level waste site. Classes

currently utilized for low-level radioactive wastes usually include: Class A, Class B, Class C, NARM, NORM, and low-level Mixed Waste. These six classes (excluding the majority of Department of Defense related wastes) originate from two basic sources or generators: utilities and nonutilities. Utility generators within the compact produce low-level wastes which are classifiable into 18 acronym categories. Non-utility generators utilize a minimum of nine basic categories which are divided into dry waste and process (wet) streams.

The proposed Texas repository currently will not accept Mixed Low-Level waste, Greater Than Class C (GTCC) waste, or Transuranic (TRU) waste (exceeding 370 Bq/g] [10 nCi/g]), thereby establishing the upper limits for disposal. At present no defined lower limit for radioactive wastes exists. This standard is essential for effective planning and the estimation of future waste streams. Its name (BRC, de minimus, Exempt) is unimportant.

Semantic problems exist. Precise definitions for such common words as processing, recycling, generation, etc. are necessary; they are not used in the same sense between utilities within the same state let alone between the states. Related waste acceptance criteria that must be addressed include: types of beneficiation of waste, validation of point of origin, consistent and easily recognizable labeling that includes an inventory, waste manifest contents, transport tracking, package standards, etc.

INTRODUCTION

In 1980 the State of Texas was required to respond to the federal mandate to establish regional low-level waste sites (Low-Level Radioactive Waste Policy Act, 42 USC 202 1b). The act required the establishment of either regional (compact) or single state (non-compact) low-level radioactive waste repositories. Six months after the passage of the act, the Texas Low-Level Radioactive Waste Disposal Authority (Authority) was established. The site selection process was initiated in February of 1983 by the Authority.

Texas was one of the states initially opting to develop its own statewide low-level radioactive waste repository (non-compact). In 1989, in response to non-compact repository problems associated with the Interstate Commerce Clause of the U.S. Constitution, the Texas legislature attached a rider to an appropriations act that directed the Authority to examine the problems and benefits of compact and non-compact status. In 1991, reports were presented and reviewed at hearings before the House Committee on Environmental Affairs and the Senate Committee on Natural Resources. These hearings resulted in the recommendation to pursue an interstate compact.

In 1993, Texas House Bill 2665 was passed which established the legislative approval for the development of the compact. In November of 1993 the legislature of Maine, on the basis of a referendum, joined the compact. The legislature of Vermont in May of 1994 became the third member of the compact. The Texas Compact will become an official entity upon its ratification by the United States Congress which is expected in 1995.

The Texas site is currently being designed as a near surface facility with engineered barriers. This option was selected over the deep geological subsurface option on the basis of simplicity of operation, flexibility, and low costs for investment and operation. This is balanced against the fact that it offers less isolation from the biosphere than the deep geological subsurface option, consumes surface land, and is more susceptible to human intrusion. Case studies of national low-level near surface and deep geological subsurface radioactive waste sites (1) lend little useful direction for the proposed semiarid Texas site.

COMPACT RESPONSIBILITY

The basic responsibility of the Authority and the Texas Compact, as well as all of the entities involved in the disposal of radioactive waste, is to ensure the protection from radiotoxicity of this and future generations by the adequate isolation of these wastes from the biosphere (2). Performance assessment is the apparent means to test the assumption of adequate isolation. It is a semi-quantitative tool that can be effectively utilized by governmental regulatory and licensing agencies.

McCombie, Papp, and Coplan (3) define performance assessment as the quantified description of the current and future behavior of a system. It is determined by the time-dependent responses of many separate, coupled and uncoupled, interdependent subsystems (e.g., duration of decreasing strength of radiation, geological setting

and events, climatological changes, engineering barrier parameters, etc.) which are fixed within a sequence of different environmental settings or scenarios. The primary purpose of the assessment is to produce a computer applicable model of the radioactive waste system.

The basic elements of the performance assessment structural database (3) include information on such parameters as: the radioactive materials present, other materials present, repository layout, materials and dimensions of engineering barriers, and site characteristics. The structural database, interacting with the chemical and physical conditions developed, form a system that must be identified and analyzed. The resulting system of interacting processes are then computer coded and converted into a mathematical model which varies with a time-dependent process. Potential environmental variations (scenarios) are then tested yielding predicted probabilities and consequences. These data are then evaluated utilizing sensitivity and uncertainty analyses to develop a justifiable level of confidence in the safety of the repository in order to meet public, regulatory, and licensing requirements. Absolute proof of the safety of any repository is not attainable nor is it required. What is always expected is a comprehensive analysis reviewed in a process of regulatory control that considers all reasonable possibilities.

The data requisites for the construction of either an adequate conceptual or utilizable mathematical model of the repository system requires an accurate, hopefully standardized, descriptions of such basic parameters as the repository engineering barriers (e.g., waste containers, backfill, etc.) as well as the more critical identification of the radioactive waste that is being disposed of in the repository. This ideal may be established by the development of adequate waste acceptance criteria, uniform terminology, and an accurate classification scheme.

LOW-LEVEL RADIOACTIVE WASTE CLASSIFICATION PROBLEMS

An example of one of the major national low-level radioactive waste classification problems is that the lower limit for radioactive wastes remains undefined. This regulatory vacuum has occurred with the loss of the Nuclear Regulatory Commission's politically incorrect BRC (below regulatory concern) designation (4).

This, in addition to other internal problems inherent in the current American approach to nuclear waste classification, would seem to indicate at least the need for revision of the system. The
adoption of a new code would seem to be an unworkable solution for a number of reasons, primarily political. It has been recommended that the appropriate governmental agencies investigate the possibility of adoption of the classification scheme developed by the International Atomic Energy Agency (I.A.E.A.) in order to exchange data globally. This concept has been discussed in detail elsewhere (5,6).

The problem of standardized low-level waste classification is pervasive throughout nuclear waste management. The widely accepted international radioactive waste classification is divided into six basic categories; they are: High-Level; Low- and Intermediate-Level (each of which is divided into a short-lived and a long-lived category); and an exemptible waste category. A contact dose rate of 2 mSv/hr (200 mrem) is generally utilized as the division between Low-Level and Intermediate-Level wastes (7)

The I.A.E.A, has been actively pursuing the development of the limits to be imposed on very low radioactivity wastes (8). They recommend that those materials which should be excluded or exempted from regulatory practice should include sources not amenable to control (e.g., cosmic rays, etc.) as well as those materials which represents so low a risk that it becomes a waste of resources to exercise control over them by the regulatory process. This philosophy yields two recommended categories: exemption and clearance. Exemption is one of the six basic I.A.E.A. categories and clearance represents a regulatory process.

Exemption includes small sources which typically are not regulated (e.g., research tracers, calibration sources, etc.). The exemption levels are determined by the International Basic Safety Standards for the protection against ionizing radiation and for the Safety of Radiation Sources (BBS). The I.A.E.A. in Safety Series no. 89 (9), concluded that for the purpose of exemption, a level of individual dose of some tens of microsieverts (a few mrems) in a year reasonably represents what is regarded as a trivial dose by competent authorities. As individuals may be exposed to radiation doses from several exempted practices, it is necessary to insure that the collective dose does not exceed the trivial dose. For this purpose the I.A.E.A. (9) suggests that individual doses be in the order of 10mSv (1 mrem) in a year for each

exempt practice.
I.A.E.A. clearance (or decontrol) includes formerly regulated radioactive materials which have been processed in such a way the materials no longer represent a radiological risk. Additionally, radioactivity lost by decay would be included in clearance process. This is an important factor in low-level waste management. Clearance would include recycled materials as well as waste materials derived from the nuclear fuel cycle and regulated facilities such as hospitals, research laboratories, industry, etc. When radioactivity reaches the level of activity or activity concentration (clearance level) regulatory control is removed, the recycled or waste material is then said to be cleared.
Low-level waste within the United States is defined as that radioactive waste which is not high-level waste, spent nuclear fuel, transuranic (TRU) waste (containing radionuclides with an atomic number exceeding 92), or other (consists primarily of uranium mill tailings) (10,11,12). This low-level system may be divided into seven basic categories; they are: Class A, Class B, Class C, greater than Class C. (GTCC), NORM, NARM, and mixed waste. Low-level mixed wastes are those wastes containing regulated hazardous chemicals in addition to radionuclides.
On national level, the Department of Energy (DOE) is currently working on an Environmental Restoration and Waste Management (EM) Programmatic Environmental Impact Statement (PEIS). The EM PEIS will be utilized in evaluating DOE radioactive wastes. Current supporting studies (13) utilizing the DOE Integrated Data Base (14) and the Waste Management Information System (WMIS) (15) recognizes six radiological categories; they are: 1) U/Th, 2) fission products, 3) induced activity, 4) tritium, 5) alpha-emitting radionuclides and 6) other wastes which are not solid or combustible. Additionally, Cooley and others (13) define ten treatability categories: they are: 1) combustible, 2) noncombustible and noncompactible, 3) noncombustible and compactible, 4) surface contaminated metal, 5) bulk-contaminated metal, 6) sludge/resin, 7) all other solid LLW, 8) aqueous, 9) organic liquid, and 10) remote handled.
Wastes may exist in conditions that can be classified as being generated, treated, stored, or disposed. These waste conditions are not only a classification problem, but also one of semantics. For clarity, they should follow the definitions used in the current DOE Integrated Data Base (16). "Generated waste" is radioactive material recently discharged from a facility production process; it has no commercial value. "Treated waste" has been chemically or physically altered to reduce its toxicity or prepare it for storage or disposal either on- or off-site. Treated waste includes volume reduction processes such as incineration and/or compaction for either storage or disposal. Cooley et al. (13) recognize eleven treatments for low-level waste; they are: incineration, solidification, vitrification, compaction, supercompaction, size reduction, decontamination, metal melting, evaporation, general aqueous treatment, and packaging. They also list ten treatments for mixed low-level waste; these are: incineration, wet-air oxidation, evaporation, solidification, thermal desorption, shredding, pelletizing, packaging, vitrification, and polymeriztion. "Stored waste" is generated waste, probably following some treatment, which is temporarily retained and monitored in a retrievable manner prior to disposal. Stored waste represents an operational convenience. It requires isolation, monitoring, and provisions for the environmental safety of the public. This category implies subsequent treatment, reprocessing, transport, or disposal of the stored waste. Its usage in low-level radioactive waste management is primarily either so short-lived radionuclides may decay to an acceptable level for clearance, or storage at a facility where waste is retained until the construction of a permanent facility (17). The interim storage problem, both on- and off-site, is being currently addressed by the NRC (personal comm., James E. Kennedy). Mixed waste will require Environmental Protection Agency (EPA) approval in addition to the NRC as it comes under the Resource Conservation and Recovery Act (RCRA). The last category, "disposed waste," refers to that waste that is in final emplacement with no plans for retrieval.
The Texas Low-Level Radioactive Waste Disposal Site currently will not accept low-level mixed waste, Greater Than Class C (GTCC) waste, or Transuranic (TRU) waste exceeding 370 Bq/g (10 nCi/g), thereby establishing the upper limits for disposal. Using the 1 mrem (10 mSv) per year as a standard, Texas has promulgated a BRC rule that allows certain radioactive materials to be deposited in a municipal landfill. Wastes meeting this standard are restricted to disposal in only the most

well-designed and operated landfills. The rule has provided an economical and safe method for the disposal of limited types of radioactive material; it could be roughly compared the IAEA's low-level, short-lived, cleared category of disposal.

PROPOSED WASTE ACCEPTANCE CRITERIA

The Authority is currently developing waste acceptance criteria, disposal container specifications, and an initial fee schedule in anticipation of a potential opening date in 1997. The disposal limit for the other compact states of Maine and Vermont will be 20% of the projected annual average of low-level waste generated in the state of Texas for the years 1995-2045. The Authority, in response to this requirement, undertook a series of studies that examine low-level waste streams and component characterization, as well as the quantities of radioactive wastes that will be generated in the states of Texas, Maine, and Vermont (18,19,20). What has emerged from these studies is that there are semantic and classification differences between the three independent low-level state regulatory bodies. The responsibility for and control of the incoming wastes lies with the host state. Authority criteria in large part parallel the I.A.E.A. generic approach (21).

The following discussion is an indication of the direction that the Texas Compact proposed waste acceptance criteria is developing. The criteria are anticipated to include specific provisions that the accepted waste must have an adequate manifest and be properly classified, document must disclose radiation quantities internally and surface dose rates externally, be in sealed and approved containers which will be placed within concrete canisters at the proposed site. Waste form must have designated stability or have been through an approved stabilization process. Free standing and noncorrosive liquids in containers shall not exceed one percent. Liquid wastes must be solidified or absorbed. If absorbed, the treated waste must have twice the necessary absorbent material present. Chelating or complexing agents should not be present. Pyrophoric waste must be treated to a level of being nonflammable. Waste delivered shall not be capable of detonation, explosive decomposition, or explosive reaction with water.

The waste shall not contain or be capable of gas generation to anyone handling the waste throughout the disposal process, except radioactive gaseous waste that is properly packaged at an absolute pressure not exceeding 1.5 atmospheres at 20oC with a total activity of less than 3.7 tetrabequerels (100 Curies).

Biological wastes must be layered with vermiculite, diatomaceous earth, or approved absorbent plus lime to reduce gas generation. The accepted ratio will be thirty parts of biological waste to one part slaked lime to ten parts absorbent. Waste containing hazardous, pathogenic, or infectious material must be treated to the maximum possible extent. Waste oil must not exceed one percent of the volume. Incineration ash and other powders must be treated so that they may not enter the air. Mixed waste shall not be accepted for disposal.

Waste will be placed in concrete canisters with maximum internal measurements of 6 ft. 8 in. vertical and 6 ft. 9 in. horizontal. This configuration accommodates a hexagonal pattern of 55 gallon drums which will be delivered on hexagonal, disposable pallets. Compliance violations of the acceptance criteria could result in several levels of fines, charges, or other actions.

TEXAS COMPACT LOW-LEVEL WASTE CLASSIFICATION

Development of a low-level radioactive waste site for the State of Texas, as well as the participating compact states of Maine and Vermont, requires a stable classification scheme for the orderly planning, development, emplacement, and closure of the Texas proposed low-level site. The Authority classifies low-level radioactive waste into six classes;: Class A, Class B, Class C, NARM, NORM, and Low-Level Mixed Waste. The proposed Texas Low-Level Radioactive Waste Disposal Site, as previously noted, will not accept mixed waste, Greater Than Class C (GTCC) waste, or Transuranic (TRU) waste exceeding 370 Bq/g (10 nCi/g), thereby establishing the upper category limits of the compact repository. The Texas BRC rule, discussed earlier, establishes the effective lower limit of disposal. These six general classes of waste, excluding Department of Defense (DoD) and Department of Energy (DOE) military wastes, originate from two basic sources or generators within the compact: utility and non-utility (18,19,20).

Operational wastes of utility generators within the compact have been divided by Shuman and others (18,19,20) into 18 common categories of low-level wastes (Table I). They define dry active waste (DAW) as consisting of both compactible (COTRASH) and noncompactible (NCTRASH) wastes which are frequently shipped together. COTRASH

normally implies material suitable for incineration while NCTRASH normally refers to metallic components and discarded equipment. High specific-activity waste (HIGHACT) is an acronym used at the Vermont Yankee Nuclear Power Station for activated stainless steel generated during replacement of recirculation piping. Non-Fuel Reactor Components (NFRCOMP) are periodically discarded components are typically composed of corrosion resistant alloys which may contain boron, cadmium, or hafnium as neutron absorbers.

Ion exchange resins (IXRESIN) are used to maintain and control water quality in reactor systems. They are typically styrene-divinyl benzene polymers. Four other recognized resin types are: (RWCUPRS) reactor water cleanup resins, (RWDMRES) radioactive waste demineralizing resins, (SSYSRES) secondary system resins, and (DECONRS) decontamination resins. Decontamination resins are used in the infrequent decontamination of primary coolant systems. All resins must be dewatered. The decontamination resins, however, may contain metallic materials from reactor grade steels in the components and cladding as well as possible high quantities of chelating agents.

Cartridge filters (FCARTRG) are typically constructed of woven or wound fabric or pleated paper supported by a stainless steel basket for the removal of suspended solids. Elements with natural fibers are subject to decomposition and oxidation through chemical attack, radiation damage and/or biological action. PROCFIL is an acronym for process filtered waste used in Texas.

Filter sludge (FSLUDGE) is waste produced by filters which include diatomaceous earth, powdered mixes of cation and anion exchange resins, and high purity cellulose filters. Three types are separately recognized; they are: (CONFDSL) condensate phase separator filter sludge, (FLDRFSL) floor drain filter sludge, and (FPFILSL) fuel pool skimmer sludges. Powered resins and cellulose sludges may produce gas. Filter sludge composed of metal oxides and dirt is referred to as crud. The classification no longer utilizes (EQDRFSL) equipment drain filter sludge and (OIL) spent oil.

Decommissioning wastes are lumped into three major waste streams by Shuman and others (18,19,20). These waste streams will require adequate classifiable acronyms in the future. Decontamination wastes are materials that are generated during reactor decommissioning operations. Neutron activated wastes consist of components that have been exposed to radiation from the reactor core. The third stream is plant dismantlement waste which is waste generated during the dismantlement of the power plant.

Non-utility generator waste (Table II) has been divided by Shuman and others (18,19,20) into nine basic categories which are divided into dry waste and process (wet) streams. The dry active waste, non-utility stream consists of five categories, three of which are utilized in utility operational wastes; these are compactible trash (COTRASH), noncompactible trash (NCTRASH), and high specific-activity waste (HIGHACT). Low specific-activity waste (LOWASTE) is an acronym used by a Vermont academic institution for a waste similar to COTRASH but with a higher density and moisture content which should probably should be placed with the process waste stream. Sealed sources (SOURCES) are low activity sealed sources used on calibration and reference standards.

Process (wet) wastes include absorbed (ABSLIQD) or bulk (BLKLIQD) liquids which may be either aqueous or organic. Biological (BIOWAST) wastes are generated primarily through medical and university research with the bulk of the material consisting of carcasses and tissues. Rare-earth processing waste (REPWAST) is material generated primarily through process metallurgy of rare-earth ores.

These subdivisions and categories also are assignable to the basic six classes. The nine categories do not necessarily cover the four remaining groups; which includes three of the six basic classes. These groups include: naturally occurring radioactive materials (NORM), naturally occurring and accelerator produced radioactive materials (NARM), mixed waste, and Process Metallurgy (Table III).

Types of beneficiation, as in the treatability of DOE wastes, (e.g., supercompaction and incineration versus untreated waste) also must be recorded. Problems that are still to be addressed include such basic items as: validation of point of origin, consistent and easily recognizable labeling that includes an inventory, waste manifest contents, transport tracking, package standards, etc.

SEMANTICS PROBLEMS

Preliminary discussions with the future participants in the compact have revealed a number of potential semantic problems. An example of this is in the description of

the waste product itself. If you use the annual projected waste as being "as generated" it means 2,290,000 cubic feet; if you use "as disposed", the volume is 384,000 cubic feet. When discussing differences between "as generated and "as disposed waste, generators within the same industry have a widely varying vocabulary. To one generator, "as generated" waste means without any treatment. To another, it means minimal treatment, or pretreatment such as 3:1 compaction. Curiously, to one generator "as disposed" meant that amount of waste that left the plant perimeter. To another, it meant the waste that was actually placed in the ground at Barnwell. To some non-utility generators "as disposed" means the waste shipped from the facility. To them, the waste is discarded - i.e., it will not return to their facility - whether it has been shipped to an intermediate processor for processing or storage, or to a disposal site for ultimate disposition.

What is required is standard, precise definitions for such very common, straightforward terms as generated, processed, produced, treated, shipped, and disposed. Equally problematic are the waste management terms (e.g., treated and untreated wastes). The reality is that the terms are not used in the same sense between operators in the same state or between the states. If this problem exists in the Texas Compact, it exists in other national compact and noncompact entities. It would seem to be the appropriate responsibility of the NRC on the national level and the I.A.E.A. on the international level to formulate such standards.

Three things remain to be resolved for the Texas Compact; they are standardized waste acceptance criteria, classification, and waste management terminology. The Texas repository is a microcosm of all low-level radioactive waste site projects. As an example, it is essential that the waste be properly packaged. Texas Class A waste containers are assumed to fail at 100 years while Class B and C waste containers are assumed to fail at 300 years (22). Knowledge of what waste is where is essential for those who will have the responsibility of monitoring the site in the future.

In order to be able to inventory disposed of waste and to know what waste is where, it is necessary that it be classified in a manner systematically rigorous enough to be useful. The six, really three (A,B,C with A being separated), major classes of low-level waste for the Texas repository would appear to be not enough while the numerous acronyms utilized may be too many.

To exemplify the problem, take the case of gas generation. We know that the processes of corrosion, microbial degradation, and radiolysis of certain wastes (e.g., resins) will generate gas (23). Additionally, Eh and pH have a profound influence on the rate of generation. The process may result in such effects as structural failure within the repository or the development of a gas-driven water flow bringing radiotoxin contaminated groundwater into the biosphere.

Participants in the Texas Compact will have to come to an agreement on the precise meaning of the words involved in low-level waste management. This problem, as stated earlier, is one that will be encountered within the other national low-level waste disposal entities. It is recommended that formalization of practical, uniform, translatable, low-level applicable acronyms be considered by the NRC, DOE, and DoD for military, utility, and nonutility generators on a national basis. The resultant data would allow comparison among current waste stream data sets and applications, enable more efficient design of repositories, maintain a national inventory, and develop future low-level waste strategies.

CONCLUSION

The purpose of classification is to reduce the number of components present within a system; conversely, it must be subdivided rigorously enough to be flexible and applicable. It is apparent that it will require the institution of negotiated parallel standards between the Texas Compact states as well as their generators. This problem is inherent in all of the national compacts and noncompact entities. The establishment of clearly defined national and/or international standard acronyms and process definitions for low-level waste is strongly recommended.

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22-5
RADWASTE MANAGEMENT SYSTEM - EXPERIENCES IN OPERATING AT THE KARLSRUHE RESEARCH CENTER

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ABSTRACT

We are operating several facilities for radioactive waste treatment comprising processing of all radioactive residues arising at the Karlsruhe Research Center and at other institutions in Germany (e. g. nuclear power plants or state collection facilities) and their conditioning into waste forms amenable to repository storage. In order to be able to observe the requirements imposed by the authorities, we are making use of the KADABRA computer code as accounting and management system. KADABRA is not only a passive accounting system but it assists the users in decision making and thus exerts a direct influence on processing steps. Besides accounting, KADABRA fulfills the following tasks:

- recording the processing streams at HDB,
- monitoring observance of authorized limits,
- calculating nuclides which cannot be conveniently measured by establishing correlations,
- assisting the users in decision making regarding the number and the quality of waste packages to be produced, taking into account the transport and repository conditions,
- preparing the documents needed for repository storage,
- administrating orders from our customers,
- recording operational data for invoicing at a later date,
- writing reports for customers and authorities.

INTRODUCTION

The Central Decontamination Operations Department (HDB) is the central disposal facility for radioactive residues and wastes arising at the Karlsruhe Research Center. We are operating a number of plants for treatment and conditioning of radioactive residues, e. g. incineration plants, compacting plants, evaporator and cementing plants as well as various decontamination facilities. HDB handles all radioactive residues generated at the Karlsruhe Research Center, by institutions located there and by external institutions, i. e. either conditioned to waste forms amenable to repository storage or transferred to an interim store until repository storage or, if possible, decontaminated with a view to reusing them.

In the course of processing radioactive residues many boundary conditions must be observed, e. g. license under the Atomic Energy Act, guideline for waste control, repository conditions as well as transport regulations. In order to conform to the requirements imposed by the authorities and to verify the diverse conditions laid down in the numerous regulations, we are using the KADABRA (KArlsruher DAtenBank fr Reststoffe und Abflle) accounting and management system.

KADABRA at present comprises about 2000 individual programs and is being operated on an IBM mainframe acquired for this purpose. About 40 terminals have been installed in the operating plants proper, another 40 in offices. The users make their inputs into KADABRA in situ and almost simultaneously with physical processing of the material so that continuous documentation updated at any moment is guaranteed.

The central tasks of KADABRA, namely compliance with the obligations to report, planning of processing steps taking into account repository conditions and transport regulations, documentation of the material and activity flows as well as operational documentation, will be described and explained in more detail below.

BOUNDARY CONDITIONS

License under the Atomic Energy Act

All HDB processing plants are being operated under a license granted according to the Atomic Energy Act. The license includes general requirements concerning e. g. obligations to report as well as plant related requirements and limits. Some requirements can be observed solely by application of KADABRA. For instance, it must be known at any moment which activity inventories are contained in the respective operating buildings, broken down by nuclear fuel nuclides and other nuclides, so

that the authorized limits are not exceeded. The respective listing must be prepared every month and submitted to the supervisory authority. Considering the multitude of residues handled here, accounting satisfying the requirements above would not be manageable at all by manual entries because the residues may be transferred repeatedly every month from one store to the next or from one processing plant to the next. Other reports, e. g. listings of all residues received or residues cleared in one quarter, can be prepared as well by input of the necessary instruction to KADABRA and subsequently presented to the supervisory authority.

Guideline for Waste Control

The Federal Ministry for the Environment (BMU) issued in 1989 the "Guideline for Controlling Radioactive Wastes with Negligible Heat Evolution Not to Be Delivered to a Central State Collection Facility." In this guideline the categories of radioactive wastes have been defined as well as the responsibilities and the number of documents required when delivering, transferring and transporting radioactive wastes. Besides, the ways and means of activity assays in radioactive residues and wastes are specified and instructions given on mixing wastes from different categories of delivering institutions. Moreover, administrative details, e. g. identification of waste packages, have been fixed. The waste categories, the choice of database fields, and the whole program logic for control of materials flows in KADABRA have been laid down in this guideline and, in addition, the annual inventory and outlook reports to be submitted to the Federal Office for Radiation Protection (BfS) concerning the current waste inventory and the amounts of wastes anticipated in the year to come. The said reports are mandatory for each institution delivering radioactive wastes and can be prepared by us and many other delivering institutions only with the help of statements of account produced by KADABRA.

Repository Conditions

At present, one repository store exists in the Federal Republic of Germany for acceptance of radioactive wastes without heat evolution, namely ERAM in the Morsleben salt mine. The KONRAD iron mine near Salzgitter to be used as a repository for the same type of wastes is now at the stage of planning, but implementation can be expected for the next years. Although there are tentative ideas regarding installation of a repository for heat evolving wastes at Gorleben, its construction is very doubtful for political reasons. For the two repositories mentioned first repository conditions having legal effect exist, based on safety analyses and design basis accidents which, however, in case of the KONRAD mine are still preliminary. According to the repository conditions waste forms are classified by different waste form groups, depending on waste form quality, and the maximum activity inventories per nuclide and package are fixed. Moreover, specifications are laid down as regards the cask sizes and safety devices. These conditions must be observed in production of waste forms because otherwise transfer to the repository will not be permitted. Prior to the production of waste forms, processing is planned with KADABRA taking into account the repository conditions in order to avoid production of waste forms not capable of repository storage. After production, KADABRA verifies all limits specified for each waste form, stores them and prepares a record. If in spite of planning waste forms have been produced with excessive activity values, they must subsequently be handled separately.

Transport Regulations

Waste forms conforming to the repository conditions described above have to comply in addition with the Ordinance on Transports of Hazardous Goods on the Road and by Railway (GGVS/GGVE), respectively because transport to the repository cannot be dispensed with. These rules and regulations are verified by KADABRA as regards adherence to the limits for LSAII, LSAIII or Type A and Type B transports, respectively.

MATERIALS FLOW

Materials Receipt

In order to comply with the guideline for waste control, a number of data must be available even prior to receipt of radioactive residues. To standardize the data supplied by different delivering institutions, a residue accompanying slip has been designed which the delivering institution must fill in for each residues containing cask, normally 200-l-drums, before delivery of radioactive residues to HDB. The accompanying slip is put into KADABRA before delivery of the material to get approval for delivery, and it passes some checking routines to ensure that all information required has been entered on the slip. After delivery the accompanying

slip is input once more into the system. By redundant input and automated intercomparison of both inputs the number of input errors is minimized. By the second input the residue is simultaneously transferred into our accounting system and in this way becomes part of the quantity for which we hold a license. All residues can be identified at any moment by their unique residue slip numbers and can thus be traced. By a transfer dialog immediately following the input dialog the residue is assigned to its storage place (location or cask). In this way, the activity of the residue is added to the database-internal activity account of the respective storage facility. If, e. g., the residue causes the authorized limit for the storage facility in question to be exceeded, this is immediately displayed as a warning.

Material Processing

Residues received cannot be further processed in accounting until they have been input into KADABRA via the input dialog. Processing of the residues without prior accounting is prohibited. Processing, both in physical and in accounting terms, is done by batches. Processing batches are unambiguously identified by the facility in which they are processed, the year and the running number of the batch.

Prior to residue processing a batch must be opened in KADABRA. The residues to be processed must be posted to this batch. For proper declaration of the waste forms to be produced it is of particular importance to coprocess materials of the highest possible homogeneity in terms of nuclide vector and activity distribution because under computational aspects the material pertaining to a batch is always homogenized anyway. To allow the correct assignment of the waste forms to the delivering institution it should be endeavored to coprocess only residues delivered by one delivering institution. KADABRA assists the user by suitable checking routines. The material is retained in the batch as long as work on it is actually done. Then the intermediate or final products are taken out of the batch via separate dialogs.

Waste Form Production

In the course of residue processing either intermediate products are generated which, after decontamination, are measured for clearance or further treated in processing batches, or waste forms are generated which are amenable to repository storage. The latter result from conditioning batches, in contrast with the pretreatment batches providing only intermediate products. Regarding conditioning batches, batch planning must be performed in advance taking into account the repository conditions and the transport regulations. For this, a dialog specifically developed for that purpose is used which allows observance of the repository conditions and transport regulations to be verified very quickly. By variation of the numerous input parameters optimum use can be made of the limit authorized for each package, with the cask filled up to the maximum level and taking into account plant induced waste form qualities and operational costs. In addition, the kind of activity assignment for the waste forms is specified during batch planning (s. Chapter entitled ACTIVITIES FLOW).

KADABRA automatically characterizes by data sets the intermediate and end products generated, from which data sets the respective shares of the possibly different sources of waste, customers, orders, etc. can be taken. Thus, detailed tracing back of the materials is possible. Waste forms which after such apportionment are composed of shares of several sources of waste are assigned by KADABRA to the source having the highest share because each waste form must not belong to more than one source of waste.

Disposal of Material

When residues, intermediate products or end products are disposed of, they are retired from KADABRA. The waste data sheets required for transfer into a repository are prepared and printed by KADABRA. In the dialog for disposal the date of disposal and the receiver are entered. In this way, the activity account of the last facility charged is credited.

Secondary Waste

When radioactive residues are processed quite a number of new residues are generated which, considering their histories, we do not call intermediate products but secondary waste. Secondary waste like all the other residues must be adequately identified. For this, KADABRA offers dialogs for recording of the secondary waste. The user has to enter solely information which the system cannot derive, e. g. the amount, weight, type, cask number, etc. The user can make his choice among various possibilities for automated determination of the activity inventory, e. g. by input

of the dose rate and computation using the nuclide vector averaged by us, or by input of a number of analysis or input of the gross activity and computation by means of the last completed batch or by a similar procedure.
Some secondary wastes experience shifts in the nuclide vector due to shifts in concentration in the waste, e. g. in water from cleaning of waste gas evolving in incineration furnaces: All alpha nuclides are left in the ashes, some volatile nuclides escape through the stack and others remain in the scrubbing water. This fact is taken into account by application of plant related correction factors. After a secondary waste accompanying slip has been prepared, the calculated nuclides are corrected using the appropriate correction table.
ACTIVITIES FLOW
Batch Input and Cross Contamination
When residues are assigned to a processing batch KADABRA adds up all nuclides and activities entered in the individual residue slips. In addition, KADABRA prepares, dependent on the processing facility, a residue slip for cross contamination and charges it to the batch. Cross contamination is understood to mean that contaminations from the preceding batch are still attached to plant components and spread over the materials of the new batch. Cross contamination is in most cases safety irrelevant, but on account of potential changes of the nuclide vector it is important to the waste form declaration for the repository. Cross contamination is calculated from the batch vector of the previous batch, multiplied by the contamination values to be measured or by the residues left in the casks. All batch inputs taken together make up the batch input activity. The nuclide vector so obtained is taken as the basis for producing intermediate and end products belonging to that batch.
The nuclide vector and the nuclide activities can undergo changes within a batch, e. g. by assignment of a laboratory analysis following which the analytical data are extrapolated to the batch volume by KADABRA and taken over. Not analyzed nuclides are calculated by establishing correlations, and the ratio of activity of an analyzed key nuclide to be specified to the initially declared key nuclide is used as the factor.
During waste form production the activity contained in the batch can be distributed among the waste forms in different ways:
  By residue slips
This method can be applied only if after a batch has been processed a statement can be made as to which residue is left in which waste form accommodating drum. After input of the respective residue slips per waste form drum KADABRA adds up the individual activities and so determines the gross activity of the waste forms produced. By this method, contrary to all the other methods, a residue can be followed up to the end product. The method is frequently used in compaction of residues in order to be able to identify the pellets in the product by the numbers of the residue slips.
  By relative mass
If we can assume that the processing batch is very homogeneous in terms of activity distribution, the activity contained in the batch is distributed in proportion to the mass fractions of the products. This method is mostly applied to cementing concentrates. As the material involved is an extremely homogeneous one and, in addition, the batch inventory had been determined in advance in a laboratory analysis, the declaration error is very small. The method can be applied also to nuclides hardly detectable by measurement, if determination via the dose rate must be ruled out.
  By relative dose rate
If in spite of the similarity of nuclide vectors the activity is inhomogeneously distributed, the activity contained in the batch is distributed proportional to the dose rates measured on the waste forms. This method is most frequently applied in processing solid wastes because the error in distribution by relative mass is unacceptably high on account of the inhomogeneity in activity distribution.
  By percent
This method serves less for true distribution than for providing the possibility of specifically deducting as product fractions of the batch inventory, e. g. packaging materials equivalent to 0.1% of the batch inventory, or of assigning the whole batch inventory to a waste form, e. g. 100% assignment if only one single waste form is produced.

Recalculation based on the average dose rate and the net weight

If the values declared on the residue slips prove to be not accurate enough, it is possible to recalculate the activity while retaining the nuclide vector of the batch provided that nuclides are contained in the residue which can be conveniently measured. KADABRA computes, after input of the mean dose rate, measured at 1 m distance from the drum, and of the net weight which activities correspond to that dose rate, taking into account the specified nuclide vector and the specified geometry.

Assignment of laboratory analyses

Another possibility of activity determination is the direct assignment of a product analysis. This possibility is rarely referred to in practice because the necessary laboratory analyses in most cases are directly performed with the whole material of a batch and are subsequently also directly assigned to a batch.

OPERATIONAL DOCUMENTATION

Administration of Quotations and Orders

KADABRA also administers orders placed by customers. This does not only serve the purpose of cost finding and invoicing at a later stage, but serves primarily to control the process. The orders are given unique numbers to be entered in the residue accompanying slips. In the process of receiving inspection the status of the order is queried. This ensures that we do not accept residues which are not linked to an order. During processing certain controlling fields are called from the order file, e. g. prohibition of mixing, incineration or packaging. In this way, the operating facilities are actively assisted in processing properly material received from the customers.

Cost Finding

Operational cost finding is performed in the course of processing in the facilities. For this, the processing expenditure for the respective material in the processing batch is entered in KADABRA via an expenditure dialog. After completion of batch processing the total expenditure for processing a batch is allocated in proportion to the respective amounts to the residues or intermediate products processed in the batch. The customer receives a cost breakdown by residue slips.

CONCLUDING REMARKS

KADABRA has been operated for nine years and its acceptance with the users is high. However, due to a lot of supplements and improvements, the database structures and the programs have meanwhile become too extensive and complicated. Besides, the guideline for waste control issued by the Federal Ministry for the Environment is to be replaced with the "Ordinance on the Utilization of Radioactive Residues and on the Removal and Disposal as Radioactive Wastes" (AtRAV) in which modified requirements will be laid down regarding the identification and documentation of radioactive wastes. As a fundamental revision of the present system is indispensable, the KADABRA II version is presently in the process of development and planned for being put into service in early 1996.

22-6

LOW-LEVEL RADIOACTIVE WASTE DISPOSAL: AN OVERLOOKED OPPORTUNITY FOR BENEFICIAL COOPERATION BETWEEN COMPACT STATES AND THE DEPARTMENT OF ENERGY

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ABSTRACT

One result of the Low Level Radioactive Waste Policy Act of 1980 and the Low Level Radioactive Waste Policy Amendments Act of 1985, has been that 45 states are now members of 10 interstate compacts governing the disposal of low-level radioactive waste materials generated within their member states (excluding Federal agency wastes). Progress in acquiring disposal facilities, however, has been very slow and difficult. Many state legislatures have been unreceptive, even hostile, to the subject, as have local governments and citizens. Currently, only two states (i.e., two compacts) have operational disposal facilities. This situation gives rise to the notion that the states and Department of Energy (DOE), which cooperate not at all where waste disposal is concerned, are overlooking an outstanding opportunity to resolve several of the states' problems by reversing their course and cooperating completely in low-level radioactive waste disposal. That opportunity involves states' use of DOE facilities. Major DOE facilities are located in 14 states, five

of which are compact host states, and DOE has a major ongoing program to clean up its sites and dispose of the wastes, mostly in permanent facilities on the sites. At the same time, the low-level radioactive wastes anticipated to be generated by the states will be very small in comparison to the volume of DOE wastes, probably less then 10 percent by volume, as will the facilities needed for disposal of the states' wastes.

There appears to be little or no point in the states having to acquire property - several hundred acres each of uncontaminated property - remove it from the tax rolls, then contaminate it with waste disposal facilities, and be required to manage the property in perpetuity, with all the attendant costs, when the wastes might be disposed of at the DOE sites, which already exist and are going to be managed in perpetuity as a responsibility of the Federal government. It certainly seems that the opportunity for close cooperative actions - to do more with less - is worth investigating. The principal benefits are 1) the states would not have to remove land from productive use and from the tax rolls; 2) the states would avoid the costs of acquiring land; and 3) the costs of developing and operating disposal facilities, then perpetually maintaining the disposal facilities after closure would be avoided.

Several actions are needed to effect this proposed cooperation. Officials in the compact states need to get together with DOE representatives to discuss exactly what would be necessary in agreements and action programs. Congress and the state legislatures need to be informed of the intent to explore states/DOE cooperation, and their support requested for cooperative low-level radioactive waste disposal programs. The Federal regulators (NRC and EPA) need to be brought into the discussions in view of their regulatory responsibilities involving the states and DOE. Time is of the essence in this proposal because the states are pushing forward with their waste disposal programs, and remedial action decisions are being made for DOE sites.

BACKGROUND

At present, 45 states are members of ten (10) interstate compacts governing the disposal of low-level radioactive waste materials generated within their member states, i.e., radioactive wastes that are not generated by agencies/operations of the Federal government. The five (5) states that are not members of any compacts must "go it alone" in managing their radioactive wastes, unless they can either become compact members or arrange to use the disposal facilities of the compacts.

The compacts are an outgrowth of the Low Level Radioactive Waste Policy Act of 1980 and the Low Level Radioactive Waste Policy Amendments Act of 1985, which require that states take responsibility for disposing of their radioactive wastes. The laws also enabled and encouraged cooperative ventures among any states that could reach agreements, i.e., form compacts for this purpose, in the belief that the low-level radioactive wastes could best be managed on a regional basis. The result has been the ten compacts noted above.

Within each compact, one state has been identified as the "host" state, meaning that state would "host" the LLRW generated by all its members by constructing and operating a disposal facility for that purpose. The typical disposal facility is expected to accept the members' LLRW for a period of 20 years, after which a new host state would be selected.

STATUS

Despite the development of the compacts, the paths to achieving the desired ends - creating the necessary legislation within the states, particularly the host states, and determining precisely where within the host state the disposal facility would be located - have not been easy. In Ohio, for example, which is the host state for the Midwest Compact Commission, the enabling legislation was just introduced last November (1994), but with the knowledge that it would have to re-introduced when the new session began last month (January 1995). The legislator who introduced it in November said at the time that he was well aware that it would have to be re-introduced, but that he wanted to give his colleagues time to "gain a better understanding of a difficult issue" (1, p. 13).

Many state legislatures have proven to be unreceptive, even hostile, to the subject, as have local governments and citizens in areas identified as candidate sites. The major reasons for the resistance are the desire not to have radioactive waste disposal sites in their "neighborhoods" and, in some cases, opposition to disposal of radioactive wastes anywhere in the state.

Nationally, the result of these problems has been very slow progress, and despite

the existence of most compacts for many years, only two host states - South Carolina and Washington - have operational disposal facilities. Of the remaining host states, the earliest in the planned schedule are California and Texas, but both have suffered recent setbacks. California had expected its operations to commence in late 1994, but has encountered roadblocks in acquiring the necessary land, known as the Ward Valley site, located in San Bernardino County some 20 miles west of Needles. The land is owned by the U.S. Department of the Interior and transfer of title to California has been held up at the Federal level. No one can be certain just when the transfer will be made. Texas had anticipated starting operations late in late 1996 at a site in Hudspeth County. However, Congress was late in ratifying the Texas Compact and opponents of the facility stated that if and when the Compact is ratified, they will take their opposition to court.

Another key problem has arisen with respect to the planned Wake County disposal facility on North Carolina. This facility is planned to take over from the Barnwell site in South Carolina, which will be closed on December 31, 1995. The Wake County facility was expected to commence operations in early 1996, but has encountered difficulty in satisfying the Nuclear Regulatory Commission's licensing requirements. The result is that the license is now not expected before June 15, 1996. Even if that date is met, the facility could not begin operating until late 1997, so the Compact members are faced with a minimum 2-year delay in having a disposal facility available.

The "Barnwell problem" does not end there, however. That facility was widely used by low-level radioactive waste generators in many states outside the Southeast Compact, but was closed to outside states on July 1, 1994. With Barnwell access gone, these organizations are experiencing a squeeze in any activities that produce low-level waste, and are being forced to curtail many of those activities. Particularly affected are medical research organizations, and in such widely separated locations as California and New York, where research involving use of radionuclides is already being curtailed. These problems will only be exacerbated as delays continue.

In the remaining Compacts and states, disposal facilities are variously planned to become operational over the period of 1997-2001, but many of those plans are not at all firm. The key reasons are variously that the required legislation has not been enacted, disposal facility sites have not yet been selected, land has not been acquired, and facility designs are incomplete or non-existent. Table I indicates the currently planned schedules for commencement of disposal site operations for the various Compacts and non-Compact states (2). The primary insight to be gained from the information in Table I is that, with the exception of the Southeast and Northwest Compacts, none of the Compacts or independent states has acquired a low-level radioactive waste disposal facility. Bear in mind that the current Federal law has been in effect for 10 years.

THE OVERLOOKED OPPORTUNITY

Throughout all of this, it appears that the states, the majority of them at least, have overlooked a golden opportunity to avoid, or perhaps minimize, the difficulties. That overlooked opportunity involves the U.S. Department of Energy. The Department of Energy (DOE) has a number of large plant facilities that, over many years of operations related to weapons production, have generated large quantities of radioactive wastes that must be disposed of, and DOE has a major ongoing program to clean up these sites and dispose of the wastes. With the end of the cold war, most of those facilities have been, or soon will be, declared surplus. For this reason, not only do the wastes from earlier production operations need to be cleaned up, but the facilities themselves - vast arrays of buildings and equipment - also need to be cleaned up, in fact, eliminated. Their elimination will generate even greater quantities of waste materials to be disposed of, and the cleanup and disposal activities will be going on for many years to come. Major DOE facilities are located in 14 states, five of which are currently designated as host states (see Table II).

This situation gives rise to the notion that the states and the DOE, which currently cooperate not at all where waste disposal is concerned, are overlooking an outstanding opportunity to resolve several of the states' problems by reversing their course and cooperating completely in radioactive waste disposal. Department of Energy sites are typically very large in terms of acreage and most, if not all, of the major sites expect to have permanent waste disposal facilities located on their properties.

There appears to be little or no point in the states having to acquire property - several hundred acres each of uncontaminated property - remove it from the tax rolls, then contaminate it with waste disposal facilities, and be required to manage the property in perpetuity, when the wastes might be disposed of at the DOE sites, which are going to be managed in perpetuity as a responsibility of the Federal government. In addition, the remediation of DOE sites and management of their wastes will be taking place over a long period of time - 2019 being the nominal target date for completion of remediation - so the schedule for disposal of the states' wastes could well be dovetailed into the DOE's schedules.

A primary reason for the attractiveness of this concept of cooperation between the states and DOE is that the low-level radioactive wastes anticipated to be generated by the states will be very small in comparison to the volume of DOE wastes, probably less than 10 percent by volume, as will the facilities required for disposal of the states' wastes.

The status in the Midwest Compact offers an illustration of the proposed concept. Ohio is the host state for the Midwest Compact, whose six member states expect to generate some 1.5 million cubic feet of low level radioactive waste over a 20-year design operating life of the facility. At the same time, DOE has three Ohio sites (at Miamisburg, Portsmouth and Fernald) that have large quantities of waste for disposal. At the Fernald site, for example, one Operable Unit alone (of five Operable Units) has 50-60 times the low-level waste expected to be generated by the Midwest Compact.

While much of the waste at Fernald is planned to be disposed outside of Ohio,preliminary plans are that a substantial quantity is expected to be disposed in facilities that would be built on-site. The size and rate of construction of these facilities would not be severely impacted if the Midwest Compact's low-level wastes were to be included.* In any case, it is most unlikely that the Midwest Compact states' low-level wastes would exceed 10 percent of the DOE wastes that would be disposed of at Fernald. Other sites in the DOE complex are much larger than Fernald, and might be able to accommodate much greater quantities of civilian low-level wastes generated within the states.

Given the relative low-level radioactive waste quantities and disposal needs of the states and the Department of Energy, it certainly seems that the possibility of close cooperative actions is worth investigating. As indicated above, the benefits are numerous. The principal benefits are the following.

The states would not have to remove land from productive use and from the tax rolls;

The states would avoid the costs of acquiring land;

The costs of developing and operating disposal facilities, then perpetually maintaining the disposal facilities after closure would be avoided;

The DOE sites are already in existence and are large enough to accommodate the states' low-level radioactive wastes readily;

The DOE sites are planned to have on-site disposal facilities in any case; and

The states' low-level radioactive wastes would be managed properly and perpetually along with the DOE wastes.

Several actions are needed to bring about this proposed cooperation. The first is that responsible officials in the compact states need to get together with DOE representatives for concerted discussions to determine exactly what would be necessary in agreements and action programs. At the same time, Congress and the state legislatures need to be informed of the intent to explore the states/DOE cooperation, so that the legislators can begin to think about changes required in existing laws or what new laws might be necessary to enable and expedite the cooperative waste disposal programs. The Nuclear Regulatory Commission and Environmental Protection Agency need to be brought into the discussions in view of their responsibilities for regulating the states and DOE.

It is recognized that time is of the essence in this proposal because the states are pushing forward with their waste disposal programs, and final remedial action decisions are being made for DOE sites. In DOE's Environmental Management Program, important new emphasis has been placed on need to improve productivity - to do more with less. There is no time like the present to start the cooperative ventures.

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Session 23 -- Technology Transfer and Partnership Experience
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23-1
PREPARING COMMERCIAL OFF-THE-SHELF TELEROBOTIC TECHNOLOGY FOR DOE REMEDIATION NEEDS
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ABSTRACT
Over the past decade commercial nuclear power plants have turned to telerobotic technology for managing and reducing the accumulation of waste products from their plants. Additionally, this technology has been applied beyond the day-to-day housekeeping requirements to emergency response and special surveillance missions. As a result, a host of telerobotic equipment suppliers have developed specialized tools that respond to the decontamination and remediation requirements of these plants - both within the confines of the radiation protection area and around the perimeter of the plant. The theme of this paper relates to the fact that the Department of Energy (DOE) has the opportunity to tune their approach to many of their site remediation challenges by drawing from the lessons learned in the commercial application of telerobotic experiences, rather than focusing only on the conceptual ideas typical of many basic research and development efforts. This paper presents particular design features of equipment which may be applicable to DOE site remediation needs. Examples include the potential use of fuel pool decontamination equipment for reactor inspection and fuel pellet retrieval; the application of large sludge removal vacuum systems for DOE pond and basin inspection; the use of mollusk removal systems for use in site remediation; how to apply the experience of autonomous standby diesel oil tank inspection systems for DOE single wall tank inspection; and, the application of remotely controlled plant inspection systems for DOE site assessment and work monitoring. Examples of applications in the United States and overseas will be presented. The paper provides a summary of lessons learned as well as design and performance criteria relevant to particular DOE environmental restoration requirements. Particular benefits associated with the use of off-the-shelf technology for DOE applications will be highlighted.
INTRODUCTION
Over the past decade commercial nuclear power plants have turned to telerobotic technology for managing and reducing the accumulation of waste products from their plants. Additionally, this technology has been applied beyond the day-to-day housekeeping requirements to emergency response and special surveillance missions. As a result, a host of telerobotic equipment suppliers have developed specialized tools that respond to the decontamination and remediation requirements of these plants - both within the confines of the radiation protection area and around the perimeter of the plant. This paper presents a brief overview of many of the design issues and lessons learned from the application of telerobotic equipment to commercial power and process manufacturing environments which have relevance to the remediation needs of the Department of Energy (DOE).
Although there may be a need to develop comprehensive guidelines for transitioning commercial off-the-shelf technology to the needs of the DOE, this paper only summarizes some of the practical experiences gained through relevant commercial

applications. The material in this paper is primarily concerned with the use of remotely-controlled devices. The remotely-controlled component of this equipment is typically a tracked vehicle. Few examples are presented which require the manipulation of objects other than the occasional retrieval of small objects such as loose parts, fuel pellets, etc.

There are a number of valuable lessons to be learned from the application of off-the-shelf telerobotic equipment to commercial remediation needs. Over the past 14 years this author has been involved in the practical application of telerobotic technology to a variety of environmental remediation challenges. Much has been learned from these field experiences. What is clear is that it is now easier to draw on lessons from events directly experienced than from those which are conceptualized. The theme of this paper relates to the fact that the DOE has the opportunity to tune their approach to many of their site remediation challenges by drawing from the lessons learned in the commercial application of telerobotic experiences rather than focusing only on the conceptual ideas typical of many basic research and development efforts.

This paper is organized into five sections. The first section deals with some of the criteria used to evaluate the commercial resources needed to complete a job. The second section discusses specific design features which may be applicable to DOE site remediation needs. These design features are collected from projects associated with fuel pool decontamination, reactor inspections, basin/sludge and pond cleaning and inspection, tank inspection and cleaning, the removal of mollusks from intake structures and related confined spaces. The third section provides examples of commercial remediation projects in the United States and abroad. The final sections provide a summary of lessons learn in the field and a review of the overall benefits associated with the use of off-the-shelf technology for the needs of the DOE.

EVALUATION CRITERIA AND COMMERCIAL RESOURCES

The structure of a typical remediation job consists of several elements. These elements include processes which occur prior to deployment, or job planning, up to and including the post-project review.

Job Planning

Commercial site decontamination projects first specify the requirements of the project in sufficient detail to develop a Work Breakdown Structure (WBS). The WBS is the heart of the planning document which defines the structure of the project and the specific tasks, including program management functions and the evaluation criteria needed to determine both the progress of the project and the quality of the results. Each of the projects cited in this paper were initiated following the development of a WBS implemented with commercial off-the-shelf project management tools. In a similar manner, commercial software tools were used to plan each activity with regard to schedule and cost. In some cases critical paths were established. Some critical paths were based on dose thresholds and safety constraints, while others may have been based upon the availability of support equipment.

Project Operations

Project operations are usually structured into stages. Although each project has its own unique requirements, most telerobotic material removal projects share similar characteristics. The removal of material can take the form of simple vacuum procedures, or mechanical removal through the use of lifting devices or shredders. Material must be conveyed in a manner that is efficient and safe to both the removal systems and the environment. Mechanical conveying can be accomplished through the use of augers and conveyer belts or through a pumping process, as long as the proper media is available for conveying the material.

Pumping methods are well understood and there are a wide variety of systems available which can usually meet specific volume and speed criteria. Interim pre-filtration storage facilities are typically needed if the processing subsystems can not keep up with the material transfer process. Special holding tanks which can be shielded and built with secondary containments are usually found on such project sites. The filtration process is also an area that has matured to a point where there is usually a variety of options available for the process engineer. After the establishment of realistic processing goals, in terms of yield (i.e., tons per day or gallons per hour), material quality (i.e., per cent solid dry material, rad levels, chemical content), decisions can be made regarding the type of material separation approach or what stabilization technology must be used. Finally, a number

of material containment and handling options are available to meet most Federal and local requirements in this area.

Safety

OSHA has established comprehensive safety guidelines and requirements. Most projects easily incorporate these guidelines, significantly reducing safety risks. Problems can occur, however, in projects that have not been previously encountered. These projects are plagued with unusual situations, which challenge the applicability of OSHA safety standards. This is especially true for confined spaces, which is why telerobotic applications are so valuable for confined space operations.

Maintenance

Perhaps the most neglected aspect of a project is the maintenance of the equipment used to complete the project. Proper pre- and post-project maintenance of equipment ensures that the equipment is ready for the next application. Significant delays can result if equipment fails to perform during a project. Consequently, special emphasis is placed on the time and costs necessary for the refurbishment of such equipment.

Costs

A common characteristic of most well-run commercial projects is the emphasis on cost control. Each phase of the job is monitored to insure that progress can also be evaluated in terms of economics.

DESIGN FEATURES OF IMPORTANCE

After 10 years of commercial environmental remediation projects employing telerobotic devices, a significant body of knowledge has been collected relative to important design features. In all cases the major lesson learned is that the more thorough the assessment of the requirements, the fewer problems encountered in the field. Below are listed many of the critical design features and their relevance to particular job requirements:

Mobility / Traction / Drive. A number of important tradeoffs must be considered in the area of vehicle transport. They include issues related to vehicle contact pressure, vehicle size and weight (i.e., whether or not the vehicle must operate on a geothermal liner, clay, or concrete). Of equal importance is the design of track pads relative to the traction requirements established by terrain conditions. There are also tradeoffs to be made relative to the power requirements (i.e., torque) and tractive forces needed versus umbilical size and weight. The vehicle must have sufficient power to do work and pull its own weight.

Accessibility. The use of telerobotic devices usually presents challenges relative to deployment. As a result, the equipment must be shaped and sized to fit access limitations in a way that also insures proper handling. In the commercial environment there are also limitations of the number of personnel available to the site. This may be due to a number of factors which may include exposure budgets, critical paths, and cost limitations. Therefore, special emphasis must be placed on accessibility issues which clearly address these limitations.

Control and Guidance. Inherent in the use of telerobotic devices is the need to know where your equipment is in confined spaces without being there. Control and guidance devices and methods must reliably provide immediate feedback so that you can determine where you are, where you need to be, and what you have accomplished.

Viewing and Control. Remote viewing in most environments is difficult, at best. This is especially true during submerged operations. Much has been accomplished in the area of mapping systems, optical and acoustic imaging, infra-red and similar nonvisible spectrum. However, remote viewing and control creates a host of unique performance and safety issues. Vision and control requirements are unique to each project, therefore, the system design must incorporate these unique characteristics.

Performance Measurement. The future actions taken by the operator depend on the quality of information related to the material processing rate of the equipment. Although certain circumstances do not allow for the real-time assessment of progress, most remediation project schedules can only be met or adjusted through timely operator performance feedback.

Cleaning / Material Removal. For cleaning applications, the nature of the material determines the selection of the pump and pump powering approach (hydraulic, electric, pneumatic). Similarly, the characteristics of the material determine how it is removed (suction head design) and whether or not the material needs to be conditioned prior to pumping (shredding). After the best method of surface removal and pumping has been selected, the appropriate hose must be selected. Important hose

characteristics include not only size, weight and flexibility, but also design issues related to interior and exterior surface texture and hose buoyancy.
Deployment / Recovery. One of the most overlooked elements of telerobotic remediation projects is the need for appropriate equipment handling devices such as cranes, come-alongs and lifting tripods. A wide range of off-the-shelf devices are available for equipment handling and can easily be configured for different terrain conditions.
Decontamination. Standard techniques used throughout the commercial nuclear power industry and chemical industries are applicable to many of the site remediation requirements of the DOE. It is clear that, in many cases, there will be obvious limitations in the effectiveness of some of these techniques. Consequently, permanent hot storage of the equipment may be necessary. This is sometimes the case in the commercial environment and, as a result, suitable methods for the temporary storage of equipment are available. Regardless, it is important to design and fabricate equipment which can be decontaminated.
Material Compatibility. The design of telerobotic equipment must take into consideration the material in the working environment. Chemical compatibility, heat resistance, pH and the corrosiveness of waste must be addressed during each phase of the design process. Additional concerns are those of the thermal stability of the material, its radioactivity and flash points. A number of lessons have been learned as a result of the loss of steel pump impellers, the breakdown of critical neoprene seals, loss of rubber tracks and the severe pitting of aluminum structures.
Handling System. An often overlooked element of system design is the management of umbilicals. The full operational weight of the umbilical must be calculated (i.e., pressurized hydraulic lines, electrical lines, full material discharge hoses, etc.). If a method is not devised to relieve the vehicle from the stresses and drag created by the umbilical ,then the tractive forces of the vehicle must compensate for this additional burden.
Laydown Area. The design of telerobotic systems must consider the site assembly and deployment requirements. Oftentimes, machines must be dismantled and reassembled inside the work area. This is true of many tanks and sumps which were originally designed for limited access. The laydown and assembly areas must take this need into consideration. The same is true for the special handling requirements of larger machines where the laydown and preparation areas can require a significant amount of site real estate and involve a complicated arrangement of contamination control.
Personnel Entry Limitations. Although the primary objective with the use of telerobotic equipment is to remove the operator from the material hazards, there remain occasions where some degree of manned entry into the work area may be required. This still represents a critical design evaluation issue and the equipment must be designed with the appropriate man-machine tradeoffs to insure the safe operation of the system. This requirement may also include the use of appropriate oxygen sensors, LEL meters and similar remote monitoring sensors.
Support Equipment. A final critical element of the system design is the interface between the telerobotic system components and their support systems (i.e., compressors, hydraulic power units, cranes, etc.). Not only must this equipment be sized to meet the particular performance requirements of the system, it must also be incorporated safely into the job site and configured to minimize contamination, meet all site safety requirements, and be configured to maximize the use of available space.

APPLICATIONS

The previous section briefly highlighted specific telerobotic system design and project features which, through experience, have been determined to be critical to the success of a remediation job. The following paragraphs summarize various projects which are representative of typical commercial applications of telerobotic equipment. In order to provide an interesting cross section of projects, examples were included from industries other than commercial nuclear. Although many of the sites have unique remediation needs, they all have the following features in common: 1) they all require operations within or around materials which are considered hazardous; 2) the equipment used was deployed to remove the operator from the hazardous work environment; and, 3) specific production and safety goals were established.

Nuclear Plant Outage Support

A telerobotic method of cleaning a reactor vessel flange was used at a commercial

nuclear power plant in the Northeast United States during a recent outage. After fuel was replaced in the reactor, and the upper guide structure installed, divers would typically enter the cavity to perform cleaning. Due to the movement of the fuel, there can potentially be radioactive hot particles on the cavity floor creating added hazards to the diver. A telerobotic device was deployed into the cavity during and after fuel movement to remove radioactive debris and hot particles, significantly decreasing the chance that the diver would come in contact with the hot particles. Additionally, during reactor refueling, a stray bolt was found in the reactor core. The bolt was identified to have either come off of the bottom of the upper guide structure, or off of the suction deflector of one of the reactor coolant pumps. There were four reactor coolant pumps, and a study of all vibration data could not determine which pump was the source of the missing bolt. A telerobotic inspection system was deployed and sent further down the cold leg to inspect the bottom of the reactor coolant pump where the suction deflector was located. The system was able to identify the source of the missing bolt on the reactor coolant pump. Had this approach not been successful, the crews would have been forced to pull and inspect each individual pump in an effort to identify the source of the missing bolt. Use of this approach saved a potential 40 man-rem exposure.

Water Treatment Plant Cleanup

A buildup of alum sediments and water purification filtration materials such as polymer additives, carbon, and sand accumulated in the neutralization sump of a water treatment plant. The sump was 50 ft. x 50 ft. and 22 ft. deep. There were 5 support columns, spaced at 17 ft. and 11 ft., at certain intervals in the center of the sump. There were two 4 ft. x 4 ft. openings to the sump used for equipment entry. The sump pump was located on the same wall as an emergency sump, but in opposite corners. There were 2 mixers located in the central area of the sump that extended from the ceiling to within approximately 3 ft. from the floor. The accumulated sludge was 17 inches and the estimated volume of sludge was 3,052 cubic feet.

The backwash recovery sump contained an accumulation of the same sediments as those found in the neutralization sump. There was a 6 ft. slope from both sides to one corner just below the two 4 ft. x 4 ft. side-by-side manway openings. The accumulated depth was approximately 7 ft. at the deepest part. There was an estimated 4,500 cubic feet of sediments. A polyurethane liner covered the entire area of the sump.

A telerobotic cleaning and surveillance system was used to vacuum sludge materials from the sumps and transfer the material to a filter press dewatering system that was located outside the water treatment buildings. The sludge was transferred from the sump through a suction hose to the 6" diesel powered pump located outside the area, then through a transfer hose to frac tanks located near the filter press dewatering system. Water was decanted from the tanks and then concentrated materials transferred to the filter press for dewatering. The filtrate liquid from the press was returned to the sump and dried materials were deposited into roll-off containers for transport to the disposal location.

Suppression Pool Cleaning

A suppression pool of a reactor (GE Mark III Containment BWR-6 Reactor) collected debris which was contaminated from a variety of sources throughout the plant. The pool was approximately 550 square meters and contained contaminated sediment which had collected on the bottom of the pool cavity, as well as some solid debris. A telerobotic system was deployed onto the bottom of the suppression pool cavity and vacuumed sediment through a 2" neutral buoyant suction hose to a submersible pump. Waste was pumped through a 2" transfer hose into the central fuel pool demin system. This direct transfer of material to the plant's demin system eliminated the handling of highly radioactive filters. In addition, a high pressure water jet attachment was used to remove contamination which was imbedded in hard to access areas of the pool floor. This material, plus the sediment removed by the system's rotary brush attachment, was vacuumed by the vehicle as it was directed across the stainless steel floor liner.

HVAC Decontamination

A government supply facilities used an insect repellent in the early 1970s for the treatment of clothing that was being manufactured for overseas personnel. The use of this repellent resulted in the contamination of the heating and ventilation and

cooling (HVAC) system in a large manufacturing facility. The chemicals accumulated in the dust and dirt sediment that normally builds up in HVAC systems. There was a need to perform a cleaning operation on the inside of the ducts in a manner which would protect the workers and minimize the amount of hazardous waste generated from the cleaning procedure. The cleanup requirement was to achieve an atmosphere concentration of contaminated materials that was less than 10 micrograms / sq centimeter. The building had several miles of HVAC duct work of varying widths. The project required the development of an approach to the cleanup operation that would achieve the safety and waste reduction goals established by the client. A telerobotic vehicle was equipped with full pan and tilt color camera and lighting systems for viewing, navigation and recording inside the duct work. Additionally, an air sparge system that operates at 150 psi at 100 cfm and a rotating power spray head operating at approximately 3,000 psi were specially designed and fabricated for the project.

The job specification included strategic insertion of the air sparge vehicle coupled with drawing negative pressure on the system with a large hepa vac system. A thorough cleaning was accomplished that exceeded specification requirements both in terms of cleanliness and time required.

Zebra Mussel Removal

A power plant on the Great Lakes had a severe buildup of zebra mussels in the intake structures of the service water pump forebay. The forebay was an area approximately 200' x 60' and 40' deep. This project was designed to 1) demonstrate the ability to deploy and retrieve a modified a telerobotic vehicle in the inlet and screen bays; 2) remove the accumulations of zebra mussels and other pumpable material from the floor; and, 3) reduce or eliminate the need for divers, thereby reducing the overall cost and danger of removing accumulations of zebra mussels.

A telerobotic system was modified from a standard off-the-shelf system. Mounds of zebra mussels, 8 to 10 feet in height, were encountered and successfully removed. There was a large accumulation of mussels where the vehicle was introduced into the screen bay, and the vehicle was capable of working its way through the mound to the bottom, and then maneuvering to remove the remainder of the mound.

Parts Retrieval

While disassembling a service water pump during a refueling outage, extraneous materials were inadvertently introduced into the reactor coolant system. The situation was complicated by extremely high radiation levels. With use of an on-board color video system mounted on a telerobotic vehicle, all foreign material was located. Three of the five items were recovered during initial deployment activities. On-site modifications consisting of a pneumatically-controlled gripper were made which enabled the operator to remove the remaining two items that had become lodged in a small, limited access drain line. Replacement of the pump continued on schedule with significant radiation dose savings.

Basin Sludge Removal

To comply with the client's deadline to re-commission wastewater treatment fossil plant basins, 465,000 gallons of sludge had to be removed from the plant in less than three weeks. The wastewater treatment system consisted of three HDPE lined settling basins. The primary and polishing basins held 500,000 gallons each and the chemical waste treatment basin provided capacity for an additional 175,000 gallons. A telerobotic vehicle was used in conjunction with a 6" Dri-Prime centrifugal pump for removal of the material from the basins. After removal, the sludge was transferred into a 21,000 gallon tank for concentration and decant of clear water.

An off-the-shelf telerobotic system was deployed at a European nuclear generating station which had two cooling tower basins which were 284' x 408' x 16' deep. The total volume of the basins was in excess of 14 million gallons of water. Every few years there was an accumulation of sediment which included phosphates, river silt, and local ground runoff. Additionally, high levels of algae growth and other organics were measured in the basins. Sludge accumulation was approximately 1' to 3'. Polymers had been added to the basin which resulted in material compositions ranging from loose & fluffy sediment to thick / heavy jelly-like sludges. Obstructions in the basin consisted of 3 polymer additive / sparger lines running length-wise in the basin approximately 18" off bottom - supported by concrete blocks every 6'.

A standard off-the-shelf telerobotic system equipped with a 3", 300' suction hose and pan & tilt color video was used to remove the sludge material. The vehicle was

driven along the bottom, vacuuming sediments and the waste was transferred to the on-shore pump. The sediments were transferred to settling tanks, loose water was decanted off the top, chemical additives were provided and concentrated material was then pumped to a mobile filter press unit located on-site. The material was dewatered, the water was returned to the basins and the dry cake collected and disposed of at a landfill.

The submersible color pan & tilt camera was mounted on the vehicle and used for navigation of the vehicle along the bottom, observing the cleaning process, inspecting the cooling tower intake / pump pit / screens, and for post cleaning verification. Previous cleaning methods required the cooling system to be shut down and all 14 million gallons of water drained. Personnel were required to enter the basins with hoses and pumps in order to flush the sediments down a common drain which emptied into the local river. This traditional approach was a labor intensive method which was considered to be one of the high risk jobs at the plant site. Earlier records in the neighboring town suggested that there was a high correlation between the draining of the basins and the increase in toxins in the neighboring water supply. Livestock and produce were considered to be threatened by this traditional process. Use of the remote acquisition and processing approach eliminated any threat to the environment in a cost-effective and timely manner.

Chemical Plants

A buildup of carbon disulfide (CS2) contaminated sludge in a waste collection tank created an operational problem for a pharmaceutical company. The sludge was approximately three feet deep and was covered with a thick layer of brine. The sludge had to be removed without exposure to the air due to the exothermic nature of the material. Simply disturbing the material on the bottom of the containment tank by hand-raking produced dangerous off-gassing problems. These vapors are flammable and immediately dangerous to health. Conventional removal attempts, such as the use of vacuum trucks, were deemed to be both dangerous and inadequate. The combined safety hazards of the material dictated the need to minimize or eliminate the use of personnel at or near the dike.

A standard telerobotic vehicle was modified with parts that were both compatible with the sludge and intrinsically safe. The project was designed to allow the telerobotic vehicle (equipped with a hydraulically driven submersible pump) to maneuver in a set pattern around the bottom of the dike to transfer sludge to two tank trucks parked nearby on the RCRA pad. The CS2 contaminated material was successfully pumped from under a cover of brine in the dike, to a brine-covered condition in the tank trucks. The operation lasted approximately three weeks. Pumping was accomplished in stages, allowing the material to settle under the brine in the tankers, decanting excess brine, and repeating the procedure until the sludge in the dike was down to minimal levels.

Mining

A particular copper mining site had two mill water recovery sumps. Each sump was 74' in diameter and approximately 15' deep. Each sump was filled with about 5' of extremely abrasive fine powder sediment (pH 12) which, when dry, had the consistency of near solid concrete. These sumps had not been cleaned for over 20 years and, due to the critical water recovery role for the milling process, could not be taken out of service. If these sumps were taken out of service, the entire milling process would come to a stop.

To avoid a forced shut down, an off-the-shelf telerobotic system was modified by replacing particular components with materials resistant to corrosive sludge. In addition, an auger cutting head attachment was installed on the telerobotic vehicle and deployed into the sumps. The hard sediment was broken up with the auger and vacuumed out of the sump through a 4" hose to a 6" diesel pump. The materials were then returned to the plant process system through a center well. The clean-up operation was completed without impacting normal plant operations, and the water recovery pumps remained in service. Maintenance records following this cleaning operation have reported reduced maintenance costs for the water recovery pumps to be more than 70 percent.

LESSONS LEARNED

Many of the lessons learned from the commercial site remediation jobs have been summarized below. In most cases these lessons emerged from design or process problems which were unforseen even though adequate planning was thought to have been demonstrated.

Requirements. Make sure that all requirements are documented and well understood. Do not base the entire remediation operation on drawings which are not current or not representative of the actual work site. If the area is difficult to access, then the representative drawings may be less than accurate for that very reason. This is especially true for tanks and poorly maintained basins.
Traction / Power. The original power and tractive force requirements change depending on the weather and the material composition of the job site. Remotely-controlled vehicle tracks, if not configured correctly, will throw tracks and lose traction due to their deterioration.
Performance Feedback/Productivity Indicators. Without any measure of progress, there will be little sense of value. Measurements of productivity (i.e., material transfer rates, volumes, radiation levels, etc.) must be either available continuously or calculated within a reasonable period of time. Information must also be available to the operator to determine or predict hose clogging, blade loss, binding, and similar performance problems.
Reliability. Select components which are proven, especially for critical path jobs. Pilot testing, especially for one-of-a-kind designs, must be considered an essential element of the project plan. Radiation tolerance of critical components such as cameras, electronics and sensors must be determined beforehand and an operational schedule developed to prolong the useful life of these components.
Robustness. Equipment which is big and heavy is not enough. Design for robustness and incorporate forms that contribute to structural integrity without adding unnecessary weight
System Configuration. Design, fabricate and select components which must operate as an integrated system. Processing and material hauling systems must have the capacity to keep up with the material removal pumping system. This is especially true for filter press operations or decanting associated with high integrity containers.
Cost. Without cost boundaries there is reduced incentive to do the job correctly.
Safety Margin. There is safety in numbers. Design and prepare for the worst reasonable set of circumstances. Most successful commercial projects have taken this into consideration. Elegant solutions are not as important as those that meet all of the requirements at the most reasonable cost.

OFF-THE-SHELF BENEFITS

The previous discussion relative to lessons learned has been gathered directly from field experience. Many of these lessons have been incorporated into a number of telerobotic devices which are in use today. Although these designs may not address all of the specific needs of the DOE, many of their features are relevant to some of their remediation needs. Aspects of off-the-shelf benefits which deserve to be highlighted are summarized below.
Commercial field proven equipment is designed to produce results. There are strong incentives for commercial operations to develop remediation solutions that can sustain a business beyond the first project. In terms of cost, many of the commercial vendors of remediation equipment have gone through the expense of determining what works and what doesn't. Although some research and development may be necessary, these organizations have strong incentives to employ novel solutions and complete projects with limited expenditure of resources. Remediation equipment can be found successfully working in some of the most harsh environments such as the ocean, mining operations and the petro/chemical industries.
Another benefit of off-the-shelf commercial remediation equipment is its availability. Although some research and development may be necessary to modify an existing piece of equipment, there are off-the-shelf components which can be used as the basic building blocks for field systems. Another benefit of existing off-the-shelf equipment is its adaptability to a variety of remediation missions. We have found it useful to incorporate design features into certain mobile systems that would have utility in a variety of missions. The decision to use particular types of materials that are impervious to a variety of environments may justify the additional initial cost.

TWO PERSPECTIVES ON A SUCCESSFUL LAB/INDUSTRY TECHNOLOGY TRANSFER

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ABSTRACT
Technology transfer from government laboratories to private business is of increasing concern in today's marketplace. Some prospective partners (on both sides) believe that technology transfer is a relatively simple process requiring little or no extra effort from the participants. In our experience this is not true and, in fact, positive results from a collaboration are directly proportional to the effort that both parties invest in the relationship. Communication, both between prospective partners before an agreement and between partners following the agreement, is essential. Neither technology nor marketing can stand by itself; the combination of the two can produce a useful and available product. Laboratories and industries often have very different ways of looking at almost everything. Misunderstandings arising from these differences can short-circuit the transfer process or result in the production of a product that is unsalable. We will cover some of our experiences, potential problems, and our solutions.

BACKGROUND
The long-range alpha detection (LRAD) technology and several applications were developed at Los Alamos National Laboratory (LANL) with DOE support over the past four years. This technology, and the five current patents covering applications of continuous LRAD detectors, are the subject of both CRADA and licensing agreements between LANL and Eberline Instrument Corporation. Eberline is currently marketing an object monitor, the LRAD-1, based on the airflow LRAD concept discussed below. Several other LRAD applications are currently under consideration as future commercial products.

TECHNOLOGY
Sensors based on the LRAD concept (1,2) detect the ions produced by alpha particles in ambient air rather than the particles themselves. Thus, unlike traditional detection methods, LRAD-based monitors are not limited by the range of alpha particles in air. The lifetime of the ions is dominated by wall collisions so that the ions live longer in larger volumes; in a small pipe the ion lifetime (3,4) is greater than 4 s, allowing for a much longer range than the several centimeters typical of an alpha particle. The LRAD-based sensors do not require thin windows, fine wires, or specialized gases, so they are rugged, reliable, and portable. Some applications dependent on one or more of these features are discussed in Ref. 5.
The group at LANL has built two broad types of LRAD sensor. An electrostatic detector (4) uses an electric field to sweep the ions onto a detection plate. The ion current is directly proportional to the amount of contamination located under the detection plate. Electrostatic detectors are best suited for measuring contamination on relatively flat surfaces such as soil (6), concrete, floors, or liquids. In an airflow detection system (4), the ions generated inside an enclosed volume are transported by an air current produced by a fan. The air passes through a charged grid that separates the ions from the ambient air. Again, the current from this grid is proportional to the amount of contamination. Airflow detectors are suited for monitoring complex objects and tools or for contamination inside a closed volume (such as a pipe, duct, or piece of process equipment).

FIRST STEPS
Technology transfer agreements such as CRADAs work best when they are industry-driven. The researchers can (and should) develop a technology until it is interesting to industry, but trying to "sell" a technology to an industry that doesn't want it is very difficult. If there is no interest, more research is probably more useful than a hard sell.
Are the expectations and goals of the researchers and the commercial enterprise compatible? Total agreement between prospective partners is unreasonable to expect, but an understanding of these goals is important. The two partners are driven by very different desires, and each needs to understand where the other is "coming from."
A concern of the research partner is whether the company can do the job and the company's concern is whether they can match the proposed product to a real market. These concerns can be combined into the question (asked of both parties) as to whether the company will do the job. If the answer is no, then either the "job" or one partner needs to be redefined. The researchers need to realize that their definition of the "job" may not be commercially viable, and the company needs to understand that they may not be the best outlet for every technology.
The desires of the end-users for the technology need to be considered as well as

those of the prospective partners. It is possible for the researcher to know more about segments of the market than the prospective partners; in any event, it is advantageous for the researcher to understand the final market and demand to help them transfer technologies that are most appropriate.

Seeking out potential end-users/customers who are willing to take a risk on the new development is essential to the long-term success of any technologically innovative product. Their "real world" problems may not be understood or easily duplicated by either the researcher or commercial partner. Thus, the experience gained in dealing with these users is a valuable part of the commercialization process both before and after signing an agreement.

CONTINUING DIALOG

Continued communication between partners is important for continued growth of the partnership. It is very easy to "each do your own thing" without any effort to know or understand the other partner. It is important to understand what the other partner's intentions are, not just to hear the words they're speaking. In particular, researchers are often surprised at the length of time it takes to develop a new product or marketplace. Most conflicts arise from misunderstanding intentions and goals rather than intentional discord.

The common goals of both partners should be recognized and strengthened. In any partnership there are inevitable disagreements and conflicts. There are common interests between the CRADA partners (because both parties agreed to work together); emphasizing these rather than the inevitable differences makes for a productive working arrangement. We don't mean to say that there should not be differences in outlook, but it is often more profitable to work on the areas of agreement than the areas of disagreement.

Laboratory researchers and commercial enterprises each have access to information sources that are unavailable to the other. Utilizing these resources jointly gives the partnership access to users and distribution channels that would not have been available to either one individually. We have derived significant benefits from this identification of mutual points of support.

It continues to be important for both parties to communicate with potential end-users. For many emerging technologies, these initial customers are investing considerable time and expense in an unproven technology. In a very real sense, these customers are a "hidden" third partner in the agreement. The customer relationship need not be a purely sales relationship (although sales can and should take place), but rather a two-way discussion, so that the developed product(s) match a true (rather than perceived) customer need. Although this is often perceived as the responsibility of the commercial partner, continuing knowledge of the end uses for a technology can be very important for the direction of research efforts.

MARKET REALITIES

Introducing a new technology to the marketplace can take longer than anticipated. Actual demonstrations are often required to show that the technology performs as advertised. This sluggishness in acceptance can make successful transfer slower than anticipated. The researchers and commercial partners can effectively display the technology's capabilities in scientific conferences and sales demonstrations respectively.

The challenge of introducing a new technology is compounded if the appropriate market is in its infancy and also requires development. Many diverse technologies will be attempting to acquire a piece of this developing market. The fact that a technology is "new" and "innovative" does not instantly guarantee commercial success. In a developing market, the partners need to identify not only traditional competitors and competitive technologies, but other expanding innovative solutions. Effectively dealing with this infant market, and the many new technologies it creates, requires an even closer collaboration between the partners.

Some technologies (such as the LRAD) lend themselves to multiple, diverse applications. Although this may look like an ideal transfer opportunity, it can lead to a number of potential strains between the partners. Researchers, eager to develop many applications, may not understand the inevitable time lag between product development and commercial sales. On the other hand, the commercial partner may not be aware of opportunities outside of its traditional markets. Good communication between partners is also very important in a multiple application/market situation.

COMMON MISCONCEPTIONS

The newfound desire to "commercialize" technologies has led to a number of

misconceptions about the process. We have included four of the most common ones and some comments on each.

"Technology Transfer Agreements are Simple"

Two common sentiments that are quite appropriate to this issue are "nothing worth doing is simple" and "you get what you pay for." If your goal is to produce an agreement quickly, it can certainly be done, but this agreement may not have been carefully thought out by both parties. Time spent in understanding the issues before an agreement is signed is well rewarded by saving (a much larger amount of) time trying to negotiate conflicts in a working relationship.

"The CRADA is a Final Goal"

A CRADA or other document may be legally necessary to proceed with a partnership but its existence should not be perceived as a goal in itself. The goals are (for the researcher) continued funding and development of an idea and (for the commercial partner) sales of devices resulting from the partnership and market acceptance of these devices. Without these final sales, documents do not accurately reflect the success of the commercialization effort.

"We Don't Have to Communicate after the CRADA"

It is often all too easy to create an "iron curtain" between ostensible partners after an agreement is signed. A common feeling among researchers is that once a CRADA is signed, all their problems are over. Similarly, the industrial partner often feels that once they receive a basic idea, all exchange should be over. In our experience, neither feeling is true: the parties can assist each other in future development. The same characteristics that caused both parties to agree in the first place can and should be used for further benefit to the partnership.

"A CRADA is a Funding Source"

Often, a technology transfer agreement or CRADA is viewed as a funding source for an otherwise unfunded development project. Why should a researcher expect commercial support for a project that no other users or funding agencies want enough to fund? A technology that can generate interest and funding on its own can do even better with the prospect of commercial involvement, but a technology that users don't want is going to be very difficult to commercialize.

CONCLUSIONS

All of our previous discussion can be summarized by the word communication. Often the researcher, the commercial entity, and the prospective end-user are all talking at each other but not to each other. The goal of partnership agreements (for all three of the interested parties mentioned above) is successful production, distribution, and use of a product. This can only happen if each of the parties understands the needs and requirements of the others.

ACKNOWLEDGEMENT AND DISCLAIMER

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A LAYERED APPROACH TO TECHNOLOGY TRANSFER OF AVIRIS BETWEEN EARTH SEARCH SCIENCES, INC. AND THE IDAHO NATIONAL ENGINEERING LABORATORY

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ABSTRACT

Earth Search Sciences, Inc. (ESSI) has been selected as a model for the layered approach to technology transfer of the Airborne Visible Infrared Imaging Spectrometer (AVIRIS) instrument and acquisition of supporting remote sensing technologies. ESSI has developed a layered approach to a costly technology development program that fully leverages diverse resources while retaining ownership, technical application and control, and management direction.

Since initial contact between ESSI and the Idaho National Engineering Laboratory (INEL) in February, 1994, at least seven proposals have been submitted in response to a variety of solicitations to commercialize, improve, and use the (AVIRIS) instrument and other remote sensing technologies. These proposals, matching ESSI's unique position with respect to agreements with the National Aeronautics and Space Administration (NASA) and the Jet Propulsion Laboratory (JPL) to utilize, miniaturize, and commercialize the AVIRIS instrument and platform, are combined with the applied engineering of the INEL. Teaming ESSI, NASA/JPL, and INEL with diverse industrial partners has strengthened the respective proposals. These efforts carefully structure the overall project plans to ensure the development, demonstration, and deployment of this concept to the national and international communities.

The objectives of these efforts include:

- developing a miniaturized commercial, real-time, cost effective version of the AVIRIS instrument
- identifying multiple uses for AVIRIS
- integrating the AVIRIS technology with other technologies
- gaining the confidence/acceptance of other government agencies and private industry in AVIRIS, and
- increasing the technology base of U.S. industry.

History

The initial contact between ESSI and INEL began in February 1994, as ESSI was involved in discussions with the Idaho Department of Commerce (IDOC). ESSI, a small business within the state, approached the IDOC with a need for assistance from a large institution with the capability to develop a miniaturized version of the AVIRIS instrument that could be deployed on a commercial airborne platform. ESSI had an agreement under the Space Act to utilize the AVIRIS instrument for mineral exploration since 1987. ESSI had been working as a research and development partner with NASA and JPL to improve the instrument, verifying and validating the results of the overflights as they were made. ESSI also concluded an agreement with NASA/JPL to re-deploy the AVIRIS from the ER-2 flight platform to NASA's C-130, fuse the AVIRIS data with that of other sensors on the C-130 platform, and to miniaturize and commercialize the instrument for use throughout the world. With an array of industrial partners in the U.S. and from other countries, ESSI assembled a world class team of potential partners who can provide unique capabilities for applications. TRW, Inc. joined the group in May 1994. In June, 1994, Tetra Tech, Inc., joined the ESSI and INEL efforts, adding their significant remote sensing and environmental reputation and expertise to the efforts. Tetra Tech is rated seventh of the top 200 small businesses in the United States by Forbes Magazine. Lockheed and several other firms followed Tetra Tech in October 1994, forming a team to use AVIRIS in Kazakhstan as a major cornerstone in the United States policy to convert defense industries to commercial entities. This has been followed by the U.S. Environmental Protection Agency participating in a demonstration of the technology for environmental assessment and monitoring of large areas in place of traditional sampling methods. U.S. commercial and agricultural industries also expressed an interest in using the same technology to assess areas prior to purchase and to monitor existing operations. Today, the education community is attracted to the field of remote sensing and is using AVIRIS as a technical cornerstone to establish an international Remote Sensing Center of Excellence accompanied by graduate degree

programs and a dedicated testing center in excess of 4,000 square miles at the University of Idaho.

Teaming

As a small, publicly-held business, ESSI has limited funds to bring to the venture which requires in excess of $10M to develop, test, and demonstrate a commercial version of the AVIRIS. Seeking funds from other partners or through joint solicitations with other federal agencies and federal entities is the primary means of securing capital. Signing Memorandums of Agreement (MOA), negotiating Cooperative Research and Development Agreements (CRADAs), and concluding Teaming Agreements with government and private industry has been the preferred route for ESSI to protect the intellectual property rights to the technology. These arrangements have increased the leveraging of funds and expanded the potential business opportunities. Business plans between team members are used to share the division of labor and profit.

PROPOSAL SOLICITATIONS

Advanced Projects Research Agency (ARPA) Technology Reinvestment Project (TRP)

This proposal was initiated to develop dual-use hyperspectral instruments for environmental, agricultural, mining, forestry and specific Department of Defense applications requiring unique signature identification. This $6M proposal involves the design and construction of a real-time, miniaturized, commercial prototype hyperspectral instrument and developing a new family of hyperspectral instruments with industrial firms, including: TRW, Inc., Earth Search Science, Inc. (ESSI), National Aeronautic and Space Administration (NASA), Jet Propulsion Laboratories (JPL), WJT Software, Integrated Spectronics PTY, CALTECH, and EG&G Idaho. A MOA between TRW, Inc, Earth Search Science, Inc., NASA/JPL, and EG&G Idaho, Inc., was concluded between the partners. Upon award, a Teaming Agreement and a Business Plan will be produced.

Rocky Flats Request for Proposals

This request resulted in another proposal entitled "Aerial Multispectral Sensor Platform for the Detection of Rocky Flats Hazardous (including Radiological) and Toxic Wastes". This $1.4M effort proposes to characterize both hazardous (including radiological) and toxic wastes surveyed from an airborne multisensor platform at hovering altitudes, mid-level altitudes (30,00 ft.), and high altitudes (>60,00 ft.). The airborne platforms will be enhanced with non-radiological sensing capability, including, but not limited to, thermal imaging to detect vegetal stress, seeps, and other hydrological features; laser induced fluorescence; and electromagnetic imaging/magnetometer/ground penetrating radar coupled to video to provide better arrays and overcome the limitations of separate sensors. Data fusion, data compression, and data management are project keystones. All data collected in the flyovers will be fused into a geographic information system (GIS) data base. These data will be used in the site-wide management information system affecting the environmental remediation of Rocky Flats Plant. A MOA was concluded between the partners. When an award is announced, a CRADA with the INEL will be executed.

Environmental Protection Agency (EPA) Environmental Technology Initiative (ETI)

This is a $700K proposal to use the Fort Hall Indian Reservation as a demonstration site to evaluate the current monitoring capabilities and development needs of hyperspectral instruments used for environmental purposes. The technology demonstration will also be used to identify the extent of recurring pollution on the reservation (nitrates, ethylene dibromide, and assorted pesticides and herbicides). Non-point-source pollution cannot be adequately addressed by traditional sampling methods. Partners include the Shoshone-Bannock Tribe, the INEL, EPA, NASA/JPL, and Earth Search Sciences, Inc. (ESSI). EPA funding is being leveraged with ESSI providing $350K. An MOA was concluded by the team to demonstrate the airborne multisensor platform and train Shoshone-Bannock Tribe personnel to use GIS to manage the reservation resources. When an award is announced, a CRADA will be executed.

EPA Requested Flyover of the San Jacinto River

In October 1994, EPA requested ESSI's assistance to use the AVIRIS to assess the environmental impact of a gasoline pipeline break in the Houston, Texas area. With demonstration of the technology for environmental purposes over large land areas, EPA has indicated that remote sensing may be substituted for traditional, expensive and time-consuming sampling procedures. This may include monitoring the environmental status of the Gulf of Mexico coast, the U.S.-Mexico border area, and other areas identified by EPA.

Retrofit of the AVIRIS to the NASA/JPL C-130 Multisensor Platform

In October 1994, ESSI obtained JPL support to retrofit the AVIRIS instrument to the NASA C-130 aircraft. This will enable the AVIRIS instrument to collect 5 by 5 meter pixel data flying at 5,000 meter AGL, a 16-fold increase in spatial resolution over the resolution available currently with the ER-2 aircraft.

International Remote Sensing Center of Excellence

The international Remote Sensing Center of Excellence's mission is to support watershed/river basin management as it affects all biota. The University of Idaho program is planned to be staffed by both university staff and remote sensing experts from industry. Idaho was selected because the state is rich in potential sensor services users and developers and offers a natural laboratory, in excess of 4,00 square miles, for sensor platform testing and evaluation. The potential client base includes agriculture, range management, fisheries, mining, forestry, wildlife management, and human engineering.

Defense Nuclear Agency Defense Industry Conversion Solicitation

This $11M proposal is to use hyperspectral and multispectral imaging to explore for minerals, perform environmental assessments, manage national industries, such as agriculture and forestry, natural resource management, and land use planning for the Republic of Kazakhstan. The teaming arrangement, which consists of Lockheed Environmental Systems and Technologies, ESSI, Tetra Tech, Inc., Ramparts Technologies, Lockheed Idaho Technologies Company, NASA/JPL, EG&G Measurements, Las Vegas, Bishop Associates, SEMTECH (Kazakhstan industry partner), and Scientech Inc., plans to begin operations in January 1995.

Latin America Initiative

Several proposals have been developed to partner with industry, universities, and State and Federal agencies to develop, package, and deliver Department of Energy and ESSI competitive advanced technology products/services. This approach provides solutions to critical environmental restoration and waste management problems while furthering national business and technology goals. The proposals include providing environmental management education and training services to build an infrastructure in Mexico and are planned for expansion to Chile and Argentina. Use of the AVIRIS and other non-intrusive remote sensing technologies provides the technical foundation for this effort.

FUTURE PLANS

Solicitation Plans

Through teaming with National Laboratories and other firms, ESSI plans to continue to respond to future solicitations. This allows them to identify possible technology applications for remote sensing which far exceed it current capability to fulfill. Current partners include several principal investigators requiring commercial support to move their remote sensing technologies from the laboratory environment to private industry and three firms based in the Washington DC area. One centers its efforts on federal government opportunities, the second focuses on international opportunities, and the third focuses on major corporations which have remote sensing requirements. Where possible, these firms are included as teaming partners in proposals. None require up front financing to obtain potential remote sensing applications. All potential partners agree to allow ESSI to establish offices with little or no overhead costs, gain access to new markets, and obtain the services of other firms' marketing capabilities of products and services.

Relocation Plans

ESSI is evaluating opportunities in relocating to southeast Idaho to more fully use national laboratory assets. National laboratories now offer a variety of services to small businesses. These range from technology transfers to CRADAs to shared use of the laboratory's staff and laboratories to being awarded contracts identified solely for small business. Ten percent of INEL's FY-95 budget is committed to small and minority businesses. With the INEL being designated as a test bed for new ways to work with industrial partners, future laboratory programs are expected to directly benefit and increase the technology base of U.S. industry. ESSI is affording itself the opportunity to participate in the INEL's innovative approaches for industry to access to national laboratory facilities and staff.

INTELLECTUAL PROPERTY PROTECTION IN US AND INTERNATIONAL ARENAS

ESSI current projects and outstanding funding proposals were reviewed by patent attorneys to determine appropriate actions to protect their intellectual property through patents, trademarks, an copyrights. Applications have been initiated to obtain patents for equipment, processes, and new applications of single and multiple

remote sensing systems. Protection of intellectual property has been included in all Memorandums of Agreement, Teaming Agreements, and proposals. Other methods are also being used, such as the CRADA which allows commercially valuable data generated to be withheld from public release for up to five years, and teaming with firms currently doing business in international arenas.

CONCLUSION

ESSI, with the assistance of national laboratories, has developed a layered approach to a costly technology development program that fully leverages the resources of others while retaining ownership, technical application and control, and management direction.

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DEMONSTRATING AND IMPLEMENTING INNOVATIVE TECHNOLOGIES: CASE STUDIES FROM THE U.S. DEPARTMENT OF ENERGY OFFICE OF TECHNOLOGY DEVELOPMENT

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ABSTRACT

This paper describes elements of success for the demonstration, evaluation, and transfer for deployment of innovative technologies for environmental restoration. The elements of success have been compiled from lessons learned through the U.S. Department of Energy (DOE) Office of Technology Development's Volatile Organic Compounds in Arid Soil Integrated Demonstration (VOC-Arid ID). The success of the VOC-Arid ID program was determined by the rapid development, demonstration, and transfer for deployment of technologies to operational sites that improve on safety, cost, and/or schedule of performance over baseline technologies. The VOC-Arid ID successfully fielded more than 25 innovative technology field demonstrations; several of the technologies demonstrated have been successfully transferred for deployment.

Field demonstration is a critical element in the successful transfer of innovative technologies into environmental restoration operations. The measures of success for technology demonstrations include conducting the demonstration in a safe and controlled environment and generating the appropriate information by which to evaluate the technology. However, field demonstrations alone do not guarantee successful transfer for deployment. There are many key elements throughout the development and demonstration process that have a significant impact on the success of a technology. This paper presents key elements for a successful technology demonstration and transfer for deployment identified through the experiences of the VOC-Arid ID. Also, several case studies are provided as examples.

INTRODUCTION

The Volatile Organic Compounds in Arid Soils Integrated Demonstration (VOC-Arid ID) was sponsored by the U.S. Department of Energy (DOE) Office of Technology Development. The VOC-Arid ID was one of several integrated demonstrations designed to support the demonstration of emerging environmental restoration technologies. The principal objective of the VOC-Arid ID was to identify, develop, demonstrate, and transfer for deployment new and innovative technologies for environmental restoration at arid or semiarid sites containing VOCs with or without associated contamination (e.g., radionuclides and metals). Technology demonstrations have been hosted primarily by the DOE's Hanford Site in southeastern Washington State. The DOE's Office of Environmental Management has recently reorganized its technology development efforts within the Office of Environmental Management around five "focus areas." The scope of the VOC-Arid ID, along with several other efforts, have been incorporated into the Contaminant Plume Containment and Remediation Focus Area (Plumes). The goal of this reorganization is to build upon the successes and failures of the integrated demonstrations and improve the technology development and implementation process.

TECHNOLOGY PROCESS FLOW

The movement of a technology through the VOC-Arid ID consisted of technology identification, demonstration and evaluation, and transfer for deployment. The process of identifying a technology was initiated through the preparation of a needs

statement that was distributed in a call for proposals made available by the DOE. The needs statement was developed based on input from environmental restoration operations staff and technical experts in the area of VOC monitoring, characterization, and remediation. Proposals received were processed through DOE program management and were then reviewed and evaluated by the VOC-Arid ID technical support groups. Using the technical support groups' recommendations and working to funding limitations, VOC-Arid ID management recommended technologies to be included in the program. Final decisions on technologies accepted into the program, and their funding levels, rested with DOE program management.

The demonstration process began immediately upon a technology being accepted into the VOC-Arid ID. A Demonstration Operations project engineer was assigned to guide the principal investigator for each technology through the rigors of the field demonstration and was responsible for all aspects of the demonstration. For the principal investigator's use, Demonstration Operations developed a guide for preparing demonstration documents (6). This guide defines the required documentation and delineates at what point in the process each document is due. Following each field demonstration, a series of reports were produced in evaluation of the technology.

ELEMENTS OF SUCCESS

The VOC-Arid ID has been successful in demonstrating innovative technologies and transferring those technologies for deployment. This has been due in large part to established organization, process, and interfaces. Several elements that have been identified as contributing to the program's success are described below. Woven throughout the elements for success is the consistent application of a structured process. Specific lessons learned of these elements can be seen later in the case studies.

Technology Selection

A key to the successful demonstration, evaluation, and transfer for deployment of a technology is the selection of appropriate technologies.

The VOC-Arid ID used technical support groups to provide technical review and screen technologies. The technical support groups consisted of recognized experts in the areas of environmental restoration, e.g., characterization and monitoring, drilling or access, and remediation. The technical support groups represented a cross-section of staff from DOE's operations contractors and national laboratories, universities, industry, and other Federal agencies from across the nation.

The technical support groups ensured that there was a real need for a proposed technology, and that the technical basis for the technology was sound. The technical support groups also monitored the development and demonstration of each technology to ensure that the expectations of the technology were being achieved.

Criteria, Objectives, and Parameters

Establishment of criteria by which technologies will be evaluated is a key to the successful demonstration and evaluation, and transfer for deployment of a technology.

Criteria that were established for use by the VOC-Arid ID incorporated guidance from the Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA). These criteria were modified to incorporate input received through stakeholder involvement activities (stakeholder involvement as an element of success is discussed later). Criteria developed through this process include the following:

- Technology performance: remaining contamination
- Technology performance: process waste
- Implementability and practicality of technology
- Does the technology function as intended?
- Cost
- Time
- Worker health and safety
- Public health and safety
- Environmental impacts
- Technology reputation/familiarity to public
- Future land uses/tribal rights
- Socioeconomic impacts
- Regulatory infrastructure requirements
- Compliance with regulatory requirements
- Overall protection of human health and the environment.

Identification of appropriate demonstration objectives and the parameters to measure those objectives is a key to the successful demonstration, evaluation, and transfer for deployment of a technology.
The structured demonstration process implemented by the VOC-Arid ID ensured that appropriate information was gathered during field demonstration, e.g., information needed to evaluate the technology against the criteria. This process included identification of appropriate demonstration objectives, and appropriate parameters to be measured. Focus on the development of objectives and parameters in planning for the demonstrations prevented the gathering of inadequate information by which to evaluated the technology.
Documentation
Complete documentation is a key to the successful demonstration, evaluation, and transfer for deployment of a technology.
Documentation required from each VOC-Arid ID principal investigator before fielding a technology included a conceptual test plan with a technology profile, demonstration objectives worksheet, and environmental regulatory checklist attached. The purpose of the conceptual test plan was for the principal investigator to formulate demonstration objectives, provide information to initiate regulatory and demonstration host site compliance activities, and provide for review of plans for field activities. The technology profile was used for stakeholder involvement activities. The demonstration objectives worksheet was a tool to aid the principal investigator in formulating objectives for the field demonstration. Regulatory experts used the environmental regulatory checklist to ensure compliance with applicable regulations.
The conceptual test plan was incorporated into an integrated test plan that was the document that guided the field demonstration. The integrated test plan ensured that the demonstration was conducted in a safe and controlled manner. The integrated test plan incorporated all aspects of conduct of operations including all operating procedures necessary for the demonstration.
Following each field demonstration, a series of reports were planned. The principal investigator was responsible for producing an evaluation report of the technology demonstration. The project engineer was responsible for producing an independent evaluation of the usefulness and potential of the technology. Independent cost effectiveness and stakeholder acceptability reports were also part of the reporting process. Drawing on all of the reports listed previously, the VOC-Arid ID would produce an overall technology evaluation summary report.
Environmental Restoration Interface
Demonstration conducted through the infrastructure of an environmental restoration project, or operational platform, is another key to the successful demonstration, evaluation, and transfer for deployment of a technology.
When possible, VOC-Arid ID technology demonstrations were conducted as systems in conjunction with an operational environmental restoration project. This provided the opportunity for the comparison of innovative technologies against one another, and against baseline technologies, under real-world field conditions. Bechtel Hanford, Inc. (BHI), is the Environmental Restoration Contractor at the Hanford Site. Having the operations of the VOC-Arid ID conducted through BHI provided for the technology demonstrations to be wrapped into environmental restoration project activities. Success was found in putting the innovative technologies in the hands of the people doing the environmental restoration work. This allowed those individuals in the field to become familiar with the technology and how it could improve on the safety, cost, and/or schedule performance of baseline technologies. It also provided for input to the principal investigator on improvements that might be accomplished during the development of the innovative technology.
Emphasis on Commercialization
Emphasizing the earliest possible establishment of a tie between the technology and a commercial partner is another key to the successful demonstration, evaluation, and transfer for deployment of a technology.
At the time a technology was brought into the VOC-Arid ID, work began on commercialization. The commercialization process was managed by the principal investigator who was required to involve industry to ensure the technology would be manufactured and made commercially available. It is generally expected that technologies will be made available by commercial suppliers of goods and/or services rather than by Federal contractors, national laboratories, or universities. The goal

was to ensure industry involvement in the development and demonstration process to enhance the technology's likelihood of ultimate deployment. Identification of a commercial partner is a strong indicator of the potential of an innovative technology.

Commercialization plans and partnership plans are used by the Office of Technology Development to ensure that projects consider commercialization early in the development cycle. A commercialization plan lays out specific actions, and a general business strategy, for commercializing the technology. This plan usually sets a path toward ultimate commercialization of a technology where an industrial partner has already been identified. A partnership plan establishes a path for projects or technologies where an industrial partner is needed but not identified. This plan includes schedules and strategies for soliciting or identifying partners to support a commercialization effort.

Stakeholder Involvement

Initiating stakeholder involvement as early as possible is still another key to the successful demonstration, evaluation, and transfer for deployment of a technology. Part of the initial documentation package required by the VOC-Arid ID from the principal investigator included a technology profile. The technology profile was the basis for many of the stakeholder involvement activities conducted in support of the VOC-Arid ID. This included providing information for ProTech, a computer-based communication system designed to facilitate public understanding of innovative technologies. The ProTech audience targeted individuals and groups with a stake in a technology's deployment including, but not limited to, regulators, special interest groups, citizens in a community where an environmental technology may be used, and potential industrial codevelopers and users of a technology. The technology profile was also updated upon completion of the demonstration and used in the reporting process.

Analysis of stakeholder acceptance was conducted by a technology acceptance task team. This team used the technology profile to provide preliminary information to stakeholders on upcoming technologies, systems, and field demonstrations. Focus groups and workshops with stakeholders (e.g., regulators, interest groups, and technologists) were conducted to identify issues and concerns they may have about a technology. In addition, the technology acceptance task team conducted interviews and presentations with a variety of stakeholders with the same objective. Establishment of the VOC-Arid ID evaluation criteria were developed in part from this process.

The Plumes Focus Area has established a management approach and strong emphasis on stakeholder involvement based on the VOC-Arid ID's stakeholder process. In addition, the DOE sites, such as the Hanford Site, are developing similar stakeholder involvement processes as part of their role within the Focus Area.

TECHNOLOGIES DEMONSTRATED

Following is a list of technologies that have been demonstrated over the past several years as part of the VOC-Arid ID. These are in addition to the technologies discussed later in the case studies.

Characterization and Monitoring Technologies:
- BoreSampler
- Cross-Hole Seismic Tomography
- Portable Acoustic Wave Sensor
- Arrayed Sampler
- Colloidal Borescope
- Unsaturated Flow Apparatus
- Odyssey Sensor
- Passive Soil Gas Monitors
- Fiber Optic Sensor
- FTIR Instrument
- Prompt Fission Neutron Logging Tool

Drilling or Access Technologies:
- Cone Penetrometer
- Directional Drilling
- ResonantSonic Drilling

Remediation or Treatment Technologies:
- In-Well Vapor Stripping
- In-Situ Bioremediation

Tunable Hybrid Plasma
Passive Soil Vapor Extraction

CASE STUDIES

After completing more than 25 field demonstrations of innovative technologies for characterization, monitoring, drilling or access, and remediation, the VOC-Arid ID conducted a rigorous review of many of these demonstration projects to identify barriers to, and attributes of, successfully deployed technologies. These case studies represented both privately developed and federally developed technologies and included systems for both VOC remediation and characterization or monitoring. The following sections summarize five of these case studies and the lessons learned. Each case briefly describes the technology and a limited history of development and demonstration, along with the findings of the review.

Membrane Separation System for Treatment of VOC Off-Gases

A membrane separation system developed and supplied by Membrane Technology Research, Inc. (Menlo Park, CA), was demonstrated for concentration and removal of vapor-phase chlorinated solvents from a soil vapor extraction (SVE) system. This technology was demonstrated to assess its viability as a cost-effective alternative to granular activated carbon for treatment of SVE off-gases. Off-gas membrane systems have been used commercially in industrial solvent recycling operations, but not for soil or groundwater cleanup operations. The anticipated benefit of the technology was a lower cost separations process that produced a liquid VOC waste stream that could be more easily and inexpensively treated relative to the offsite shipment and regeneration requirements of activated carbon.

A field demonstration of a pilot membrane system was conducted in the spring and summer of 1993 at the Hanford Site, in conjunction with an ongoing CERCLA cleanup operation. Planning for the demonstration began nearly a year before the field activities through the development of a detailed test plan (3). Operations staff interested in off-gas alternatives were selected to lead the field demonstration working directly with a lead technical engineer to lease, install, and test the pilot unit. The ongoing cleanup operation allowed the membrane system to be tested without significant regulatory compliance permitting by leveraging off a previously approved system. A split stream from the SVE system was drawn into the membrane unit. Effluent from the test system was returned to the SVE system for treatment through the baseline activated carbon system, thereby reducing any potential regulatory concerns.

The membrane system demonstration successfully accomplished its objectives and provided both technical performance and cost data to support an analysis of the technology's cost benefit (4). Results indicated that the system operated well, performed at high separation efficiency, but only produced significant cost savings over the baseline at VOC concentrations above approximately 600 ppm carbon tetrachloride by volume under the conditions tested. These conditions included an assumption that the VOC from this waste site could not be recycled, therefore the cost benefit associated with this technology's usual application in solvent recovery was not possible. Although the technology was not successfully implemented following the field demonstration, the effort was considered successful because it provided valuable data to the Environmental Restoration Program. This data essentially prevented an unnecessary expenditure of capital and operating funds that may have occurred if a system was procured or leased for full-scale operations or testing based on performance of systems used in the solvent recovery area.

The following are several key attributes of this demonstration and lessons learned. Staff responsible for operations of an actual cleanup were intimately involved from the beginning in selecting, planning, and operating this technology demonstration. Test objectives and performance criteria were clearly established before the demonstration to ensure that all operations costs and technical performance issues were addressed during testing.

The operation staff's interest in demonstrating the system was enhanced by 1) the desire to consider off-gas treatment alternatives to reduce the cost of the baseline system, 2) the fact that it was a commercial technology with existing data on previous applications (i.e, track record) even though they were under different conditions, 3) the technology's attributes of being a skid-mounted, fully-engineered system with simple operational controls (i.e, could be easily integrated into existing systems), and 4) the fact that it was a commercially available system that could be purchased or leased quickly to meet immediate needs.

Halosnif Fiber Optic Spectrochemical Sensor

The Halosnif sensor was developed by researchers at Battelle Pacific Northwest Laboratory (Richland, WA) for detecting chlorine or fluorine-containing compounds in air, such as trichloroethylene or carbon tetrachloride. The sensor's attributes include a wide dynamic range of detection (e.g, 10 ppmv to 10,000 ppmv), and continuous, real-time detection. Over several years, development efforts improved the sensor's detection limit, portability, and stability. The resulting sensor system used an improved commercial power supply and could be easily set up and operated as a continuous monitor.

The sensor was demonstrated at several DOE sites, functioned well, and compared favorably against other baseline techniques for chlorinated VOC detection (7). The demonstration efforts accomplished their objectives and demonstrated the technology's performance; however, this system has yet to be successfully commercialized or implemented. Although the system has several favorable attributes, it has not satisfied the requirements of end users. Primary user needs for VOC sensing are centered around lower detection limits (e.g., 1 ppm) for health and safety monitoring, and hand-held instrumentation for field monitoring. The Halosnif system is too large and bulky for hand-held applications. The system is well suited to process monitoring; however, this is a different market and there are other commercial systems that can adequately, and more effectively meet the process monitoring and control needs.

A number of lessons were learned as this technology advanced through development and demonstration.

Commercialization, market assessment, and technology transfer are critical components of successful deployment, and must be considered very early in the development cycle. A more thorough market assessment and user survey may have identified the functional requirements of a new sensor system, and competing commercial systems that any VOC sensor would have to outperform to ensure commercial success. Commercialization plans that set a path forward and identify key decision points for a technology's development effort are critical to the process of either redirecting, redesigning, or ceasing development efforts.

Several field demonstrations were conducted over 2 years in an effort to assess the technical performance of the sensor. These tests were very successful in identifying the technical areas needing improvement. However, these test did little to define the right market, target problems, or technical attributes (i.e., size, portability, detection limit, etc.) that would ensure user needs were being met. If development objectives and criteria are adequately defined, they can be key elements in the decision process for continuing development. Commercial research and development typically assesses the business or market factors early in the development cycle. In this case, those assessments were incomplete and did not adequately identify the market for this type of sensor system.

User needs are often vague, and ill defined. This sensor system was developed based on a defined, but general need identified by environmental restoration staff. Without functional requirements, or clear performance goals associated with each need, technologies can be developed that appear to meet the requirements. Only after years of development and demonstration does the developer realize the need has either changed, or was never adequately defined.

In Situ Permeable Flow Sensor for Groundwater Flow Measurement

A groundwater flow sensor developed by researchers at DOE's Sandia National Laboratories (Albuquerque, NM) was developed and demonstrated under DOE's VOC in Non Arid Soils Integrated Demonstration (VOC Non-Arid ID) in 1993 (2). In 1994, this technology was used at DOE's Hanford Site by environmental restoration staff as part of remedial investigation efforts within the Site's 100 Area. This deployment of the in situ permeable flow sensor represented a successful technology transfer from a research and development program to a cleanup effort.

This sensor uses heating elements and temperature sensors to indirectly measure water flow, and is deployed by placement in an uncased borehole below the water table. The formation is allowed to collapse around the sensor, resulting in a system that measures flow within the natural sediments rather than within a significantly disturbed borehole or well. The goal is to achieve accurate measurement of flow that is not likely possible with conventional methods. The system can be used for aquifer characterization or monitoring before, during, or after remediation of the aquifer.

Development, demonstration, and technology transfer of the in situ permeable flow

sensor was successful for the following reasons.

The technology was demonstrated multiple times as part of the VOC-Non Arid ID and its performance thoroughly documented.

Environmental restoration staff responsible for remedial investigation/feasibility study and cleanup actions at the Hanford Site learned about this innovative sensor system through direct involvement as technical reviewers within DOE's Technology Development Program. Although information on many new technologies is disseminated through publications, conferences, and technology databases, direct, one-on-one contact between the developers or suppliers and the operations staff is critical. In addition, operations staff involvement in review and evaluation of technology demonstrations resulted in immediate, complete, and unbiased information on performance of the sensor technology.

Characterization and monitoring technologies are more easily transferred because they are generally less expensive to deploy and/or test than remediation systems. In addition, they are typically not held to the rigorous performance standards of remediation technologies (especially for screening applications), and do not usually require lengthy regulatory review or approval prior to deployment.

At the time of deployment at the Hanford Site, this sensor system was not commercially available. Therefore, deployment of the sensors was accomplished through a direct contract between the operations program and the original investigator from a DOE national laboratory. Although it is generally felt that technologies are more readily accepted if commercially available, in this case the presence of an industrial supplier was not important to the environmental restoration staff. If hardware could be procured easily and technical support obtained from the developer, a viable industrial supplier was not important. However, broad application of the technology to multiple customers would clearly require an effective commercial partner.

Six-Phase Soil Heating for Enhanced Removal of VOCs from Soils

A variety of soil heating methods are being developed and demonstrated by industry, DOE, and other Federal agencies as a means of enhancing volatilization and subsequent removal of VOCs from soils. Six-phase soil heating is an electrical resistance or "ohmic" heating method developed by researchers at Battelle Pacific Northwest Laboratory. Field testing was accomplished by the VOC-Arid and VOC-Non Arid IDs at DOE's Hanford and Savannah River Sites, respectively. The technology was tested at pilot-scale in an uncontaminated area at the Hanford Site as part of shakedown testing, and then tested at the Savannah River Site's M-Area for enhanced removal of trichloroethylene and perchloroethylene from a clay lens (1). The technique uses six electrodes in an array, with six-phase alternating current power to resistively heat soils in situ, thereby drying the soil and volatilizing contaminants. A soil vapor extraction system is used in conjunction with the heating system to recover and treat released vapors.

The demonstration was successful in heating soils to approximately 10C (50F) within a 6-m (20-ft.) dia. zone. In addition, post-test analysis of soils within the heated zone indicate that greater than 95 percent removal of contaminants was accomplished in the 2-month duration test. The demonstration resulted in increased interest from both industry and environmental restoration staff at other DOE sites. Specifically, the Environmental Restoration Program at DOE's Rocky Flats plant in Colorado requested support in remediating several operable units with VOCs and dense nonaqueous-phase liquids (DNAPLs) present in low conductivity soils. Efforts are being planned for use of soil heating at Rocky Flats. Development activities for six-phase soil heating are being finalized this year with the completion of a design tool to support deployment of the technology and commercialization.

Although deployment within the Federal and private sectors has not yet been accomplished, the outlook is very positive. Several lessons learned have resulted from this development and demonstration effort.

A well-designed, phased approach was used to develop this heating technology over approximately 4 years. Several sources of funding, primarily from DOE's Office of Environmental Management, were obtained during this time, and at times leveraged with other funding, to accomplish all of the lab-, bench-, pilot-, and field-scale tests necessary to scale-up the technology.

Commercial interest in the technology was observed early in its development cycle; however, serious interest from industry was not realized until after the Savannah River field demonstration. This is likely a result of two things. First, industrial

enthusiasm was dependent on solid evidence of the effectiveness of the technology in a field test. Until an initial contaminated site test was completed, there was little proof that the technology could be effectively used. Secondly, commercialization of the technology was not a primary focus of the funded activities. The developers focused on technical development in conjunction with the establishment of appropriate intellectual property. Identification of one or more commercial partners was somewhat delayed until intellectual property was protected, and proof of performance was obtained. Commercial interest and partnering may have been possible and beneficial earlier in the development cycle if it had been a priority. The DOE and the developers may have sacrificed some intellectual property, but the timeline for deployment may have been reduced.

SEAMIST Membrane Instrumentation and Sampling System

Science and Engineering Associates (Santa Fe, NM) developed an inflatable membrane system that is used as a continuous packer system in uncased (i.e., open) boreholes and as a platform for deploying a variety of instruments and sampling tools. The purpose of this technology is to provide for more representative sampling of soil gas, VOCs, and other soil contaminants, as well as for in situ air permeability measurements. The SEAMIST system is commercially available and is being used for site investigations and monitoring. The technology is now owned by Eastman-Cherrington Co., Albuquerque, NM.

SEAMIST was demonstrated at a variety of sites, including DOE sites such as Savannah River and Sandia National Laboratories. Environmental restoration staff at the Hanford Site became aware of the technology several years ago and purchased a system to be used in characterizing sites contaminated with VOCs (5). However, SEAMIST use at the Hanford Site was significantly different than the originally intended purpose. For example, operations staff requested a system to be used in large diameter wells, to depths of 76 m (250 ft), without predetermined sampling ports. Because of the geology at the Hanford Site, temporary casing is needed to maintain borehole stability during drilling. Open boreholes are not possible for any significant depth. Hanford Site contractors desired a system that could be deployed during drilling to collect samples at the current depth of the borehole. Therefore, a modified SEAMIST system was used where the soil vapor sampling was accomplished through a sampling line dropped down the well to depth, and the SEAMIST system was used to create a continuous packer and ensure vapor sampling occurred only from the formation adjacent to the end of the sample line.

SEAMIST is currently being used at the Hanford Site in routine characterization efforts and is gaining additional applications through continued use by environmental restoration staff. In addition, the developers are expanding applications of the technology to other contaminants and analyses.

This technology was successfully deployed at several sites and is still being used for additional applications. Several key features resulted in the successful deployment of the SEAMIST system.

Environmental restoration staff had a well-defined, immediate need (e.g., collection of representative soil gas samples during drilling) and were prepared to use their resources to find and test new approaches to sampling. Existing techniques for sampling had shown wide variability and provided inadequate results. In addition, end users gained knowledge of the details of the technology's operation, recognized applicable features, and identified features that would tailor the technology to the specific needs of the site.

End users took ownership of the modification, industrial interface, and demonstration of the technology. Direct and significant involvement from the site user resulted in a product that met the needs of the site.

The SEAMIST system was commercially available, and was supported by a company eager to modify, adapt, and redesign the system to best support the needs of the customer.

CONCLUSIONS

The ultimate goal of DOE's Technology Development Program is deployment of new technologies and technology systems that improve on safety, cost, and/or schedule of performance over baseline technologies. Field demonstrations are a critical component of successful development and deployment of environmental technologies. However, field demonstrations alone do not guarantee successful transfer for deployment. There are other key elements throughout the development and demonstration cycle for any technology or technology system that have significant

impact on the likelihood of deployment. The VOC-Arid ID established a development and demonstration process designed to enhance the likelihood of successful deployment following field demonstration. This process includes clearly defined methods for technology selection, performance evaluation involving stakeholders in the development and demonstration cycle, and commercialization. Many keys to successful technology deployment were identified in lessons learned from more than 25 field demonstrations. Also, several key attributes for successful deployment were identified through case studies. The case studies further highlight the applicability of the development and demonstration process established by the VOC-Arid ID, and identify areas where additional emphasis must be placed.

In summary, key attributes include the following:

Well-defined needs that clearly describe performance objectives and requirements for technical solutions. These needs are then used in the selection of appropriate technologies.

Well-designed development efforts with phases and tasks that identify key decision points. Clearly defined performance objectives and evaluation criteria are needed for each stage of development. A broad set of criteria is needed to ensure that as many issues can be addressed as early as possible in the development cycle. Projects unable to meet these key performance goals are redirected or terminated early, allowing more promising methods to continue toward demonstration and deployment.

Clear and complete documentation of all phases of the development, demonstration, and performance evaluation process. Thorough reporting of results with wide distribution. Effective communication of demonstration results, especially to a variety of potential end users, is critical to deployment.

When possible, conduct field demonstrations in conjunction with environmental restoration projects. This allows the opportunity for end users to become familiar with the technology and affect its development. This also provides the opportunity to view the innovative technology under field conditions along side baseline technologies.

Early efforts are needed to assess commercial markets and identify commercial partners. Industrial involvement early in the development cycle can enhance marketability, funding for development, and availability of the technology immediately after demonstration.

End user and other stakeholder involvement in the development, demonstration, and/or adaptation process for technologies and systems. This involvement includes management, oversight, and technical involvement in the development and demonstration efforts, as well as significant involvement in the development review and evaluation process.

The DOE's Office of Environmental Management has recognized the strengths and weaknesses of its past technology development and demonstration program, and has recently initiated a "New Approach" involving focus areas. This new process incorporates the key attributes identified above and in the case studies. Specifically, the focus areas are emphasizing direct user and stakeholder involvement in every facet of the development cycle, along with technology selection that is targeted more directly to complex-wide, but site-specific problems or needs. The demonstration efforts and lessons learned from the VOC-Arid ID, along with other development and demonstration efforts, have been incorporated into the Contaminant Plume Containment and Remediation Focus Area (Plumes) and are providing a building block for the New Approach.

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TECHNOLOGY DEMONSTRATION PLANNING - THE KEY FOR SUCCESS IN DEMONSTRATING PRIVATE SECTOR INNOVATIVE TECHNOLOGIES AT DOE SITES

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ABSTRACT

The Department of Energy (DOE) Office of Environmental Management (EM) is actively pursuing the development of innovative technologies from the private sector to contribute to the many waste management and environmental restoration problems across the DOE complex. Before any innovative technology can be applied at a site, it must be demonstrated to both the site and the regulators that the technology will perform according to expectations. Prior to such a demonstration, there are a number of key planning steps that must be taken, including the careful identification of the most appropriate site, a commitment on the part of the site to host the demonstration, preparation of a detailed demonstration plan prior to the demonstration, clear delineation of duties and responsibilities, consideration of regulatory issues, etc.

This paper describes an approach to selecting the most appropriate site at which a private sector technology developer should demonstrate, and a discussion of the preparation of the demonstration test plan which should be prepared prior to the demonstration.

INTRODUCTION

The Department of Energy (DOE) Office of Environmental Management (EM) is actively pursuing the development of innovative technologies from the private sector to contribute to the many waste management and environmental restoration problems across the DOE complex.

Before any innovative technology can be applied at a site, it must be demonstrated to both the site and the regulators that the technology will perform according to expectations. Due to the specific nature of problems at some DOE sites, a technology that has been demonstrated at a non-DOE site usually must also be demonstrated at a DOE site before it can be accepted. Thus, the demonstration becomes the critical element in determining the success of the technology development.

Prior to such a demonstration, there are a number of key steps that must be taken, including the careful identification of the most appropriate site, a commitment on the part of the site to host the demonstration, preparation of a detailed demonstration plan prior to the demonstration, clear delineation of duties and responsibilities, consideration of regulatory issues, etc. There have been a number of technology demonstrations at DOE sites as part of the Office of Technology Development (OTD) Integrated Demonstrations and Integrated Programs. As these technologies have been developed at the sites as part of these programs, there often is an easier path to demonstration than with outside parties (including small businesses) who are often unfamiliar with the nature and culture of the DOE sites. This paper presents an approach to selecting a site for demonstration, and the elements that should be included in the demonstration test plan that must be developed prior to conduct of the demonstration.

SITE SELECTION

Demonstration site identification is a key decision in obtaining programmatic success since it impacts on all aspects of each project, it guides the planning in the initial project phase, provides a basis for Go/No-Go decisions for continuation of the project, and drives the planning for subsequent phases. Therefore, site identification, negotiation, and planning must be made as early in the project life cycle as possible.

Site identification involves the consideration, evaluation, and weighing of a

variety of issues, goals and objectives, and must be directed by appropriate and sound criteria. Seven primary criteria have been delineated for identification of a demonstration site. While all of the seven criteria are considered to be important, they are not listed in order of priority. An overriding order of priority has not been established, because the relative importance of any criterion may vary from project to project, depending on the nature of the demonstration project and/or the technology involved. Thus, prioritization of site-selection criteria will receive consideration upon initiation of the site identification process. The seven criteria are:

1. There must be a defined and documented site need for the technology.
2. The technical approach must be applicable and feasible for efficient performance at the site.
3. The regulatory status of the facility and compatibility of established regulatory schedules and constraints that apply to the site must not unreasonably limit the feasibility of a successful project demonstration.
4. Technical, scientific, commercial and operational factors that lead to the preference of a specific site by the technology development contractor must be considered.
5. The potential for the demonstration to show applicability of the technology at other sites or throughout the DOE complex must be determined, and considered.
6. The costs associated with conduct of demonstration at alternative sites must be considered in order to assure cost effectiveness. The availability and source of funding must be established.
7. The site must demonstrate its willingness, technical ability, and adequate organizational structure to support the demonstration project.

The following discussion elaborates on each of the above criteria:

CRITERION 1. There must be a defined and documented site need for the technology.

In order to support a decision to commit funding and resources to an on-site small-scale and eventual full-scale demonstration, it is necessary to establish that the basis of the proposed project will provide for resolution of an identified need. By clearly defining the needs of the end user, clear objectives and performance goals by which the success of the demonstration can be measured will be established. A major criterion must be that the technology can be applied to resolution of an identified problem. The test plan can then be developed so that the maximum probability of transferring a proven technology to a practical field application is assured.

CRITERION 2. The technical approach must be applicable and feasible for efficient performance at the site.

The technical applicability and feasibility of demonstrating the project technology at the site must be considered and verified. A technology that potentially will work very well at one site may well be a failure at another due to differences in site conditions. Differences in site conditions could include, for example, waste matrices, range of contaminants, contaminant concentrations and characteristics, geological and meteorological conditions, and clean-up levels or goals.

CRITERION 3. The regulatory status of the facility and compatibility of established regulatory schedules and constraints that apply to the site must not unreasonably limit the feasibility of a successful project demonstration.

The regulatory status of the potential site and the compatibility of the technology development project schedule with the site schedules for regulatory compliance, including characterization and remediation, must be evaluated and properly weighed in site identification decisions. The need and ability to obtain the necessary regulatory permits, whether the site is under CERCLA or RCRA, the requirements for NEPA documentation, and the requirements of a Federal Facility Compliance Agreement will all affect site identification.

CRITERION 4. Technical, scientific, commercial and operational factors that lead to the preference of a specific site by the technology development contractor must be considered.

In certain cases, a technology development contractor may have previously established a strong understanding of the requirements of a specific site remediation program and the prevailing site conditions, and may have previously structured his research and development program to address the specific concerns. This situation is particularly apparent if a contractor has been working at a site utilizing the existing state-of-the-art, and has developed more advanced concepts. A

contractor may also make the case that the conditions at the proposed site offer the greatest potential for developing a commercially viable (profitable) piece of equipment, instrument, or treatment process.

CRITERION 5. The potential for the demonstration to show applicability of the technology at other sites or throughout the DOE complex must be determined, and considered.

An important consideration in site identification is the multiple applicability of a technology to the DOE complex resulting from a demonstration at a given site. Would the demonstration results be applicable to a variety of needs at a number of sites, or would it only solve a limited or specific problem? To the extent that multiple applicability has been identified, site identification should lead to the development of a Demonstration Test Plan that will ensure the generation of adequate data to support a determination that the technology has applicability to other sites across the DOE complex, and that the results of the demonstration can be applied to other sites.

CRITERION 6. The costs associated with conduct of demonstration at alternative sites must be considered in order to assure cost effectiveness. The availability and source of funding must be established.

It is recognized that life cycle costs associated with the conduct of a demonstration may significantly differ between sites. Such differences may derive from costs for utilities, logistics, permitting, sampling and analysis, mobilization, etc. These differences may affect the cost-effectiveness of the demonstration. In addition, the site-specific demonstration costs must be evaluated in comparison with the availability of budgeted funding.

CRITERION 7. The site must demonstrate its willingness, technical ability, and adequate organizational structure to support the demonstration project.

Successful demonstration of a technology project requires active and effective support from DOE management at the facility, the Management and Operating contractor, and other support organizations. Accordingly, a particular site's technical or scientific background relative to the nature of the problem and the proposed technology application must be considered. The facility's technical and administrative resources and past experience in demonstrating projects of similar scope and complexity will be considered to be an important element in the ultimate success of the demonstration project.

SITE IDENTIFICATION PROCESS

In order to ensure that the application of new technologies will receive the acceptance and support of all affected parties, it is essential that a process of open discussion and involvement in the process is adopted. The active support of many organizations across the DOE complex facilitates and accelerates the successful demonstration of a technology development project. The comments and recommendations of all affected parties should be actively sought throughout the demonstration site identification process. Affected parties include, but may not be limited to: the technology development contractor, Environmental Restoration or Waste Management division at the DOE Operations Office, Technology Development division at the DOE Operations Office, EM-30 or EM-40, EM-50, Site M & O or ERM contractor, and Stakeholders and Regulators.

In particular, it must be recognized that the Regulators and the Stakeholders will have a significant impact on the final selection, since the ultimate use of the technology at a particular site must receive their approval. It also must be recognized that there are significant differences in the administrative approach of the different regions of the U.S. EPA, and in the states towards the adoption and implementation of innovative technologies at DOE sites.

DEMONSTRATION TEST PLAN (DTP)

Once the proposed demonstration site has been identified, the DTP is used for documenting the details of agreement by the involved parties for preparing, conducting, and reporting the results of a technology demonstration. The purpose is to ensure the effectiveness of the demonstration. The DTP should provide a means for establishing defensible data on the subject technology, thus satisfying the needs of DOE, technology developers, end users, regulators and additional stakeholders. Accordingly, the DTP should be prepared with appropriate signatory approvals prior to the onset of demonstration activities.

The DTP should be written in the context of the selected host site for the demonstration. That is, the plans for the demonstration should be integrated with

the cleanup program already underway at the site. The reason for the integration is two fold: environmental technology development projects are undertaken to help make environmental restoration programs more effective, and coordinated efforts are necessary to enhance DOE credibility.

The following are critical elements that should be included in the DTP:

1. Approval Signatures

Approval of the DTP concept is necessary to ensure that the demonstration will be conducted in a manner acceptable to all stakeholders who have a vital role. Vital roles are identified by responsibilities to conduct or support the demonstration or to pass judgement on the acceptability and/or validity of the results. Signatories include parties identified in the DTP having responsibility to perform the demonstration, having regulatory authority to accept or reject the results, or parties identified as representing concern on issues of acceptability.

The following signatories may be required to approve the DTP: the technology development contractor, the DOE host site (or other host site) representative, the DOE site contractor(s) representative, the applicable regulator(s) representative, and local or regional stakeholder(s).

2. Introduction/Project Description

This section should include a general introduction and provide a concise description of the project. This section also establishes the context of the technology to be demonstrated, including the perspectives of technical development, the ability of the technology to address DOE Focus Area needs, host site environmental and regulatory processes, and stakeholder interests. Examples of DOE EM concerns are policies, such as waste minimization, and cleanup goals for the affected sites.

This section should establish the host site background and physical setting for demonstration. The background should identify how the technology can be applied to a particular site need (i.e. Operable Unit characterization or remediation), consent order, impending Record of Decision (ROD) for remediation, or other concern which establishes a need for the subject technology. This section should also establish the site physical setting (i.e. topography, physiography, geology/hydrogeology, drainage, etc.) and, if applicable, historically establish how the site need occurred (i.e. production process, contaminant spill, etc.).

In addition to background information, the following should be included as elements of the project description for the host site: regulatory setting and special issues; the technology need with regard to DOE EM concerns; and the relationship to subsequent efforts (e.g., other phases or promotion and commercialization).

3. Technology Description

The description of the technology should address the specific EM Focus Area (i.e. Plume, Mixed Waste, etc.) addressed as well as specific technical aspects of the subject technology. This section explains the nature of the general class of technology (e.g., a type of treatment) and typical benefits (e.g. better, faster, safer or cheaper) as well as technology challenges and limitations. It describes the particular technology in general terms, then sets it apart from the general class of technology.

This section should provide a process flow diagram and narrative description, if applicable, which defines the conceptual design and indicates parameters such as influent and effluent waste streams, residuals production, major technology components/platforms, etc. which define unit operations of the technology system. This information helps to ensure that all potential impacts and support needs are identified (i.e. Health Physics) for potentially involved parties.

4. Demonstration Objectives

This section defines the objectives, their basis, and provides information on acceptability issues as well as technical issues, which were identified in the Introduction/Project Description and in the Technology Description. This section should address demonstration tasks individually and collectively and refer to the Quality Assurance Project Plan (QAPjP), if appropriate.

Additionally, the following items should be discussed: demonstration tasks, intended use of demonstration results, demonstration objectives, data quality objectives, expected results, measures of success, and criteria for evaluation.

5. Experimental Design and Procedures

The design and procedures are based on the developer's technology and the objectives of the demonstration. This section identifies the systematic aspects for the conduct of the demonstration, which will be used to develop the desired confidence in the

results. It satisfies all of the Demonstration Test Objectives and anticipates reasonable contingencies. Along with Site Preparation, Equipment, and Materials (Section 6), this section provides site-support participants adequate information to prepare for the demonstration.
This section may refer to specific appendices or documents that contain all of the details and should address, but is not limited to, the following aspects of the demonstration:
Preparation of samples (i.e. in situ, surrogate, test bed, etc.),
Definition of the volumes or amounts to be tested (influents, effluents and anticipated residuals),
Definition of test procedures and methodologies,
Definition of critical parameters and levels of testing,
Definition of demonstration limitations, and
Identification of instrumentation/controls/equipment and control strategy as well as standard operating procedures.
6. Site Preparation, Equipment, and Materials
This section describes the demonstration area at the host site, presents the details of equipment mobilization/demobilization, describes set up of the demonstration, lists the items to be used during the demonstration, and describes the immediate post-demonstration management of the equipment and site. Along with Experimental Design and Procedures (Section 5), this section provides site-support participants adequate information to prepare for the demonstration. This section should refer to Residuals Management (Section 11) and it may also refer to a Residuals Management Plan or a Demobilization Plan for the detailed information.
This section should address preparation, equipment, and materials in a comprehensive manner, including equipment lists, models and specifications. Additionally, setup prior to demonstration and demobilization procedures should be detailed.
7. Sampling and Analysis Plan
This section explains when, where, and how information will be collected (sampled or measured), including both field and laboratory procedures for sampling and analysis. Specific methods and procedures for implementing field activities should be contained in the Sampling and Analysis Plan (SAP). The SAP should include a Field Sampling Plan (FSP) that describes sampling locations, procedures and handling. The SAP should also include a QAPjP that describes sample analysis, management, custody and data validation. The SAP should be prepared in accordance with applicable local (host site), state, or regional (e.g. EPA) regulations and directives. This section should also comply with any applicable Consent Orders or Administrative Orders for the host site.
Additionally, all equipment used for measurement and sampling should be detailed in the SAP, including equipment models and operating procedures. This section should identify what quality control sampling is performed regarding each type of sample or measurement, according to the specified Data Quality Objective (DQO) level. The section refers to the FSP, the QAPjP and related standard operating procedures for detailed information.
8. Data Management
This section describes how collected (sampled or measured) data are recorded, handled, and transferred for validation and subsequent analysis and interpretation. It identifies what quality control sampling is performed regarding each type of sample or measurement, according to the specified DQO level. This section refers to the FSP and the QAPjP and related standard operating procedures for detailed information. The section needs to address data from both the field and the laboratory.
9. Data Analysis and Interpretation
This section describes how collected data are validated, interpreted, presented, and reported. It identifies what quality control sampling is performed regarding each type of sample or measurement, according to the specified DQO level. The section describes how results are assessed compared with the Demonstration Objectives. This section refers to the FSP and the QAPjP and related standard operating procedures for supplemental or detailed information. A detailed discussion of data analyses, including graphical presentation, modeling and interpretation should be provided here.
10. Health and Safety
This section addresses the health and safety of all those who might become involved

with the planned demonstration activities. Elements addressed include, but are not limited to, health physics, hazards assessment, precautions, emergency response, and training. The section refers to the Health and Safety Plan (HASP) and other standard operating procedures for supplemental or detailed information.

11. Residuals Management

This section accounts for all residuals resulting from the demonstration, including side stream effluents, unused influent waste streams (not subjected to testing), and investigation-derived waste (i.e. soil cuttings, groundwater, etc.). Non-inclusive examples are treated wastes, treatment residuals, used containers and expendable, contaminated clothing and debris, laboratory samples and sample extracts, contaminated equipment and materials. Non-inclusive examples of issues to be addressed are generated types and quantities of residuals, disposition criteria (i.e. waste manifests, waste characterization, etc.), and decontamination/demobilization. The section may refer to a supplemental Residuals Management Plan, Environmental Compliance Plan, Demobilization Plan, or other standard operating procedures for supplemental or detailed information.

12. Public Participation

This section describes the historic identification of, and interaction with, the community and other stakeholders (e.g., local citizenry, tribes, regional regulators) concerning the planned technology demonstration (e.g., site identification and evaluation process). This section shows the interaction activities that will occur during and following the demonstration. It identifies activities to achieve stakeholder consensus on the DTP, if applicable. Other topics addressed should include stakeholder identification, information releases, exchanges of ideas and information, identification of issues, and the selection of signatories to the DTP. This section may refer to a Public Participation Plan for supplemental or detailed information.

13. Reports

The reports section describes the documentation on the results of the technology demonstration, and all of the supporting records of activity and measurements. This section addresses the measures and standards developed in the Demonstration Objectives and Data Analysis and Interpretation. The discussion should identify all deliverables and their respective content and organization. This section may also refer to the QAPjP regarding records management for supplemental or detailed information. A discussion should be provided concerning the detailed presentation of capital and operating costs associated with the demonstration. Cost information will be the basis for technology economic analysis.

14. Schedules

This section provides the start and end date and dependency of each task. The period of applicability ranges from the drafting of the DTP, including a readiness review for the commencement of the demonstration, through to end of the present demonstration phase. The end of the phase is characterized by the report of the demonstration, review by those stakeholders/regulators or others who are to validate data or pass judgement on the demonstration results, and the disposition of residuals and wastes.

The section accounts for interfaces, reviews, and analytical turnarounds. It also includes reasonable slack time to preclude a miss of the project completion date. This section addresses all of the responsibilities identified under Project Organization including those associated with the Experimental Design and Procedures, SAP, HASP, Residuals Management, Community Relations, Reports, and Regulatory Compliance. This section may refer to a supplemental Readiness Review Plan for detail. The section should include a detailed Gantt Chart indicating the start date, stop date, dependencies and overall project schedule.

15. Regulatory Compliance

This section addresses regulatory compliance with respect to the overall demonstration and any applicable site cleanup process or host-site directives for the entire demonstration; prior, during and after test activities. The section particularly addresses regulatory compliance needs to be satisfied prior to the readiness review and commencement of the demonstration.

A description of how the demonstration fits into and satisfies the governing requirements of the site cleanup process (i.e. CERCLA, RCRA, etc.), including post-demonstration regulatory action on the results should be included in this section. Additionally, applicable non-environmental directives: local, state, or

federal such as security, site-access, and labor-related requirements should be discussed here.
This section should contain a discussion which accounts for conformance with other potentially applicable environmental requirements before conducting the operation, such as permits, licenses, and satisfaction of NEPA. It identifies requirements to be satisfied during the demonstration and upon cessation, such as reporting residuals to be disposed. The section may refer to a Regulatory Compliance Plan or Readiness Review Plan with supplemental or detailed information.
16. Project Organization
This section addresses management and staffing, responsibilities, support and site service requirements for the demonstration activities. It identifies roles and responsibilities of the technology development contractor, host site organizations (i.e. DOE, contractors, etc.), vendors, and regulators or stakeholders, who are expected to pass judgement on the results. It addresses all of the tasks identified under the Experimental Design and Procedures, Sampling and Analysis Plan, Health and Safety, Residuals Management, Community Relations, Reports, and Regulatory Compliance. Task managers and their responsibilities should be identified here.
17. Supplemental Material
Additional information in the form of appendices or adjunct documents and references is normally needed to provide supplemental and detailed information to complement the body of the DTP. The additional information is typically in the format of a plan or procedure. Typically the additional information necessarily conforms to a program already in place at the site where the demonstration is to occur. Typically, a particular site is selected because of involvement in an environmental restoration program (i.e. CERCLA, RCRA, etc.) with applicable Consent/Administrative Orders. The following are examples of supplemental material for a DTP:
Field Sampling Plan (FSP): (component of the SAP)
Quality Assurance Project Plan (QAPjP): (component of the SAP)
Health and Safety Plan (HASP)
Residuals Management Plan
Public Participation Plan
Readiness Review Plan
Regulatory Compliance Plan
Standard Operating Procedure(s).

Session 24 -- Geochemical Considerations in the Disposal of Radioactive and Mixed Waste
Co-chairs: Tjalle T. Vandergraaf, AECL;
Robert S. Rundberg, LANL
24-1
SIMULTANEOUS INVESTIGATION ON THE GENERATION OF COLLOIDAL SODIUM, CHLORINE GAS AND STORED ENERGY IN GAMMA-IRRADIATED ROCK SALT
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ABSTRACT
In Germany rock salt formations are envisaged as potential hosts for disposing of high level radioactive waste. One possible technique is the emplacement of vitrified HLW containers in deep boreholes, resulting in an exposure of the host rock to high doses and and temperatures. Gamma irradiation of rock salt leads via a complex reaction scheme to radiation damage with accumulation of colloidal sodium and chlorine gas in the sodium chloride lattice. The formation of the molecular irradiation products is associated with the storage of energy. Many studies have addressed this problem in the past, but in most of these studies, either the radiation-induced formation of only one of the products was determined or only rather low radiation doses have been used. In order to provide the missing data, the temperature dependence of the radiation-induced formation of colloidal sodium, molecular chlorine gas and the associated deposition of energy in rock salt was

determined between 100C and 250C. At each temperature the irradiation dose was varied between 106 and 108 Gy. After irradiation, colloidal sodium was measured via hydrogen evolution upon dissolving the salt in water. Chlorine gas was determined both in the gas phase above the salt and in the bulk. Stored energy was determined by DSC on selected halite specimen. The present study confirms the equivalence of the radiation-induced formation of the different products, i.e. colloidal sodium, molecular chlorine, and stored energy, over a substantial dose range and temperature range of interest for a repository. From the data it is concluded, that at temperatures above 150C, colloid formation starts to saturate with increasing dose. At 100C about 0.7 mol% sodium were detected for 108 Gy but no saturation was indicated. Most of the molecular chlorine after irradiation is found to be in the bulk, but chlorine evaporation from the salt increases with temperatures. The stored energy follows the trends outlined for the molecular radiation products, i.e. it increases with dose. A conversion factor of about 70 J/g per mol-% colloidal sodium in the rock salt is indicated.

INTRODUCTION

Besides Germany several other countries around the world envisage rock salt formations as potential hosts for disposing off high level radioactive waste. One possible technique is the emplacement of vitrified HLW containers in deep boreholes. Initially, this would expose the salt in the vicinity of the waste to dose rates in the order of 1000 Gy/h or less and temperatures between 150 - 200C, depending on the cooling time of the spent fuel elements. Dose rate and temperature decrease with time due to the radioactive decay of the waste. A total dose of 108 to 109 Gy will be absorbed in the salt surrounding such an emplacement borehole.

Gamma irradiation of rock salt leads via a complex reaction scheme to radiation damage with accumulation of colloidal sodium and chlorine gas in the sodium chloride lattice. The formation of the molecular irradiation products is associated with the storage of energy (1,2). In order to evaluate the long-term safety of a repository in rock salt, among other things quantitative information as to the amount of radiation damage is required.

The formation of radiation defects depends on the parameters total dose, dose rate, temperature, microstructure of the salt, and the level of impurities. This has been the subject of numerous studies in the past. In most of these studies, however, the radiation-induced formation of only one of the products was determined, one of the notable exceptions being the investigations of Jenks et al. (1975) (3), but the temperatures were below 150C. Also, in many studies only rather low radiation doses were employed at which the coagulation of molecular radiation products has just begun.

As yet no detailed data are available for the temperature dependence of the formation of radiation damage in the temperature region above 150C, which meet the following criteria: firstly, both molecular radiation products are determined chemically in the bulk of natural salt crystals, and, secondly, the stored energy is determined in the same samples. In order to provide the missing data, an experimental program was carried out.

EXPERIMENTAL METHODS

Irradiations

The irradiations were carried out in a cooling basin for spent fuel elements at the High Flux Reactor (HFR) at ECN in Petten. In order to be able to irradiate about 200 sample ampoules within two years, two special facilities were designed and built, so that the samples could be exposed between spent fuel elements which have been used in previous reactor cycles. The irradiation facility consisted of an aluminum container covered by a watertight lid with an O-ring and an aluminum sample holder housing five irradiation ampoules. The sample holder was equipped with an electrical heater and five thermocouples for measuring and controlling the temperature in the range between 80 and 250C.

Two of these irradiation containers were placed in a special fuel storage rack normally suitable for storing 42 spent fuel elements. In two positions the space for four elements has been modified for positioning the irradiation containers. A cadmium tube was utilized in order to absorb thermal neutrons. As spent fuel elements are used as irradiation source, the gamma intensity decreases and the energy spectrum changes with time. Typically, the gamma intensity of the fuel elements decreased from about 240 to 40 Gy/h within 30 days. For irradiation periods more than 28 days the fuel elements were changed every 28 days. The

radiation doses absorbed in the salt samples were individually determined in each ampoule via two different solid state dosemeters. One type of dosemeter was based on a thermoluminescence measurement of LiF while quartz powder was used in the other dosemeter, which was analysed by ESR. The dosemeters were encapsulated in small glass vials (outer diameter 8 mm, length 35 mm) and placed inside the ground salt. Both types of dosemeters were especially developed to meet the temperature and dose rate requirements of the HAW in situ irradiation.
The salt samples were irradiated in glass ampoules having a length of 250 mm and a diameter of 49,9 1 mm. These ampoules were filled with 300 grams of ground salt via a glass tube of 8 mm inner diameter. Altogether 250 ampoules were prepared. In order to remove the residual air, the ampoules were evacuated to about 1 mbar and then atmospheric pressure was re-established using either helium or a synthetic air-mixture (oxygen/nitrogen 20:80 v/v), respectively. The process was repeated three times. The ampoules were then sealed gastight by a glass blower.
Throughout this study natural rock salt from the Asse Salt Mine having a grain size of 1 to 3 mm was used . Its mineralogical composition was determined chemically and mineralogically and found to be

halite	97,5 wt.-%
anhydrite	2,0 wt.-%
polyhalite	0,45 wt.-%
total water content	0,03 wt.-%

Thus, these samples are largely representative for the envisaged repository in salt domes in Northern Germany.
Quantitative Determination of Colloidal Sodium
The amount of colloidal, metallic sodium in the crystal lattice was quantified chemically. About 10 g of salt were weighed into a teflon flask (volume about 250 ml) which was sealed with a screw cap equipped with two septa. The weighing in was carried out in a nitrogen atmosphere in a glove box in order to prevent any reaction between the metallic sodium and air humidity. Then 100 ml water were added through a septum and the salt was dissolved. The metallic sodium reacts with the water yielding hydrogen according to

$$Na_{met.} + H_2O \rightarrow Na^+ + OH^- + H_2$$

An aliquot of the gas phase in the teflon flask was withdrawn through a septum using a gastight syringe and analysed gas-chromatographically for hydrogen. For each irradiated ampoule three independent hydrogen measurements were performed. The standard deviation of these determinations was typically 10%.
The amount of colloidal sodium in irradiated salt is typically given in mol-%. It is calculated from the measured gas phase concentration of H2 as follows:
Eq. (1)
with:

mcol.sod.	yield of colloidal sodium [mol-%]
GPC	measured gas phase concentration [ml/ml]
GPV	free gas phase volume of teflon flask [ml]
mNaCl	amount of dissolved rock salt [g]
2	stoichiometry factor, since each colloidal sodium generates H2
MWNaCl	molar weight of NaCl [58,44 g/mol]
MVH2	ideal molar volume of H2 at 298,15 K [24465 ml/mol]

Quantitative Determination of the Radiolytical Chlorine
Molecular chlorine, Cl2, was determined chemically via reacting it with dilute sodium hydroxide solution and subsequent iodometric titration of the resulting hypochloride as follows:

$$Cl_2 + 2\,NaOH \rightarrow NaCl + NaOCl + H_2O$$

Hypochloride reacts with KI

$$OCl^- + 3\,I^- \rightarrow I_3^- + H_2O + Cl^-$$

and the resulting ion I3- is titrated against a redox electrode using 0.01 N sodium thiosulfate solution.

$$I_3^- + 2\,S_2O_3^{2-} \rightarrow 3\,I^- + S_4O_6^{2-}$$

The change in the redox potential is used in order to detect the equivalence point. Chlorine is volatile and may already evaporate from the salt crystal during the irradiation, thus resulting in an underdetermination, if the product was only measured in the bulk. Therefore, chlorine was quantified both in the head space of the glass ampoule and in the bulk of the salt.
As chlorine is extremely reactive, all glass surfaces which come into contact with

the gas sample were treated with hypochloride solution in order to destroy chlorine-consuming compounds (e.g. detergents). Afterwards, the flasks were rinsed thoroughly with Millipore-filtered water and dried.

Determination of Chlorine in the Gas Phase

After irradiation the ampoules were connected to a pump stand, consisting of a glass transfer tube with several connectors to which four glass bulbs, each equipped with a stop-cock and a septum, were attached. The whole system was evacuated using a membrane pump and sealed. Then the break seal at the irradiation ampoule was opened by dropping a small teflon-coated magnet on it, thus allowing the gas sample to enter the void volume. Nitrogen was added in order to adjust a final pressure of 1.5 bar. All pressures, i.e. after evacuation, after opening the irradiation ampoule, and after adding the nitrogen, were recorded for control purposes. Via the septum 5 ml of dilute sodium hydoxide solution (0,01 N) were injected into the glass bulb containing the gas sample. An aliquot of the solution was then withdrawn with a syringe and injected into the titration flask. The exact amount was determined by weighing. Afterwards 70 ml H2O, 20 ml HCl (2 mol/l) and 1 ml KI-solution (300 g/l) were added and the solution was kept dark for 5 minutes. The solution was then titrated. The mass of chlorine gas present in the head space of the irradiation ampoule is calculated as follows:

Eq. (2)

with:

mCl2(g.p.) mass of molecular chlorine in gas phase [mval/kg]
VStd consumption of standard solution [ml]
fStd titer of thiosulfate standard solution.
mNaOH amount of NaOH solution being redox titrated [g]
GPVtotal volume of gas phase in total system [ml]
GPVbulb volume of glass bulb [ml]

Determination of Chlorine in the Bulk of the Salt Crystals

The aqueous solution, that was prepared for determining the amount of colloidal sodium (see above), was also employed for measuring the amount of molecular chlorine in irradiated solid rock salt. After the hydrogen measurement between 5 to 30 ml were removed from this solution, weighed, and analysed for its hypochloride content using the same recipe as described above. The mass of molecular chlorine present in the rock salt is calculated according to:

Eq. (3)

with:

mCl2(salt) mass of chlorine in rock salt [mval/kg]
Vstd consumption of standard solution [ml]
fstd titer of thiosulfate standard solution.
mNaCl mass of dissolved rock salt [g]
msol mass of solution being redox titrated [g]

Then, the total chlorine yield in irradiated rock salt is calculated via

Eq. (4)

with:

mCl2(total) total mass of chlorine being produced upon irradiation [mol-%]

Calorimetric Measurements of the Energy Stored in Rock Salt

The determination of the energy storage in irradiated rock salt was carried out by differential scanning calorimetry (DSC) using the microcalorimeter SCERES B-900 S, that allows the application of sample masses of several hundred milligrams. Halite samples of 260 - 290 mg were weighed into small glass ampoules and heated two times from 30C to 400C at a constant heating rate of 1 K min-1. The reference ampoule contained pure sodium chloride (NaCl, suprapure) which had been heated for 3 weeks at 300C in order to recover defects in the crystalline structure. During the first heating the halite releases the stored energy and the colour of the sample changes from black to white. The second heating process is necessary in order to obtain a baseline under identical measurement conditions but without any reaction peak. In the following data evaluation the baseline (second heating curve) is subtracted from the reaction curve (first heating curve). As a result of this procedure one gets the 'true' exothermic peak without any twist in the baseline.

RESULTS

The g-radiation-induced generation of colloidal sodium depends strongly both on dose and temperature. This is shown in Fig. 1, which presents the data for irradiation

temperatures 100C, 150C, 200C, and 250C, respectively. The data points for high dose irradiations have all been set to 100 MGy. This value was estimated from recorded decay curve for the gamma intensity of the spent fuel elements, as the solid state dosemeters were not calibrated for such a high dose range at the temperatures being used. Recalibration of the solid state dosemeters is presently performed.

Significant colloid formation is observed for 100C and 150C, while only small amounts are produced at 200C and hardly any colloid formation is found for 250C. The data suggest that damage saturation occurs at the two higher temperatures. In contrast, clearly no saturation is obtained yet at 100C and 150C. As expected, the gas atmosphere during irradiation has no influence on the colloid yield. Identical results are otained both for irradiations carried out under nitrogen/oxygen (80/20 v/v) and under helium (data not shown).

At each temperature and dose, colloid formation is accompanied by the formation of equal amounts of molecular chlorine (on a molar basis). Figure 2 presents the respective data for all our irradiation experiments. The scatter at very low yields (i.e. data for low dose irradiations and/or high temperatures) is most likely due to a decrease in the measuring accuracy as the lower detection limits of the hydrogen and hypochloride detection are approached. In most cases slightly lower chlorine yields than colloid yields are obtained. The relative difference is higher at lower absolute yields. This finding is attributed to small chlorine losses during the detection procedure, which inevitably occur even though extreme care has been taken in handling and preparing the equipment used.

The relative portion of chlorine gas which is released from rock salt already during irradiation and, thus, is found in the gas phase increases with temperature. The data for high dose irradiations (duration 80 days, dose about 100 MGy) under helium are given in Table I. At temperatures of 150C and less only insignificant amounts of chlorine are released during the experiment.

The stored energy was measured in some of the samples. The available data are displayed in Fig. 3 versus the amount of colloidal sodium that was detected in these samples. Interestingly, no stored energy could be measured in samples that were irradiated at 200C (triangles in Fig. 3), even though both colloidal sodium and molecular chlorine were clearly detected.

The two lines in Fig. 3 represent conversion factors between stored energy and the amount of colloidal sodium that are published in the literature. The upper line corresponds to a conversion factor of 125 J/g stored energy per mol-% colloidal sodium in rock salt. This number was taken from Groote & Weerkamp (4). The lower line corresponds to a value of 70 J/g stored energy per mol-% colloidal sodium, that was derived by Jenks et al. (3). Clearly all our data fall close to the latter value.

Each data point in Fig. 3 represents the mean of three independent DSC-measurements. In some cases it was observed that the individual results varied considerably (up to 40%). This is likely due to the small rock salt specimens used in the DSC-measurements being different in their mineralogical composition from the bulk of the sample. If the small specimens used in the DSC-measurements contain more sulfate minerals than the bulk, a lower value for the stored energy is expected, since the radiolysis mechanism in ionic crystals depends on the charge of the ions. Currently this hypothesis is looked into experimentally.

DISCUSSION

The present study confirms conclusively that the irradiation of rock salt leads to the formation of equal amounts (on a molar basis) of the molecular products colloidal sodium and chlorine gas over a broad temperature and dose range. This is one of the few investigations, in which the various products have been determined for the same sample. Notable exceptions are the work of Jenks & Bopp (3) and of Groote & Weerkamp (4). However, in these studies the irradiation temperature was restricted to a range below 150C. In order to quantify the formation of molecular chlorine, both the amount present in the crystals and the amount released into the gas phase above the salt must be determined. This is particularly true for temperatures above 150C. A time dependence for the chlorine release is conceivable but was not investigated in the present study.

The formation of radiation defects and of molecular products in NaCl depends, in a complex manner, on the parameters dose, dose rate, and temperature. Also microstructure of the salt and the level of impurity were suggested to have an

influence. In principle, three phases can be distinguished in which the formation of defects is dependent on radiation dose. In an induction phase no significant growth of sodium colloids or other molecular irradiation products occurs. This is followed by a phase in which a more or less linear relatioship exists between defect formation and dose. Under specific conditions a saturation of defect formation occurs. When considering waste disposal it is ultimately the extent of radiation damage to the salt which is of interest. The linear phase and the possible saturation level are therefore of most significance. In the present study saturation is clearly observed at a temperature of 250C (at a very low level of damage). At 200C the start of saturation is indicated. No saturation occurs at 100C and 150C in the investigated dose range. This is in accordance with the extended Jain-Lidiard (JL) theory (5-7), which predicts that at a dose rate of 104 Gy/h damage saturation begins above 300 MGy. Unfortunately, under the boundary conditions of our experiments this dose range is only accessible using prohibitively long irradiation times. The absolute yield of radiation damage observed in our experiments is also in the expected range. At a dose of about 100 MGy, a damage level of about 0.80.1 mol-% was determined. For comparison, using the JL-theory values of 1,4 mol-% and 0,6 mol-% are estimated for dose rates of 104 Gy/h and 105 Gy/h, respectively, for that dose.

The amount of energy that is deposited in the salt due to the formation of molecular radiation products is of particular importance when the thermo-mechanical consequences of a spontaneous release of the energy is considered. Our data seem to indicate that one mol-% colloidal sodium corresponds to a stored energy of 70 J/g. This value agrees well with the one given by Jenks et al. (3). From their data Groote & Weerkamp concluded a value of 125 J/g (4). The reason for this difference is presently not clear. However, one reason may be the different type of samples used in the experiments. While natural rock salt samples were used in our experiments, Groote & Weerkamp employed melt-grown salt crystals.

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WATER ROCK INTERACTIONS IN SALT FORMATIONS AND SAFETY ASPECTS OF REPOSITORIES

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ABSTRACT

Two very common water rock interactions in salt formations are discussed in the view of the safety considerations for underground repositories. These processes, the "Hartsalz" and "Carnallitit" dissolution were studied in two large scale in situ experiments. The results are presented and compared with the results of the geochemical modelling with the computer code EQ3/6. Geological field observations and mineral assemblages agree well with the results of the geochemical modelling employing the Pitzer formalism along with the Harvie, Mller and Weare database. The chemical compositions of the high saline solutions resulting from these processes are given along with the quantitative reactions which lead to such compositions. It is concluded that once the mechanisms of the chemical reactions are well understood it becomes possible to evaluate realistically whether such processes, when encountered in the repository, are still active or whether they are

finished. It also becomes possible to estimate the volume changes associated with the reactions and thus the impact of these reactions on the integrity and the geomechanical stability of the salt formation. The intimate knowledge of the reaction mechanisms during water rock interactions is a useful tool which helps to distinguish between geologically old and young brines, between brines from the outside or the inside of the salt formation, i.e. between hazardous and harmless brines in the repository.

BACKGROUND

In the safety assessment of a repository in salt formations - according to the German concept - the investigation of the hypothetical accident scenario of water or brine inflow is of particular significance. The water path is considered to be the most important way for the mobilization and release of radionuclides from the repository. The volume and composition of the contaminated brines is largely dependant on the accident scenario which is considered in the safety assessment. Two different accident scenarios are discussed in Germany:

- the limited scenario with a volume of maximum 1000 m3 of intruding brine
- the unlimited scenario with a brine volume equal to the volume of the repository (complete flooding)

The results of the safety assessment with these two different scenarios differ significantly. Therefore good reasons are needed for the selection of the most realistic scenario. The brine volume which has to be considered depends mainly on the origin of the brines and the access ways into the repository. Brines genetically linked to the origin of the salt formation generally have a small volume, whereas brines from the overburden have an unlimited reservoir and can completely flood the repository, provided that access ways exist. The knowledge of the origin of brines provide information on possible pathways within the salt formation and possible connections to the overburden.

Old brines formed in the geological processes which led to the formation and partial transformation of the salt formation have not only a relatively small volume but generally are trapped within the salt formation. Such brines have no connection with the overburden. They can enter the repository if they are encountered during the exploration and construction phase of the repository. But as they have no connection with the overburden their reservoir is restricted. No major brine inflow has to be expected. The practical experience made in the salt and potash mining industry shows that such brines generally dry up after a short period of inflow. If they are encountered during the operational phase of the repository they can easily be collected and removed. Once they have dried up no further brines from that reservoir must be expected. Such brines are harmless.

The situation is completely different if the intruding brines come from the overburden. Generally they don't dry up. Often they dry up in one place and reappear quickly in another place in the mine. Eventually they may completely flood the repository. Such brines are dangerous. It is essential to have tools at hand to distinguish such brines from harmless brines. Often the two different types of brines have a similar chemical composition.

This discussion shows how essential for the safety of a repository it is to collect reliable information on the origin of brines encountered during the exploration phase of the repository. In addition the discussion illustrates the limited value of the limited accident scenario in the safety assessment. It is simply not possible to demonstrate that using the limited scenario is a conservative approach.

Geomechanical processes may lead to an inflow of harmless brines with a small reservoir in the postoperational time of the repository. In this case the limited scenario would be the more realistic one. But it can not be excluded that geomechanical processes can cause the inflow of large volumes of brines from the overburden. This would require the consideration of the unlimited accident scenario. As no supervision of the repository in the postoperational time is required by the German concept such processes and the intrusion of brines would occur unnoticed. As nobody can tell in this case which kinds of brines will enter the repository the only conservative approach is the unlimited accident scenario.

In order to be able to distinguish during the operating time of a repository between hazardous brines from the overburden and harmless brines from the interior of the salt formation a thorough understanding of the geochemical reactions during water rock interactions is needed.

In the following chapters the results of two large scale in situ experiments of

relevant water rock interactions in salt formations, "Hartsalz" and Carnallitit" dissolution are discussed and compared with results of the geochemical modelling with EQ3/6, using the Pitzer-formalism (1) and the Harvie Moller, Weare database (2).

WATER ROCK INTERACTIONS IN SALT FORMATIONS

"Hartsalz" Dissolution (H_2O + sylvite + kieserite + halite + anhydrite)

"Hartsalz" is the german term for a common and valuable secondary potash formation within the Zechstein salt sequences in Northern Germany. There are different types of "Hartsalz", like kieseritic, anhydritic and langbeinitic "Hartsalz". The main components of all "Hartsalz" types are sylvite and halite. Minor components are different sulfates like kieserite, anhydrite and langbeinite which give the name of a specific "Hartsalz" type. All "Hartsalz" types were formed by the interaction of water with the primary potash formation "Carnallitite" which in turn was formed by the primary precipitation from the evaporating sea water. As the kieseritic "Hartsalz" is the most common "Hartsalz" type in the Zechstein salt formations the dissolution behaviour of this type was studied in detail.

A large scale in-situ experiment was carried out in the potash mine Hope, near Hannover in Northern Germany from 1984 to 1992. In the Hope mine "Hartsalz", consisting of 61.4% halite, 31.7% sylvite, 3.4% kieserite and 1.7% anhydrite was mined. The abandoned mine was flooded with a NaCl saturated brine. A research programme was set up to record and evaluate data on the geochemical, geomechanical and geophysical processes occurring during and after flooding. The geochemical program monitored the changes in the chemical composition of the charged NaCl solution in contact with "Hartsalz" over a period of 7 years. Preliminary results of the experimental program and of the geochemical modelling with EQ3 and EQ6 were published by (3 and 4). Later a more complete evaluation of the results and a detailed comparison between experimental data and geochemical modelling was carried out by the author. Some of the results of this work are illustrated in Fig. 1 and 2. Figure 1 shows the chemical evolution of the brine, the amount of rock affected by the dissolution process and the time needed for the solution to reach different concentrations. The symbols represent samples taken at different times from different locations in the mine, whereas the lines show the evolution as calculated with the geochemical code EQ3/6. The vertical line in Fig. 1 marked with R1 indicates the reaction step for which a quantitative reaction comprising the initial solution, the resulting solution, all reactants and all products, was calculated using the results of the geochemical modelling. The experimental results cover only a small part of the entire reaction. The line R1 in Fig. 1 corresponds to R1 in Fig. 2. The lines R1 - R7 indicate the steps for which quantitative reactions were calculated as well. As we know the composition of the initial and resulting solutions and the reactants involved in the reaction and considering the good agreement between experimental data and computer modelling at each step of the reaction, we conclude that we can extrapolate the results reliably beyond the limits of our experimental data. Figure 2 demonstrates, that during the reaction water is continuously consumed and the reaction terminates when no water is left. The water is consumed during the reaction primarily via the formation of the mineral kainite. During the reaction (R7, Fig. 2) 1 kg of water in the initial solution can theoretically affect about 90 kg of "Hartsalz". About 27 moles of sylvite, 22 moles of kieserite and 11 moles of anhydrite can be dissolved and 5.5 moles of carnallite, 5.7 moles of halite, 11 moles of kainite and 5.5 moles of polyhalite can precipitate. Between R6 and R7 the chemical composition of the solution does not change anymore. It has reached the invariant composition of an IP 21 solution in the six component system Na-K-Ca-Mg-Cl-SO_4. This solution is saturated with halite, sylvite, carnallite, kainite and polyhalite. In spite of the solution staying constant in composition between R6 and R7, the dissolution continues, as the solution is not yet saturated with kieserite and anhydrite. The dissolved amounts of $CaSO_4$ and $MgSO_4$ are consumed by the formation of new kainite. This process finally stops when no water is left. In Table I the quantitative reaction at point R6 of the entire reaction path is given.

Whereas in flooded "Hartsalz" mines the resulting solutions normally reach the composition of an IP21 solution (see final solution in the reaction R6 in Table I, b), it is very unlikely that the reaction will proceed under natural conditions to the very end (R7 in Fig. 2) where no water and no solution is left. The reaction will stop whenever one of the reactants is exhausted or has no more direct contact

with the solution. Therefore it is very important to have natural analogues, i.e. flooded potash mines, in order to be able to show which steps of the theoretical reaction can be reached under natural conditions (different mineralogical compositions of affected potash rocks, different outcrops etc.). It is equally important to have a thorough theoretical understanding of the expected reactions. Provided a good agreement between experimental results and geochemical modeling as in this example, the modelling can give a deep insight in the mechanisms of the reaction.

"Carnallitit" Dissolution (H_2O + carnallite + kieserite + halite + anhydrite)

"Carnallitite", a potash rock consisting mainly of carnallite, kieserite and halite, is very common in the Zechstein salt sequences in Northern Germany. The dissolution behavior of this salt formation, which is very different from that of "Hartsalz", has been studied in detail as well. Whereas "Hartsalz" is a secondary potash formation, "Carnallitit" was formed by the simultaneous precipitation of carnallite, kieserite and halite from the evaporating sea water. The rock "Carnallitit" can contain minor amounts of sylvite, anhydrite and clay minerals.

The dissolution of "Carnallitit" was investigated in a vertical borehole drilled into the "Carnallitit" formation in the Asse Mine, near Braunschweig. The borehole, 40 cm in diameter and 4 m deep, was filled with a NaCl saturated brine. The change of the chemical composition of the brine was measured over a period of 4.5 years. The experimental results and the results of the geochemical modelling are shown in Fig. 3 and Fig. 4. As in the "Hartsalz" experiment, a good agreement between experimental and theoretical data was obtained. In the "Carnallitit" dissolution experiment the boundary conditions were more favorable, which led to a more complete coverage of the theoretical curves with experimental data. i.e. most of the total reaction path was covered by the experiment. As we have a good agreement between the two sets of data, this experiment is even more valuable in terms of demonstrating the ability of the EQ3/6 code to reliably predict the dissolution reactions in salt formations.

For the safe disposal of hazardous wastes in salt formations it makes quite a difference if "Hartsalz" or "Carnallitit" is present in the openings of the repository. Whereas in the dissolution reaction of "Hartsalz" the initial amount of water is gradually consumed by the formation of hydrated minerals in the dissolution reaction of "Carnallite" the initial amount of water increases (compare Fig. 2 and 4). The line R4 in Fig. 3 marks the composition of an invariant solution of the six component system Na-K-Ca-Mg-Cl-SO_4 called IP19. This solution is saturated with halite, carnallite, kieserite, kainite and polyhalite. It is not yet saturated with anhydrite. The saturation with anhydrite is reached at line R5. The chemical composition of the IP19 solution and the quantitative reaction which leads to the composition of an IP19 solution (R4) is given in Table II.

Beside the differences in the dissolution behaviour of the two potash formations "Hartsalz" and "Carnallitit" there are also similarities. The resulting solutions IP 21 and IP 19 are saturated with halite, carnallite, kainite and polyhalite and have consequently very high Mg concentrations (above 100.000 mg Mg/kg H_2O).

CONCLUSIONS

By describing two relevant water rock interactions in salt formations and comparing experimental results with equilibrium calculations it was demonstrated, that we have a good understanding of these processes. The close correspondence between the calculated and experimental results of the large scale field experiments, the "Hartsalz" and "Carnallit" dissolution, indicates that the geochemical computer code EQ3/6 with the Pitzer-formalism and the Harvie, Moller, Weare database yield reliable predictions.

It can be assumed that the long term mineralogical changes establish an equilibrium between the brines and the surrounding minerals. Such equilibria can be calculated accurately for the six component system Na-K-Ca-Mg-Cl-SO_4 at 25C. In spite of the general good agreement of the results of in situ experiments and geochemical modelling, differences are noted in the sulfate saturation of these natural, young brines and the calculated compositions of these brines. In older brines the sulfate supersaturation tends to disappear. Sulfate supersaturation should not be present in very old brines, which are remnants from the long term geological processes. It is worthwhile to investigate if these differences can be used in order to distinguish reliably old from recent brine compositions. It is especially important, therefore to study not only the mineral assemblages of the long term geological processes but

also the fluid phases.
From this investigations we conclude that once the mechanisms of the chemical reactions are well understood it becomes possible to evaluate realistically whether such processes, when encountered in the repository, are still active or whether they are finished. It also becomes possible to estimate the volume changes associated with the reactions and thus the impact of these reactions on the integrity and the geomechanical stability of the salt formation. The intimate knowledge of the reaction mechanisms of the short and long term changes in the mineralogical assemblages and the associated brine chemistry is a first prerequisite for the correct evaluation of the origin of brines and essential for the correct evaluation of the hazards which brine inflows may pose for the safety of a repository in salt formations.
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24-3
GEOCHEMICAL ATTENUATION: A COMPONENT OF ENGINEERING WASTE DISPOSAL CELLS
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ABSTRACT
The primary function of waste repository sites is to isolate waste or waste-derived leachate from the underlying soil and ground water. The beneficial, attenuating properties of geologic materials have been long recognized and considered during siting of mining and radioactive waste repositories. To attain isolation of waste from the environment, engineering designs often include installation of a synthetic or natural clay bottom liner with a leachate collection system and a low-permeability cover. The engineering performance standards addressing isolation systems are mandated by regulations; however, regulations also recognize natural factors retarding or blocking the migration of contaminants away from a waste repository. In the United States, the owner of the waste repository may be exempted from stringent performance standards if the attenuating capacity of the soil present between the landfill and ground water is demonstrated. In Europe, investigation must include a study of the possible purifying power of soil and subsoil. Therefore, an efficient engineering design must take into account the natural ability of underlying soils to preclude migration of leachate components.
This paper discusses methods of identifying and quantifying geochemical barriers, both in the field and in the laboratory, and of integrating the barriers into repository engineering design. Also presented are two case studies where natural geochemical barriers were shown to mitigate the migration of heavy metals and redionuclides.
INTRODUCTION
Mining and radioactive waste, when contacted by water from precipitation, will generate mostly acidic leachate. Migration of leachate into the strata underlying waste will first contaminate the vadose zone and, if not retained in the vadose zone, will eventually contaminate the underlying ground water. Mechanisms by which percolation of leachate occur are illustrated in Fig. 1.
Engineered top covers, bottom liners, and leachate collection systems are a first line of defense against leachate migration. The second line of defense (the focus of this paper) is the geochemical barrier created by natural attenuating properties of soils underlying the repository.
The beneficial, attenuating properties of geologic materials have been long recognized and considered during siting of radioactive waste repositories. A prime

example is the ongoing investigations of Yucca Mountain, Nevada as a potential site for an underground, high-level nuclear waste repository. The presence of abundant zeolitic tuffs underlying the region was a major consideration in selection of this site for study. The zeolites are highly selective for certain cationic radionuclides and could potentially be a natural barrier to radionuclide migration (Broxton et al., 1987).

The ability of natural clay materials to retard the movement of radionuclides also has been recognized (Proust et al., 1990). The Boon clay formation in Belgium has been under consideration as a potential host formation for wastes from nuclear power production. The low permeability and high ion exchange capacity of the clay formation are considered to provide an efficient natural barrier against the release of radionuclides (Patyn et al., 1989).

Clay minerals also have been shown to attenuate heavy metals, such as cadmium, lead, and zinc (Griffin, et al, 1976). Sandstone, siltstone, and shale were found to immobilize arsenic, chromium, lead, silver, thorium, and vanadium from uranium mill tailings leachate (Serne, et al, 1983). Organic compounds are known to be attenuated by carbonaceous matter in soil (Schellenberg, et al, 1984). These and similar studies suggest the appropriateness of incorporating natural attenuation into the design of hazardous waste repositories.

Geologic materials provide natural barriers to radionuclide migration because numerous radionuclides are highly adsorptive. For example, 90Sr, 137Cs, 60Co and 226Ra are considered moderately to strongly adsorbed in most types of clayey or silty sediments (Cherry et al., 1979). Cesium-137 can be used to measure soil erosion rates because of its strong adsorption onto soil particles (Ritchie and McHenry, 1990).

Sorption ratio (Rd) values determined for geologic materials provide further evidence of their natural attenuating capacity. For example, Rd values determined in the Yucca Flat area, Nevada, indicated that the sorptive capacity of natural valley fill material could retard transport of actinide radionuclides in vadose water by two or more orders of magnitude in comparison to the movement of water (Winograd, 1981). The valley fill contained significant content of zeolitic, clayey, and other tuffaceous detritus.

Because of the importance of natural attenuation, a legal obligation to define geochemical barriers has been incorporated into environmental regulations of the United States. This regulation [40 Code of Federal Regulations (CFR) 264.301 (b)(3)] provides for an exemption from a costly liner and leachate collection system if "alternative design and operation practices, together with location characteristics, will prevent the migration of any hazardous constituents into the ground water or surface water at any future time" (emphasis added). In European Community countries, "a study of the hydrogeological conditions of the area concerned, and of the possible purifying power of the soil and subsoil," must be done (Johnson and Corcelle, 1989, p. 71).

Consideration of a geochemical barrier to leachate migration in the design of a new waste repository offers simplified and less-costly construction. Savings may exceed 30 percent of the cost of "full-scale" Resource Conservation and Recovery Act (RCRA) repositories. In the case of existing or proposed repositories, demonstrating the presence of a competent geochemical barrier may reduce efforts to obtain required permits.

RECOGNITION OF GEOCHEMICAL BARRIERS

Most soils will provide some geochemical barrier, but certain soil materials are typically much more effective. These include clays, calcareous units, iron or manganese soil coatings, and carbonaceous materials. These materials in underlying soils are the first indication of the presence of a geochemical barrier.

The geochemical reactions causing geochemical barriers are governed by the characteristics of the soil as related to the characteristics of the leachate. The change of leachate pH from acidic to neutral or alkaline due to its reaction with soils is of utmost importance in migration prevention. Also of importance are the absorptive properties of clay, especially when clay and other soil particles are coated with iron and/or manganese oxides, not an uncommon phenomenon.

Figure 2 schematically shows the various reactions as the pH of the leachate increases. The important reactions are: a) the co-precipitation of heavy metals and other ions along with iron and manganese found in the leachate; and b) adsorption of the ions on the surface of precipitated and co-precipitated oxides.

For the non-chemist, simplified definitions of co-precipitation and adsorption are as follows:

Co-precipitation involves metals becoming "entangled" in the lattice of oxides and hydroxides of iron and manganese. When the iron and manganese precipitate, other metallic elements are retained within the metal oxide and hydroxide precipitates.

Adsorption occurs when heavy metals, such as iron and manganese, precipitate and form solids with charged surfaces. Other metals are attracted to these surfaces and are held by sorption mechanisms on the surface. Henceforth, further transport of the elements ceases unless geochemical conditions are altered.

In addition to looking for the presence of calcareous, carbonaceous or clay materials, investigation of geochemical barriers should also include quantification of total organic carbon (TOC) in soils. Organic material has a large attenuating capacity for some metals.

Figure 3 is an example of the separation of leachate components as a result of a geochemical barrier. In this case, the leachate was an acidic plume from mine wastes. The leachate originally contained several metals and sulfate, but the caliche underlying the mine waste served as an effective geochemical barrier. Sulfate, which is least affected by interaction with the caliche, traveled the farthest and was located outside the hatched area shown on the figure (i.e., in "unaffected soil"). The metals in the leachate were retarded relative to the sulfate and remained within the hatched area. The confined area containing metals relative to sulfate distribution is evidence of the geochemical barrier.

QUANTIFICATION OF GEOCHEMICAL BARRIERS

Quantification of the capacity of geochemical barriers requires investigation of critical properties of the soil which typically begins with an examination of the available boring logs to obtain information on the mineralogy and content of organic matter.

Laboratory tests can provide quantitative data about geochemical barriers. Although column tests produce reliable results, the test can be time-consuming. Therefore, sequential batch tests can be used to speed the testing process by discretizing the column process into distinct sequential batches. Sequential batch tests simulate continuously-leached columns wherein successive seepage solution comes into contact with soil (Houle and Long, 1980).

A variety of information can be obtained from the sequential batch tests. For example, the weight of the constituent retained can be compared to the weight added to determine attenuating capacity (Houle and Long, 1980). Contaminant partitioning ratios between soils and water can be calculated for use in transport modeling. Also, the necessary thickness of attenuating material required to provide an adequate geochemical barrier or, conversely, the attenuation capacity of the actual material present can be calculated.

CONSIDERATION OF GEOCHEMICAL BARRIERS IN ENGINEERING OF WASTE REPOSITORIES

If quantitative analysis of a geochemical barrier demonstrates that underlying strata attenuate leachate migration, consideration of the geochemical barrier can be incorporated into the repository design. In accordance with CFR 264.301, the design requirements for the bottom liner of new repositories built in the United States after January 1992 call for construction of a two-component liner system. The upper component of the bottom liner is designed to prevent the vertical migration of hazardous constituents into the bottom liner. The lower component of the bottom liner is designed to minimize vertical migration of hazardous constituents should the upper component of the bottom liner become breached. The typical RCRA-mandated design is shown on Fig. 4, Detail A.

If the site-specific geochemical and/or hydrogeological data indicate that the native soil has sufficient capacity to attenuate migration of hazardous constituents, the lower components of the liner may be eliminated or minimized. Figure 4, Detail B, depicts the liner modified for favorable geochemical conditions. Table I shows a cost comparison for construction of a conventional bottom liner system and a modified bottom liner system for a hypothetical five-acre repository. Elimination of the lower component of the liner may result in a net capital cost reduction of about 35 percent.

CONSIDERATION OF GEOCHEMICAL ATTENUATION IN PERMITTING PROCESSES

Demonstrating the presence of a geochemical barrier can also be helpful to a regulatory process relating to assessment of liabilities from past operations or to permitting new facilities.

Case Study 1 - The operator of a large copper mine and mill in the southwestern United States had disposed of acidic copper tailings over an area of approximately 90 acres. Disposal continued for more than 100 years. The investigation to determine the impact of tailings disposal revealed the absence of metals in ground water. To ensure that metals would not contaminate the ground water in the future, the regulatory agency directed the characterization of the geochemistry of soils underneath the impoundments.

The soil samples from borings were analyzed for soil pH, acid neutralizing potential (ANP), cation exchange capacity, carbonate content (as CaCO3), and metals abundance in tailings, including cadmium, chromium, copper, iron, lead, manganese, and zinc. Background soils were found to be alkaline with high ANP. The thickness of this soil between the surface and the ground water table was 500 feet.

When calcareous soil was contacted by acidic tailings leachate, the ANP of some of the soils was compromised, yet metals were retained in soils prior to reaching the ground water. This investigation demonstrated that only five to ten percent of the geochemical barrier separating tailings from ground water had been used in the course of more than 100 years of tailings disposal. This demonstration, along with the documented absence of metals in ground water, alleviated the primary concerns of regulatory agencies in the process of site permitting.

Case Study 2 - An application of the method relying on measurement of iron and manganese coatings was used during design of a repository for flue dust from an abandoned copper smelter in the northwestern United States. Extractable iron (representing iron coatings on soil) ranged from 220 to 2,520 milligrams per kilogram of soil.

Based on expected leachate composition, arsenic was the only chemical of concern. A published study (Pierce and Moore, 1982) was used to obtain information about the attenuating capacity of iron oxide with relation to arsenic. The resulting calculation indicated that the soil had the capacity to adsorb a minimum of 160 milligrams arsenic per liter volume of soil, which was well above the expected loading. Since the repository is positioned 80 to 100 feet above the water, a breach of the geochemical barrier over the course of time is not likely.

The final decision for the flue dust disposal was to solidify the flue dust to pass the Toxicity Characteristic Leaching Procedure test. The geochemical barrier below the repository provided a redundant protection of underlying ground water.

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24-4
THE SORPTION BEHAVIOR OF CS & CD ONTO OXIDE AND CLAY SURFACES
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ABSTRACT
The sorption of Cs and Cd on model soil minerals was examined by complementary analytical and experimental procedures. X-ray photoelectron spectroscopy (XPS) and nuclear magnetic resonance (NMR) spectroscopy were used to characterize the chemical and physical nature of Cs-reacted soil minerals. Cd and Cs adsorption isotherms for kaolinite were also measured at variable pH and temperature to establish likely reaction stoichiometries, while atomic force microscopy (AFM) was used to characterize the microtopography of the clay surface. XPS analyses of Cs-exchanged samples show that Cs is sorbed at mineral surfaces and at the interlayer site of smectite clays, although the spectral resolution of XPS analyses is insufficient to differentiate between basal, edge or interlayer sites. 133Cs MAS-NMR results also show that Cs is adsorbed primarily in an interlayer site of montmorillonite and on edge and basal sites for kaolinite. Cd adsorption isotherms on kaolinite were found to be additive using Al2O3+SiO2 Cd binding constants. AFM quantification of kaolinite crystallites suggest that edges comprise up to 50% of the BET surface area, and are consistent with NMR and surface charge results that Cs and Cd adsorption occur primarily at edge sites.
INTRODUCTION
Effective disposal of radioactive and mixed waste in near-surface low level radioactive waste (LLW) facilities will ultimately depend upon the geochemical interactions of dissolved aqueous waste with common oxide and silicate minerals. These minerals, which range in complexity from quartz to smectite-like clays, are found in the adjacent soils, if not in the engineered waste facility. The geochemical controls for sorption of hazardous chemical and radioactive wastes onto minerals is often governed by the structure and reactivity of those mineral surfaces (1,2). A fundamental understanding of those mineral-fluid interactions is important for proper evaluation of the waste disposal and retention scenarios, as the site licensing agency must complete a performance assessment (PA) of the proposed LLW disposal site in accordance with regulations. The key to PA is the ability to model transport and retardation of dissolved radionuclides from a LLW disposal site. Current geochemical models utilize retardation factors (KD) to handle the uncertainties involved in radionuclide transport (3,4). These models work reasonably well when the site is well characterized, but mixing of groundwater and waste leachate may produce conditions not easily modeled by the KD method. Although theoretical models exist for modeling sorption processes on minerals, experimental verification of those models lags, as does atomistic characterization of the retardation mechanism.
This study aims to examine the fundamental mechanisms and processes by which soil mineral phases absorb metal cations, specifically non-radiogenic Cs and Cd. Because sorption onto the surfaces of structurally and chemically heterogeneous soil

minerals is complex and not very well understood, we have emphasized complementary experimental and analytical studies of metal sorption using model soil minerals. Specifically, we have employed modern spectroscopic techniques, such as X-ray photoelectron spectroscopy (XPS), solid state magic-angle spinning (MAS) nuclear magnetic resonance (NMR) spectroscopy, and atomic force microscopy (AFM), to characterize surface and bulk properties of reacting soil minerals. Adsorption isotherm measurements were also done at variable pH and temperature to establish likely reaction stoichiometries and bonding enthalpies.

METHODS

Analytical and experimental studies of the cation sorption processes were performed using both natural and synthetic soil mineral samples, including quartz (SiO_2) and corundum (Al_2O_3), gibbsite ($Al(OH)_3$ from Ouro Preto, Brazil), kaolinite ($Al_2Si_2O_5(OH)_4$ from Warren Co., Georgia), and montmorillonite (Ca-interlayer clay from Apache Co., Arizona). These minerals were chosen to provide increasingly complex structures and chemical sorption sites whereby the process of metal sorption onto soil minerals might be determined. For example, quartz and corundum are simple anhydrous oxides consisting of corner-sharing Si tetrahedra or edge-sharing Al octahedra, whereas gibbsite consists of a partially-occupied Al layer which is octahedrally coordinated by hydroxyl ions. Kaolinite is a 1:1 layer type clay mineral comprised of a tetrahedral SiO_4 layer linked to an octahedral gibbsite layer (TO structure, see Fig. 1). Montmorillonite is a complex 2:1 layer type smectite clay (TOT structure), composed of an Al-deficient octahedral sheet sandwiched between two inward-pointing SiO_4 tetrahedral sheets, where interlayer cations satisfy charge balance. Characterizations of powdered minerals and reacted samples were made using standard analytical techniques (AA, DCP, EDS, ICP-MS, TEM, and XRD) in order to verify the structure of starting materials and reacted phases, and to quantify solution compositions.

After a series of Cs sorption tests, several surface-sensitive spectroscopic techniques were used to characterize the extent and site-specific control of Cs sorption onto these minerals. These techniques include X-ray photoelectron spectroscopy (XPS), which is sensitive to the near-surface (<<20) composition and chemical state of elements, and nuclear magnetic resonance (NMR) spectroscopy, which yields structural and bonding information at the molecular level. In these tests, approximately 1 g of the mineral was reacted with 100 ml of 0.1 or 0.01 M CsCl solutions in polyethylene bottles for 5 days at 25 and 50C. A shaking bath was used to maintain constant temperature and suspension of the sample. After the test, samples were separated by filtration. The XPS analyses were performed under vacuum (~ 2 x 10-9 torr) using a Perkin Elmer PHI 5400 instrument employing Mg Ka X-rays and a take-off angle of 45. Adventitious carbon (C 1s binding energy) provided an internal standard for correcting energy shifts resulting from sample charging. The 133Cs NMR spectra were collected at room temperature and at 65.5 MHZ using a spectrometer with a 11.7-T superconducting magnet. Spectra for most samples were collected at room humidity, however, a few kaolinite and montmorillonite samples were collected at 100% relative humidity (RH) in order to evaluate interlayer site occupancy and exchange. Some samples were also cooled or heated in order to evaluate thermal effects upon site occupancy. The 133Cs chemical shifts are reported in parts per million (ppm) relative to an external 0.1 M CsCl solution at room temperature. Cs and Cd adsorption onto kaolinite was measured as a function of pH at 25 and 50C. The titrant for the adsorption measurements was 0.1 M NaOH and the pH electrode was calibrated using NBS standards at the temperature of interest. Four g of kaolinite powder (BET surface area 10 m^2/g) were titrated in 50 mL of 0.1N $NaClO_4$ using a Mettler DL12 titrator. Slurries were continuously stirred and N_2 gas was pumped directly into the solution prior to and during each titration to purge CO_2. Temperature was maintained by adjustment of a hot plate and water bath in which the reaction vessel was immersed. Certified Cs and Cd AA standards were diluted to ppm levels and used as stock solutions. A 0.1 M solution of $NaClO_4$ was used as the background electrolyte in these adsorption experiments because ClO_4^- complexes with neither Cs nor Cd. Total metal concentrations in sampled solutions were measured by DCP or ICP-MS analytical methods. Cs and Cd adsorption was determined by monitoring metal concentrations as the solutions were titrated to pH ~9. The dissolved metal and pH data were regressed using FITEQL (5) to calculate best fit values of pKx and site density for Cs and Cd as a function of temperature.

Relative proportions of basal and edge surface area for untreated kaolinite were

determined using AFM techniques. Kaolinite was ultrasonically suspended in water for 1 minute. 50 mL of this solution was placed on freshly-cleaved muscovite and air-dried. Images were obtained in contact mode using a Park Scientific Instruments AutoProbe LS (5 mm scanner) and 0.6 mm thick Ultralevers (silicon cantilever and tip). Scanning speeds ranged from 0.5 to 2 Hz. Particle dimensions and step heights were measured on raw images of a few representative grains. These dimensions were used to calculate relative amounts of edge and basal surface area. Estimated errors for calculated surface areas are 10% based on an averaged 5% uncertainty in a single height or length measurement.

RESULTS

A Cs XPS signal was not observed for any of the starting materials, although several Cs peaks were observed for all reacted samples but gibbsite. EDS analyses acquired during XPS analysis indicate that powdered corundum sorbed ~0.3 mole % Cs; the kaolinite slab exhibited Cs sorption at the 1 mole % level; and powdered montmorillonite sorbed ~2 mole % Cs. Sorbed Cs concentrations were reduced by 50-75% after Ar sputtering of the near surface. Based upon these results, the relative Cs sorption capacity for these minerals is montmorillonite >> kaolinite >> corundum >> gibbsite. In addition to near-surface compositional profiles, XPS can measure shifts in the binding energy of core electrons resulting from a change in chemical environment (chemical shift). These shifts result from a change in the nearest neighbor, oxidation state or crystal structure. We did not observe a chemical shift for Cs sorbed onto montmorillonite nor kaolinite. Multiple surface or interlayer sites for Cs could not be resolved from the single peak Cs spectra observed for clay mineral samples (Fig. 2). These same samples were also characterized by 133Cs magic angle spinning (MAS)-NMR spectroscopy. There was no observed 133Cs MAS-NMR spectral signal from the corundum or gibbsite samples. The NMR spectra for the montmorillonite samples contain one broad peak at -18 to -12 ppm, whereas the kaolinite samples contain multiple peaks (Fig. 2b), including a large, relatively narrow peak at a more shielded chemical shift (-40 to -25 ppm) along with several other weak peaks. Higher concentrations of Cs in the reacting solution, as well as lower temperatures and higher humidity, resulted in less shielded chemical shifts. There was increased peak broadening for kaolinite and montmorillonite with decreasing temperature, although separate peaks were unresolved even at -80C.

Cs and Cd adsorption measurements onto a kaolinite clay substrate show that metal binding is favored at high pH, although Cd is more strongly bound to kaolinite relative to Cs (Fig. 3). Generally, association of metals with hydroxylated mineral surfaces parallels metal-hydroxyl association at high pH in aqueous solutions. The weak binding of Cs relative to Cd therefore, arises from its proportionally weaker tendency to hydroxylate. Potentiometric titrations were done on Al2O3 (6) and SiO2 (7) to resolve pH-dependent surface charge and calculate Al and Si surface site acidities. Adsorption measurements were subsequently done with Cd and regressed to calculate metal binding constants for each oxide. Potentiometric titration measurements were also made with kaolinite slurries at 25 and 50C to evaluate the pH-dependence of multi-site surface charge as a function of temperature. Proton donor-acceptor reactions were found to occur simultaneously on the Si and Al sites exposed at basal planes and edges. We found that the Si site acidity at the kaolinite-solution interface differs minimally from that of pure SiO2, whereas Al sites became appreciably more acidic with substitution into the kaolinite matrix. Increasing temperature causes both Al and Si sites to become more acidic, the Si sites more so than the Al sites. Calculated site densities increase with increasing temperature suggesting appreciable surface roughening. The combination of increasing site acidity and density points to kaolinite having a greater sorptive potential at higher temperatures.

Particles of untreated kaolinite ranges from 0.1 to 0.8 mm in diameter and 100 to 1200 in total thickness. Edge to basal surface area ratios range from 15 to 50%. Edge surface area from small steps on the surface (Fig. 1b), typically 1 to 10 unit cells high (7 to 70), increases the edge area obtained from the gross shape of the grain by a minimum of 10%. The additional contribution of step edge area depends on the number of and average height of the steps. The potentiometric titration results require a site density of 2.25 sites/nm2 to explain the amount of adsorption assuming that the entire BET surface area sorbs hydrogen or hydroxyl ions. Crystallographic site densities are estimated to be 3 to 6 sites/nm2. Therefore, if adsorption occurs only at edge sites where the density of charged sites is highest,

then a minimum of 38% of the BET surface area must be edge surface. This amount of edge surface is approximately the median of the range gleaned from limited AFM observations.

DISCUSSION

One of the primary goals of this study is to identify the specific sites for metal adsorption. Two spectroscopic techniques were employed to characterize Cs-sorbed minerals. XPS surface spectra of reacted montmorillonite samples suggests almost complete exchange of cesium for the original interlayer calcium. Mineral structures without exchangeable cation sites yielded spectra characterized by the presence of strong Cs signals, indicative of Cs sorption on mineral surfaces and along grain boundaries. The presence of single, unresolvable XPS peaks also suggests that the chemical shifts associated with Cs adsorption on edge or basal sites are not very strong. NMR spectra obtained from Cs reacted montmorillonite indicate the presence of a range of interlayer sites that are dynamically averaged at temperatures above approximately 20C. Room temperature 133Cs MAS-NMR spectra for the kaolinite samples, however, show the presence of as many as four distinct sites. The bulk of these sites are thought to be edge and basal plane sites, in addition to some smectite-like 'exchangeable layer' sites. The latter probably result from the presence of small quantities of smectite-like layers on the kaolinite crystallites. Given the sheetlike structure of kaolinite (see Fig. 1), exposed Si and Al sites at the crystal's edge would provide likely sorption sites suggested by the 133Cs NMR spectra.

We also sought to determine if soil clay KD's could be estimated from simple oxide component reactivities. Cd sorption onto kaolinite at 25C is shown in Fig. 3b as a function of pH. The line through the results was independently calculated from the Cd binding constants measured on the single oxides Al2O3 and SiO2. The quality of this fit is exciting because it suggests that metal binding on clay minerals is additive from the oxide components. Note that metal adsorption isotherms have already been measured on the most important (hydr)oxides in soils. The goodness-of-fit of the two-site model suggests that, of the four sites indicated by the NMR work, only two dominate the overall sorptive capacity of kaolinite. However, interlayer cations and multivalent cations commonly found in smectite clays will probably pose special problems when predicting additive adsorption properties for those minerals given the compositional and surface charge variability of expandable 2:1 layer clays. We are, therefore, in the process of extending our efforts to specifically account for these effects. The end result of this work will, ideally, be a methodology by which KD's can be predicted for soils.

AFM examination of the untreated kaolinite demonstrates that edges may comprise over 50% of the available sorbing surface. It is known that metal sorption onto clay surfaces is a function of the charge distribution on the surfaces as well as the absolute proportions of exposed planes of differing surface charge. Although total surface areas can be measured using routine BET methods, the best way to determine proportions of edge and basal surface is to use an imaging technique where all three dimensions can be determined on individual particles. The easiest method for doing this is AFM (8,9). Basal kaolinite surfaces often contain numerous steps of one to a few unit cells in height. AFM examination of a few kaolinite grains revealed at least a 10% increase of edge surface area from such steps beyond that estimated from average particle dimensions. Better statistical characterization of surface area proportions could affect significantly the interpretation of both NMR and experimental adsorption data.

CONCLUSION

XPS analyses of Cs-reacted soil minerals show that Cs is sorbed on the surfaces of corundum, kaolinite, and montmorillonite, but not on gibbsite. The XPS spectra suggest that Cs is sorbed on the interlayer site of montmorillonite and possibly on kaolinite, although the spectral resolution of XPS analyses is insufficient to differentiate between basal, edge or interlayer sites. 133Cs MAS- NMR results demonstrate that Cs is adsorbed in a motionally-averaged interlayer site of montmorillonite. For kaolinite, most of the adsorbed Cs is found on edge and basal sites. Measured sorption isotherms for Cd were found to be additive for simple mixed-oxide clays, suggesting a potential mechanism for predicting the sorptive capacities of other clays. A physicochemical rationale for component oxide acidity shifts in multi-oxide silicates remains an important obstacle. Quantification by AFM of edge and basal surface area for kaolinite showed that edges can comprise as much

as 50% of the total BET surface area. This is consistent with interpretation of H+/OH- adsorption occurring primarily at charged edge sites. By implication, the combined AFM and NMR results suggest that Cs and Cd adsorption also occur primarily at edge sites. These results provide a generalized and systematic method for evaluating adsorption of metals on soil minerals, and could lead to innovative techniques promoting the fixation of the metal (radionuclide) species.

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FIGURE TITLES

Fig. 1. Kaolinite clay ($Al_2Si_2O_5(OH)_4$: 1a) Schematic of kaolinite structure along (100); note that the SiO_4 tetrahedral layer is apically bonded to a gibbsite-like octahedral layer; the basal spacing is 7.1 ; 1b) AFM image of kaolinite crystallites with (001) orientation; note the well-developed hexagonal habit. Particle at upper right contains surface steps ~800 in diameter. Step thicknesses average 7-8 unit cells in height; average particle thickness excluding surface steps is 290 . Assuming particle is hexagonal in shape and that similar steps occur on both basal surface, the total edge surface area is 0.048 mm^2; total basal area is 0.325 mm^2.

Fig. 2. XPS and NMR spectra: 2a) High resolution XPS spectra for unsputtered kaolinite and montmorillonite after reaction with 0.1 N CsCl solution at 50C for 120 hours; the single peaks do not identify the presence of multiple surface or interlayer sorption sites; 2b) 133Cs MAS NMR spectrum for kaolinite after reaction with 0.1 N CsCl solution at 25C for 5 days; one broad and three minor peaks have been assigned to edge, surface and 'interlayer' Cs sorption sites.

Fig. 3. Adsorption isotherms for Cs and Cd onto kaolinite: 3a) Measured Cs and Cd adsorption isotherms as a function of pH; note the strong sorption of Cd at high pH relative to the weak binding of Cs; 3b) Adsorption isotherm of Cd (solid line) calculated from Cd binding constant of SiO_2 and Al_2O_3 is plotted against Cd adsorption measurements (filled circles) at 25C.

24-5

TRACER STUDIES OF RADIONUCLIDE MIGRATION IN FRACTURED ROCK AT A SCALE OF 1 METER

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ABSTRACT

The Large Block Tracer Test consists of a series of experiments to study the migration of radionuclides through fractures in rock at the scale of one meter. The separate effects being considered are sorption onto minerals within the rock matrix; diffusion of radionuclide species through the rock matrix, diffusion and hydrodynamic dispersion within the fracture; and the effect of heterogeneity in the fluid flow field (also known as macrodispersion or channeling). The rock fractures that are being used have natural fractures or artificial fractures with engineered heterogeneity. These tracer experiments will provide data with well-defined geometry and conditions for use in code validation. The experiments also provide an experimental framework to test inverse methods. Results are presented for a series of migration experiments using conservative tracers in artificial fractures with near parallel plane and near wedge-shaped fractures. The results are compared with those predicted with transport code TRACR3D (1). The fracture is treated as an equivalent porous medium with a "cubic law" permeability and a porosity that is proportional to the aperture. The results show good agreement, both between experimental results and those predicted by TRACR3D, but also between the distribution of the dye tracer in the fracture and the elution profiles. This suggests that the transport of a tracer through a fracture can be inferred from elution profiles.

INTRODUCTION

The modeling and prediction of radionuclide migration from an underground nuclear waste repository are a complex problem involving many physical and chemical processes. The coupling of these processes depend on the spatial distribution of hydrologic and chemical properties of rock. The exact details of these distributions will probably not be known. Approximations and some form of stochastic modeling will be used. The Large Block Tracer test allows the testing of sorption/retardation models in complex water flow systems. The experiments also provide an experimental framework to test inverse methods.

The primary purpose of Large Block Tracer Test is to study the effects of diffusion and adsorption on the migration of radionuclides through fractures in rock at the scale of one meter. The one meter scale is an intermediate between smaller laboratory scale experiments and the field scale. One meter scale fractures are experimentally more readily characterized because the fluid volumes are on the order of 100 milliliters rather than 100 microliters. The greater fracture volumes allows the fluid velocity to be controlled over a broader dynamic range with standard pumps. The first set of experiments (those described in this work) involves artificial fractures with simplified geometry. This was done to reduce the uncertainty in the fluid flow field (i.e., macrodispersion) so that the effects of diffusion and adsorption could be isolated.

Three different engineered fractures are being used to provide data for validation of transport models and to test inversion algorithms for inverse problems:

1. a fracture with a regular, rectangular cross section to study the effects of matrix diffusion and sorption without a need to consider channeling dispersion,
2. a fracture with grooves at 1 cm centers parallel to the flow along the fracture, to study asymptotic Fickian dispersion at low transport solution velocities, and
3. a wedge shaped fracture with the wedge dipping across the flow field. This fracture geometry has a covariance function for the permeability field that resembles the geostatistical model used by Gelhar (2) to describe stratified aquifers. The wedge shaped fracture approximates an aperture distribution where the correlation length is greater than or equal to the scale of the experiment. This situation has channeled flow and a large macrodispersivity. The results obtained with this type of fracture can therefore be used to model contaminant transport in natural fractures under field conditions.

These fracture geometries have been constructed from granite slabs and have begun to

provide experimental data. Tuff blocks have been excavated in the vicinity of Yucca Mountain and are being prepared for experimentation.
The first phase of experimentation, now completed, is intended to study only the fluid flow in the fracture. The fluid movement is studied visually using fluorescent dye. This phase tests our ability to construct idealized fracture geometries to an acceptable tolerance. Techniques were developed to determine deviations from design and compensate for the effects of those deviations.
EXPERIMENTAL
The Large Block Radionuclide Migration Facility (3) is equipped to handle blocks of rock with a maximum dimension of 1 m and is designed as an IAEA Type B laboratory. Migration experiments and post-experiment radiochemical analyses, including two-dimensional gamma scanning, alpha scanning, and autoradiography of the fracture surface can be performed in this facility.
Slabs of pink granite were obtained from the Cold Spring quarry on the south eastern flank of the Lac du Bonnet batholith near Lac du Bonnet, Manitoba, Canada. To develop the techniques required to establish uniform flow through a fracture with a regular cross section, artificial fractures were assembled from a 4 inch thick, 24 x 24 inch slab of polished pink granite and a 4 inch thick, 24 x 24 inch acrylic sheet, separated by 1/16 inch-thick strips of stainless steel. The surface of the granite slab was polished at the quarry with a mixture of iron oxides and spray painted white to increase the visibility of the colored dye in the fracture. Deviations from flatness were measured both for the acrylic sheet and granite slab. The deviations were determined using a certified, calibrated straight edge and feeler gauges.
To create uniform flow across the width of the fracture with the regular aperture, acrylic inlet and outlet plena were attached to two sides of the fracture; the other two sides were sealed with a silicone sealant (Canadian General Electric RTV-108). Porous polyethylene membranes with uniform pore size of 40 micrometers were inserted between the plena and the fracture to decrease the transmissivity between the inlet plenum and the fracture which could lead to an uneven introduction of the contaminant into the fracture. The outlet plenum was divided into eight equal sections to enable flow from each different area of the fracture to be collected into individual fractions. The parallel plane fracture was used to calibrate the flow through the fracture. Since the transmissivity through a fracture with apertures on the order of 1/16 inch is very high. slight differences in transmissivity in the fracture result in large changes in flow through the fracture. To eliminate this problem, the individual chambers of the outlet plenum were equipped with lengths of very fine bore tubing with an outside diameter of 1/16 inch. By trimming the ends of this tubing, uniform flow could be achieved across the fracture.
Distilled deionized water was used as the transport solution and was pumped through the fracture at a flow rate of 30 mL/hr, giving a residence time in the fracture with the rectangular cross section of approximately 20 hours. A 2 x 10-4 mole/L uranine concentration was used as the conservative dye. These migration experiments were performed over periods of 48 hours. Images of the dye in the fracture were obtained with a SONY Model AVC-P7 CCD video camera connected to a Data Translation Model DT 3851 frame grabber installed in a 80486 personal computer. The camera is triggered by an external trigger using a DAS-16 I/O board installed in a portable 80386 personal computer. The images are stored as .IMG files on the hard drive of the 80486 computer and enhanced with Data Translation Global Image software to produce .TIF files. Five milliliter samples of effluent from each of the eight outlets were taken by automatic fraction collectors.
RESULTS
The results described were obtained from the first set of a series of migration experiments in artificial fractures with a regular, well defined aperture width.
1. TRACR3D calculations
Forward calculations of the transport of both sorbing and nonsorbing tracers have been made using the code TRACR3D. These calculations were made to examine the effects of fracture geometry, adsorption, and sorption kinetics. The calculations have also been used to assist the design of the fracture. The porous material on the inlet and outlet of the fracture, used to make the flow field with the fracture less sensitive to small changes in pressure at the sampling locations, was included in these calculations. The code was used to predict the movement of the nonsorbing

uranine dye. The deviation from flatness was incorporated into the fracture permeability distribution.
The uranine profiles for the parallel plate fracture and for the wedge shaped fracture (shown in Fig. 1) were calculated. Sorption and matrix diffusion were set to zero because the granite face had been coated with epoxy paint The agreement between the calculatedprofiles and observed fronts is good. Only minor artifacts are evident near the boundaries of the fracture.
The wedge shaped fracture was modeled as an equivalent porous medium whose permeability is a function of the aperture (k = w2/12). The porosity of each node was proportional to the average fracture aperture (porosity = local width / maximum width). The permeability at each node is the average of the cubic law permeability for the fracture within the boundary of the node.
The target average flow velocity was 10-3 cm/s. The pressure gradient needed to provide this velocity is found by integrating the velocity across the breadth of the fracture of the fracture, i.e.,
Eq (1)
where u is the average velocity, Lx, is fracture width, Lz is the fracture length, wo is the maximum aperture, Dp is pressure difference across the fracture, and is the viscosity. From this relationship the pressure gradient needed is 0.04 millibars. This calculation shows why it is experimentally difficult to achieve a uniform flow field given small variations in permeability across the components of the inlet and outlet of these experiments.
The uranine concentrations in each of the eight outlet ports in the wedge shaped fracture were calculated using the same aperture distribution. These results are shown in Fig. 2. The earliest breakthrough is predicted in ports 3 and 4 in agreement with the concentration profiles.
Calculations have also been made for the grooved fracture for nonsorbing tracers, sorbing tracers, and high molecular weight nonsorbing tracers. The results of these calculations showed that matrix diffusion tended to reduce the effect of channeling on the tracer concentration in the fracture fluid. Sorption retarded the migration of tracer. The most significant effect of the grooved fracture was observed for the high molecular weight tracer. The effect of channeling appeared to be enhanced for high molecular weight (low diffusivity) tracers.
2. Uranine profiles
The shape of the front of the injected uranine indicated that the flow through the center of the fracture was slightly higher than at the edges. This is due to a slight concavity of the polished granite slab and the acrylic sheet. Since the linear flow velocity through the fracture varies with the square of the fracture aperture, a slight deviation from a uniform cross section is sufficient to create differences in the linear velocity through the fracture. In the case of the rectangular fracture, the aperture al the edges was 1/16 inch. Six hours after the start of the uranine injection, the leading edge of the dye had moved approximately 6 inches along the edges of the fracture, while the center of the front had moved 12 inches. This corresponds to a deviation of 1/38 inch from a perfectly flat surface. After 24 hours the dye injection was discontinued and the experiment continued with deionized water at the same flow rate. For the wedge-shaped experiment, the spacer was removed from one side of the block and the spacer on the other side was replaced by a 3/16 inch spacer. This kept the same average permeability for the fracture. The dye injection was again discontinued after 24 hours and the experiment continued with deionized water at the same flow rate. The resulting profiles are shown in Fig. 3 for 3, 6, 24, and 30 hours, respectively. The wedge shaped fracture geometry dominated the flow pattern but the effect of the cupped granite surface was still evident.
The effluent from each of the eight outlet plena was sampled using fraction collectors. The sample concentrations were analyzed using spectrophotometry. The concentrations in the eluted fractions are shown in Fig. 4. The effect of concavity of the fracture is again evident with the earliest breakthrough occurring in outlet ports 3 and 4. The breakthroughs were delayed (as expected) in the narrower part of the fracture. The dispersion (as evidenced by the lower slope of the leading edge) increased as the fracture became narrower. This is due to the greater deviation (across the portion of the fracture sampled) from the average flow in the portion of the fracture sampled.
Figure 2 Concentration in outlet plenum as a function of time and position across

fracture.
The results show good agreement, both between experimental results and those predicted by TRACR3D, but also between the distribution of the dye tracer in the fracture and the elution profiles. This suggests that the transport of a tracer through a fracture can be inferred from elution profiles.
An algorithm for inverting two dimensional integral equations has been developed. The method applied is that of Tikhonov (4) regularization. This code will be used to sharpen the images obtained by 2-D gamma scanning of radionuclide tracers retained on the surfaces of the fracture at the completion of the experiments. The code has been tested by reconstruction a regular image that had been distorted by a gaussian filter. The recovered image was sharpened considerably and was in good agreement with the initial image. This algorithm has also been incorporated into the TRACRI (5) code. TRACRI will be used to infer permeability fields from tracer experiments.
Transport of a nonsorbing tracer in the wedge shaped fracture has been calculated with TRACR3D and with one dimensional approximations. The stochastic model of Gelhar agrees very well with the TRACR3D calculation. The differences are small for nonsorbing tracers. Dispersivities inferred from migration experiments using nonsorbing tracers have been found to be too small to adequately predict the dispersion of sorbing tracers (6). Future experiments will be used to test this observation.

CONCLUSION

Construction of 1 meter scale artificial fractures from granite slabs has been completed. Calculations have been completed that predict the effects of matrix diffusion and sorption on the transport of radionuclides through the fractures. Experiments are starting to provide data for validation of the TRACR3D code.
The experiments have shown that:
1. It is experimentally difficult to engineer an ideal fracture geometry but deviations can be accounted for in the modeling.
2. Simple fractures, i.e., a wedge shaped fracture provides a realistic model of macrodispersion because the covariance function of this fracture is similar to some geostatistical models. This is validated by the excellent agreement between calculated and measured concentration profiles.
3. The fluid dynamics in these simple fracture geometries can be predicted with sufficient accuracy so that the effects of adsorption and diffusion can be studied unambiguously.
We are most interested in determining under what conditions the spatial distribution of fracture aperture and connectedness (i.e., the permeability) can be found from flux and transport measurements. This is an inverse problem and, in general, such problems cannot be solved uniquely to any arbitrary level of resolution. However, it is possible to resolve the unknown structure down to some spatial scale with an uncertainty in resolution inversely related to the scale. given sufficient observations of the system's response. In our present experimental design, with fluxes measured at eight outlet ports, the flow paths through the wedge fracture can only be coarsely estimated. This uniqueness difficulty can be alleviated by adding constraints. In this case, we have the concentration profiles of the uranine dye at each outlet port. This information can be used to reduce the magnitude of uncertainty in the fracture geometry. If we add one further constraint, that the geometry has a self-affine structure, for example, further improvement in resolution is possible. Through a sequence of simulations with our inversion model TRACRI, we show how addition of each of these data sets and constraints improves the resolution of the fracture geometry.
The results of these preliminary studies look promising for the development of a better understanding of the consequences of channeling dispersion to radionuclide migration through rock.

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IMPLEMENTATION OF BLT-EC: A COUPLED GEOCHEMICAL TRANSPORT CODE FOR ASSESSING THE RELEASE OF CONTAMINANTS FROM NEAR SURFACE DISPOSAL*

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ABSTRACT

In assessing the performance of low-level waste disposal sites it is often found that the release of many long-lived radionuclides is controlled by solubility and interactions between the surrounding media and contaminant in solution. BLT-EC is a computational tool that directly incorporates the effects of chemistry on the release and transport from LLW disposal facilities. The capability to calculate the changes in chemistry over time due to interactions with the engineered barrier, containers, and wastes makes BLT-EC an extremely powerful tool for performing detailed calculations within the disposal region. Results from these detailed calculations can be used to support selection of parameters (solubility limits and distribution coefficients) used in performance assessments. As an illustrative example, the BLT-EC code has been applied to analyze the chemical conditions in LANL Area G pit 37. Pit 37 wastes contain a heterogeneous mix of metals, cellulosics, concrete, soils and sludges. The initial geochemical assessment focuses on estimating the range of environmental conditions that can be expected in the pit. Emphasis is placed on estimating the pH and Eh changes due to corrosion of metal components and dissolution of concrete.

INTRODUCTION

Under most shallow land disposal conditions, the ground-water pathway is the most common and important pathway for the release of radionuclides to the accessible environment. For low-level waste (LLW) disposal, release may be controlled for periods of hundreds of years by the presence of engineered barriers (disposal unit, containers, and waste form). However, at longer times and away from the engineered barriers, control of the movement of contaminants is obtained through chemical processes (sorption, precipitation, solubility, etc.).

In assessing the performance of LLW waste disposal sites it is often found that the release of many long-lived radionuclides is controlled by solubility and interactions between the surrounding media and contaminants in solution. In general, the role of chemistry is treated in an ad hoc manner by modeling porous media/contaminant interactions using an empirical parameter known as the distribution coefficient. This coefficient is the equilibrium ratio of the amount sorbed on the porous media to that in solution. For many contaminants the value of the distribution coefficient can range by several orders of magnitude depending on the chemical conditions (pH, Eh, presence of competing ions, etc.) and the properties of the sorbing media. Similarly, solubility limits are chosen from literature values or, whenever possible, from site-specific information. Again, the value for the solubility limit of a contaminant depends strongly on the existing

chemical conditions.
The chemical conditions in a disposal unit are expected to be quite different than those in the surrounding undisturbed soil. In particular, corrosion of metallic materials and leaching of cement based waste forms and engineered barriers are likely to control the pH, Eh, and local chemistry. Since radionuclide sorption and solubility properties are strongly dependent on the chemical conditions, choosing appropriate values for these properties without knowledge of disposal unit chemistry may in some cases be highly problematic.
In an attempt to provide a tool that uses a more direct approach at incorporating the effects of chemistry on the release and transport from LLW disposal facilities Brookhaven National Laboratory, BNL, has developed a coupled geochemical transport code, BLT-EC (Breach, Leach, Transport - Equilibrium Chemistry) (1). During the past several years BNL has been developing a family of computer models that predict the release and ground-water transport of radionuclides from LLW disposal facilities. These models are capable of predicting waste container degradation, waste form leaching, and radionuclide migration with retardation and first-order decay in unsaturated or saturated porous media. BLT-EC improves upon these models by including chemical processes. These chemical processes, under the assumption of chemical equilibrium, are represented by a set of nonlinear algebraic (mass action and mass balance) equations that describe how the various chemical constituents are distributed among the solid, dissolved, adsorbed, and gas phases present in the pore space. Oxidation/reduction, precipitation/dissolution, complexation, and sorption reactions may be simulated.
BLT-EC combines segments of three existing computer codes, BLT (2), HYDROGEOCHEM (3), AND MINTEQA2 (4). The container and waste-form performance models (Breach and Leach) in BLT are used to estimate the time at which release begins and the rate of release of contaminants from the waste forms. The transport model in the BLT-EC computer code is based on a modified version the hydrological transport module contained in the finite-element code HYDROGEOCHEM (3). The geochemical model used to solve the chemical equilibrium equations in BLT-EC is a modified version of the geochemical computer code MINTEQA2 (4). The fact that this geochemical computer code has a large user community was an important factor in the selection of this code for BLT-EC. In addition, the code's extensive thermodynamic data base is continually being expanded for applications in radioactive waste management.
The capability to calculate the changes in chemistry over time due to interactions with the engineered barrier, containers, and wastes makes BLT-EC a powerful tool for performing detailed calculations within the disposal region. Results from these calculations can be used to support selection of parameters (solubility limits and distribution coefficients) used in performance assessments.
The ability of BLT-EC to simulate interactions between chemical processes and transport was demonstrated on an example field-scale problem (5). In this paper, an illustrative example of the use of BLT-EC in estimating the pH and Eh conditions due to corrosion and leaching of metal and cement based materials in an existing Los Alamos National Laboratory (LANL) LLW disposal facility will be provided. Emphasis will be placed on determining the range of values that can be expected and their potential impact on the mobility of uranium.
This paper is outlined as follows. We begin by providing a description of the LANL LLW disposal facility examined in this study. A geochemical conceptual model is then presented that outlines the important chemical and transport processes likely to occur in the facility. BLT-EC is subsequently used to predict spatial and temporal variations of pH and Eh in a particular LLW disposal pit. This information is then used to assess the potential importance of in situ chemical conditions and their impact on the mobility of important radionuclides. Ultimately this information will be used to help select appropriate radionuclide distribution coefficients and solubilities for the performance assessment of the LANL LLW disposal facility.
Facility Setting and Description
The disposal facility is located at LANL's Technical Area 54 (TA-54), area G on Mesita del Buey, a mesa of the Pajarito Plateau, approximately 4 km (2.5 mi) southeast of the town of Los Alamos. Mesita del Buey is a narrow, gently sloping mesa bounded by Canada del Buey to the northeast and Pajarito Canyon to the southwest. It is at an elevation of approximately 2030 m, 30 to 40 m (98 to 130 ft) above Canada del Buey and Pajarito Canyon, and 260 m (850 ft) above the main aquifer.

LANL began disposing of LLW at TA-54, Area G in 1957. Since then 35 pits (trenches) and nearly 200 shafts have been excavated into the native Bandelier tuff and filled. Pits are used to dispose of "routine" LLW, which generally consists of refuse from laboratory processes (e.g., paper, glassware, small benchtop equipment), facility upgrades (e.g., ductwork, conduit, large machine-shop equipment), and clean-ups (e.g. building an environmental media). Routine LLW is contaminated at such low radionuclide concentrations that it poses inconsequential radiological risk during disposal operations. Shafts, on the other hand, are used to dispose of "non-routine" waste that requires special handling, containment, or disposal due to either its regulatory status or radiological characteristics.

Pit 37 is the most recently filled disposal unit at TA-54, Area G. Disposal in this pit began in 1990 and ceased in 1994. During this time, enhanced waste characterization records were maintained a result of recently implemented waste generator and disposal facility requirements. As a result, Pit 37's inventory is considered the most complete of Area G's disposal units. The material inventory includes (in descending order of volume) metals, cellulosics, concrete, soil, and sludges. A graphical representation of material volumes is shown in Table I. These waste forms are generally disposed of in plywood boxes (metals), cardboard boxes (cellulosics), bulk (soil and concrete), and metal drums (sludge). Radionuclides of concern include americium (Am), carbon (C), Cesium (Cs), cobalt (Co), neptunium (Np), Strontium (Sr), plutonium (Pu), technetium (Tc), tritium (3H), and uranium (U). In Pit 37, most of the activity is associated with U-contaminated metal (8.16 Ci) and Pu-contaminated cellulosics (4.37 Ci).

The dimensions of pit 37 are approximately 18 meters deep by 25 meters wide by 223 meters long, providing over 43000 m3 of volume. Waste was emplaced in nine layers, each approximately 1.5 meters thick, and separated by at least .3 meters of compacted crushed Bandelier tuff derived from native soils. Crushed tuff is also placed between containers as backfill, then compacted in place to minimize void space and reduce the potential for subsidence due to container failure. The resulting waste-to-backfill ratio is approximately one-to-three.

GEOCHEMICAL PROCESSES

As ground water infiltrates through disposal pit 37, its chemistry will change due to mixing with pore water and reaction with various waste form, container, contaminant, and soil constituents. The reactions that take place will modify the existing chemical form of the contaminants in the disposal facility and therefore alter their leachability and mobility. This degree of alteration will largely depend on two factors; the chemical properties of the infiltrating water and the evolution of water chemistry as the infiltrating water passes through the disposal unit. Analytical results of ground water collected at LANL's Water Canyon Gallery are provided in Table II. This native ground water is likely to be representative of recharge (noncontaminated) within pit 37. Concentrations of Cl, NO3, and PO4 are less than 1 mg/kg suggesting that anthropogenic species are not present. The presence of dissolved carbonate species may be important since they can enhance the mobility of some contaminants. For example, uranium solubility tends to be enhanced by the presence of carbonate species and the subsequent formation of Uranyl carbonate complexes, particularly if the uranium is hexavalent. These complexes are often negatively charged which tends to enhance their mobility.

Metal corrosion and leaching of cement based materials can significantly modify the chemistry of infiltrating water by generating ferrous iron and calcium hydroxide. The degree of modification in pit 37 may be substantial considering the large volumes of metal and concrete disposed (recall Table I). Corrosion of steel materials provides an appreciable source of iron to the aqueous phase which helps buffer oxidizing conditions in the region around the containers. The hydrolysis of iron also tends to decrease the pH of leachate. A simplified description of these processes is as follows (6). Assuming that the contributed iron from corrosion is FE(II), its oxidation consumes dissolved oxygen and hydrogen ions according to the following reaction:

$$Fe^{2+} + 0.25\ O_2 + H^+ \rightarrow Fe^{3+} + 0.5\ H_2O$$

The aqueous concentration of FE(III) increases as oxidation continues until saturation with respect to FE(III) oxyhydroxide is reached and precipitation occurs according to the hydrolysis reaction:

$$Fe^{3+} + 3H_2O \rightarrow Fe(OH)_3 + 3H^+$$

The result of both reactions is the net production of two moles of H+ for each mole

of Fe(II) oxidized, which tends to increase the acidity of the pit leachate. In general, several other iron redox and hydrolysis reactions will also occur but the above two reactions convey the essential point. It should be noted that corrosion products (iron oxyhydroxides) may also provide a good substrate for sorption of radionuclides.

The impact of iron corrosion on pore water acidity can be counteracted by the leaching of calcium hydroxide (Ca(OH)2), which can comprise up to 25% of the volume of cement materials (7). The generation of Ca(OH)2 helps buffer the acidic conditions by providing a source of OH- ions.

Knowledge of pH conditions in pit 37 is important because of its effect on solubility, sorption, and speciation characteristics of radionuclide species. The concentration of hydroxides, for example, directly affects the solubility of metal hydroxide and radionuclide species; as pore water becomes more acidic the concentration of the hydroxide ion decreases, causing an increase in radionuclide concentration and solubility. Moreover, a soil's capacity to adsorb contaminants is frequently influenced by pH. For example, amorphous precipitates such as iron oxyhydroxides have a zero point of charge (ZPC) of approximately 8.5 (8). At pH levels below the ZPC, the surface charge is positive. Thus, anionic complexes of U, Am, Np, Pu, and Tc will have a tendency to adsorb to a lesser extent than cationic complexes of the same elements at pH values greater that the ZPC for specific adsorbents. Cationic species of the elements of concern adsorb onto smectite and ferric oxyhydroxide to a greater extent than anionic species at pH 7 and higher. Many radionuclides are also susceptible to changes in speciation because of changes in pH. For example, many radionuclides form carbonate precipitates at higher pH levels while they do not form carbonate compounds at lower pH levels.

Redox conditions are also important in pit 37 because many hazardous and radioactive chemicals, such as the actinides, have multiple valences and are subject to redox reactions which yield species that have sharply contrasting solubility and sorption characteristics. For example, anionic complexes of U, Am, Np, Pu, and Tc tend to adsorb to a lesser extent than cationic complexes of the same elements at pH values greater that the pH ZPC for specific adsorbents. Cationic species of the elements of concern adsorb onto smectite and ferric oxyhydroxide to a greater extent than anionic species at pH 7 and higher.

Based on the above discussion, it is evident that pH and Eh conditions will control the mobilities of several key radionuclides. Moreover, these conditions are strongly dependent on the balance between the impact of infiltrating water and its constituents, acidity generated by ferrous iron oxidation reaction, and the buffering effect of cementitious materials. These processes are examined in the next section using BLT-EC.

Application of BLT-EC

Each of the package and waste materials in Pit 37 have the potential to affect the chemical environment in the disposal unit to a degree that may impact leach and/or transport rates of important radionuclides; in particular:

- Oxidized metal will alter the reduction-oxidation potential and tend to lower pH conditions.
- Cement leaching will provide a buffering effect on pH conditions.

Although chemical reactions certainly occur in the Area G disposal units, it is difficult to know whether these reactions should be considered in source term modeling. The primary objective of this work is to gain a preliminary understanding of the significance of metal waste form corrosion and cement leaching on the chemical conditions in Area G disposal units.

The potential interactions between corrosion and cement leaching in Pit 37 are examined here by considering transport with reaction in a vertical, one-dimensional column representation of the pit. This approach is based on the following simplifying assumptions:

- The metal and cement based materials are homogeneously distributed throughout the pit.
- Water infiltration through the pit is unidirectional and downward at a constant rate.
- The release of iron (Fe(II)) and calcium hydroxide (Ca(OH)2) due to corrosion and cement leaching occur at constant rates.

As noted above, the problem domain is taken to be a one-dimensional column. The ground surface is located at 0.0 meters and the downward direction is positive with

the column reaching a depth of 25 meters, 7 meters beyond the bottom of the pit. The column has a porosity of 0.5, a bulk density of 1.2 g/cc, and a dispersivity of 0.5 meters. For each simulation, the column was partitioned into fifty finite elements of size 0.5 m x 0.5 m. Simulations were conducted for 150 years using a constant time step of 0.5 years. One iteration between transport and reaction calculations was allowed.

Eleven chemical components were considered; their initial pore water concentrations (mole/l) in the column were as follows: H+ = 8.65x10-4, Ca+2=1.73x10-4, F-1=3.15x10-6, CO3-2=7.31x10-4, Mg+2=1.15x10-4, Na+1=2.13x10-4, PO4-3=7.37x10-7, SO4-2=2.97x10-5, Zn+2=7.65x10-7, Fe+2=0.0, and Fe+3=1.0x10-8. This composition is similar to the native ground water given in Table II, but with some of the lesser important constituents neglected to reduce computational effort. The partial pressure of oxygen was held fixed at a constant 0.2 atmospheres during the course of the simulations. At the top of the column (ground surface), the composition of the incoming water was set equal to the initial pore water composition. The initial pH and Eh values under these conditions are 7.6 and 770 millivolts, respectively. Finally, concrete and metal materials are assumed to be emplaced between the depths of 2.5 meters and 18 meters.

In situ degradation of metals and concrete materials are complex processes and are not well understood. These processes are strongly dependent on chemical conditions such as pH, availability of oxygen, and sulfate concentrations in addition to the degree of system drainage (9). Typical corrosion rates for steel range from as high as 7 x 10-2 cm/yr for carbon steel (9) to as low as 3 x 10-8 cm/yr for 316 stainless steel (10). Concrete degradation rates can be as high as 1 millimeter/yr (11). For the base case simulations, we selected the steel corrosion rate as 3 x 10-8 cm/yr and the concrete degradation rate as 0.01 millimeter/yr. In order to convert these rates to source terms for Fe2+ and Ca(OH)2 having units of moles per unit volume per unit time it is necessary to estimate the surface area per unit pit volume for both the metal and concrete materials. Estimates were obtained by assuming that the ratio of material surface area to the cross-sectional area of the pit is equal to the ratio of material volume to the pit volume. Surface areas for the metal and concrete were estimated to be 92 m2 and 11 m2, respectively. In addition, we assumed that the concrete materials are comprised of 25% Ca(OH)2. Based on these assumptions, the Fe2+ and Ca(OH)2 source terms are QI = 1 x 10-7 moles-Fe2+/m3-yr and Qc = 2 x 10-6 moles-Ca(OH)2/m3-yr, respectively.

A total of four simulations were performed for this study. Two simulations were performed using the source term values noted above; two additional simulations were performed using increased source term values. The four simulations are described in Table III as follows.

Simulation results for these cases showing the distribution of pH and Eh at different times are presented in Figs. 1, 2, 3, 4. As shown, Figs. 1 through 3, the pH values throughout the column initially increase and then gradually decrease with time as Fe3+ hydrolysis reactions become increasingly significant. At the lowest source term rates, only minor variations occur in the Eh and pH. Figure 1 displays this for Case 1. Case 3 with the higher flow rate is similar and not shown here. The largest variations in pH occur in cases 2 and 4, Figs. 2 and 3, the cases with the largest source terms. The pH drops below 7.0 in these two cases, with the majority of the pit being slightly acidic at 150 years in the simulation with the lower flow rate, case 2. Comparisons between cases 2 and 4 also show that an increased infiltration velocity tends to enhance the reduction in pH.

Eh values are oxidizing and range between 700 and 820 millivolts, Fig. 4. The results also indicate a linear relationship between pH and Eh; note that the pH and Eh curves for each case are inverted images of each other, compare Figs. 3 and 4. For this reason, other Eh figures are not presented here. These results are not surprising. Although Eh values were computed directly by BLT-EC during the course of solving for chemical equilibrium, their relationship with pH in this problem is determined by the redox limit of aqueous systems and is given by (12):

$$Eh = 59.2(20.78 - pH + 0.25 \log PO2)$$

The simulation results show the potential importance of metallic corrosion and cement degradation processes on solution chemistry within pit 37. Although lower bounds were used for corrosion and degradation rates, these processes still had a significant effect on the evolution of pH and (Eh) in the pit. At both flow rates, the general trend was for the pH to increase to about 8.7 during the first fifty

years and then gradually decrease over the next 100 years to values near 7.0. At higher metallic corrosion rates the reduction in pH would likely be much more substantial and the pH may fall well below 7.

Role of pH on Uranium Mobility

As noted earlier, pH can have a significant effect on the solubility and sorption characteristics of contaminants. In oxidizing environments, sorption is generally a more important control on uranium mobility than precipitation of uranium minerals (13). Hsi and Langmuir measured adsorption of uranium in well characterized systems and found that at pH values above 6, dissolved uranium species are strongly adsorbed onto amorphous ferric oxyhydroxide. This sorption behavior may be significant in pit 37 because of the large quantities of amorphous ferric oxyhydroxide precipitates likely to be present due to corrosion.

Two MINTEQA2 calculations were performed to determine the speciation of uranium for pH values of 6 (Eh = 882 mV) and 8 (Eh = 769 mV). The composition of the water corresponded to the native ground water described in Table I. The initial concentration of uranium was 10-6 moles/liter of UO2 2+. At the lower pH value approximately 90% of the uranium exists as neutral uranyl dicarbonate (UO2CO3), whereas at the higher pH approximately 90% of the uranium is predicted to exist as UO2(CO3)3-4. The neutral species will have a lower affinity for sorption than the anionic species.

Summary and Conclusions

A preliminary analysis of the influence of metallic corrosion and cement degradation on pH and Eh has been performed for the conditions found in Los Alamos National Laboratory's disposal pit 37. Two water flow rates were used for the simulations: 0.02 m/yr and 0.002 m/yr. The analysis modeled the corrosion process through a constant source term of Fe2+ and cement degradation as a constant source of Ca(OH)2. A lower bound was determined for the release rates of these contaminants. Below these release rates, substantial changes to the ambient chemical conditions will not occur. These lower bounds correspond to corrosion rates on the low end of the range of measured values for stainless steels in soil systems. Therefore, it is likely that the actual corrosion rates will be higher and metallic corrosion will play a substantial role in determining solution chemistry within the pit.

Increasing the cement degradation rates and the metallic corrosion rates by one order of magnitude above the lower bounds resulted in changes in the pH by as much as one order of magnitude. At the lower flow rate, 0.002 m/yr, changes remained localized within the waste region of the pit. At the higher flow rate, 0.02 m/yr, the incoming rain water moved the contaminants out of the pit region in sufficient quantities to alter pH several meters beneath the pit. At both flow rates, the general trend was for the pH to increase to about 8.7 after 50 years due to the dissolution of Ca(OH)2. At later times, the pH decreased due to the corrosion of Fe and the interactions with the incoming rain water. The model predicts the pH will be near 7.0 after 150 years in the pit at the low flow rate. At the high flow rate, the minimum value of pH within the trench is 6.8.

The analysis is a first step in the evaluation of the evolution of chemical conditions in Pit 37. Several refinements to the model would be necessary to better define the chemical conditions in the pit. These refinements would include:

Better characterization of the metallic components to improve the estimates of available surface area and potential corrosion rates. A more sophisticated cement degradation model to account for release of calcium silicates.

Inclusion of the chemical properties of the Bandelier tuff used as backfill

Simulation of the solubilities and sorption properties of key radionuclides as a function of the pH and Eh.

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A NATURAL ANALOGUE FOR HIGH-LEVEL WASTE IN TUFF: CHEMICAL ANALYSIS AND MODELING OF THE VALLES SITE

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ABSTRACT

The contact between an obsidian flow and a steep-walled tuff canyon was examined as an analogue for a high-level waste repository. The analogue site is located in the Valles Caldera in New Mexico, where a massive obsidian flow filled a paleocanyon in the Battleship Rock Tuff. The obsidian flow provided a heat source, analogous to waste panels or an igneous intrusion in a repository, and caused evaporation and migration of water. The tuff and obsidian samples were analyzed for major and trace elements and mineralogy by INAA, XRF, X-ray diffraction, and scanning electron microscopy and electron microprobe. Samples were also analyzed for D/H and 39Ar/40Ar isotopic composition. Overall, the effects of the heating event seem to have been slight and limited to the tuff nearest the contact. There is some evidence of devitrification and migration of volatiles in the tuff within 10 m of the contact, but variations in major and trace element chemistry are small and difficult to distinguish from the natural (pre-heating) variability of the rocks.

INTRODUCTION

The U.S. Department of Energy has selected the Yucca Mountain Site (YMS), located in Nevada, as a potential repository for disposal of high-level commercial radioactive waste. Yucca Mountain is composed of rhyolite tuffs, in various states of welding and devitrification. To determine the suitability of YMS for safe storage, it is necessary to predict the chemical and physical response of the tuffs to local and site-wide heating for 10,000 years. For over a decade, scientists have used geochemical, thermal-hydrologic, and geomechanical codes to model the interaction of the tuffs, groundwater, and the radioactive waste. Most models focus on the degradation of the waste containers, dissolution of radionuclides in the waste, and the eventual diffusion, advection and sorption of the dissolved contaminants (1). However, the dissolution of the tuff minerals, and consequent reprecipitation in cooler regions, generation of halogen-rich gases, and polymorphic transitions must also be modeled, since these processes affect permeability and structural integrity of the host rocks.

Uncertainty in the models arises from the need to extrapolate from equilibrium conditions, or small-scale experiments performed on the bench top over days or months, to meter- and km-scale processes occurring in the repository over thousands of years. Radionuclide solubility and sorption calculations are based principally on equilibrium thermodynamics, and a limited number of laboratory tests to assess the rate of important reactions. The accuracy of the extrapolations can have a dramatic effect on the predicted safety of the repository. For example, some models predict that clinoptilolite, which strongly absorbs Cs and Sr, will form as the tuffs alter; other models predict the formation of less-sorptive mordenite, or even a non-sorptive feldspar-quartz assemblage from the same rocks (2,3).

One means to reduce this uncertainty is to study natural analogues -- large masses of geologic materials that were subject, over hundreds to millions of years, to processes analogous to those expected for the repository. The Valles Caldera, located in the Jemez Mountains of northern New Mexico (Fig. 1A), provides a good natural analogue for many of the processes expected at YMS. In the southwest corner of the caldera, the Banco Bonito obsidian (BBO) flow filled a steep-walled canyon cut in the Battleship Rock tuff (BRT). The obsidian, initially at temperatures in excess of 850C , heated the porous tuff in the canyon walls and vaporized much of the pore water. The goals of the Valles Natural Analogue Project were to: 1) search for evidence of chemical and mineralogic changes in the tuff -- specifically changes that occurred in response to the heating event; 2) provide a well-characterized example for testing chemical migration models; and 3) provide guidance for future analogue studies and code development.

Fig. 1. Study sites for the Valles Natural Analogue project. 1A: geologic map of Valles Caldera. 1B: vertical contact between pink-gray tuff spire (left) and blocky, blue-black obsidian (right). 1C: horizontal contact.

GEOLOGICAL BACKGROUND

The Valles Caldera formed following the eruption of the Otowi and Tshirege Members of the Bandelier Tuff, at 1.45 and 1.1 Ma (4). Subsequent to the last major eruption of the Bandelier Tuff, the caldera collapsed and a lake formed in the central depression. The Deer Canyon Rhyolite erupted in the central portion of the caldera, accompanied by landslides off the topographic rim, so that approximately 600 m of fill was deposited over the foundered cauldron floor. As lacustrine deposition waned, a resurgent dome (Redondo Peak) with 900 m of relief formed in the central portion of the caldera. From the beginning of resurgence to 540 ka, numerous rhyolite domes erupted along the ring fracture. The last volcanic activity in the region included eruption of the BRT. A deep, steep-walled canyon was eroded into the porous BRT, and the canyon was subsequently filled with 150 m of rhyolitic obsidian (the BBO flow). The ages of these units are uncertain because of ambiguities inherent in current dating methods (5); however, 40Ar/39Ar ages obtained for this study show both units are 400 to 250 ka old. After the BBO filled the ancestral canyon, the modern-day San Diego Canyon was eroded to the west, exposing much of the BRT-BBO contact. For the purposes of this study, the BRT is analogous to the tuffs at the YMS, and the BBO provides a heat source analogous to repository waste panels or a disruptive igneous intrusion. The contact region between the two units is the focus of our analyses.

SAMPLING AND METHODS

Initial samples were taken from VC-1, a vertical core hole drilled through a lobe of

the BBO. However, the VC-1 samples suffered several limitations. First of all, the core samples are small and potentially unrepresentative. More important, the BRT-BBO contact in the corehole is represented by a thick breccia zone, smearing and adding ambiguity to any observed chemical trends. Therefore, additional samples were obtained from the cliffs of the present-day San Diego Canyon, where the BRT-BBO contact is exposed in outcrop (Fig. 1B and 1C). These outcrops are near the very edge of the flow, where the flow stopped against the steep walls of the ancestral canyon; consequently the breccia zone is small to non-existent. Samples were taken from "horizontal" contacts (Fig. 1C), which provided long sampling paths, and vertical contacts (Fig. 1B) where there is little possibility of contamination by a soil layer.

Bulk chemical and mineral analyses were obtained by a variety of techniques, including: instrumental neutron activation analysis (INAA), for trace elements such as U, Th, Cs, Rb, Ba, Ta, Hf and rare earths; glass disc X-ray fluorescence (XRF) for major elements; pressed-pellet XRF for Cl and S; ion chromatography (IC) for leachable anions; fluoride ion-specific electrodes; and X-ray diffraction (XRD), electron microprobe (EMP), and scanning electron microscopy (SEM) to identify and analyze mineral phases and the abundance of glass. To test effects of heterogeneity, the tuffs were broken down into fine-grained matrix components and clasts (including xenoliths from older units), and heavy mineral separates. Differences between matrix and whole rock analyses proved to be small, and it was determined that the abundance of heavy minerals (including oxides and sphene, as well as micas, pyroxenes and amphiboles) was too small to account for significant variations in rare earth, F, and U or Th contents.

RESULTS

Figure 2 gives a sample of chemical variations observed at two outcrop sites as a function of distance from the BRT-BBO contact. Figures 2A and 2B show representative volatile contents at site 12, a vertical contact, and site 13, a "horizontal" contact (the latter has a horizontal exposure, but the BRT-BBO interface dips into the cliff face at 45). Note the large difference in the distance scale for the two sites. In all sites examined, F and Cl have an antipathetic behavior, with Cl depleted near the contact, and F enriched in the same region. Physical effects of the heating are more pronounced at site 12, where the tuff nearest the contact is deformed and obsidian clasts are oriented parallel to the contact; evidently, the contact tuffs were raised above the glass softening point (ca. 670C). At site 13, the tuffs nearest the contact have lower glass contents (are more devitrified), and are depleted in Cl and water; there is also a slight but statistically significant increase in dD near the contact. In contrast, non-volatile trace elements such as Ce, U and Ta (Fig. 2C and 2D) show little or no systematic variation. Cs may increase slightly toward the contact, but the increase corresponds to systematic changes in CaO, FeO and Na2O contents, and may reflect original variations in tuff composition. Variations of most other analytes were quite small; for example, bulk silica in the site 13 tuffs varied from 70% to 72%, with standard deviation less than 1%.

Fig. 2. Chemical variations at sites 12 and 13 as function of distance from BBO-BRT contact.

Mineralogical changes in the tuffs, as a consequence of heating, are slight. The voids of pumice shards (that comprise the tuff matrix) are lined with delicate, sub-mm lathes of albite and less abundant silica polymorphs; qualitatively, the amount of void-lining drops off with distance from the contact. Near the contacts, the matrix glass shards have partly devitrified to micro-porous, extremely fine-grained intergrowths of feldspar and silica. The Rb/Cs, a potentially sensitive indicator for clay/zeolite development, shows little variation (Fig. 2C). Neither XRD nor SEM analyses showed unambiguous development of zeolites or clays in the outcrop samples. Attempts to beneficiate fine-grained zeolites and clays, by standard settling techniques, yielded no clay or zeolite peaks in XRD spectra.

GAS TRANSPORT EXPERIMENTS

The chemical analyses above suggest we should seek mechanisms for transport and concentration of volatiles, particularly F and Cl. We developed a simple thermal model for the obsidian flow-tuff system, based on the Stefan solution (6,7), that accounted for evaporation, convection and radiation at the top of the obsidian flow, and boiling and conduction at the BRT-BBO contact. Because the tuff is very porous and has a comparatively low conductivity, temperatures well above the glass

softening point could be sustained at the contact for hundreds of years, consistent with our observations of plastic deformation in the vertical contacts. In addition, the boiling front may have been tens of m from the contact for up to a thousand years. Thus it is important to examine the behavior of the halides in steam, and determine if gaseous transport, alone or in concert with some other mechanism, could lead to the observed variations in F and Cl abundances (Fig. 2).

A simple flow-through system was constructed to measure release rates of F, Cl and metals from tuff as a function of temperature and time. Sample of crushed, washed tuff were packed between wads of silica wool in a 2.5 cm diameter, 50 cm long silica tube, and the tube was placed in a tube furnace. By means of a silica carburetor and low-volume peristaltic pump, steam was fed through the system, condensed, and collected for INAA and IC analysis (apart from Na, no trace elements were detected by INAA). Two experiments were performed; in the first, the temperature was ramped from 125 to 800C, and held at each increment for several hours; 1383 g of steam was passed through 33.2 g of tuff in 15 days. In the second, the sample was held at 400C for 54 hours; 172 g of steam was passed through 33.8 g of tuff.

Results of the two experiments are shown in Fig. 3A,B. Figure 3A gives the apparent gas/rock KD . While there is divergent behavior of Cl and F at T <600C, the apparent KD is 1 for both elements in this range, suggesting that steam-rock partitioning alone is a poor mechanism for causing the antipathetic behavior of Cl and F (Fig. 2A), and is probably inadequate to explain the F concentrations observed near the BRT-BBO contacts. However, the interaction can produce steam containing tens of ppm F, which may condense and interact with tuff elsewhere. Figure 2B, for the second experiment, shows the decrease in loss rate with time. The fitted curves assume a pure diffusion model for release of F and Cl from the glass shards, assuming the glass septa are 10 mm thick (consistent with SEM examination). The regression line for F yields an initial matrix F of 277 ppm, consistent with our chemical analyses, and a diffusion coefficient of 1.510-12 cm2/sec. The solid line fitted to the Cl data uses all points; the dashed line omits the first 3, which are potentially contaminated with the system blank. These two fits yield initial Cl of 2.7 and 4.2 ppm respectively, and diffusion coefficients of 6.310-12 and 1.610-12. The calculated initial Cl contents are clearly wrong, suggesting the control on Cl loss is not pure diffusion. However, the calculated 400C diffusion coefficients for both F and Cl are consistent with literature estimates (8,9).

Fig. 3. Results of flowing gas experiments.

CHEMICAL VARIATIONS INDUCED BY CAPILLARITY/EVAPORATION

The existence of a boiling front can create complex compositional variations in heated tuffs. Travis and Nuttall (10) calculated the distribution of SiO2 and Cl in fractured and porous tuff, as functions of time and distance from the waste canister heat source. A hot, dry zone forms immediately around the canister; the boiled water condenses several m away, yielding a concentric zone of saturation. Capillary action draws the water outward from the condensation zone; not only is the water drawn away from the heat source, out into the cooler tuff, but it is also drawn back towards the heat source. At a given time, whole rock SiO2 and Cl tend to peak on both sides of the condensation zone. On the hot side, evaporation tends to concentrate the solutions and cause SiO2 precipitation, whereas in the cool region, the inherent decrease in SiO2 solubility with decreasing temperature also causes a SiO2 buildup. These calculations suggest one should search for regions of silica and halide enrichment as proof of the capillarity/evaporation model, yet are inconsistent with the lack of significant trends in bulk silica content at the Valles analogue. However, Travis and Nutall did not consider the dissolution of other silicates, nor the possibility that the silicate deposition would be lost in the larger matrix variations.

A simple, heuristic model was developed to predict mineralogical and bulk F and SiO2 variations in the boiling region. From SEM examination of the relative volumes of voids, pumice solids, and the crystalline void fillings, we estimate an effective water/rock volume ratio ranging from 100/3.5 to 100/30 for 50% to 100% saturation. The walls of the voids and the fine-grained (0.3 mm width), high-surface-area crystallites lining the voids are assumed to be the most chemically active part of the system, and the denser clasts and glass shards (which show little sign of alteration) are assumed to be inert. We assume there is a relatively large, wet source region where water equilibrates with tuff at 100C, and this water is subsequently pulled into the boiling region by capillarity, where minerals

precipitate, existing minerals are altered by interaction with the solution, and the aqueous phase becomes enriched in components such as Na and F. The calculations were run with the react code, assuming atmospheric O2 and CO2 fugacities. In some runs precipitation of all silica polymorphs except chalcedony was suppressed to allow clinoptilolite to form, which has a dramatic effect on the sorption of Cs but very little effect on the major element chemistry of the bulk precipitates.

Figure 4A shows how the void mineralogy changes for a sample calculation, expressed as g of solids per kg of water. The relative amounts of major silicates change little, especially after about 30 pore volumes have evaporated. However, the F content of the precipitates increases steadily as fluorite precipitates. The predicted change in the bulk rock chemistry (including contributions form the inert matrix) is +0.1% for silica, and +867 ppm F after about 60 pore volumes are evaporated. The change in silica is completely insignificant compared to the 0.6% variation observed for pairs of adjacent samples taken far from the contact. However, the change in F content is several times the native F content of the tuffs, and is consistent with the F enrichment levels seen at site 12 (Fig. 2A).

Fig. 4. Mineral and chemical variations predicted for capillarity/evaporation model. Mineral abundances expressed as (g of precipitates)/(kg pore water).

Figure 4B illustrates that complex trace element variations can arise from this simple model. Translated into bulk rock concentrations, the precipitation of Na2U2O7 could elevate the bulk U content by 10 ppm in the intermediate stages of evaporation. However, as the system becomes more alkaline with increased evaporation, the U solids redissolve.

CONCLUSIONS

The BRT shows remarkably little alteration from the heating event. There is some evidence for devitrification, loss of volatiles, precipitation of albite-silica in voids, and perhaps redistribution of fluorine within 10 m of the contact. There has been little development of sorptive phases such as zeolites and clays, and there is no evidence for extensive silica remobilization in the bulk analyses. These results are consistent with a simple model for concentration of solutes and precipitation of minerals in the boiling zone.

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Session 25 -- Recycling and Reuse of Radioactive Materials and Facilities-I
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U.S. DEPARTMENT OF ENERGY'S ENVIRONMENTAL RESTORATION RECYCLING PROGRAM

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ABSTRACT

This paper outlines the U.S. Department of Energy's (DOE) proposed recycling program under the Environmental Restoration Program. The proposed "Recycle 2000" policy calls for recycling radioactively contaminated scrap metal generated by environmental restoration activities into containers for storing and disposing of wastes within the DOE complex. The paper describes the policy, DOE's approach to developing it, and the status. Supplemental information includes estimates of the supply of radioactively contaminated scrap metal and the need for waste containers, as well as brief descriptions of site-specific reuse and recycle projects.

INTRODUCTION

Millions of tons of potentially recoverable materials have accumulated over the years at U.S. Department of Energy (DOE) sites. The scrap pile continues growing as surplus DOE facilities are taken out of service through decommissioning. Surplus items in this pile include scrap metals, timber, concrete, and other structural materials that can either be reused in their present form or recycled into new products.

Some of these materials, principally high grade metals, are in demand in the marketplace. Beneficial reuse and recycle of materials that DOE no longer needs offer opportunities for responsible materials management and environmental stewardship.

REUSE AND RECYCLE PROJECTS AT DOE SITES

Many of DOE's environmental restoration site managers have recognized these opportunities. As shown in Table I (on the following page), which highlights a representative sample of site-specific reuse and recycle projects, several sites already are finding uses for materials the Department no longer wants or needs.

As the table demonstrates, these projects vary widely in purpose and scope. For example, the Savannah River site is recycling surplus heat exchangers into stainless steel containers for waste storage and disposal. The Hanford site is extracting rebar from dismantled buildings for resale and crushing concrete from these buildings for use in road construction on the reservation. Each of these sites has taken an approach to reuse or recycle suitable to its own needs. Now, DOE is looking at the prospects for reuse and recycle from a national perspective.

FOCUS OF THE NATIONAL RECYCLE PROGRAM

Many of the recoverable materials at environmental restoration sites possess significant market value, recycle potential, or both. This is particularly true for metals. Estimates of the quantity of scrap metals already available in the DOE complex vary, but the most conservative estimate puts the total on hand at approximately 150,000 tons (1). Other estimates of the current total quantity on hand range up to nearly 400,000 tons (2).

The inventory of scrap metals on hand will rise in the coming years as more decommissioning projects are started at the hundreds of facilities already identified as surplus. Estimates of scrap metals to be generated by decommissioning projects in the future are on the order of one million tons, in addition to what is now available.

THE "RECYCLE 2000" POLICY PROPOSAL

Given the quantity of metals already in the Departmental scrap pile and the expected volume to be generated by decommissioning in the future, DOE is considering setting a national policy for recycling these excess materials. This policy proposal is known as "Recycle 2000."

The Recycle 2000 policy proposal would set a goal that, by the year 2000, at least 50 percent of the waste storage or disposal containers used by DOE's Environmental Management Program would be fabricated from DOE-generated, radioactively contaminated scrap metals (RSM). The waste containers would be used only once and only within the DOE complex. If insufficient metal is available to meet the 50 percent goal, the policy would be to refrain from burying potentially recyclable steel and to use all available containers made from RSM. This proposal is

intentionally limited so as to be manageable and achievable by the year 2000.

APPROACH TO RECYCLE 2000 POLICY DEVELOPMENT

DOE decided to involve stakeholders in the formulation of the Recycle 2000 policy from the beginning, because broad-based stakeholder support will be critical to successful implementation of a recycle policy. The Recycle 2000 concept was raised first with a small group of representative stakeholders in July 1994. That group identified a broader group of stakeholders to involve in the policy development process, including local citizen groups; environmental action and public interest organizations; unions; industry; local, state, and tribal governments; federal regulatory agencies; DOE operations offices and sites, including DOE contractors; and DOE Headquarters.

DOE decided, with the encouragement of the small group of stakeholders, to involve the broader group of stakeholders in a workshop. The workshop offered an opportunity to discuss the Recycle 2000 proposal, to obtain stakeholder input on the idea, and to identify stakeholder issues and concerns.

RESULTS OF THE RECYCLE 2000 WORKSHOP

The Recycle 2000 workshop took place in early December 1994 and included representatives from 26 different organizations. The participants first had an opportunity to review and discuss relevant background information. This included research results on the supply of radioactively contaminated scrap metal and the demand for waste management containers, current disposal versus reuse/recycle practices, the scrap metal recycling industry, scrap metal reprocessing, the regulatory environmental for RSM recycling, and economic factors pertaining to RSM recycling.

Discussion of issues at the workshop centered on disposal versus recycle as RSM management options; on-site versus off-site processing and fabrication; implementation at the local versus regional versus national level and other implementation issues; waste commingling and secondary waste; RSM regulation; economics of recycle and disposal; and continued stakeholder participation in Recycle 2000 decisionmaking.

At the conclusion of the workshop, participants expressed their support for recycling RSM into waste containers, if the process is:

- Protective of public and worker health and safety
- Developed through an open, credible process
- Economic compared to other viable waste management options
- Employed for RSM unlikely to be released for unrestricted use
- Equitable for sites and states
- Scheduled according to a definite timetable
- Environmentally responsible, without compromising clean-up nor compounding existing problems
- Designed not to preclude further recycle.

Workshop participants and DOE alike agreed that more, and better, information is needed to ensure that recycling objectives and these conditions can be met.

RESOLUTION OF ISSUES FOR POLICY DEVELOPMENT

DOE has initiated a series of research and analytical efforts to address the issues discussed at the Recycle 2000 workshop and to complete the groundwork for a sound policy decision about whether or not to go forward with Recycle 2000. These efforts include an economic analysis of the Recycle 2000 proposal, analyses of technical issues associated with fabrication of RSM into other forms, the physical properties of the finished forms, and examinations of liability and regulatory constraints that may apply to recycle of RSM for DOE waste containers.

Economic Issues

The economic feasibility of the Recycle 2000 proposal depends on the answers to the following questions: What are the quantifiable costs of recycle? What are the quantifiable costs of disposal? What intangible or non-quantifiable factors need to be considered? How should these factors be weighted relative to each other and to the quantifiable costs?

To answer these questions, DOE is developing an economic model and examining available data for possible use in the model. This includes information on projected needs for waste containers, the current and future supply of scrap metals, and current and projected waste disposal costs.

DOE recently conducted a study to identify hazardous and low level radioactive waste containers used and purchased for the storage of solid, liquid, and sludge wastes

associated with operations and clean-up activities at 28 DOE sites. Table II presents the total volumetric equivalent of containers actually used each year from 1992 to 1994 and the total volumetric equivalent of containers projected for low level waste use annually during 1995 to 1997.

The economic analysis will need to link the projected demand for containers to the projected inventory of RSM to determine the prospects for achieving the goal of the proposed Recycle 2000 policy to use 50 percent or more of containers made from RSM. Several studies in recent years have examined the current scrap metal inventory and projected additions to it. Comparative analysis of these study results is complicated by differences in definitions and assumptions as well as incomplete or conflicting reports from several sites. Conservative estimates, however, put the future total for scrap metals to be decommissioned at more than one million tons (1).

Identifying the true costs of disposal also will be critical to this analysis. As reported by participants at the Recycle 2000 workshop, disposal fees at DOE facilities range from $7 to $10 per cubic foot at the Nevada Test Site to $70 per cubic foot at Hanford. Many of the workshop participants were skeptical that current disposal fees truly capture all costs associated with disposal, including future landfill management and monitoring. These costs will need to be built into the economic model for accurate comparison of relative costs for recycling and for disposal.

Technical Issues

Of particular concern are risks to workers from handling RSM, including the risks that may arise while it is processed into containers. DOE has taken a preliminary look at radionuclide partitioning during metal melt to determine contaminant fate. Recent literature on the partitioning of contaminants during melting shows the following:

- Elements remaining in the melt include Co-60, Mn-54, Ni-63, and Fe-55
- Elements that split between the melt and the slag include Ce-144
- Elements that oxidize and partition to the slag include Sr-90, U-238, U-235, and Pu-239
- Elements that vaporize and end up in the off-gas system include Cs-134, Cs-137, and Zn-65.

Once the RSM has been fabricated into waste storage or disposal containers, worker exposure risks may arise during transportation, waste loading, placement at disposal sites, or monitoring after land disposal. To address these and other potential risks, DOE plans a series of technical analyses to address the physical properties of finished containers, including grain boundaries and radiation embrittlement, and what happens between melting and final fabrication, including welding, machining, and sorting.

Liability and Regulatory Issues

Industry participants at the Recycle 2000 workshop indicated that, owing to concerns about liability, DOE would need to maintain ownership over the RSM fabricated into waste containers so that liability would remain with the Government. The Department has begun examining this issue to determine how liability concerns may affect development of the proposed policy.

At present, no regulatory standard exists for volumetrically contaminated RSM. Absent standards, DOE controls recycling and reuse on a case-by-case basis using the requirements contained in DOE Order 5400.5, "Radiation Protection of the Public and the Environment." These requirements include an appropriate evaluation under the National Environmental Policy Act (NEPA), a complete analysis of hazards and benefits of the proposed action (including addressing "as low as reasonably achievable", or ALARA, issues), and coordination of the release or reuse activity with the appropriate external regulator.

Both the U.S. Environmental Protection Agency (EPA) and the Nuclear Regulatory Commission (NRC) are contemplating regulations that address recycling radioactively contaminated materials. Both organizations are a year or more away from having draft standards available for public review. In NRC's case, the regulations, once promulgated, would apply only to NRC licensees. Restricted recycling within the DOE complex, as proposed for Recycle 2000, may not lie within EPA's purview. Regulatory parameters thus remain highly uncertain.

CONCLUSION

Stakeholders and DOE agree that recycling RSM into waste containers for use within

the DOE complex is preferable to disposal of these contaminated metals, subject to the conditions established by the stakeholders. DOE is proceeding with development of the proposed Recycle 2000 policy.
The Department expects to complete the analysis of economic, technical, and regulatory issues over the summer and to have a draft policy decision package ready for DOE management review by the end of September 1995. DOE will continue to involve stakeholders in the policy development process to ensure that their viewpoints, values, and concerns are addressed.
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METAL RECYCLING TECHNOLOGY AND RELATED ISSUES IN THE UNITED STATES AND UNITED KINGDOM
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ABSTRACT
Radioactively contaminated metallic materials comprise a large part of the potential waste products which result from nuclear facility repair, refurbishment, and decommissioning. United States Government (Departments of Energy and Defense) facilities, U.S. nuclear power plants, and other commercial nuclear fuel cycle facilities have large inventories of radioactive scrap metal which could be decontaminated and recycled into useful radioactive and non-radioactive products. Residual radioactivity and recycling criteria are needed to avoid the high cost of disposal and the waste of natural resources. The standards must be practical, measurable, and science-based. In the United Kingdom, the gaseous diffusion plant at Capenhurst has been decommissioned. A large fraction of the metallic scrap has been recycled into the metals market. Other structural materials have also been released as uncontaminated scrap. U.K. release criteria for residual radionuclide contamination have been applied to these operations. A variety of techniques were utilized to size reduce large components, to remove radioactivity, and to survey and release these materials. These methods and the application of release criteria has a direct relationship to methods which would be applicable in the U.S. and in other countries.
INTRODUCTION
This paper provides a current status of metal recycling technology and issues related to the release of recycled or decontaminated radioactive scrap metal in the US and UK. Radioactively contaminated metallic materials comprise a large part of the potential waste products which result from nuclear facility repair, refurbishment, and decommissioning. United States Government (Departments of Energy and Defense) facilities, US nuclear power plants, and other commercial nuclear fuel cycle facilities have large inventories of radioactive scrap metal which could be decontaminated and recycled into useful radioactive and non-radioactive products. Recycling this material into useful products is needed to avoid the high cost of disposal and the waste of natural resources and to reduce the impact to the environment and energy consumption for new material processing.
This paper describes the specific U.K. technology and experience in the decontamination, recycle, and release of scrap metal. It also describes the U.S. environment for metal recycle, including the volumes and levels of contamination, and the current and proposed release criteria. Comparisons are presented between the U.S. and U.K., both in technology and methodology for recycle and in regulatory criteria for residual radioactivity and material release. The paper then provides suggested approaches and criteria for U.S. recycling.
U.S. METALS PROBLEM
In the U.S. there is a large quantity of radioactive scrap metal (RSM) which could be recycled. The sources of this scrap metal are presented below along with approximate quantities of these metals.

Metal Sources

There are three main sources that represent significant quantities of metals suitable for recycle:

1. Nuclear Power Plants represent the largest current source of RSM. Nuclear power plants currently generate from regular plant overhauls radioactive scrap metal such as old heat exchangers, piping, pipe hangers, and spent fuel racks. Since this metal is not generally free-released directly by the plants to the open market, few choices exist:

a) Bury the metal at an approved low-level waste landfill, such as Barnwell, where the current cost is from $3.28/lb to $8.00/lb, depending on status as either a sited or unsited waste generator with the Southeast Compact.

b) Send the metals to a licensed radioactive scrap metal processor such as SEG or Quadrex (American Ecology).

c) Dispose directly after specific approval from the NRC (10 CFR 20.2002 -formerly 20.302). About 25 such approvals have been obtained.

The total volume of metal generated from the 109 operating nuclear power plants in 1993 has been estimated at 29 million pounds (1).

2. DOE Facilities. There currently exist stockpiles of metal at various DOE sites across the country. Significant quantities of additional metals will come from facilities as they are decommissioned and dismantled. The current estimate of the DOE facilities for all metal including that which is in stockpiles is 1.7 to 3.6 billion pounds! As facilities start to be decommissioned/dismantled and the metal is sent to be processed, the annual quantity should increase to 50-90 million pounds/year. DOE disposal costs vary from $5 to $10 per cubic foot at Nevada (not the life cycle cost) to about $50 per cubic foot per year at Idaho for low-level waste. The latter reflects the continual costs of maintenance of the disposal/storage facility.

3. Nuclear Plant Decommissioning. As power plants reach the end of their current 40 year operating licenses, they will be decommissioned. There are currently 109 nuclear power plants. The first plants to be decommissioned include Yankee Rowe, Trojan, and Shoreham. This market might be stretched if some of the plants successfully extend their NRC operating licenses beyond the 40 years of if the power plants get acceptance to put the majority of the facility into mothballs. It is currently estimated that 1.5 billion pounds of steel exist in the nuclear power plants that could be processed. The quantity of this material should start to be significant by 2003. It is estimated that the average annual flow of metal should be over 25 million pounds per year (1).

Timing

Sources of RSM are currently increasing in all three segments. The nuclear power plant RSM is expected to level off in five years and will start to decrease in the year 2003 as more plants are decommissioned, eliminating the generation of metal waste from operations. Depending on the decommissioning approach taken, plant decommissionings could be completed by 2042.

Metal from the DOE is a difficult source to predict because of the problem DOE has with free release and the fact that the Nevada Test Site is currently charging between $5 -$10/cu ft to dispose whereas the actual life cycle cost is more like $35/cu. ft. or more. Until the DOE decides on a workable free release policy (including liability issues for their management and operating contractors), the material for processing will be limited to acceptable internal end uses, such as containers, and to limited free-release from specific sites.

The third segment, RSM from the decommissioning of nuclear power plants, will be erratic in nature until about 2014. The flow of metal should continue until 2053 when the final plants will be decommissioned completely.

DOE Scrap Metal Volumes and Locations

The quantities of DOE scrap metal vary by type and location of these materials vary by site with large quantities at Oak Ridge (K-25), Paducah, Portsmouth, and Nevada (2). The specific quantities actually in inventory vary depending on which survey you believe. Total quantities in inventory are estimated from 400,000 tons to 1,800,000 tons (3).

Current U. S. Release Criteria

The Energy, Environment, and Resource Center at the University of Tennessee was directed by the DOE to research the establishment of effective environmental standards for RSM. In a draft report published in May 1993, they gave a summary of

the current status (4).
There currently are no nationally applicable standards that facilitate the treatment and recycle of RSM. Existing standards by the NRC are for surface contamination only. These are not adequate guidance in the treatment of RSM as there is currently no standard for release of volumetrically contaminated materials.
The following standards by various agencies exist:

The International Atomic Energy Agency (IAEA) - Basic Safety Standards (BSS) for specific isotopes of radioactivity follow ALARA principles. In addition, there is "The Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities" which provides safe levels for the recycle and re-use of steel, aluminum, and concrete. This set of principles includes volumetric and surface contamination levels, but is based on an individual dosage of one mrem per year, which is below what is considered practical and achievable on an economic basis.

The International Commission of Radiological Protection (ICRP) - original advocates of ALARA.

The Environmental Protection Agency (EPA) has developed standards for the safe levels of radioactivity, as they apply to drinking water, and air quality. There are also some recent guidelines pertaining to the treatment of soils and debris which might have a bearing on RSM.

The Nuclear Regulatory Commission (NRC) developed the Regulatory Guide 1.86 in 1974. This guide deals with surface contaminated materials that are nonactivated. It is this guideline that the nuclear waste processors have used in their licensed approval to free release metals from nuclear power plants. The values are in DPM/100cm2. The Energy Act of 1992 was to encourage the NRC to initiate the Enhanced Proposed Rule (EPR), an effort to establish cleanup standards for the decommissioning of nuclear power plants. This effort could result in changes or at least clarifications in how RSM is to be processed and released for unrestricted release. So far, this effort is only for decommissioning release standards for facilities.

DOE Order 5400.5 is, in part, based on NRC Regulatory Guide 1.86, and deals only with surface contamination. Unfortunately, a few years ago the director of DOE's Office of ERWM issued a memo stopping all release of DOE metals for recycle, which was done in response to a specific incident. Even though this specific problem has been corrected, some of the sites are hesitant to resort back to Order 5400.5. The only exception is Fernald, which has been sending steel to service vendors to be made into shield blocks and for decon and free release. Fernald plans to place a contract for the decontamination and release or fabrication into boxes of over 700 tons of mild steel from Building 7 (5).

U.K. EXPERIENCE IN METAL RECYCLE

Capenhurst Diffusion Plant Decommissioning

The Capenhurst Diffusion Plant was built in the early 1950's, at which time it was the largest industrial building in Europe under a single roof, measuring 1,000 yards in length and 160 yards in width. It was originally built to produce highly enriched uranium for military purposes, but this came to an end in the early 1960's when it was converted and extended for low enriched uranium production for civil use. After a further twenty years of life, the plant was shut down in 1982. By this time centrifuge enrichment plants were built and operating at Capenhurst and the diffusion plant was no longer economical. Since that time a program of decommissioning and dismantling has been in progress with an process plant material having been recycled or put in the appropriate form for disposal. Part of the building has already been demolished and a new centrifuge plant constructed on the same land. The remainder will remain in tact so long as there is economic use for it.
Post operational work to empty and cleanup the plant was carried out to leave as little residual contamination as possible by the use of a fluorinating agent to convert solid deposits to volatile fluorides which were pumped away. Further cleanup operations were carried out on the static plant to locate and deal with any significant buildup of solid or gaseous pockets of contamination remaining within the 4,800 process stages and 1,200 miles of interconnecting pipework.
A safe system of working was then established for the dismantling of the plant, which included survey measurements throughout the plant to ascertain what amounts of activity were left which would have to be dealt with at the decontamination and disposal stages. The initial phase of dismantling involved the cutout, removal, and

storage of large numbers of components, including compressors, coolers, valves, large diameter pipework, and the large process stage units.
Since 1984, up to 6,500 tons of major plant components have been successfully stored outdoors, including 850 tons of steel stage units (6). The 18,000 tons of structural steel have been dismantled and disposed of as clean scrap. Process pipework and the large process stage units have been cut up into lengths and sections suitable for the decontamination process. A variety of methods have been employed in cutting up the process plant. These include both hot and cold cutting, automated wherever feasible. Robotic-controlled plasma cutting has been employed to volume reduce a total of 5,300 tons of large aluminum stage units. Steel shells have been separated from a further 1,000 tons of stage units using remotely-controlled oxyacetylene methods. Both processes have been carried out in specially developed ventilated enclosures. Most of cold cutting was done by automated metal cutting techniques (including band sawing).

Decontamination Technology

Chemical treatment for the removal of uranium and uranium breakdown products is a well established process and significant quantities of aluminum and steel have been cleaned and recycled to the metals market. However, a large section of the plant had been exposed to reactor recycled feed material and contaminated with Np-237 and Tc-99, the latter being particularly difficult to remove. One disposal option was land burial at Drigg (near BNFL's reprocessing plant at Sellafield), the only currently available low-level waste site in the U.K. The large volumes of material involved (some tens of thousands of cubic meters), equivalent to several years of remaining Drigg lifetime, made this option unacceptable. Furthermore, costs would have been very high. It was decided to develop a procedure for effective decontamination to very low, deminimis levels at which it could be sold as uncontaminated scrap metal. An extensive laboratory and pilot plant investigation was successfully carried out and a full scale decontamination plant has been built and is in operation. Over 9,000 tons of aluminum and steel has been decontaminated to date and recycled to the scrap metal market. The flowsheet design for the plant was based on the plant discharges having a negligibly small impact on the environment and on the U.K. statutory regulations for recycling scrap metals to the open market. The process is one of successive stages of decontamination in process liquors which are subsequently treated by ion exchange methods. This results in the activity being transferred to relatively small and manageable volumes of solid and liquid residues and low activity solid and liquid wastes.

U. K. Scrap Metal Release Criteria

Criteria for free release of scrap metal was established for the U.K. based on the U.K. Radioactive Substances Act, 1960 and revised 1988 (7). Specifically, all radionuclides (total alpha, beta and gamma) must be less than 0.4 Bq/gm (0.011 nCi/gm). There is an exemption for naturally-occurring uranium (alpha) of less than 11.1 Bq/gm (0.3 nCi/gm).

Results from the Capenhurst Metal Recycle Program

In summary, the major achievements to date are as follows:

Over 99.5% of the decommissioning materials (160,000 tons) was recycled as clean material.

Disassembled a complete diffusion plant comprising 4,800 process stage units, 1,200 miles of pipework, and associated plant items, including 3,500 tons of electric motors.

Management and control of the outdoor storage of 6,500 tons of contaminated plant components for periods of time up to nine years.

Developed hands-on and remotely-controlled systems for volume reducing 8,000 tons aluminum, 23,000 tons steel, 70 tons copper, 300 tons aluminum bronze, 200 tons cupro-nickel, and 320 tons of nickel.

Since 1983, over 5,000 tons of aluminum (including 950 tons from the highly enriched section of the plant) have been prepared, decontaminated, and sold for recycle into the metals market. Some 23,000 tons of steel (including 1,100 tons from the highly enriched section) have been prepared and sold directly to the metals market as "clean" scrap, without the need for full decontamination.

The 3,500 tons of contaminated electric drive motors (including 800 tons from the highly enriched section of the plant) have been successfully treated and sold for recycle in the metals market.

Developed a process for separation of contaminated aluminum heat exchange sheaths

from cupro-nickel liner cooling tubes.

Ancillary buildings, including 11 large cooling towers (13,000 tons), pump houses, electrical substation, and part of the main diffusion plant building were completely raised to the ground producing 46,000 tons of concrete rubble for off-site disposal.

Five bays of the plant have been demolished to greenfield status. The recovered site (of approximately six acres) was made available for the construction of a new centrifuge enrichment building.

The 850 tons of low-level contaminated waste have been dispatched to the U.K. radwaste disposal site at Drigg, operated by BNFL.

Over 88,000 cubic yards of material was able to be disposed of by recycling or free release thus avoiding disposal as radioactive waste. At $2000 per cubic yard ($74 per cubic foot, much less than U.S. commercial rates), this resulted in a savings of over $180 million. Additional savings came from not having to use radioactive materials rules for much of the dismantling activity and not having to package and transport materials as radioactive. This savings is more than the cost of the actual decommissioning activity.

U.S. / U.K. REGULATORY COMPARISON

The rules and regulations under which the nuclear industry operates in the U.K. have both similarities and differences with the rules and regulations in the U.S. One area of significant difference is that in the U.K. the same agencies regulate both radioactive and hazardous wastes. This eliminates areas of conflict of jurisdiction such as occurs in the U.S. with the handling of mixed wastes. In the U.K. there is no classification such as mixed waste. Also in the U.K., the nuclear waste category called intermediate-level waste is between low-level waste and high-level waste. This intermediate-level waste is currently being cement grouted and stored for the eventual long-term burial in an underground repository.

There are also some differences in the definition of requirements and their application. In general, the U.S. requirements are prescriptive and extremely detailed, while in the U.K. the onus is placed on the owner of the waste to provide detailed protective criteria to the satisfaction of the regulator. In some cases, these risk-based criteria are more extensive than equivalent U.S. legal requirements.

Quantitative criteria for nuclear protection, upon which requirements in the two nations are based, are the same. They are the International Committee on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA). However, these organizations' guidance and rules are applied differently. In the U.S. isotopic discharges are regulated by concentration at the source, while a U.K. site is limited to an annual total discharge quantity. In the U.K. deminimis levels of radioactivity are defined which allow innovative clean-up and recycling of materials; while in the U.S., deminimis is effectively limited to that which can be shown to result in a dose to an individual of less than 0.02 mSv/hr. Reference 7 lists the discharge limits by radionuclide for gases and liquids in the U.K. and the U.S.

On the environmental side, the U.K. has certain environmental release regulations based on the use of surrounding arable lands for food production and water for drinking and fishing. Since the U.K. has a relatively small land mass, it is more densely populated and farmed than the U.S. This makes the environmental requirements more stringent, but reasonably based on real risk rather than concepts in which no risk is acceptable.

PROPOSED U.S. REGULATORY CHANGES AND COMMENTS/SUGGESTIONS

The EPA and the NRC are currently in the process of rulemaking actions to set radiological standards for decommissioning. The EPA is also performing risk and economic evaluations in preparation for developing standards for surface and volumetric contamination release standards for recycled materials. Currently, the EPA appears not to be considering free release for DOE scrap metal, but only restricted release (8). Standards for free release are necessary to fully implement a scrap metal recycle policy.

Practical standards are needed for recycling and decommissioning. To promote their use, these standards must be useable and understandable. A separate set of criteria which is for "restricted use" should be developed and allowed which will permit a decommissioning to restrict the use of the facility/site without having to maintain an active license for the site. For example, rules should allow the use of a former nuclear facility for an industrial plant, with certain obvious restrictions.

Material recycling and free release standards are needed for volumetric contamination in addition to surface contamination. Much of the DOE scrap metal has been volumetrically contaminated. These standards must also be measurable with current technology cost-effectively. Where possible, the regulatory agencies should establish release numbers which are directly measurable; e.g. nCi/gm or DPM/100cm2. This approach is preferable than requiring each radioactive material user to develop their own release numbers from site specific analyses for exposure to the public. The experience of others should be utilized in developing these limits, especially current scrap metal recycle facilities and decommissioned facilities. Finally, the standards must be based on a body of recognized and authoritative, scientific evidence and knowledge, such as that published by NCRP and ICRP, and not on an arbitrary assumption of what may be acceptable to some interest groups.

DERIVED BENEFITS

This paper has discussed the waste forms that are generated in the U.K., how they are handled and the specific handling of radioactive scrap metal at Capenhurst. Next, a discussion was presented on the regulatory requirements in the U.K. and how they are similar and different from the U.S.

Now that these subjects have been discussed, it is appropriate to tie them together in a discussion of benefits the U.S. may derive from the U.K.'s nuclear industry experience with scrap metal recycle and waste management.

First of all, the U.K. is actively collecting, treating, and at least properly storing for disposal all of its nuclear waste as it is being generated. In the U.S. this is not the case, especially in the DOE communities. For the U.S. to get to this practice they need to expedite the collective process of standardizing the discharge limits for all nuclear waste materials as their release is related to the public health and the protection of the environment. Risk-based analysis can result in risk-based criteria that can be restrictive relative to the personal exposure and not on the absolute values.

Secondly, the U.K. has been able to actively recycle significant quantities of metal back into society. The DOE is currently stalled with this effort because of their policies on release of DOE metals. Huge quantities of metals are not being processed and recycled. In the U.K. the Capenhurst facility has been able to generate jobs for the nuclear industry and reduce the cost of D&D by recycling while protecting the public and the environment.

Thirdly, in the U.K. land is at a premium and the country cannot afford to contaminate and not clean up for reuse the land and facilities that have ceased their intended operation. Capenhurst is a very small site and is surrounded by communities. Similarly, Sellafield is located on only 700 acres and also has close neighbors. By comparison, the Savannah River site is on 192,000 acres of land with its closest communities about 20 miles away. Therefore, in the U.K. at the Sellafield Complex, when a new facility is to built, it is designed with a specific life expectancy. It is also designed to be more easily decommissioned and dismantled at the end of its useful life. When a new facility is to be built, an old facility must be dismantled completely; including any soil and ground preparation. The cost to do the eventual D&D is factored into the facility as a life cycle cost and is accrued over the life of the facility so that when the facility ceases to operate, there are sufficient funds available to perform the proper D&D.

Lastly, in the U.K. there is an integrated effort to treat the public and the environment in a proper perspective. Rules and regulations are designed along the principles recommended by the ICRP. These basic principles are:

Any practice involving radiation exposure must be justified in relation to its benefits.

Any necessary exposures must be kept as low as reasonably achievable, economic and social factors being taken into account.

Radiation doses must not exceed recommended dose levels.

In addition to the U.K. experience there are benefits derived from U.S. recycling and decommissioning experience. The cost savings of having rules that can be complied with is true not only for the owner, but also for the public. We have numerous occurrences in the U.S. of companies which declared bankruptcy, forcing the public to pay for the cost of cleanup whether it was low-level radioactive, or high-level waste. Having reasonable rules will also avoid and/or reduce dismantling, packaging, transportation and land disposal; increasing the safety to the public and to the workers.

There are additional benefits to recycling which should be considered. Most of these benefits are true whether we are talking about radioactive scrap metal or domestic scrap, such as aluminum soft drink cans. These benefits include the overall economic benefit, reduced energy consumption, and minimizing the environmental impact of new mining and material processing.

CONCLUSIONS AND RECOMMENDATIONS

Recycling is an important part of the overall safe management of radioactive materials. Reasonable standards for material recycle and free release, including radiological standards for decommissioning, will assist in compliance and will be an overall benefit to the public, workers, and industry. It is also the best use of our natural resources. Specific volumetric and surface contamination standards for recycle and decommissioning should be established such that they are practical, definitive, measurable with current technologies cost-effectively, and are based on good science.

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25-3

THE STAINLESS STEEL BENEFICIAL REUSE INTEGRATED DEMONSTRATION

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ABSTRACT

In 1992 there was a desire to determine the ultimate disposition of 68 Process Water Heat Exchangers at the Savannah River Site (SRS). Each Heat Exchanger weighs about 100 tons. Since the Heat Exchangers are radioactively contaminated they could be classified as radioactive waste and disposed through shallow land burial on site. The cost for such a disposal would exceed $10 million. The Heat Exchanger material being over 95% 304 stainless steel would represent a commodity value of several million dollars on the commercial scrap market. Unfortunately, the metal is volumetrically contaminated, a situation for which there is no "de minimis free release" level, thereby preventing recycle of the metal into the commercial market place. However, the metal could be recycled back to the DOE in a "controlled release" manner. The radioactive scrap metal (RSM) could be reprocessed into new reusable products which are returned to the DOE for use within the DOE Complex. The new products would not be used within the public arena.
In 1994 the DOE Office of Technology Development initiated a demonstration to recycle contaminated stainless steel by melting 60 tons of RSM and refabricating it into containers (55 gallon drums and 100 cubic feet boxes) that could be used for long term, temporary storage. The Demonstration covers the entire recycle chain from the accumulation of the feed stock (through the disassembly of excess components) to

the receipt and utilization of the final products. The disposition of waste generated during the recycling process is also a part of the Demonstration. The actual melting of the radioactive scrap metal and the fabrication of the final products is accomplished through subcontracts with private industry. The Demonstration is a precursor to the establishment of regional stainless steel recycle facilities at or near major DOE sites. Radioactivity levels of the RSM to be shipped have been provided to the melt/fabrication subcontractors for their utilization in obtaining regulatory permit modifications.

Delivery of final products is schedule for 1995 and 1996. The activity level of the RSM to be supplied to industry has been determined to be less than one curie in total. The average specific activity level of the Cobalt-60 which will be volumetrically imbedded in the final products has been estimated to be of the order of 117 pico curies per gram (4.31 becquerels per gram). This level is expected to cause no difficulty in the utilization of the drums and boxes as containers of other waste forms.

THE INTEGRATED DEMONSTRATION

The Savannah River Site (SRS) has a large amount of radioactive scrap metal (RSM) which includes approximately 6800 tons contained in the form of excess heat exchangers. These heat exchangers have reached the end of their productive life and are normally declared Low Level Radioactive Waste. As such the heat exchangers would normally be disposed by burial at SRS at a cost of over $10 million. If however, the metal could be melted and refabricated into items such as waste containers, the material would be beneficially reused, eliminating the need to add an equivalent amount of clean metal to the contamination stream. In effect a potential liability would be changed into an asset. Figure 1 displays the concept.

To demonstrate the feasibility of this concept, the DOE (Office of Technology Development) is funding an Integrated Demonstration at SRS, in which a small amount of stainless steel RSM (60 tons) is to be melted, and refabricated into drums (55 gallons each) and boxes (100 cubic feet each). The melting and refabricating will be accomplished by private industry. The products will be sent to SRS for use. The secondary waste stream, consisting primarily of a small volume of clippings and possibly slag, will be returned to SRS for disposal as LLW. A total of approximately 485 drums and 85 boxes are expected to be produced. Delivery is scheduled for the 1995 - 1996 time frame.

The two private industrial companies participating in this Integrated Demonstration are:

1. Carolina Metals Inc. of Barnwell SC (a subsidiary of Nuclear Metals Inc.), and
2. Manufacturing Sciences Corp. of Oak Ridge TN.

The finished products to be produced by the private industrial participants are identified in Table I.

The numbers of boxes and drums to be fabricated are approximate since the actual values will depend on the efficiency of the (yet to be implemented) manufacturing processes.

EXPOSURE TO THE FINISHED PRODUCT

An important part of the Integrated Demonstration is to establish that the fabricated products will be sufficiently low in activity so as to allow worker proximity.

The total activity contained in the 60 tons of RSM to be sent to the melters is less than one curie. Most of the activity, being tritium, will off-gas. Of specific interest is the amount of cobalt-60 which will be volumetrically distributed in the remelted steel. This isotope will dominate the exposure from the fabricated boxes and drums. Samples taken from the 60 tons of RSM indicate the total cobalt-60 activity is of the order of 0.00636 Ci. If homogeneously distributed in the refabricated products, the concentration will be of the order of 117 pico curies per gram (4.33 Becquerels per gram).

A spectrum of RSM taken from different components will be used as feed metal. Table II groups the specific RSM selected and the cobalt-60 activity associated with each Group.

The melters, of course, will not homogeneously mix all of the recycled RSM to make each Heat at an activity of 117 pCi/g. Each Heat will be different. Figure 2 was generated assuming two ton Heats, starting with the highest activity feed RSM and proceeding to the lowest activity feed RSM. The 60 tons results in a total of 30 Heats.

It is important to know the exposure one might receive from the fabricated boxes and drums. One Heat of 2 tons will result in approximately four boxes or forty drums. Figure 3 depicts the geometry of the calculation of the maximum exposure dose rate to an individual at 100 centimeters from two boxes or eight drums fabricated out of the highest activity Heat (511 pCi/g).
Under this calculated maximum exposure a worker continuously standing in the vicinity of the products would receive a maximum of 0.39 mrem per hour (or 3420 mrem per year). A worker spending 40 hours a week for 50 weeks a year at the position depicted would receive a yearly dose of 780 mrem per year (which represents about 15% of the 5000 mrem per year dose set by the Nuclear Regulatory Commission for occupational workers). It is important to realize that these are calculated values which represent a conservative approximation of reality. The melt activity will most likely result in a mix of the various groupings in a manner which best suits the requirement to melt the actual geometry of the RSM at hand. This will cause the maximum exposure to be less than calculated. In reality it should be difficult to detect any radiation beyond background in the vicinity of the products. Table III identifies the calculated exposures at 100 centimeters for the various Heats shown in Fig. 2.

ESTABLISHMENT OF A VIABLE STAINLESS STEEL RSM INDUSTRY

To assure the commercialization of this Beneficial Reuse Industry will require a commitment to procure sufficient quantities of product over a multi-year period. The DOE has a need for large numbers of waste containers. A commitment by the DOE to buy product would allow industry to obtain financing for production equipment. Because it will also benefit the commercial nuclear industry, this activity is expected to survive on its own once it is up and running. Figure 4 depicts the steps required to establish a viable industry.
The stainless steel RSM recycle industrialization program (see Fig. 4) has essentially three phases: 1) The first phase is this Integrated Demonstration or Convergence Phase in which Government, industry, and other stakeholders converge on the best relationship for manufacturing products; 2) The second or Commitment Phase will require private industry to invest to produce products, most economically, for a government supported market. A partnership of government and industry during this Phase may be necessary; 3) The third or Completion Phase will be a self sustaining RSM recycle industry operating without the need for explicit government support. Regional RSM recycle centers on or close to a number of the DOE Sites may eventually exist.

PROGRAM BENEFITS

The recycle of any quantity of RSM eliminates the need to obtain an equivalent amount of new metal which would itself become contaminated. Also the cost of and need for burial space for the recycled RSM is avoided. If the program is successful, potential environmental liabilities can be turned into assets. Resources will be conserved and waste minimized.

25-4

APPLYING COMMERCIAL RECYCLING PRACTICES TO DEPARTMENT OF ENERGY SCRAP METALS

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ABSTRACT

This paper reviews the process by which commercial generators of radioactively contaminated scrap metal transfer such material to commercial recyclers and assesses whether the same process is applicable to radioactively contaminated scrap metal originating from DOE sites. Three issues that currently prevent DOE sites from using recycling as an alternative to long-term storage or disposal of surface contaminated metals are also reviewed. The issues are:

- Moratorium on transportation of radioactive materials off-site;
- Materials shipped off-site must have "no added radioactivity"; and
- DOE radionuclides cannot be sent to commercial sites for burial.*

There are significant quantities (thousands of tons) of scrap metal already accumulated at DOE sites, with larger quantities projected to be available for

recycling from decontamination and decommissioning surplus buildings from the nuclear weapons complex. While commercial utilities and other NRC-licensed facilities have been recycling scrap metal, including copper, admiralty brass, stainless and carbon steel, and aluminum, for as many as ten years, DOE has done little to reduce the burden of either storing these materials or disposing of them as radioactively contaminated wastes.

This paper reviews the process by which commercial companies utilize vendor services for the decontamination and recycling of contaminated scrap materials. The approach DOE would use for this same process, to recycle with subsequent free release or to accomplish waste minimization, is also examined.

COMMERCIAL RECYCLING PRACTICES

Presently there are two methods by which NRC licensees may have metals containing radioactive surface contamination decontaminated for recycling. They are:

Contracting with a firm that brings specialized equipment on-site and decontaminates the material to the free release standards published in NRC Regulatory Guide 1.86.

Packaging the material in accordance with DOT and NRC regulations, and shipping it to an NRC-licensed recycler.

This report focuses on the issues relating to transportation, shared liability and transfer of title, which do not need to be addressed for materials decontaminated on-site, but are important when transferring materials to an off-site recycling facility. Both decontaminated materials and the waste products generated by on-site decontamination techniques are handled by the site licensee in the same fashion as any other on-site generated materials.

In either case, if the state in which the work is being performed is an NRC agreement state, the state may have instituted release limits that are more stringent than those found in Regulatory Guide 1.86. Regardless of the release limits, if the material is not suitable for decontamination, or if the cost of decontamination is not economical, other options such as incineration, size reduction, or super-compaction are available to substantially reduce the cost of disposal.

Recycling companies require that the licensee provide a general description of the material and its history in addition to more specific information, such as physical dimensions, weight, volume, and materials of construction (with drawings if applicable). A radiological characterization is also needed, including an isotopic analysis with activation and fission products and transuranics in abundances down to 1% or less. For undetected radionuclides the lower limits of detection must be listed unless it is known they are not present, in which case they can be identified as "not present."

When a licensee decides to ship contaminated metals to a recycling center, they normally issue a request for proposal. The nature of the contract will be dictated by whether the licensee intends to make a one-time shipment or multiple shipments. For multiple shipments, where specifics of each shipment may not be known in advance, the proposal may contain a pricing sheet.

When the contract provides for multiple shipments, the licensee may ship materials to the recycling company any time sufficient material is accumulated. The licensee may utilize a trucking company with which they have a contract for transportation services or request the recycling company to transport the material if transportation is included as part of that contract. The shipping manifest, which lists the material, its composition, radionuclides and their concentrations, also acts as a basis for the billing. The transportation cost is based on shipping distance per shipment, not weight, making it desirable for the licensee to ship as much material per shipment as possible without exceeding normal shipping weights.

If the licensee arranges for transportation, shipments may be sent at the licensee's convenience. However, the licensee must notify the recycling company of the shipment's arrival, usually 24 to 48 hours in advance. If the recycling company handles the transportation arrangements, they will inform their staff of a shipment's expected arrival.

Commercial recycling currently is limited to metals. SEG will accept virtually any ferrous and non-ferrous metal in any physical configuration which meets their Waste Acceptance Criteria (WAC) for radiological (<1,000 mR/hr on contact) and hazardous characteristics (trace only, no constituents regulated under the Resource Conservation and Recovery Act [RCRA]); however, attempts are made to limit aluminum,

galvanized metals and lead content. Surface contaminated metals accepted by SEG for decontamination for free release include ferrous and stainless steel, lead, tin, copper, aluminum, brass, and bronze, as well as refractory metals - including zirconium, tungsten, tantalum, and molybdenum - which have melting points above 3,000F. The latter four metals are accepted for surface decontamination only because their melting points are higher than the maximum temperature attainable in the SEG melter. Quadrex will accept many of the same materials, but notes that they will not accept RCRA heavy metals, toxic metals, activated metals or pyrophoric metals for decontamination.

Under special arrangement with DOE, SEG produces shield blocks for resale back to DOE. Metals are utilized which cannot be decontaminated for free release; these include stainless steel, carbon steel, iron, nickel, chromium and ferrous alloys with melting points at or below 3,000F. Small quantities of copper, aluminum, brass, bronze and stellite are acceptable as long as they do not exceed 1% by weight of the total mass. A premium is charged for materials with surface contact dose rates above 200 mrem/hour because additional shielding is required. Materials that require transportation in shipping casks are not accepted.

A recycler may want to inspect a shipment before it is sent. For a new customer or one with a one-time only contract for unusual materials, the recycler would inspect the materials prior to shipment. If, however, the customer has a long-standing general services contract and has previously shipped materials to the recycling center without incident, pre-shipment inspections are not necessary. In either case, only materials that comply with the recycling center's WAC may be shipped.

Recycling companies do not maintain detailed records on the physical form of recycled materials, but estimates of quantities are available. SEG estimates that, over the past ten years, they have decontaminated approximately 7,200 ft3 of lead weighing over 487,000 pounds, more than 75,000 ft3 of brass condenser tubing weighing more than 3 million pounds, and approximately 50 million pounds of other metals. Of this, 16 million pounds were fabricated into shield blocks for DOE, with 90% of the remaining 34 million pounds sold to the commercial scrap metals market. In the past ten years, Quadrex has accepted roughly 95 million pounds of metal, of which they have recycled (free released) about 81 million pounds or about 85% of the materials accepted. For both SEG and Quadrex, materials that could not be decontaminated effectively (or that ended up as slag in the SEG melters) were disposed of as low-level radioactive waste with ownership still vested in the generator.

Selection of a decontamination method by the recycler depends upon the composition and physical shape of the material or item to be cleaned and on the nature of the contamination. Surface contamination is considered either smearable or fixed. Smearable contamination can be removed by smearing, or wiping, the surface with a cloth. Removal can be enhanced by using several chemicals ranging from commercially available bathroom cleaners to proprietary chemicals.

Fixed contamination can be removed by either chemical or physical means. Chemical methods include acid etching, electropolishing, and complex, multi-reagent processes, some of which contain chelants. The use of multi-reagent processes containing chelants should be minimized; wastes containing these chemicals require special treatment to meet disposal criteria. Quadrex limits the use of electropolishing because of the large volume of contaminated phosphoric acid generated as a secondary waste; this material also requires special treatment.

Physical techniques usually involve impingement of the contaminated surface with particles, commonly referred to as blasting. Steel shot, solid carbon dioxide pellets, abrasive grit, ice crystals, sponge balls and high-pressure water have all been used successfully for various applications. Because of the potential for creating airborne contamination during blasting, these operations are restricted to being performed inside containment booths or glove boxes. Also, because of the need to minimize secondary waste volumes, materials like sand are rarely used. Tumbling, in large tumblers similar to those used for polishing gem stones, is used to clean scaffolding clamps.

The success of decontamination methods is highly dependent on geometric and surface features. In addition to porous materials, which are difficult to decontaminate because of the migration of contamination into the surface, materials with complex features, crevices and other inaccessible surfaces cannot be decontaminated by mechanical cleaning techniques. If such areas can be cleaned but not surveyed, the

material cannot be released for recycling.
Companies that provide on-site decontamination services have developed special equipment that vacuums up grit and dust, eliminating most, if not all, airborne contamination.
Standards used to determine when materials have been adequately decontaminated for free release are incorporated into a facility's license conditions. Facilities licensed directly by the NRC must satisfy NRC Regulatory Guide 1.86. If a facility is licensed by an "agreement state," the agreement state has the right to establish standards more restrictive than those established by the NRC. In specific cases, higher limits than specified in Regulatory Guide 1.86 can be approved in response to a petition for a license amendment incorporating appropriate risk analysis and exposure pathway modeling. In such cases, the agreement state may permit unrestricted release or restrict the use of the material to be recycled. Such restricted use normally is specified in the application.
Recycling companies use state-of-the-art radiation detection equipment, including large area-flow proportional detectors, GM detectors, plastic scintillation detectors, or intrinsic germanium detectors to adequately verify decontamination. In addition to their fixed contamination monitoring program, the SEG Unconditional Release Program includes requirements for taking smears of items to be released to determine the levels of removable contamination.
Both the SEG and Quadrex facilities are licensed by the State of Tennessee, which is an NRC agreement state. License reporting requirements under Title 10 CFR Part 20 (10 CFR Part 20) only apply to the secondary wastes generated by these facilities. These facilities are not required to report to the NRC on the quantities of materials decontaminated. Materials decontaminated at an originator's facility by an on-site services contractor are not reported under 10 CFR Part 20 since they are not part of a reportable waste stream; only secondary waste shipped off-site would be reported.
Facility inspections are performed by the State of Tennessee. Both routine and non-routine inspections may be performed from once a year to four times a year to confirm adherence to the facility license requirements. The principal concern is the nature of the activities conducted rather than the quantity of materials being handled. The NRC may participate as an observer on some inspections.
Title to recyclable materials resides with the licensee having most recent control over the material. Thus, for material shipped to a recycler by a carrier not owned by the recycler, the originator of the material holds title while it is in transit. The recycler takes title when it is determined that the material meets the recycler's WAC. This can occur at the originator's site if the carrier is owned by the recycler or at the recycler's site when the material is accepted. Material not accepted by the recycler, because it fails to meet applicable WAC or due to a Department of Transportation (DOT) violation, is returned to the originating facility at the originator's expense.
The recycler holds title of material decontaminated to release levels specified in the recycler's permit until the material is purchased and removed by a scrap dealer or other recipient.
The licensee with title to the material is responsible for compliance with applicable NRC and DOT regulations regarding storing, manifesting, labeling, packaging, and transporting the material.
Title to radionuclides removed from the material in the decontamination process is dependent upon the traceability of the radionuclides in the waste streams. Waste traceable to a specific originator after the decontamination process is considered distinguishable. Title to such waste remains with the originator until it is accepted for disposal at a licensed low-level waste burial site. Wastes generated during the decontamination process which cannot be linked to specific originators are deemed indistinguishable. Title to indistinguishable waste lies with the recycler until it is accepted by a licensed low-level waste disposal site.
An example of distinguishable waste is incinerator hearth ash. Material being fed into the incinerator can be temporarily stopped without an incinerator shutdown. The corresponding interruptions in the discharge of hearth ash can be used to determine the source of incineration ash.
Indistinguishable waste is best exemplified by bag house ash. It is not feasible to clean or remove the ash from off-gas system filters after materials from a particular originator are incinerated. Ash in these filters will generally contain

radioactivity from incineration of material from several licensees. Thus, radionuclides cannot be associated with specific originators.
Liability for problems that may occur during the transport of contaminated material to a recycler's facility is shared by those entities involved in the transportation process. Who arranges for and who provides the transportation will determine with whom liability is shared, be it the originator, the shipper or the recycler. Each of these entities is covered by insurance under the Price Anderson Act through American Nuclear Insurers.
Liability for alleged harm from the recycling of contaminated material can lie with every entity associated with moving the materials to the consumer. The originator of the material, the recycler, the transporter, the scrap dealer, and the eventual fabricator can be named as possible defendants in a law suit raised by the consumer. Each of these entities can also sue any predecessor(s) associated with the material. The basis of these legal claims would be that harm was done to one entity through the negligent activity of another. The alleged harm could be personal injury or equipment damage. Two things must be proven for the claim to succeed: that harm was done, and the material which was recycled was the cause of the injury.
Under the present standards associated with the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), originators may also be liable for their distinguishable and indistinguishable wastes after disposal at a licensed burial site. Originator liability for damages caused by the site and costs of remediating it are covered by the site's perpetual care fund. However, if site remediation costs and damages exceed the fund, it is feasible, under CERCLA, that all generators of wastes found at the site could be liable for their distinguishable wastes. Moreover, under CERCLA's present joint and several liability structure, the portion of remediation costs attributable to indistinguishable wastes would be shared among identified generators.
The terms contained in a recycler's permits can have a regulatory impact upon originators by prescribing the flow of materials from originators to recyclers. The primary terms of a recycler's permit are the standards identified for the free release of materials. Until NRC free release standards are established, permits adopt the guidelines of NRC Regulatory Guide 1.86 plus ALARA (As Low As Reasonably Achievable) as the release standard. The guidelines in Regulatory Guide 1.86 plus ALARA take on the force of law by incorporation into a recycler's operating permits.

Release limits, established in a recycler's permit, are important in the recycling process because they will likely determine the cost effectiveness of recycling materials. Materials will not be recycled if the cost of decontaminating them to the established release limits is greater than the cost of disposal.
Recyclers' permits may also contain conditions which specify their amount of storage space for contaminated materials, the radionuclide inventory they may carry, and the amount of time they can hold materials before processing. These terms will have implications for the volumes and types of materials originators may wish to recycle and the recycling schedule for the materials.
APPLYING COMMERCIAL PRACTICES TO DOE SCRAP METAL
No restrictions exist to prevent DOE from recycling surface contaminated materials. The same companies involved in decontaminating radioactively contaminated materials from commercial companies are more than willing to perform similar service for government facilities.
To a limited extent, recycling has been performed at the Fernald Environmental Management Project (FEMP) by the Fernald Environmental Restoration Management Corporation (FERMCO). This initial recycling project (1) was a competitive procurement won by SEG with Quadrex as a subcontractor. Under this procurement, SEG segregated the scrap metal piles at FEMP into carbon steel and non-carbon steel piles. The carbon steel materials were size-reduced (i.e., cut up into small enough pieces to fit into sea-land containers) on-site and packaged for shipment to SEG's facility in Oak Ridge, Tennessee. The non-carbon steel metals were also size-reduced and packaged for transportation, but shipped to the Quadrex facility, also located in Oak Ridge.
The carbon steel taken to SEG was melted in a furnace at 2,800F. The slag produced during melting contained approximately 90 percent of the radionuclide contamination, with the remaining 10 percent evenly distributed throughout the metal. While melting results in a significant decontamination of the metal, it cannot be released for

commercial use because there is no Federal regulatory standard which addresses volumetric contamination. SEG cast the steel into shield blocks for use by DOE under controlled conditions. If clean, virgin metal were used as shield blocks it would eventually become radioactive by activation. Therefore, recycling contaminated material eliminates the contamination of new, uncontaminated material.
The non-carbon steel, i.e., stainless steel and aluminum, was shipped to Quadrex, where it was decontaminated to the free release standards contained in their license. The method used was a series of proprietary chemical baths, after which the materials were surveyed and released to a local scrap metal dealer.
Both SEG and Quadrex created secondary wastes that required disposition. SEG generated both distinguishable and indistinguishable waste. The distinguishable waste was in the form of slag from the melter. As a distinguishable waste attributed to the Fernald scrap metal, ownership, or title, was still vested in DOE. This material was packaged and shipped by SEG to the Nevada Test Site as DOE waste. Other waste generated at SEG included dust collector wastes, dry active wastes (DAW), trash, wipes, rags, and booties; contamination of these materials could not be specifically attributed to the Fernald scrap metal. Being indistinguishable in their origin, these wastes were considered to be SEG wastes under SEG's Tennessee radioactive materials license. These wastes were solidified and buried at the commercial low-level waste disposal facility in Barnwell, South Carolina.
The secondary waste generated by Quadrex consisted of sludge resulting from the treatment of the decontamination solutions. As with DAW, contamination in the sludge could not be specifically attributed to the Fernald scrap metal processing and was therefore considered to be Quadrex-generated waste. This waste was disposed at the Barnwell facility after solidification with cement in 55-gallon drums.
With the success of the Fernald recycling program, an obvious question emerges: "Why aren't we doing more of this?" Several reasons have been identified, including the following:
People do not know where to start, what information is needed, or whom to contact.
DOE has a ban on the off-site shipment of radioactive materials.
DOE has a ban on the transfer of materials with "added radioactivity" to non-DOE facilities.
DOE cannot send radioactive materials to a commercial recycling facility because waste would be disposed of at a commercial burial site.
As shown by the Fernald experience, these issues do not prevent DOE from utilizing private contractors for decontaminating radioactively contaminated scrap metal for free release into the private metals market. The following discussion focuses on apparent misconceptions surrounding various DOE Orders and directives so that their true value can be understood, that is, laying a solid technical foundation by which these materials can be reused in general commerce without any hazard or adverse impact on the public.
1. Determine if the material being considered for recycling is appropriate. The flow chart in Fig. 1 is intended to assist in this decision process.
2. Assemble needed information on physical, chemical and radioactive properties of the scrap material. In addition to specific parameters previously discussed, further requirements can be identified by contacting recycling companies, including the following.
Quadrex, Oak Ridge, TN.
Scientific Ecology Group, Oak Ridge, TN.
Alaron Corporation, Beaver Falls, PA.
Bartlett Services, Inc., Plymouth, MA.
The assumption that off-site shipment of radioactive materials is prohibited is incorrect. The Environmental Management (EM) program has implemented a ban, or more precisely a moratorium, on the transportation of certain materials to facilities not licensed to handle radioactive materials. Those materials include the following:
radioactively contaminated materials which can only be shipped to a licensed facility under any circumstances, and,
material which is not known to possess any radioactive contamination but originates in areas where radioactive contamination is known to exist, such as a Radioactive Materials Management Area specified by DOE, is commonly referred to as "suspect material."
This transportation moratorium does not affect shipments of contaminated or suspect materials to a licensed recycler if the material being shipped complies with a

facility's WAC. In some cases, materials may not be contaminated above trace quantities.
The "no added radioactivity" statement is not contained in any DOE regulation or DOE Order. The term comes from the April 1993 draft of the Performance Objective for Certification of Non-Radioactive Hazardous Waste, which states that the performance objective is:
"To assure that hazardous/toxic wastes shipped from DOE facilities to commercial treatment, storage or disposal facilities have no bulk or volume radioactive contamination added as a result of DOE operations and are in compliance with DOE Order 5400.5 criteria for surface contamination unless the receiving facility is specifically licensed to manage radioactive waste."
This statement confirms the fact that current DOE regulations and Orders do permit the transportation of surface contaminated materials to licensed recycling companies for decontamination and subsequent free release. Similarly, DOE may release the material under DOE Order 5400.5 after verifying that the surface contamination limits have been met. This applies whether decontamination is performed on-site by a DOE contractor or a sub-contractor.
Any waste material with radionuclides distinguishable as originating with DOE would continue to be DOE's responsibility and either returned to DOE for disposal or sent directly to a DOE disposal facility as DOE waste. Secondary wastes that are indistinguishable as to their origin are considered to originate at the recycling facility and would be disposed of at a commercial facility.
As previously discussed, title to the contaminated material is vested in the organization under whose license the material is being controlled at any given time during the decontamination process. Radionuclides, however, if distinguishable, remain the responsibility of DOE as originator.
To avoid co-mingling of indistinguishable waste in the form of incinerator ash from the bag house filters, one commercial recycling company is constructing a low-level waste incinerator that will burn only DOE waste. Dedicating facilities to decontaminate government material may be a useful approach at several DOE sites.
DOE requires that the shipper and the recycling facility assume all liability for problems that may arise during transport to the recycling facility. Shippers and recycling companies do not like this arrangement; still, it has become accepted. Should a problem arise during transport, such as an accident resulting in contamination, DOE would respond as required by internal regulations.
"Suspect material" or "suspect contamination" is material which is suspected of being contaminated but has not been verified by any analytical method as being either contaminated or uncontaminated. At a 1992 meeting on the Waste Shipment Moratorium, a DOE representative noted "that a previous review of compliance with 5400.5 guidelines revealed that very few sites were capable of demonstrating compliance." (2) While DOE sites have the necessary equipment to characterize materials for compliance with DOT transportation regulations, it appears that some sites may lack equipment with the added sensitivity to demonstrate compliance with DOE Order 5400.5. Under these circumstances, materials could be shipped to a recycler for examination. Materials that are above the limits of either DOE Order 5400.5 or NRC Regulatory Guide 1.86 could be shipped back to DOE for disposal; materials with contamination levels below these limits could be recycled without further decontamination.
At Fernald, the cost associated with the recycling of this material - $4,755,920 - was 18 percent higher than the estimated costs of packaging, transporting and burying the same material at Nevada Test Site (NTS) - $4,021,440, based on a disposal cost of $10 per ft3 (1). However, taking into account the value of the recycled metal (an average of $800 per ton), the cost of the recycling process is reduced to $3,030,430, roughly 25 percent less than disposal. Even without the credit for the value of the recycled metal, recycling becomes cost effective when disposal costs are based on full life cycle costs, which is not the case at NTS. EPA (3) has estimated that a burial cost of $15 per ft3 at NTS is the breakeven cost for recycling. In addition to reduced cost, there are other benefits to be gained from recycling these types of materials. Issues that commonly affect waste disposal operations - for example, liability, site closure, and long-term controls - are substantially reduced.
Any distinguishable secondary waste is the originator's responsibility; disposal costs associated with these secondary wastes must be considered in a cost analysis.

By selling the cleaned scrap metal, the recycling company offsets disposal costs associated with indistinguishable secondary wastes.

CONCLUSIONS

There are no present DOE regulations, policies or procedures that prohibit DOE facilities from contracting with commercial recycling companies for services to transport, decontaminate and recycle surface contaminated materials. If detailed characterization of the material at the DOE sites is not practical, then general information, including suspected radionuclides and contact dose rates, can be used to ship materials to a recycling facility as "suspect material" for final determination by the recycler. The originating DOE facility will also have to make arrangements to receive shipments of distinguishable secondary wastes generated by the recycler for appropriate disposition.

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DEVELOPMENT OF MAJOR PROCESS IMPROVEMENTS FOR DECONTAMINATION OF LARGE, COMPLEX, HIGHLY RADIOACTIVE MIXED WASTE ITEMS AT THE HANFORD SITE T PLANT

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ABSTRACT

As part of the decontamination/treatment mission at the Hanford Site, Westinghouse Hanford Company, under contract to the U.S. Department of Energy, conducts decontamination activities at the T Plant complex. Currently, the 221-T canyon High-Level Waste Decontamination Facility and the 2706-T Low-Level Waste Decontamination Facility capabilities are limited because upgrades are needed. Major process improvements must be developed to decontaminate large, complex, highly radioactive mixed-waste items. At the T Plant complex, an engineering team process was used to project possible solid mixed-waste feed streams and develop a preconceptual system to decontaminate and treat the waste. Treatment objectives and benefits were identified. Selected technologies were reviewed and improvements required to implement a preconceptual system at T Plant were considered. Decontamination facility alternatives were discussed in conjunction with ongoing and future decontamination activities at the Hanford Site, including efforts to enhance overall decontamination operations and capabilities.

INTRODUCTION

The Hanford Site (currently owned by the U.S. Department of Energy) was acquired by the Federal Government in 1943 to house facilities for the World War II production of plutonium. For more than 20 years, the facilities were dedicated primarily to plutonium production and management of the associated wastes. In later years, programs were diversified to include advanced reactor research and development of renewable energy technologies. In recent years, the mission has changed from production of special nuclear materials to primarily waste management and environmental restoration. Westinghouse Hanford Company is the current Management and Operations Contractor for the Hanford Site, which comprises approximately 1,450 km2 of semiarid land located in south-central Washington State.

T Plant, located in the 200 West Area of the Hanford Site, was constructed from 1943 to 1944 as a wartime plutonium separations facility. It began chemical processing using the bismuth phosphate process in 1944 and operated under that mission until August 1956. In 1957, T Plant was converted into a decontamination and waste management facility and continues in this mission today. The T Plant canyon is a reinforced concrete structure 259 m (850 ft) long by 22.5 m (74 ft) high by 20.7 m (68 ft) wide. The canyon building contains 40 cells, each of which has a 1.8-m- (6-ft-) thick concrete cover block with 78.7-cm- (31-in.-) high protruding cover

bails. Decontamination operations occur above these cells on the canyon deck. The T Plant canyon is currently conducting only limited decontamination activities because upgrades are necessary before items can be accepted that have detectable alpha contamination or dose rates that exceed 100 mrem/hr on contact.
T Plant complex support facilities include the 221-T canyon, 2706-T Low-Level Waste Decontamination Facility, 214-T Storage Facility, 271-T Office Facility, 211-T Chemical Storage Facility, and mobile offices and change rooms. Figure 1 shows the T Plant complex layout and a cross-sectional view of the 221-T canyon.
The 2706-T facility, constructed in 1960 and upgraded in 1994, provides decontamination services for large transportable equipment having low levels of radioactive and hazardous contamination. The facility is used to decontaminate railroad equipment, plant process equipment, vehicles, and tools. The facility is equipped with a bridge crane, work pits for cleaning rail and highway rolling stock, steam cleaners, ice blasters, and various tanks. The facility cannot routinely accept items with detectable alpha contamination or items with dose rates exceeding 100 mRem/hour on contact.

WASTE STREAM EVALUATION

A waste stream analysis was performed to identify all potential feed streams to a centralized decontamination facility (1). It was estimated that over 350,000 m3 of contaminated equipment waste could potentially be generated over the next 30 years. This estimate included over 1,800 pieces of long-length equipment from tank farms (equipment items requiring size reduction for disposal in a standard burial box), a substantial mixed waste portion of the waste stream (over 35 percent of the total waste stream), and significant transuranic contaminated material.
To date at the Hanford Site, the mission to decontaminate and treat large complex equipment items and components that are currently installed in high-level nuclear waste storage tanks has received the greatest attention. These components (e.g., pumps, instrument trees, and air lances) can be up to 21.3 m (70 ft) long, 167.6 cm (66 in.) in diameter, and weigh over 70 tons when installed in the shielded transfer containers. These items are highly radioactive (up to 66 Rem/hour on contact) and are also designated as radioactive mixed waste because of the presence of a variety of chemical contaminants.
Uncertainties of the projected waste stream have a significant effect on the development of a long-term decontamination/treatment strategy and the development of a definitive processing approach. Further analysis of the preliminary waste stream projection indicates that a large portion of the contaminated equipment may never be removed from existing facilities and will either remain in place in a stabilized manner or be decontaminated/treated in the field. Recent studies indicate that only 200 pieces of long-length contaminated equipment will be retrieved over the next 10 to 15 years instead of the 1,000 pieces originally projected.
If a substantial feed stream develops, a centralized decontamination facility will likely be required. However, if the key generators at the Hanford Site perform their own decontamination and treatment in the field or obtain acceptance from the regulators to stabilize contaminated equipment in place, capital funding for a new treatment facility may not be warranted.

DECONTAMINATION/TREATMENT OBJECTIVES

Decontamination/treatment of surface-contaminated equipment as an alternative to reduce solid waste volumes and storage requirements has been deemed an option which should be evaluated further.
The benefits directly associated with the decontamination/treatment of the equipment can be summarized as follows:

- Significantly reduces radioactive mixed waste volumes and costs associated with packaging and subsequent disposal (Low-level disposal over radioactive mixed waste disposal is more economical by an approximate factor of three) (2).
- Provides a means for disposal of land-disposal-restricted hazardous waste (defined in 40 CFR 268 (3) and WAC 173-303-140 (4)), which requires treatment before disposal, mandated by the Resource Conservation and Recovery Act (RCRA) of 1976 (5).
- Contributes to waste minimization, conserves natural resources, and reduces personnel radiation exposure
- Reduces the cost for direct replacement of failed equipment
- Reduces the area required to store contaminated equipment.

By achieving these objectives, considerable cost savings can be realized through the reduction of waste volumes, mobility, and toxicity. Also, the possibility of reuse

or recycling can minimize waste storage and disposal costs. Figure 2 illustrates the desired end states of the material following decontamination/treatment.

TECHNOLOGY/PROCESS EQUIPMENT SELECTION

Team Engineering Approach

Central to developing major process improvements at T Plant is the process for selecting decontamination technologies and process equipment. To tackle this effort, T Plant personnel used a team engineering approach to define and evaluate waste and equipment treatment, handling, and volume reduction technologies and processes. This team, known as the T Plant Treatment and Disposal Evaluation (TRADE) team, was tasked with identifying technologies that could be implemented to support the waste and equipment decontamination mission. Also, the team was to identify the most viable technologies and the processes for application of those technologies.

To accomplish the process development objectives, the TRADE team formed several focus groups to examine containment systems, handling technologies, size-reduction technologies, treatment technologies, existing plant systems and capabilities, and waste stream identification and characterization methods. Throughout this process, the TRADE team followed the general methodology outlined below:

Review existing plant capabilities to determine what is usable, acceptable, and viable or could be made so through improvements (e.g., retrofit or operational changes)

Establish contact with industry experts, manufacturers, and suppliers to obtain information on existing and new technology developments and processes.

Evaluate data in a team format to determine the most viable treatment/process options and the best methods for incorporating the options into plant functions in a manner consistent with "debris" rules and accepted by cognizant regulatory authorities.

Provide recommendations for option selections to management and develop conceptual engineering information for implementation of the selected options.

Decontamination/Treatment Process Development

The team engineering efforts of T Plant's TRADE team resulted in the development of a process flow referred to as the primary decontamination module/secondary decontamination module (PDM/SDM). The PDM/SDM process flow will provide sufficient throughput capability to meet the expected contaminated equipment generation rates over the next 10 to 15 years. The concept includes two decontamination modules. In the first module, size reduction and chemical decontamination to reduce radiation levels will be performed. In the second module, abrasive cleaning and waste packaging will be performed to meet RCRA hazardous debris rule requirements. It is intended for the PDM/SDM process to be primarily remote-handled; however, waste packaging operations and material handling between the PDM and the SDM can include some contact-handled activities. A process flow schematic is provided as Fig. 3. Conceptual layouts for the PDM/SDM are shown in Fig. 4.

The following is a brief description of the PDM/SDM process:

Upon transfer into the facility, the equipment item will be remotely removed from the container using manipulators and an overhead crane. The container will be decontaminated using a water spray or by steam cleaning in the 2706-T facility. These decontamination methods will be adequate because the containers are expected to have only slight amounts of smearable contamination.

The equipment item will be transferred by conveyor to a mechanical cutting apparatus for size reduction. Size reduction will accomplish two objectives, 1) long-length items can be reduced to lengths that can be packaged in standard burial boxes, and 2) equipment can be cut and separated into portions having different levels of contamination. A typical tank farm pump can provide an example of the advantage associated with portioning contaminated items before decontamination/treatment. Highly contaminated portions of the pump that have been submerged in the tank waste could be cut and separated from less contaminated portions of the pump exposed only to the airspace between the surface of the waste and the tank dome. Another advantage would be the capability for separation of mixed waste parts of a piece of equipment from parts designated as low-level waste.

Following size reduction, The equipment pieces will be decontaminated using chemical dip tanks. Current chelating, oxidizing, and acid/base chemical technologies provide excellent radionuclide removal efficiencies. Three dip tanks will be used, each with its own rinse tank. Secondary containment will be provided for each of the dip and rinse tanks. A filtration and treatment system, an integral

part of this equipment, will provide treated recycled chemical fluids to the process.

The chemically treated equipment pieces will be moved into the secondary decontamination module where abrasive cleaning will be performed by carbon dioxide pellet blasting. This activity will remove most of the surface and embedded contamination and will also be very effective on hazardous waste such as lead-based paint. The CO2 pellet blasting operation will be performed in a self-contained module to ensure independent ventilation from the high rate of off-gas generation.

An inspection will be performed at an inspection station following abrasive cleaning and before final waste packaging. This inspection station satisfies the RCRA hazardous debris rule requirement for visual inspection. Equipment pieces with remaining radionuclide or hazardous waste contamination in excess of allowable quantities will receive repeated treatment (by chemical and/or abrasive methods) until they are deemed acceptable at the inspection station.

After passing inspection, the waste pieces will be moved to the waste packaging station. The waste packaging station includes a grout/stabilization station to meet RCRA requirements, a burial box packer, and a drum/drum overpack packing station. These various packaging stations provide flexibility for compliance with final waste disposal packaging requirements.

Facility Upgrades

To accomplish decontamination of large, complex, highly radioactive mixed waste items at T Plant in a manner compliant with all Federal and State environmental and safety standards, major process improvements to T Plant are required. These upgrades were identified after a thorough review of T Plant's viability to perform a long-term decontamination mission. These improvements include upgrading the electrical, ventilation, and liquid storage and handling systems at T Plant; upgrading a variety of other systems; and installing the new processing modules.

Alternative Evaluations

Alternatives to the installation of new decontamination/treatment process modules in the 221-T canyon are being considered. One alternative is installing the process modules in a annex to the 2706-T facility. Another alternative is performing treatment/stabilization activities in the field. The process of developing new decontamination/treatment capabilities at the Hanford Site is in its infancy and will be an evolutionary process driven by the cleanup mission and regulatory requirements.

CONCLUSIONS AND FUTURE ACTIVITIES

The evaluation and understanding of potential solid waste feed streams and the processes necessary to treat and decontaminate these streams is an ongoing, iterative process at the Hanford Site. Each step in the engineering process brings forth additional information and knowledge in the processing of unique waste types. Several decontamination facility alternatives have been developed and evaluated. These alternatives considered a full range of possibilities from no action through development of full-scale facilities.

Based on the analysis of data accumulated and the evaluation of alternatives, there is not sufficient justification to proceed with a centralized high-activity decontamination facility.

Long-length highly contaminated equipment from the waste tanks may not be a viable near-term waste stream; treatment in the field is currently being evaluated.

The waste streams are highly uncertain and will be formally evaluated in detail in the next six months.

The capital costs are relatively high.

If the treatment of long-length contaminated equipment in the field does not prove to be viable or acceptable, or if another waste stream is identified with a high degree of certainty, the use of a small annexed facility should be pursued in conjunction with the low-level decontamination capabilities at the 2706-T facility. It is vital that the low-level decontamination activities at the 2706-T facility continue and be enhanced as new or emerging technologies are identified. Any upgrades needed to maintain 2706-T should be pursued. In addition, efforts should continue in treating and disposing of the equipment and material already in the 221-T canyon and using the 221-T canyon for limited decontamination operations. The facilities and personnel associated with the 2706-T facility and the 221-T canyon can provide critical support in the areas of low-level decontamination, repackaging, and technology development. If the decision is made to decontaminate/treat material

in the field, T Plant personnel (both at 221-T and 2706-T) can provide a valuable service by developing, demonstrating, and testing decontamination tools and techniques that can be transported to the field.
A single facility or approach for dealing with the diversity and volume of possible waste streams is not practical. The optimal solution will likely involve a combination of alternatives and will be phased in as the site decontamination/treatment needs evolve and regulations change.
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25-6
RECYCLING RADIOACTIVELY CONTAMINATED MATERIALS: EXPERIENCE AND EXPECTATIONS
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ABSTRACT
Just in recent years, Federal Agencies, especially the Department of Energy (DOE); the Department of Defense (DOD); the Environmental Protection Agency (EPA); and the Nuclear Regulatory Commission (NRC), as well as the commercial nuclear enterprise, have begun to consider radioactively contaminated materials as resources for beneficial reuse rather than wastes. Most outstanding among these materials are metals and concrete which have become radioactively contaminated in various nuclear enterprise activities.
The DOE Oak Ridge Operations Office (ORO) began to move in the early '70's to manage, to a recognizable degree, contaminated scrap metals as resources rather than wastes. In the mid '80's, ORO involved private industry in responding to DOE needs in dealing with scrap metals. From among several industrial firms, the Scientific Ecology Group, Inc. (SEG) was selected to deal with the specified ORO scrap metals inventories.
SEG has Metal Processing Facilities in Tennessee, where effective technologies and techniques are used to process radioactively contaminated metals to remove contamination from both surface and volumetrically contaminated metals and form them into items for beneficial reuse. Currently, feeds for these facilities are provided from commercial nuclear power plants, DOE, and DOD.
Customized products from volumetrically contaminated scrap metals are cast shielding, canisters, boxes, and reinforcing rebar mats and fibers for use within the DOE complex. Decontaminated items which meet the criteria of the State of Tennessee, NRC Regulatory Guide 1.86, and the SEG approved procedures are "free released" to the industrial community for "unrestricted use."
Several million tons of radioactively contaminated scrap, currently in inventories, have been identified. Programmatic redirection and decommissioning of nuclear facilities are producing many millions more. Meeting identified and expected needs for metals will require the use of these amounts.
SEG is processing these metals to meet the prescribed requirements using innovative technologies and techniques to reuse a tremendous national resource. In the SEG metal processing facilities, beneficially reusing contaminated metals from DOE, DOD, and commercial sources is achieved. Customized cast shielding is produced in four sizes of shield blocks for use in nuclear physics and high energy research. In this area, needs for more than 200,000 tons have been identified. 65,000 tons of recycled metal shielding are being produced for the stated needs of DOE. Canisters for high level glass logs are to be made from SEG smelted stainless steel. Smelted stainless steel ingots will be available when required for use in fabricating containers for wastes. Metal boxes and other specified containers for low-level wastes are being fabricated. Construction activities and container fabrication of

recycled concrete will be provided rebar, mats, and fibers.
SEG stands ready, in current and planned facilities, to respond appropriately to dealing with recycling-contaminated materials, especially metals and concrete. Other companies are beginning to enter this endeavor. The DOE is moving positively toward making these recyclable materials resources for recycle and beneficial reuse. A trade association has been formed composed of private firms with capabilities and interests in processing metallic scraps from the federal and commercial nuclear enterprises.

INTRODUCTION

This paper focuses upon opportunities and necessities to reuse valuable materials for tremendous cost savings, waste minimization, pollution prevention, conservation of important resources, and providing socio-economic benefits. Identification of needs, resources, responding technologies and techniques, products, and impacts will be discussed.

A very important consideration is the sheer volume of the large amounts of metal that have been, and will be, generated as a result of DOE decontamination and decommissioning activities. The preliminary estimate of metals generated from decommissioning activities over the next 15 years is in excess of 1.5 million tons. This includes metallic equipment, piping, structural steel, and other metal components that are housed in buildings that no longer have functional value to the DOE in this Post-Cold War Era.

DOE considers this metal as both an asset and a liability. If this metal were to be buried, it would consume six million cubic feet of scarce disposal space. At a minimal cost of $50.00 per cubic ft., this results in a disposal cost of $300,000,000.00. The value of the 1.5 million tons, if it can be recycled and beneficially reused, is in excess of 1.5 billion dollars.

Of special interest are results of an ongoing execution of a Program Research and Development Announcement Project (PRDA). The objective of this project is to demonstrate how contaminated metal can be utilized to produce storage, transport, and disposal containers for use within the DOE system for the disposal of radioactive waste. Containers have been produced and are in use. These disposal containers have been specially designed to meet transportation and disposal requirements of both the Department of Transportation (DOT) and the DOE. Several other containers made with recycled metal and concrete are still in process.

The PRDA program involved transporting contaminated carbon and stainless steel materials from the Fernald Environmental Management Project (FEMP) to SEG where it was decontaminated, melted, rolled into sheet, and fabricated into 4 x 4 x 6-ft. containers. These containers were designed not only to meet today's requirements but also to comply with new anticipated requirements expected from the DOT some time in 1995. The containers are also designed for a higher payload in order to meet DOE's need to transport more dense material, such as the soil, metal, and concrete which will result from environmental restoration and decommissioning activities.

These containers made from recycled steel are destined to be returned to the Fernald Site where they will be utilized to transport and dispose of radioactive materials at the Nevada Test Site (NTS). In addition to the FEMP, several other DOE facilities have indicated an interest in receiving prototypes of this container for an independent review and evaluation of their use.

Development and production of these containers from recycled metal is significant because this success has shown how the DOE and private industry can work together to produce valuable products from an asset which would otherwise be a liability. This project has shown recycling metal to be economically feasible, DOE could avoid the cost and need for the disposal of the metal, while creating a whole new business and market within the private sector that will create jobs across the country. This program which will arise from this project can utilize surplus DOE facilities and equipment as part of the program and extend their usefulness and longevity which will strengthen the American economy.

Combining the beneficial reuse of recycled metals with supply of needed waste containers is now a successful technological and sociological achievement in DOE privatization venture.

More than a million tons of radioactively contaminated metals, considered scrap and waste, have been identified at the many sites under the purview of the DOE and many more with the DOD, and in the commercial sector. Many more quantities will be coming from the decontamination and decommissioning activities. Several concerns exist

regarding the disposition of existing inventories and potential generations of contaminated scrap metals. In the past, good potentially reusable metal has been buried as waste. Time has come, and is much overdue, for that practice to cease. Capability to process these scraps to needed products exists, and responding facilities are in operation.

In the late eighties, the DOE's Oak Ridge Office (ORO) pioneered the move to involve private industry in dealing with the contaminated scrap metal under its purview. Consequently, SEG emerged as the leader in processing contaminated metal for beneficial reuse.

Now, DOE has an active scrap metal management program. Materials, equipment, and facilities that exceed their useful life or are no longer necessary to support DOE's mission are either stored on-site, decontaminated and released, or processed and sent to disposal sites. Considerable development has been performed by DOE contractors on the metal-refining (smelting) of contaminated metals. SEG has enhanced and implemented those technologies and developed new technologies and techniques at its own expense. Some decontamination processes are acceptable for unrestricted sale (e.g., aluminum), while others do not meet regulatory limits. DOE supports efforts to find effective means of recovering the value and utility of contaminated materials.

Using effective technologies and techniques for processing radioactively contaminated metals, SEG has built with private funds and has in operation in Oak Ridge, Tennessee a Metal Processing Facility (MPF). This MPF is used to process radioactively contaminated metals, rid them of the majority of contamination, and form them into customized shield blocks and other beneficial use items. Significant volume reduction for scrap metals (in excess of 20 to 1) is achieved with metal melting services.

This facility has the capability to decontaminate metals, including lead, carbon steel, stainless steel, copper, aluminum, and various alloys. Feed for the MPF is currently coming from commercial nuclear power plants, DOE, and DOD.

STATUS

In the early days (1940's, 1950's and into the 1960's) of the Nuclear Enterprise, the principal thrust under Clinton Laboratories, Manhattan Engineering District, and the United States Atomic Energy Commission (AEC) was production of fissionable materials for military uses, with consideration of nuclear science for peaceful purposes being secondary, but recognizably emphasized. During that period, surplus equipment and scrap materials contaminated with any degree of radioactivity were moved to storage, scrap piles, and waste disposal areas with little attention beyond getting the materials and items out of the way of research and production and to make room and convenience for new equipment for expanding and new programs. Minor effort was made to make a few facilities and some equipment and materials available for uses other than they were originally intended.

In the mid 1960's and early 1970's, focus was turned upon peaceful uses of nuclear energy and the associated activities. Serious consideration began to be given to the psychological, physiological, and economical impacts of all aspects of both the public and private involvements in nuclear energy and its applications. Initially, mild interest, then funded activities, arose to examine the enterprise's little used and surplus facilities, inventories of materials, scraps, and wastes. Special attention was given to those which were contaminated with radioactivity acquired in research, development, and production of nuclear materials, parts for weapons, and fuels for nuclear reactors.

The moves toward identifying and decontaminating materials and items which were contaminated with uranium, fission products, and other radioactive materials were first motivated by desire to recover uranium, when that material was scarce in available form. Although tempered with the federal desire to get the most production for the least dollar-cost with little regard for other impacts, other motivations entered. Among those motivations were declassification, maximum utilization of little used and surplus items, decommissioning, control of nuclide migration from scrap yards, aesthetics, waste management, and public concern. This led to a few alert and astute management considerations in the late 1970's and early 1980's responding to obvious and anticipated needs revolving in somewhat ineffective ways around utilization of surplus materials, program changes, aesthetics and public concern, imagery improvement and clean-up, decontamination, waste containment, volume reduction, waste minimization, pollution prevention and known and anticipated

needs in scrap and waste management and facility decommissioning and transition. Beginning in the mid-1980's, the acts began to come together. That move carried to implementation such watchwords as "Conservation of Strategic National Resources," "Maximum Utilization," "Decontamination," "Waste Minimization," and "Privatization." During that evolution of management concern, the DOE and its predecessors and contractors began to recognize outstanding needs for dealing with its massive amounts of contaminated scrap metal and metallic waste materials. In 1983, based upon earlier research, surveys, findings, and recommendations, positive funded action was initiated to respond to the needs in management of scrap metals. This resulted in positive actions to involve the private sector in demonstrating technologies and techniques and to implement approaches for processing those materials to acceptable forms for beneficial use or disposal. Among those actions was a request to private industry to assist in characterizing, sorting, and volume-reducing a portion of DOE scrap metal. Subsequently, interested commercial firms were invited by DOE to participate in a two-phase effort to decontaminate, process, and remove the respective scraps for beneficial reuse or appropriate waste disposal. SEG responded to that invitation and in partnership with DOE demonstrated its technologies and techniques appropriate for dealing with all aspects of the thousands of tons of scrap composed of ferrous, nickel, copper, aluminum, and miscellaneous scraps.

Now, these materials are being moved from DOE, DOD, and commercial sources for processing to beneficial reuse and appropriate disposal of any resulting secondary waste. SEG is acquiring that scrap to process it, using proven environmentally safe and regulatory acceptable technologies and techniques in accordance with DOE desires, to move the maximum amount possible into beneficial use.

Of particular concern and high motivation to SEG is arranging for the ferrous scrap materials (and some copper and aluminum) to be fabricated into customized shielding blocks for use in DOE's high-energy and nuclear physics programs. Those programs, particle acceleration and beam projects, require high tonnages of high density shielding. Since the early planning stages of SEG's response to this scrap project and continuing to date, SEG has been in contact with DOE contractors who have defined and anticipated needs for large quantities of molded-to-design shielding materials for use in funded and proposed major experiments. They have great interest in SEG's supplying their needs from at least five justification standpoints.

Customized Shielding

Much conservation of personnel, energy, and time, structure flexibility, facility efficiency, and engineering and construction accuracy are afforded by made-to-order shielding. At the potential users' requests, SEG has spent much time in planning and design efforts for current and planned shielding requirements and is producing four types of customized shield blocks.

Resource Utilization

Many tons of valuable steel are required for these experiments. The shielding materials, when in place in active experiments, acquire high induced radioactivity. This condition does not significantly reduce shielding efficiency, but will present a very troublesome problem when disposal is required. If virgin steel were used, additional radioactive waste would result and tremendous cost avoidance would not be realized.

Waste Minimization

Since the DOE scrap is only slightly contaminated and as such might be difficult to decontaminate to meet regulations for free-release to American public and noncontaminated steel shielding would become radioactive in the experiments, using DOE scrap material would reduce the waste potential by the amount of recycled materials used in shielding.

Pollution Prevention

The propagation of waste migration and expansion for several miles is reduced.

Economics

The scrap metal, an excellent source of shielding, is already owned by DOE and can be processed by SEG to the users' requirements.

The identified near term shielding needs show more than 65,000 tons of customized shielding blocks. Continuing annual requirements have been indicated and estimated, showing an increasing requirement.

The development of this beneficial reuse of scrap steel has involved: extensive negotiations with the interested parties, modification of the processing that will

have to be performed, and the development of close tolerances for the fabrication of the shielding blocks. SEG has conducted these negotiations with the principal laboratories.
This beneficial reuse methodology is independent of any proposed regulatory change; it directs the metals toward a substantial beneficial reuse with virtually no chance of inadvertent release to the public. This method will also provide a beneficial reuse of future quantities of metals.
Other uses for the metals have been found such as metal boxes, canisters, and drums for DOE and commercial radioactive waste; target material and armor plate for the DOD; and reinforcing rebar and mats for use in the nuclear enterprise. Metal and concrete containers have been developed, tested, and are ready for use.
Your attention is invited to Tables I and II for a perspective of quantities, values, and potentials of a representative portion of our national resource of inventories and requirements. Table I gives selected scrap metal volumes and values; Table II gives SEG products and quantities of specific materials needed for required items.
Advantage of reclamation or reuse of these DOE materials are many; a few of which are:
1. The beneficial reuse of valuable metal thereby reducing the potential environmental impacts and health and safety risk inherent in the mining and refining of virgin ore and the production of power required to obtain and refine the ores.
2. The beneficial reuse of nonreplenishable scarce resources.
3. The beneficial reuse of volumetrically contaminated metals in radioactive environments, thereby reducing the nonradioactive metals that would otherwise become radioactive and increase the total amount of radioactive material requiring disposal.
4. The potential reduction in cost of high energy physics or other nuclear research programs requiring large volumes of ferrous metals that can be volumetrically contaminated. (This customized 35,000 tons of shielding for Los Alamos National Laboratory alone is estimated by DOE to save twenty million dollars.)
5. Generation of the least amount of waste requiring disposal or any other ultimate solution alternative. Thus, potentially reducing the amount of disposal space required and the amount of land removed from productive use.
6. The potential reduction in costs of radioactive material management from both "3" and "5" items previously listed.
7. The potential to turn an apparent liability (i.e., stored contaminated scrap metal) into revenue or equivalent U. S. Budget savings.
8. The removal of an apparent and/or potential contamination source term from the DOE, DOD, and commercial sites.
9. The permanent isolation of a majority of contaminating radionuclides from people and the environment.
10. A proactive response to a need (the disposition of the scrap metal) that has existed for over thirty years.
11. Resolution of the disposition of the scrap metal in accordance with all existing statutory regulations.
12. Providing the most outstanding waste minimization endeavor.
In addition to customized shield blocks, fabrication equipment makes possible production of other steel components such as sheet steel, centrifugal cast containers, rebar, and fiber reinforcing mats for use in meeting requirements of the nuclear enterprise.
Cast shielding is being produced by SEG for use in high energy and nuclear physics studies. Needs and requirements for more than 200,000 tons of products have been identified for those studies. Soon, stainless steel ingots are expected to be produced and stockpiled for use in making canisters for high level waste glass logs and possibly for use in fabricating containers for transuranic wastes. Moves are underway to fabricate boxes for containing low level wastes for use in both public and private sectors. Extrusion capability can be used for producing rebar for use in construction activities. Reinforcing mats will be produced using a new fabrication technique.
All in all, the MPF is a valuable asset for meeting the many needs for the total nuclear enterprise and simultaneously beneficially reusing a tremendous national resource through innovative technologies and techniques.
Recycle of radioactively contaminated metals is underway but needs the incentive

that can be provided by Congressional understanding, endorsement, and emphasis and by properly motivated groups. A few very minor efforts have been effected at some DOE installations resulting in low tonnage being decontaminated on-site and released to the public. SEG has produced approximately 8,000 tons of customized shield blocks and decontaminated more than 1,000,000 pounds of lead and more than 6,000,000 pounds of brass. SEG is obtaining more than 68,000 tons of scrap from the DOE Oak Ridge Operations Office including more than 9,300 tons of nickel ingots. SEG has and is currently processing many tons of scrap metal from the FEMP site. SEG has a contract with Martin Marietta Energy Systems to remove scrap metal from the Oak Ridge site waste streams being sent to SEG for processing. More than 8,000 tons of scrap metal have been received for processing by SEG from DOD. Scrap metal is being received from more than 100 nuclear power plants.

Since 1980, the commercial nuclear industry has decontaminated more than 150 million pounds of metal.

DOE has not recycled metal to the extent of the commercial industry or other government agencies. Some of the barriers limiting or delaying DOE participation are:

- unrealistically low burial costs which do not reflect total life cycle costs or the projected costs of new disposal sites. However, action seems to be underway by DOE to adjust this situation.
- extended Procurement cycles
- no release criteria for volumetrically contaminated bulk material
- reluctance to certify secondary waste for disposal at Nevada Test Site and Hanford Reservation
- lack of mission orientation

SEG has appropriate permits and licenses to deal with all scrap metal requirements and stands ready to process DOE scrap metal or scrap from any other source. Further, SEG offers a full menu of services dealing with all aspects of Environmental Management. Other qualified companies are showing interests and experience and offer services in processing scrap metal. A Trade Association has been formed composed of private firms with capabilities and interests in processing metallic scraps from federal and commercial sources.

There is and will continue to be a great challenge for recycling metals from the nuclear and related enterprises. Let's respond intelligently.

25-7

CASE STUDY: EVALUATION OF A SCENARIO FOR THE REUSE OF STRUCTURES IN THE PRODUCTION AREA AT FERNALD

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ABSTRACT

The potential for the reuse of uncontaminated structures at federal facilities that are being remediated should be evaluated. Although various factors often limit the viability of such reuse, it may be economically attractive to reuse selected structures. Consideration of a hypothetical reuse scenario for the U.S. Department of Energy's Fernald, Ohio facility shows that the reuse of selected buildings that were not significantly contaminated by production activities at the site may be considerably less expensive than the construction of new ones. The cost of removal of existing buildings is a major factor influencing the relative advantages of these two options. For Fernald, no need for the facility's buildings has been identified; however, the reuse of structures may be a viable option at other facilities.

INTRODUCTION

The U.S. Department of Energy (DOE) and other federal agencies are currently

planning the remediation of numerous contaminated facilities. Some of these facilities are quite large and contain many buildings and other structures having varying degrees of contamination. What is the potential for the reuse of such structures? This paper will address issues related to this question by examining the potential for the reuse of structures at one large DOE facility that is in the early stages of remediation. No plan exists for the long-term reuse (i.e., use not associated with remediation) of any structures at the facility; however, development of a scenario for the hypothetical reuse of structures at the facility allows the identification and evaluation of issues associated with the reuse of structures at large, contaminated federal facilities that no longer are carrying out their original missions.

The facility considered in the hypothetical reuse scenario is the Fernald Environmental Management Project (FEMP), located near Cincinnati, Ohio. The FEMP was used by DOE and its predecessor agencies to produce high-purity uranium metal from 1952 until 1989. The former production area of the FEMP contains over 200 structures, including about 100 buildings. Most of the structures at the FEMP date from the 1950's and many of them and portions of nearby soil and groundwater are radiologically contaminated as a result of the production activities at the facility. It is currently planned that all the structures at the FEMP will be removed during remediation. Remediation is being carried out in accordance with the requirements of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), as amended. Decontamination and dismantling of structures from the production area will be managed using an interim remedial action being carried out for Operable Unit 3, which addresses improvements located in the 136-acre former production area. Environmental media are addressed by a different operable unit. Remediation will continue well into the next century.

This paper considers three major topics related to the hypothetical reuse of structures at the FEMP: 1) What are the major factors that would limit the potential for the reuse of structures at a facility such as Fernald? 2) What opportunities are available for the conservative reuse (i.e., reuse that minimizes the potential for any future human exposure to contamination) of structures at Fernald? and 3) What are the expected costs of a conservative reuse scenario for the facility?

FACTORS LIMITING THE REUSE OF FEMP STRUCTURES

Is long-term reuse of some of the structures in the Fernald production area a reasonable alternative that should be considered? Although a large number of structures potentially suitable for use as offices, warehouses, or manufacturing facilities are available, their reuse is generally limited by a number of factors:

The presence of contamination in most structures, particularly those that were actually involved in production activities.

The presence of transite in the walls and roofs of many structures. Transite is made from cement and asbestos and, although the asbestos is not friable, buildings constructed with this material are not considered suitable for reuse.

The age of the structures. Most of the structures date from the 1950's.

The potential for obstructing the remediation of environmental media if the structures are left in place.

The location of structures in the footprint of any likely on-property disposal cell and the potential for conflicts with the construction of any such facility. Also, structures may be utilized to support remedial activities and may not be available for other uses.

The potential for future liability if undetected contamination is present in reused structures.

No identified need for the structures by the federal government. Assuming continued DOE control of the facility, future use is likely to be limited to DOE or other federal activities.

These factors are expected to be relevant to the potential reuse of many structures at other major DOE facilities also.

At the FEMP, many of the factors limiting the potential for the reuse of structures apply primarily to those structures that were used for actual production activities. Various other structures that have been used for administrative activities have fewer constraints on their reuse because of their generally uncontaminated state, the nature of their construction, and their location outside the likely footprint of any on-property disposal cell. The following section considers opportunities for the conservative reuse of structures at the FEMP and develops a reuse scenario for the

property. The costs associated with the scenario are then evaluated in the subsequent section.

DEVELOPMENT OF A REUSE SCENARIO FOR THE FEMP

A reuse scenario for the FEMP was developed for the time period following the completion of all remedial action activities, except for the restoration of the Great Miami Aquifer, the major groundwater resource underlying the facility. Following remediation, decontamination and dismantlement activities addressed by the interim remedial action would be complete, contaminated soils and construction debris would be removed, and it is assumed that the majority of these contaminated media and wastes would be placed in an on-property disposal cell.

Structures defining the reuse scenario for the FEMP were selected to provide a refurbished property that would be suitable for future industrial or commercial development, as well as for federal government use. Candidate structures for reuse were selected from the approximately 100 buildings available; buildings are the most valuable structural resources at the FEMP. To be considered for reuse, buildings must:

Have only limited or no radiological contamination.

Have no transite used as a construction material.

Be permanent structures in sound condition.

Not be connected to non-reusable structures.

Have a floor area greater than about 900 square meters (about 10,000 square feet) (unless connected to another structure that would result in a total combined area exceeding that amount). Small isolated structures are assumed to have limited value.

Be currently envisioned to present no conflict with site remediation (i.e., not limit access to contaminated environmental media and not obstruct disposal cell construction or operation.)

Buildings meeting these requirements will require relatively limited effort to make them available for reuse. Since only buildings with limited or no contamination are considered for reuse, only minimal soil contamination is expected to be present around or under them, limiting the potential for any conflicts with the remediation of environmental media. The following paragraphs discuss the development of a reuse scenario on the basis of the constraints established by expected post-remediation conditions at the FEMP and the application of the criteria listed above for selecting buildings.

Following remediation, the property is assumed to include an engineered disposal facility, a network of approximately thirty groundwater extraction wells in groups of five to eight wells each, a groundwater treatment facility, and a set of refurbished buildings consistent with a reasonable reuse scenario. The existing parking lot is assumed to be present but greatly reduced in size to be more consistent with the work force required for the reuse scenario.

The engineered disposal facility required to accommodate site-wide needs is anticipated to be approximately 670 m (2,200 ft) by 490 m (1,600 ft) by 12 m (40 ft) high. As presently envisioned, the disposal cell is expected to hold an estimated 1.8 million cubic meters (2.3 million cubic yards) of contaminated soil and construction debris. Due to the large footprint of the disposal cell, which includes land utilized as a lay-down area, a significant portion of the structures in the FEMP production area would have to be removed to accommodate an on-property disposal alternative. The disposal cell is anticipated to be located in the northeast corner of the FEMP. A stormwater control channel around the disposal cell is anticipated to be required to control stormwater runoff.

The groups of groundwater extraction wells are assumed to be located predominantly on the western and southern portions of the FEMP, where there are no current or planned improvements. The groundwater may potentially be treated before being discharged to the nearby Great Miami River. The groundwater treatment facility is scheduled for start-up in early 1995 and is located in building 51.

A reuse scenario was developed using the criteria and constraints discussed above. The approximately 100 existing buildings were first screened to determine those that have only limited or no contamination and that are not constructed of transite. Next, the remaining buildings were evaluated to determine whether they are permanent structures in sound condition, are not connected to non-reusable structures and do not conflict with any anticipated remedial activities, such as the construction and operation of the engineered disposal cell. As noted above, small isolated buildings were assumed to have limited value and were eliminated from consideration.

The structures selected for the reuse scenario are the service building (building 11), administration building (14A), laboratory building (15), security building (28A), human resources building (28B), safety and health building (53A), in-vivo monitoring building (53B), finished products warehouse (77), plant 6 warehouse (79), and the receiving and incoming materials building (82). A simplified perspective view of the FEMP is provided in Fig. 1, with the buildings selected for the reuse scenario highlighted. Assuming that these buildings are refurbished for use as offices and warehouses, the FEMP would potentially have about 32,500 square meters (350,000 square feet) of available floor space. The buildings retained would be the former administration buildings located on the southern portion of the production area and some warehouses located on the eastern edge of the production area but south of the disposal cell footprint. These buildings are readily accessible to the parking lot and main access road. Among the buildings selected, only buildings 11, 15, 77, and 79 would require some decontamination before reuse. A simplified perspective view of the FEMP following remediation is shown in Fig. 2, assuming an on-property disposal cell and a groundwater treatment facility, plus implementation of the hypothetical reuse scenario involving the buildings identified in this paragraph.

EVALUATION OF COSTS FOR THE REUSE SCENARIO

If no actual uses can be identified for the space provided by the buildings included in the reuse scenario, it would be most reasonable to remove them rather than provide maintenance and security for unneeded buildings. No such uses have been identified and current plans call for all buildings at the FEMP to be removed. However, if a potential for reuse actually existed for them, then the major factor determining whether their reuse is reasonable is the relative cost of reuse versus the cost of providing acceptable alternative space. The evaluation of the reuse scenario presented here consists of a consideration of these relative costs. Issues other than those involving costs are considered above in the development of the reuse scenario.

The estimated costs for the reuse of existing buildings and the estimated costs for their replacement with new ones are summarized in Table I. The estimates in the table are provided to allow a rough comparison of the two cases considered and to allow identification of the major factors contributing to differences in their costs. The cost estimates provided were developed to allow these evaluations to be conducted and are not intended to be any more definitive. The cost comparison in Table I does not account for the fact that refurbished buildings would likely be worth less than newly constructed buildings and would have a shorter expected period of use. A more detailed evaluation of the potential for the reuse of structures would account for these factors. No cost is assigned to the land needed for any new construction because it is assumed that any new buildings would be located on uncontaminated areas of the FEMP outside the former production area.

The unit costs for construction used in Table I are approximate values for new construction in the Cincinnati area (1). The cost of refurbishing the existing buildings is based on industry standards (2), assuming that the labor and material components of the total refurbishment cost are 60% and 40%, respectively. The refurbishment cost is estimated to range between 15% to 50% of the costs of new construction for the building types being considered. Given that many factors (e.g., custom fabrication of components to match existing construction) can significantly increase the costs for refurbishment, it was conservatively assumed that refurbishment costs are 50% of the costs of new construction.

Reuse of an existing building may involve removal of any asbestos that is friable or in poor condition. A small amount of asbestos-containing material is present as piping and ductwork insulation in building 82; however the costs of its removal were determined to be negligible in comparison with the other costs for building reuse.

The cost for surface decontamination was estimated assuming that all interior surfaces in buildings with some contamination present (buildings 11, 15, 77, and 79) were cleaned using a high-pressure power washer and that a fixative coating was then applied to all the surfaces. Recent experience at the FEMP indicates that such surface decontamination can reduce removable contamination levels by a factor of fifteen (3). Application of a fixative coating (acrylic latex paint) is used to "lock down" any remaining loose surface contamination. The costs for surface decontamination were estimated from (4) and exclude treatment costs for the resulting wash water.

The cost of providing a replacement for the existing buildings includes the cost of removing the existing buildings. If not reused, these buildings must be removed. If it were decided to construct new buildings on the FEMP to provide space that could be provided by existing buildings, then the cost of removal of the existing buildings is a cost that must be added to the cost of new construction. If buildings are reused, then the cost of their removal is avoided.

The costs for removing buildings were developed using a "bottoms-up" cost methodology. Contaminated buildings are assumed to be decontaminated before being dismantled. Direct costs associated with decontamination and dismantlement include containment of potential airborne contaminants, surface decontamination by water washing as needed, disassembly and dismantling, wrapping, and transporting waste materials to interim storage areas. A non-productive time allowance for the use of personal protective equipment was included in all applicable activities. Factors for the various indirect cost components (e.g., overhead and profit, contingency) were applied to complete the cost estimate for the individual buildings. The costs were estimated for each building and combined to give the total in Table I. Values presented here should be considered to be conceptual with an estimated overall level of accuracy of +50 percent/-30 percent.

The costs for decontamination and dismantlement of the buildings were developed for this study and do not represent official estimates for the FEMP. Note that the estimated costs given in the table apply only to the specific buildings considered in the scenario; the costs do not apply to any other structures.

Disposition costs associated with the wastes produced by dismantlement of the buildings were estimated on the basis of a preliminary cost estimate developed for an on-property disposal cell (5). The incremental costs of on-property disposal are about $84 per cubic meter ($64 per cubic yard) and the disposal volume for the wastes generated from the ten buildings would be about 33,000 cubic meters (43,000 cubic yards), yielding a cost for on-property disposal of about $3 million.

The cost of providing replacement space in new on-property buildings that would be equivalent to that provided by buildings identified in the reuse scenario (Case 1 in Table I) is estimated to be about $47 million or about $1,400 per square meter (about $130 per square foot) of floor space. Reuse of the buildings (Case 2) would cost about $11 million or about $340 per square meter (about $30 per square foot). The relative cost advantage of reuse of the buildings over replacement is not sensitive to uncertainties in costs for refurbishment or decontamination. Even if the refurbishment costs are as high as the costs of new construction, reuse may still be advantageous from a cost perspective because of the high costs of dismantling all the structures compared to the costs of decontaminating only several. Decontamination costs are estimated to be only a small fraction of the total costs associated with building reuse and very large increases in decontamination costs would be necessary to significantly affect total costs for reuse. For example, doubling decontamination costs would increase total costs for reuse by less than 10%. Therefore, if additional decontamination such as scabbling of concrete or metal is required in some areas, the effect on total cost for building reuse would be small.

Costs associated with the eventual removal of new or refurbished buildings are not included in the cost estimates developed. Including such costs would decrease the relative cost advantage of building reuse. However, such costs likely would not be incurred for some decades and, particularly in terms of a present-worth analysis, the total costs for the reuse of the buildings would remain well below those for their replacement.

The most critical factor related to evaluating the reuse scenario is the cost of removing existing structures. If uncontaminated structures are dismantled using the same methods as those used for contaminated structures (except for worker protection), then expected costs will be high. The CERCLA Record of Decision for the interim remedial action (6) does not specifically discuss methods that might be used, but always discusses "dismantlement" as opposed to "demolition." Practices at other DOE facilities also appear to favor dismantling as opposed to demolition, even for uncontaminated buildings, if the buildings are located in generally contaminated areas (7). Defining the approach that will be used to remove buildings with little or no contamination is critically important to understanding any cost advantages associated with reuse. If the costs of removing the existing buildings constitute a significant fraction of the cost for the replacement case, then avoiding

dismantlement costs by reuse of buildings could potentially result in substantial cost savings, assuming a need for such space is identified.

CONCLUSIONS

In general, the potential for the reuse of buildings and other structures at the FEMP is limited by the lack of an identified mission for them, by the current condition of most structures, and by the possibility for interference with the remediation of environmental media. These conclusions apply in particular to the buildings that were used for production activities.

Opportunities for conservative reuse may exist for some selected buildings at the FEMP that together contain several tens of thousands of square meters (several hundreds of thousands of square feet) of floor space, that are in sound condition, and that have limited or no contamination. These buildings are administrative and warehouse buildings located outside the major areas of contamination and in areas that would not interfere with the construction of an on-property disposal cell. If a need existed for such space, it might be economically preferable to reuse these buildings as opposed to removing them and constructing new ones. However, the attractiveness of reuse is sensitive to dismantling costs. High dismantling costs favor reuse.

The methods to be used for the removal of FEMP structures having little or no contamination need to be better defined. Such information is important both for evaluating any potential reuse of the structures and for evaluating the total costs for the interim remedial action that will remove them.

Arguments favoring the reuse of some buildings on the basis of cost alone are insufficient to justify retaining such structures in the absence of any specific, identified need for the structures. For federal facilities that are planning remedial activities, the viability of reuse of any buildings will depend on the levels of contamination in the buildings, the level of any contamination around and under the buildings, the state of the buildings, and the future land use anticipated for the facility. However, if constraints involving these factors are not present, long-term reuse of the buildings may be economically attractive. Reuse of structures with limited or no contamination should be considered as a serious option at federal facilities if a need for such structures can be identified.

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FIGURE CAPTIONS

Fig. 1. A Simplified Perspective View of the FEMP, with the Buildings Selected for the Reuse Scenario Highlighted.

Fig. 2. A Simplified Perspective View of the FEMP Following Remediation, Assuming an On-Property Disposal Cell, a Groundwater Treatment Facility (51), and Selected Buildings Retained for Reuse.

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ABSTRACT

To achieve an optimized solution of the overall material management problem, the options of radioactive wastes management, the options for recycling and reuse and, last but not least, the options for disposal of the material as non-radioactive waste should be considered.

A survey is provided on the current status of these options available in Germany. Special emphasis is given to the possibilities of reuse and recycling. Criteria for clearance of material are available for metals. As a principle, the recycling of scrap into products for the nuclear industry like waste containers or shielding plates is the preferred way. If this is not feasible or economically justified, material can be released either without restrictions or with conditions on its further treatment.

Furthermore, the present planning work regarding the management of material arising during decommissioning is presented; aspects of final disposal of radioactive waste are finally addressed in more detail.

INTRODUCTION

In the Federal Republic of Germany, nuclear electricity generation has a share of more than 30 % of the overall production of electrical energy. Most of the nuclear power plants, presently in operation, are expected to end their service life not before some decades from now. Based on an operational time period of 35 years, decommissioning and dismantling will start in the next century. According to current plannings, shut-down of commercial scale power plants will commence in the year 2002. Service life will be followed by preparation for safe storage and a safe storage period of 30 years. The dismantling and demolition period of a modern light-water reactor is estimated to last up to 8 years.

In Germany, a total of 15 nuclear power plants were finally shut down and are in the decommissioning phase or have already been dismantled. It is therefore substantial experience available with decommissioning of nuclear power plants.

Shut-down, dismantling and demolition of research and prototype reactors, installations of the nuclear fuel cycle and research facilities have to be planned as well. There is ongoing planning, e.g. the decommissioning of a fuel fabrication plant, or the ongoing planning for the dismantling of the WAK pilot reprocessing plant in Karlsruhe.

During backfitting and decommissioning of nuclear installations, and to a more limited extent also during normal operation, a considerable amount of radioactive residues, dismantled parts and equipment arises. The management of this material is an important task. In the Atomic Energy Act, the basic modes of material management are set out. The arising material - either residues or dismantled parts - can be recycled or reused without harmful effects or be disposed of as radioactive waste; interim storage before final disposal is possible.

The purpose of this papier is to provide an overview of German policies, regulations, experiences and status of planning for the future tasks of decommissioning of nuclear installations and the corresponding material management.

OPTIONS FOR MATERIAL MANAGEMENT

Overview

In the Atomic Energy Act it is distinguished between radioactive residues and dismantled radioactive parts, which can - according to an article law passed on April 29, 1994 -either be reused or recycled without harmful effects or disposed of as radioactive waste. Before the article law was passed, reuse and recycling were given priority as compared to disposal. Now these two basic options are equally ranked.

An overview of the different possibilities for disposal and recycling or reuse available in Germany is provided in Fig. 1.

Radioactive waste is finally disposed of in deep geological formations. Shallow land burial of radioactive waste is not practiced in Germany. However, if radioactive waste fulfills the condition of "very low specific activity" it can be categorized as non-radioactive material and be disposed of like conventional waste, which is

either incinerated or disposed of in a municipal landfill.
A more complicated pattern is formed by the possibilities of reuse and recycling. In the case of reuse, items from the nuclear field are used elsewhere in their original form and function. Recycling means that the material is taken as input for the production of "new" material. A typical and practically very important example is the recycling of dismantled metallic scrap for the production of waste containers or shielding plates.
Material to be disposed of as conventional waste or recycled or reused will in general be subject to an act of clearance from the system of regulatory supervision. This clearance can be unconditional in the sense that there are no restrictions in terms of reuse, recycling or disposal after clearance. Conditional clearance would for example be steel scrap released for recycling, the condition being that scrap is indeed processed and melted for the production of new steel. Disposal as waste or reuse of items would have to be excluded in such a case. Obviously, criteria for conditional clearance need not be as restrictive as for unconditional clearance.
On the other hand, material has not necessarily to be cleared for recycling or reuse if it can be further treated or used under regulatory supervision. Examples would be the transfer of one piece of equipment from one nuclear installation to another or the melting of radioactive scrap from a nuclear facility in a licensed melting facility.
Radioactive Waste Disposal
According to the concept developed in Germany, radioactive waste is disposed of in deep geological formations. Interim storage before final disposal is possible.
Presently, the Morsleben facility is available for disposal of short-lived low and intermediate level radioactive waste with rather low alpha emitter concentrations (1). This repository constructed in a salt formation has resumed its operation in mid-January 1994. Since then radioactive waste is disposed of in this facility. Operation is presently planned to continue until the end of June in 2000 when the operation license expires. 40,000 m3 radioactive waste shall be disposed of in the Morsleben repository in this period.
The Konrad repository, a former iron ore mine in Salzgitter, is characterized by very dry conditions and substantial layers of clay as a geological barrier (2). The results of a research and development programme on behalf of the Federal Government confirmed that Konrad is in principle qualified as a disposal site for radioactive wastes with negligible heat generation. The licensing procedure was initiated in 1982, and waste disposal can commence at the end of the century provided the license is granted until then. Operation can continue for more than 40 years, with 650,000 m3 waste package volume being disposed of in that period.
Since 1979, an extensive site exploration programme is conducted at the Gorleben site. Presently, two shafts are being sunk for underground exploration to provide necessary information on the geological structure and on the available repository volume in the salt dome being necessary for the assessment of the site's suitability. Start of operation of the Gorleben repository is expected after the turn of the century.
Recycling, Reuse and Disposal as Non-radioactive Waste
Early practices related to recycling, reuse and conventional disposal were based on regulatory guidance issued in 1979 (3). However, criteria have been reconsidered since IAEA has issued basic recommendations on acceptable public exposure in Safety Series 89 (4). On this basis, a number of recommendations has been issued by the German Commission on Radiological Protection (Strahlenschutzkommission - SSK) which are the basis of present licensing.
The first recommendation "Radiological Protection Principles for the non-detrimental Recycling and Reuse of slightly radioactive Iron and Steel from Nuclear Power Plants" was published in 1987 (5).
It is related to the clearance of metal scrap and components leaving a nuclear power plant, which are valid for activity spectra from light water reactors (LWR). Such spectra are dominated by Co-60 and Cs-137. Furthermore, the recommendations are explicitly not to be used if a-contamination is suspected. The recommendations for the release of ferrous and non-ferrous metal give clearance levels for the total activity and are summarized below.
The first priority is to reuse or recycle within the nuclear industry, if this is not feasible then several release options are available.
1. Unconditional release is possible if the mass specific activity is below 0.1 Bq/g

for each piece and the surface activity is below the limits prescribed by the German Radiation Protection Ordinance (RPO).
2. Release of scrap on the condition of melting is allowed if the mass specific activity is below 1.0 Bq/g for each piece and the surface activity is below the limits prescribed by the RPO.
3. Controlled recycling can be considered when the mass specific activity is higher than 1 Bq/g and/or the surface activity higher than the limits prescribed by the RPO. Within controlled recycling scrap is not released but rather melted under license. After melting the product can be unconditionally released if it meets the requirements put forth in point 1.
On a case by case basis the licensing authorities can allow products with a mass specific activity between 0.1 and 1 Bq/g to be used in applications where an increased radiation exposure is not expected. This option is not of great practical importance.
The surface contamination clearance levels of the RPO require the prove, for the entire surface, that the total surface activity (fixed plus non-fixed) is less than
0.05 Bq/cm2 for Alpha-emitters (e.g. Pu-239, Am-241),
0.5 Bq/cm2 for Beta/Gamma-emitters (e.g. Co-60, Cs-137) and
5.0 Bq/cm2 for weak Beta/Gamma-emitters (e.g. Fe-55, Ni-59).
These limits are averaged over any 100 cm2 of the surface.
This recommendation was later extended to non-ferrous metals using essentially the same principles and clearance levels (6).
Based on a model developed by BfS for the incineration of waste and disposal on a municipal landfill, criteria have been derived for conventional disposal (7, 8). A corresponding recommendation by the SSK will be issued in the near future.
A study has been conducted on the recycling and disposal of concrete and building debris (9), but has not resulted in regulatory guidance up to now.
It can be summarized that regulatory guidance for the clearances required for material management has been developed to a fairly complete system so that a basis has been provided for future licensing for decommissioning.

DECOMMISSIONING OF NUCLEAR FACILITIES

Decommissioning Projects

In the Federal Republic of Germany, 21 nuclear power plants are in operation which roughly contribute one third to the overall electrical power generation. On the other hand, 14 units including demonstration and prototype plants, have been taken out of service (Table I).
In 1995, the complete dismantlement of the KKN plant is expected. Decommissioning of the KRB-A plant is undertaken and licensed in the steps turbine building, contaminated equipment in the reactor building and finally dismantling the activated parts. Presently work on step two is underway. Valuable experiences and data for future planning were collected (10), especially regarding radiological and economic aspects of decontamination and dismantling.
Different approaches are taken in terms of decommissioning strategy, e.g., for KWL safe storage for a 30 year period has been implemented.
Decommissioning of the units at the Greifswald site is a major task for the next decade. First applications for decommissioning licenses have been made.

Material Management for Decommissioning

Based both on the criteria for radioactive waste disposal and for recycling, reuse and conventional disposal as well as on the experiences already gained in decommissioning and dismantling projects, the licensees have performed adequate plannings for future decommissioning and dismantling projects. To provide an example, in Table II the amounts of radioactive waste and the materials that can be recycled are presented according to the planning for the dismantling of a 1000 MWe LWR.

Radioactive Waste Arising from Future Decommissioning Projects

Information on radioactive waste arisings is collected by BfS and used in disposal planning. An estimate of radioactive waste originating from decommissioning projects can be outlined in the following way:

A total of about 50,000 m3 of waste package volume is expected from research, demonstration and prototype reactors and from nuclear power plants already finally shut-down as well as from the WAK reprocessing facility at Karlsruhe. According to preliminary plannings, this waste volume is to be expected until the end of 2010.

The 21 nuclear power plants in operation will contribute a total volume of about

57 700 m3. Based on a safe storage period of 30 years before dismantling an average rate of 1900 m3/a is projected for the period from 2033 to 2063.
These planning figures are subject to regular updates by BfS based on information provided by the licensees.

SUMMARY AND CONCLUSIONS

Present material arisings during operation of nuclear power plants and even more those from present and future decommissioning necessitate a well balanced concept for material management. This concept must cover both the criteria for disposal of radioactive waste and for clearance derived from safety assessments. On this basis, optimization of material management can be achieved taking into account economic aspects.
In Germany, the regulatory basis for material management has been developed. As far as recycling, reuse and conventional disposal are concerned criteria have been derived and formulated in SSK recommendations, which are consistent with the fundamental guidance document Safety Series 89 issued by IAEA/NEA. Conditional clearance criteria for disposal, recycling and reuse are available. The derivation of criteria for unconditional release is a major task that is being discussed on the national and international level. Harmonized solutions for transboundary movement of material are urgently needed.
A very attractive option is recycling without any clearance from the nuclear industry. In this case, the new products (e.g. waste containers) are fabricated in a licensed company and subsequently reused in the nuclear field. This is established in Germany on an industrial scale. It must however be recognized, that the need for this type of products could turn out to be rather limited.
The German concept for disposal of radioactive waste is exclusively based on deep geological disposal. One repository licensed for disposal is available at Morsleben at least until the end of June in 2000. Licensing for the Konrad repository which will be limited to radioactive waste with negligible heat generation is in a very advanced state. Planning for the salt dome repository at Gorleben continues.
The utilities are planning the future material management on the described basis formed by the options of radioactive waste disposal and recycling, reuse and conventional disposal.
The BfS is committed to further extend and optimize the regulatory guidance in this important field. It participates actively in the international process to achieve harmonized solutions for transboundary movement of cleared material.

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Session 26 -- Correcting Plutonium Vulnerabilities at EM Sites: Complying with DNFSB Recommendation 94-1
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27-1
ALPHA CHARACTERIZATION OF CONCRETE-SURFACES AT D&D SITES
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ABSTRACT
Waste minimization and worker protection at decontamination and decommissioning (D&D) sites requires continual and extensive characterization for radioactive contamination. Contamination detectors that are sensitive, rugged, fast, and capable of covering large areas are needed. The concrete surface monitor (CSM) developed at Los Alamos National Laboratory supports both characterization and waste minimization efforts at D&D sites containing large concrete surfaces and structures of different forms and shapes. This report describes the CSM design and the results of several field experiments.
INTRODUCTION
Typical radioactive contaminants consist of uranium or plutonium. These isotopes emit alpha particles. An alpha particle can travel approximately 4 cm through air; with such a short range, traditional direct detection methods are often ineffective. Instead of detecting the alpha particle directly, the concrete surface monitor (CSM) detects the ion pairs created as alpha particles interact with air molecules. As a 5-MeV alpha particle interacts with air molecules, it will generate an average of 150,000 ion pairs, resulting from the 35-eV energy loss for every ion pair production (1). An electrostatic field in the CSM transports these ion pairs to a detection plate. The ion pairs generate a small current, proportional to the alpha source strength, that is detected by an electrometer. As compared to traditional detectors, the CSM can operate effectively at a much higher level of sensitivity while allowing real-time and in situ operation.
Using the 300-cm2 CSM, large concrete surfaces can be characterized effectively. An active detection surface ranging from 300 cm2 to 2500 cm2 allows CSMs to monitor a larger surface area with a single measurement. By characterizing concrete surfaces before, during, and after decontamination and decommissioning (D&D) work, the amount of radioactive waste could be minimized. The use of the CSM to support waste minimization will effectively cut the cost of D&D.
CSM DESIGN
The CSM is based on the electrostatic Long-Range Alpha Detector (LRAD) (2). The current generated by the CSM is detected by the sensitive electrometer developed by Guy Arnone at Los Alamos National Laboratory (LANL) specifically for LRAD detectors. The reading can either be sent to a portable computer or read out directly on a portable oscilloscope for a visual interpretation. There are two types of surface monitors; the 2500-cm2 monitor includes background subtraction technology while the 300-cm2 and the 1431-cm2 CSMs do not. The LRAD Surface Monitors have already been described in more detail in several reports for soil surface applications (3,4). The CSM was built specifically for field demonstrations and characterization of

concrete block edges at a D&D site at LANL. Due to the large concrete block surface area, we designed a CSM with an aluminum box monitor 27 cm x 11 cm x 10 cm with an opening on the bottom, yielding a 300-cm2 detection surface. An adjustable clamp was used to fasten this monitor to the block creating a firm contact between instrument and detection surface with varying degrees of freedom.

A background-subtracting surface monitor is shown schematically in Fig. 1. In this detector the alpha particles emitted from the concrete surface create ions only in the lower chamber, and hence a signal only from the first signal plane. All of the ions created by surface contamination will be transported to the lower signal plane; none pass into the upper chamber. Radioactive radon gas that enters the lower chamber from the concrete surface is not electrically charged, so it can freely pass into the upper chamber. This gas will decay equally in both chambers, resulting in an identical contribution of signals on both the first and second signal planes. To speed up the mixing of radioactive gas into both chambers, a small fan is inserted between the chambers (5).

A CSM with an active volume of dimensions 50 cm x 50 cm x 10 cm, and a 2500-cm2 detection surface, allows measurements over larger surfaces with a single reading. Again, the response time for both CSMs is between 30 and 120 s per reading depending on the strength of the contamination.

In the final analysis, the current was measured from the top chamber (instrumental background at that level) and subtracted from the current measured from the bottom chamber (signal from floor plus instrumental background at that location). The calibration factor for the background subtraction CSM was measured at 6 dpm/fA. Two instrumental background readings were taken on the aluminum plate. The results of the instrumental background readings are in Table I, showing both the upper and lower chamber activity.

This data for both the upper and lower chamber shows essentially the same reading.

FIELD DEMONSTRATIONS

Concrete Block Measurements

On July 15, 1994, the first-generation CSM was used to measure the residual alpha contamination on concrete blocks that came out of trenches at Technical Area-21 (TA-21) at LANL. The dismantling of trenches is a part of the D&D effort currently in progress (6).

The 300-cm2 CSMs were used due to their appropriate size and attachment clamps. Prior to the demonstration, the detectors were calibrated using a set of National Institute of Standards and Technology traceable 239Pu sources ranging from 100-5620 dpm. The sensitivity of the 300-cm2 CSM is 7.1 dpm/fA. The sensitivity of other CSMs ranges from 5.5 to 7.1 dpm/fA.

The concrete blocks to be monitored were cut out of trenches. The surface to be monitored was relatively flat, although some areas were rather rough and irregular. These blocks were placed with varying angles, hence at some locations a clamp attachment was used to secure the detector and create an air seal between the surface and the detector volume. For background measurements, an aluminum plate was used. The aluminum plate underneath the detector prevents penetration of any alpha radiation from the concrete or ground into the detector. This gives an instrumental background measurement consisting of leakage current, electronic noise, leakage of outside air into the monitor, and the ambient level of gamma radiation but does not measure any alpha emissions from the surface.

Ten readings were taken from various locations on three different concrete blocks. For each point the instrumental background was subtracted from the measured reading to give the surface activity. For comparison, readings from concrete surfaces around the town of Los Alamos were taken from nine different locations. These locations included an apartment complex, sidewalks in front of businesses, an historical landmark, the Aquatic Center, a church, a library, and the local Post Office totaling 27 readings. In Table II the results from both experiments are compared. For comparison, readings within the city of Los Alamos range from 62 to 93 dpm/100 cm2 above background due to the natural activity in concrete. The measurements were taken from apartment complexes, sidewalks of businesses, an historic landmark, the Aquatic Center, a church, a library, and the local Post Office.

A potential error of 20 dpm/100 cm2 was determined to be a direct result from the portable oscilloscope and the concrete. Concrete has an intrinsic activity level (even when concrete is "uncontaminated") and these readings may be showing variations in the "natural" activity of the concrete.

Water Tower

On May 31, 1994, the 1431-cm2 CSM was used to characterize an 18-m2 area under a water tower at TA-21. A total of 24 points were taken at 50-cm intervals. Instrumental background was again taken on an aluminum plate. Relative to this background, the highest observed points were 160 dpm/100 cm2. This area is relatively clean based on the Department of Energy public release limit of 300 dpm/100 cm2 for transuranics. Figure 2 shows the water tower at TA-21 prior to demolition and removal to a dump site.

Most of the readings were between 40 and 80 dpm/100 cm2 above the instrumental background. Figure 3 shows the contour plot covering the accessible area. The result yielded a maximum count of 160 dpm/100 cm2. This area is relatively clean. Independent measurements performed by the site's health physicist with a commercialized detector confirmed our evaluation.

Previous measurements using similar LRAD soil surface monitors found a naturally occurring alpha emission background around 200 dpm/100 cm2; therefore, most of these readings above instrumental background can be ascribed to naturally occurring activity.

Figure 2. The water tower at TA-21. The area under this tower was characterized.

Figure 3. A contour plot of the measured activity level under the water tower.

Utility Pads and Concrete Floors

On August 30, 1994, both the background-subtracting and the 1431-cm2 hand-held CSM were used to measure the residual alpha contamination on the floors of Building 146 and surrounding concrete pads at TA-21. Depending on the size of the area to be monitored, appropriate CSMs were used. The background-subtraction CSM was used on the floor in Building 146 because it has the larger detection area. The utility pad and water tower outside Building 146 were sampled with the hand-held CSM. Instrumental background was checked using the aluminum plate at a regular interval. The collected ions were read out using the LANL electrometer and a portable oscilloscope.

Figure 4 shows the 24 readings taken from the concrete floor. Some regions were inaccessible as shown by the single-hatched marks.

We took 42 readings on a 61-cm grid on the utility pad. The measured activity varied from 0-40 dpm/100 cm2 above instrumental background for most readings. The concrete floor was measured using the background-subtracting CSM. With an active-detection surface area of 2500 cm2, each measurement covered a larger area than the other detector.

Generally the activity ranges from 0-50 dpm/100 cm2. Three of the 24 readings show higher levels of alpha activity. One reading shows 193 5 dpm/100 cm2; visually, this location shows a fairly large grease spot. Two other locations close to this spot gave readings of 51 3 dpm/100 cm2 and 83 19 dpm/100 m2. Wherever possible, the site health physicist verified the high readings to provide an independent confirmation.

In addition to the Building 146 field experiment, the background-subtraction CSM was also used to monitor the concrete floor in Building 324 at TA-21. Instrumental background was recorded at 62.2 fA and 26 readings were taken over an 89-m2 area. The results show activity ranging from 0-30 dpm/100 cm2. Again, this is equivalent to "normal" concrete.

Figure 4. A contour plot of alpha activity on the concrete floor at TA-21 (Building 146) D&D site.

CONCLUSION

The task of waste minimization of concrete surfaces and structures requires detectors that are sensitive, rugged, operate in real time and in situ, and capable of measuring large surfaces. CSMs have been designed to meet those requirements in addition to being cost effective. Using these CSMs, concrete surfaces and structures can be characterized effectively and efficiently.

ACKNOWLEDGMENTS

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27-2

ALPHA CHARACTERIZATION INSIDE PIPES USING ION-TRANSPORT TECHNOLOGY

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ABSTRACT

Many DOE facilities have several miles of waste pipe systems that are internally contaminated with various and often undetermined radio nuclides. Unfortunately, currently acceptable alpha detection technologies are inefficient, time consuming, and do not address the problems presented by small diameter or curved pipes. In general, the problem of detecting alpha contamination on the inside surface of pipes is complicated by the fact that alphas do not penetrate the pipe walls. Unlike their conventional counterparts, alpha detectors based on ion transport technology sense alpha particles by collecting the ions created in ambient air as the particle loses its kinetic energy. The ions inside the pipe are transported by a fan-generated air current to an electrode inside the detector, which is attached to one end of the pipe. The collected charge at the electrode is proportional to the number of ions created inside the pipe, which in turn is proportional to the number of alphas emitted.

Typically, monitoring for alpha contamination inside pipes or ductwork involves disrupting the operation to access as much surface area as possible for standard alpha monitoring. The detector based on ion transport technology effectively minimizes such disruption and in many circumstances will allow for in situ monitoring of a system that might otherwise not be practically accessible to standard methods.

INTRODUCTION

According to a recent call for decontamination and decommissioning (D&D) proposals by DOE, "A method is needed for performing non-destructive, in situ measurements to detect radioactive contamination inside enclosed volumes. These volumes may include, but are not limited to, pipes, ducts and process equipment." Currently acceptable technologies cannot address the problems presented by alpha contamination located in small-diameter pipes, complex process equipment, and inaccessible volumes. Yet every DOE facility has waste pipe systems or equipment or both that is internally contaminated with possibly undetermined radio nuclides. In addition, gaseous-diffusion plants, like the one at Oak Ridge, Tenn., will require significant internal radiological characterization of their process equipment. Clearly, some kind of internal volume monitor (IVM) is required.

Los Alamos National Laboratory (LANL) is currently developing various IVM detectors, based on ion transport technology (1), which use airflow for ion collection. For such detectors, an air current (generated in the enclosed volume by an external fan) transports the ions to the electrode inside the IVM. The current that is measured is directly proportional to the number of ions (either positive or negative) in the enclosed volume which, in turn, is directly proportional to the total contamination level on the interior surfaces.

Understanding the relationship between the following six parameters is crucial to the future success of monitoring both large and small volumes with IVMs: ion lifetime, grid voltage, airflow, pipe geometry, and detector response. These

parameters are empirically investigated in this paper for one specific IVM currently under development at LANL: the pipe IVM. This detector is designed to non-intrusively detect contamination in pipes and ducts ranging in size from less than 1 cm to greater than 1 m. Most gases can be used for ion transport so pipes and ducts filled with various gases, as well as with ambient air, can be monitored effectively. In addition, because ions can pass through almost any convolution in the pipe, complex plumbing can be readily monitored. This document summarizes the results from laboratory tests performed at LANL in 1994. Over this time, the relationship between the six parameters mentioned above was studied using a prototype pipe IVM.

EXPERIMENT SETUP

The basic mechanical components that constitute a pipe IVM include the grid and associated standoffs, a fan, a spacer to isolate turbulent back flow, and a filter. This hardware is assembled into a detector system and attached onto the end of a length of possibly contaminated pipe as shown in Fig. 1. A spacer filled with flow-straightening straws is required to prevent turbulent back drafts from the fan from affecting the collection of the ions at the grid and effectively lowering the response. A 3M FiltreteTM filter (G-100) at the end of the pipe opposite the detector is also necessary to prevent ions from the room from getting inside the pipe volume. (For stronger sources, such a filter was not necessary because of the large difference between detector response and ambient background.) The single-grid pipe monitor used for collecting the data presented here is shown in Fig. 2.

Two basic pipe IVM designs were tested: the single-grid IVM and the double-grid IVM. As shown in Fig. 2, the grids used for ion collection are basically perforated circular copper disks. For the single-grid design, the signal is taken directly off of the lone grid, to which a nominal 300 V is applied. In contrast, for the double-grid design, the ions are collected on one grid that is kept at ground while the other grid receives the 300 V necessary for sweeping ions onto the collection grid. In both cases, primary ion transport is accomplished by fan-generated airflow through the entire length of the pipe. Comparisons under like conditions (e.g., same airflow and pipe length) between the single-grid and double-grid designs yield a single-grid response that is approximately twice the double-grid response. (The pipe used for this test was 2 ft, or ~ 61 cm, long.) For this reason, the single-grid design was chosen as a benchmark for studying the relationships between response and voltage, airspeed, pipe geometry, and source strength. In addition, the single-grid pipe monitor was used to estimate ion lifetime and study the effect of pipe geometry and airflow on the lifetime of ions within the pipe.

RESPONSE VS VOLTAGE

This test was performed using a 20-ft-(~600 cm) long straight pipe with the goal of determining the optimal voltage for ion collection. The schematic diagram of the experimental set-up is shown in Fig. 1. For this test we used a 217,642 dpm 238Pu source and varied the voltage from 25 V to 1100 V. The airspeed at about 6 in. (~15.2 cm) from the grid was kept at a constant 180 fpm (.35 m3/min) and the source was placed 18 ft (~548 cm) from the detector as the voltage was varied. Plotting response in fA (1x10-15 A) vs voltage, yields a curve that plateaus at approximately 600 V. The experiment confirmed predictions that if voltage is too high, an unusable large signal results from grid-voltage-induced ionization. Traditionally, ion transport detectors have used 300-V batteries because they are commercially available; however, the test results indicate that the optimal voltage that does not ionize surrounding air, yet collects the ions efficiently, is approximately 600 V. This optimal voltage is a function of grid material and geometry.

RESPONSE VS SOURCE POSITION

This data was taken to see how the response from a given source varied as a function of the distance between the grid and the location of the source within the pipe. Figure 3 is one such data set, which clearly illustrates the powerful utility of ion transport technology. As shown in Fig. 3, a clear distinction can be made between a 1000 dpm source and a 2500 dpm source from a distance of 18 ft (~548 cm). Another feature that is apparent in this plot is that, as expected, the sensitivity is lower at 18 ft (~548 cm) than at 2 ft (~61 cm) corroborating the notion that most of the ion losses are caused by interactions with the walls of the pipe.

ION LIFETIME

Similar data involving detector response and source distance was taken using a 217,000 dpm source to calculate the lifetime of the ions. For our purposes, ion

lifetime is defined as the time (distance/air speed) at which the response falls to half the value of the number of ions obtained when the source is placed directly next to the grid inside the pipe. For an airflow of 180 fpm (~.35 m3/min), ion lifetime was calculated to be 8.4 s. Figure 4 shows one such plot of the ion lifetime and a fit obtained using the function

Eq. 1

where N0 is the initial number of ions, t is the life time of the ions and t is the time. Ion lifetime varies as a function of air flow and pipe geometry. Initial experiments indicate that ion lifetime is also a strong function of the radial position of the source within the pipe. A source placed at the center line of the pipe will generate alpha particles that will last much longer than those generated from a source placed at the bottom or top of the pipe; a result which makes sense considering that the velocity profile inside the pipe is maximum at the centerline and minimum at the outer diameter walls.

RESPONSE VS AIRSPEED

For the next set of tests, the grid voltage and source strength were held constant and the airspeed was varied. Figure 5 shows one data set from these tests in which 300 V was applied to the grid using a 217,000 dpm 238Pu source. As shown in Fig. 5, the highest airspeed used was 400 fpm while the highest response was obtained at a setting of 180 fpm (~0.35 m3/min). This data seems to indicate that after a certain high velocity, the ions are simply getting swept through the grid without being detected. The previous data sets discussed have indicated that the probability of capturing an ion on the grid is proportional to airflow and applied voltage. The optimal value of each of these parameters has been determined using the single-grid pipe IVM. We suspect that response is also directly proportional to the grid surface area and that increasing the surface area will yield a higher optimal value for the airflow (and perhaps grid voltage). It is our thought that using a parallel plate "venetian blind" grid design will counter the effect of response degradation at higher airspeeds and also enable detection distances greater than 18 ft (~548 cm). Whether the surface area directly facing the flow ("wet" surface) or the total surface area in contact with the flow is of primary interest will be determined in future tests using the second prototype pipe IVM with a venetian blind grid. Experiments conducted using a monitor fitted with a parallel-plate, venetian blind grid design do indicate increased detection efficiency so application of the concept to pipe monitoring may prove very beneficial.

In another set of similar test runs we observed that there were ranges of velocity over which detector response was constant. This range was fairly wide at 2 ft (~61 cm) and narrower at 18 ft (~548 cm). This suggests that in the limit as source distance decreases to zero, primary ion transport is no longer from airflow, but rather from the electrostatic field around the grid.

RESPONSE VS PIPE GEOMETRY

The next set of data involved placing a 217,000 dpm 238Pu source at four different positions along a pipe with a 90 bend in it. The pipe used was 4-ft long with a 90 bend in the middle and four holes drilled into it. The first and second holes, which were before the bend, were 1 ft (~30.5 cm) and 2 ft (~ 61 cm) away respectively. The third and fourth holes, which were after the 90 bend, were 2 ft 8 in. (~81.2 cm) and 4 ft (~122 cm) away from the detection grids respectively. A grid voltage of 300 V was used along with a 180-fpm (~.35 m3/min) airspeed. A plot of response vs distance is shown in Fig. 6. The decrease in the response between 2 ft (~61 cm) and 2 ft 8 in. (81.2 cm) mainly comes from the increased distance between the detector and the source rather than the bend in between. This particular observation is crucial for facilitating monitoring in bent pipes or process equipment and suggests that such monitoring may be limited by the traditional parameters discussed earlier (e.g., airflow and grid voltage) rather than pipe geometry if laminar flow can be maintained in all sections. In other words, wall recombination effects do not appear to be any more significant in curved pipes than in straight pipes if laminar flow is achieved.

FUTURE DIRECTIONS: SOURCE LOCALIZATION

We are entertaining two ways of localizing sources within pipes. The goal is to be able to pinpoint the location of a point source within a pipe. This may ultimately lead to the ability to detect large concentrations of accumulated material in a pipe containing an otherwise uniform distribution of contamination.

The first method attempted involved shuttering a source to determine the time for

the signal to reach the grid. Knowing the airflow and measured time, we can then calculate the distance from the source to the grid. The accuracy of this measurement will depend on the time resolution of our data acquisition system and the accuracy of the airflow measurement. A simple proof of principle test using a fan and the pipe IVM was performed to investigate the ability to localize the position of a source inside a pipe. A 238Pu source was placed at the end of the pipe and the time between source placement and signal observation was recorded. This time was then used to back-out the approximate position of the source in the pipe. The calculated distance of 16 ft compared very well with the actual distance of 20 ft considering the coarseness of the experiment. These results seem to suggest that with a more refined setup, source positions could be determined to within 2 ft using this preliminary method. Prior to conducting this experiment, a similar attempt was made at determining the source position by pulsing the fan at a very low airspeed (~170 fpm). The negative spike caused by pulsing the fan made it very difficult to see the negative signal caused by the source. If source localization is to be made field-ready, such problems will have to be addressed. Two possible solutions include digital signal processing of the grid signal to separate out fan effects from the actual source response or a double monitor configuration utilizing a detector at each end of the pipe. Such solutions will be investigated and the general feasibility of source localization with temporal data will be researched further. The second method involves taking data from one end of the pipe and then moving the detector to the other end of the same pipe and taking data again. By looking at the responses at the two ends one should be able to tell at what section of the pipe the source is located. This particular method will localize contamination but it might pose practical drawbacks in real life. For example in some D&D scenarios the pipe might be accessible from only one end. Therefore we believe that further exploration into both methods of localization is quite necessary at this point. Also differences in the response from different ends of the pipes will vanish when we start using higher airflow with venetian blind designs for grids. Thus depending on the application we might want to retain a single-grid design for pipe monitors along with the pipe monitors with venetian blind grids.

CONCLUSIONS

In conclusion, with further development we believe that we can provide nondestructive, in situ and real time measurements for contamination within the pipes. A fully developed pipe monitor technology should be capable of characterizing pipes ranging in diameter from 1 cm to very large pipes. In addition to the straight pipe, we should be able to give reliable information on curved pipes. Moreover, localization of the contamination appears possible to some degree. It is important to note that the ion collection technique using airflow within pipes is the simplest case for detecting contamination within more general internal volumes. Other possible volumes include glove boxes (2) and truck trailers filled with cargo. (In fact, we are studying assay of glove boxes.) In addition the pipe monitors shown in Fig. 1 have been used to assay weapon cartridges (3).

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27-3

A PORTABLE ALPHA DETECTOR FOR FAST, LOW-LEVEL, REAL-TIME MONITORING

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ABSTRACT

We have developed a portable Medium Object Monitor (MOM) that is based on Long-Range Alpha Detection (LRAD) technology at Los Alamos National Laboratory. This monitor is

a very fast, mobile, sensitive alpha detector. We have begun a series of laboratory and field tests with the monitor. We will present data detailing the monitoring of contaminated objects in a laboratory setting, along with some comparative results from traditional monitoring techniques. We will also present data derived from the monitoring of "clean" material, stressing the expediency of an LRAD-based detector.

INTRODUCTION

Introduction to LRAD-Based Technology

At Los Alamos National Laboratory (LANL), the Environmental Applications (EA) section has been developing an alpha particle detection system (Long-Range Alpha Detection, LRAD) based upon ionization detection. It is well known that the mean energy expended in a gas per ion pair formed is approximately 35 eV/ion pair (1). As such, a typical 5-MeV alpha particle will produce about 140,000 ion pairs per disintegration. We have found that if these ions are transported onto a charged grid the current produced is proportional to activity (2).

An LRAD-based detection system has many advantages over traditional alpha monitoring methods. The short range of the alpha particle implies that all of its energy produces local ionization. If the ionization is contained within an enclosed area, then it becomes easy to transport the ions onto a detection grid.

Amplification of one alpha particle to 140,000 ions provides numerous benefits. Ions are much longer lived than alpha particles. One experiment determined an ion cloud half-life to be about 8 s when the ions are being propagated down a 3.5-in. diameter pipe at 180 ft/min (3). Only a fraction of the ions will recombine in air or onto a surface. The remaining ions allow effective monitoring on convoluted objects or interior surfaces such as beakers, nuclear instrumentation modules, or pipes. The large number of ions ensures that the signal responds rapidly to activity. Once an object has been placed into the detection chamber, and it has been sealed, only a few seconds are required to clear the chamber of ambient ionization and observe the signal rise (if there is a signal to observe) over background. Alternatively, if the object being monitored is not contaminated with alpha activity, only a few seconds are required for the signal to return to its background level. The automated nature of LRAD-based systems ensures consistent results, eliminating the variability of hand monitoring.

Ion Collection Methods

Ion collection to the charged grid can be facilitated two ways. With an electric field, the ions can be electrostatically accelerated onto a grid. Ions can also be swept onto the grid via airflow.

Electrostatic Monitors

An electrostatic detector is typically a five-sided rectangular or square box with a shallow profile. The open face can be placed over the surface of interest with the sides of the detector forming a seal to the surface. The charged detection grid, usually held at 300 VDC, is about 4 in. away from the grounded surface creating an electric field from the ground to the grid. The charged grid is mounted to the detector with Teflon or lexan standoffs to reduce leakage current. Electrostatic monitors are best used for monitoring two-dimensional media, such as the earth, floors, walls, or ceilings. LRAD-based detectors are very good at providing accurate activity levels when monitoring flat sources of activity. This has been proven in many field monitoring demonstrations at Fernald, Sandia, and Los Alamos monitoring surfaces such as concrete, grass fields, laboratory flooring, dirt lots, and New Mexico desert. Electrostatic monitors are the most efficient monitors of an LRAD design. A typical electrostatic monitor will detect 75% to 95% of the ions that are liberated into the electric field (4). It is thought that the static nature of the electrostatic monitor makes it ill suited for monitoring objects that have complex convolutions.

Airflow Monitors

Airflow monitors use airflow to sweep the ions onto a charged grid. An airflow monitor has, on the intake end, a fan manifold and two filtration systems: one for particulates and one for ions. The fans force air through the system, creating the airflow that is essential for ion transport in this type of detector. The particulate filter screens out dust, dirt, and other airborne debris that could contaminate the chamber interior or act as a surface upon which ions could plate out. The electrostatic filter is a charged grid between two grounded screens that filters out ambient ionization from the air. The air flowing into the detection chamber should be relatively free from ions and particulates. The detection grid, on

the outtake end of the detector, is usually held at 45 or 90 V and segregated from the detector with Teflon or lexan guarded stand-offs. The grid is sandwiched between two grounded screens. Ions are swept out of the contaminated media by the air stream and onto the detection grid. Airflow detectors are useful for objects and other convoluted items because the airflow can sweep the ions out of the structure. Although sensitive, airflow monitors are not effective assay devices; the non-uniform distribution of activity on three-dimensional objects makes it very difficult to precisely calibrate the detectors. For example, if a cube of metal were being monitored and the contamination were on the upwind face of the cube, many of the ions would be swept into the block and recombine. If the contamination were on the top of the block, all of the ions would be liberated into the airstream immediately. If each scenario had identical activities, the second scenario would yield a higher current flow on the detection grid. Airflow detectors are not as efficient as the electrostatic designs, able to optimally detect 40% to 50% of the liberated ions.

Combination Electrostatic/Airflow

Another type of detector is a combination airflow/electrostatic detector. These monitors are identical to airflow monitors, except the electric field of the detection grid extends into the detector system to assist in the ion transport, whereas airflow monitors rely only upon the airflow to transport the ions. This type of detector could offer improved efficiency over the standard airflow designs, because of the additional advantage of the electric field. These detectors are ideal survey devices, able to quickly monitor objects of many shapes and sizes. They are also very sensitive, able to quickly (15-20 s) detect source activities of less than 100 disintegrations per minute (dpm). However, these detectors have a disadvantage because it is more difficult to precisely assay objects for precise activity levels.

Intent

This paper presents data from laboratory experiments using a variety of convoluted objects with an imbedded source. We compare the response efficiency of an electrostatic, airflow, and combination detector. Where applicable, we compare these with a traditional hand-held probe. We also present data obtained while monitoring noncontaminated objects, focusing on expediency and accuracy.

While laboratory tests are important, the real test of any radiation detector is how well it works in the field under uncontrolled conditions. We have had the opportunity to monitor several objects, including a contaminated sweater. We discuss the monitoring of this sweater.

PROCEDURE

Detectors Used

The laboratory experiments used three detectors. We used an electrostatic monitor, an airflow monitor, and a combination detector, all described above. The electrostatic monitor is a floor monitor mounted on a hand cart; the detector has a footprint of 50 cm x 50 cm, or 2500 cm2. The grid was charged with 300 VDC, and the detection grid was about 4 in. from the floor. The electrometer was a Kiethley 617; our data acquisition system provided current levels with full statistical accompaniment. The airflow detector was a commercially available Eberline LRAD-1. The LRAD-1 is a rectangular detector with a sample chamber volume of about 76 L. The LRAD-1 only provides an activity level in dpm, with no statistical information. The quoted activity levels have a confidence level of 95% (5). The combination detector is the Portable Object Monitor (POM), a prototype portable detector designed and constructed for the Department of Energy. It has a cylindrical detection chamber that has a volume of about 2.2 L. The electrometer is a LANL-developed current-to-voltage proportional electrometer. The POM used the same data acquisition system as the electrostatic detector.

Experimental Hardware

We placed an 1100 dpm 239Pu source into three convoluted objects: an electronics module, a square metal tube, and an aluminum pig with and without a cap on it, viewed in Fig. 1. The aluminum pig is a block of aluminum with a small (1.3-cm diameter) hole bored through its length with a larger (2.5 cm) hole drilled into the top to provide access for the source. The pig is 9.8 cm long, 6.4 cm high, and 3.8 cm deep, with about 164.2 cm2 of area or an effective contamination level of 382 dpm/100 cm2. The cap is a small copper plug that fits over the source access hole. The square aluminum pipe is 18.1 cm long and 2.5 cm square. It has an internal area of 181 cm2 and an external area of about 181 cm2 for a total of about 360 cm2. The

effective activity is about 300 dpm/100 cm2. We placed the source into the middle of the tube. The electronics module is a box 10.8 cm long, 6.4 cm high, and 4.1 cm deep for a surface area of about 279.3 cm2 and an activity of 197 dpm/100 cm2 with a screw-on lid and perforated aluminum grids on each end. The source was placed into the approximate center of the electronic module. Assuming that the "contamination" were fixed transuranic material, the release limit would be 300 dpm/100 cm2. The "activity" on the pig exceeds that limit. The activity on the electronics module and the aluminum tube is below that limit. The pig, with the cap on, has the most constrained geometry, being the most difficult to monitor. In each case, when the object was placed into an airflow-style detector it was placed so that the airflow was perpendicular to the open end of the object to facilitate the most efficient ion sweeping.

Calibration

Each detector was calibrated using a 239Pu National Institute of Standards and Technology traceable source set. The LRAD-based detectors have a source response that is linear with activity, as such, the following calibration coefficients are optimum:

Electrostatic Activity = .180 dpm/fA x (current)
Airflow Activity = N/A
Combination Activity = .109 dpm/fA x (current)

Therefore we are able to predict current for a given activity or the activity for a given current. Based upon the listed calibration constants, we know that an 1100-dpm source should produce the following results for the following detectors:

Electrostatic Current = 198 fA
Airflow Current = 1100 dpm
Combination Current = 120 fA

Results

The first object that we tested was the pig without the cap on it; results are shown in Fig. 2.

This configuration allowed ions to drift out of the top of the pig because the copper plug did not cover the source access hole. This configuration also proved to be the most balanced, with each detector able to monitor about 25%-30% of the optimum signal. This was the only configuration for which a handheld probe detected any activity.

The configuration in Fig. 3 offered the tightest geometry, and the most "contamination," with an activity level of about 380 dpm/100 cm2. The hole through which the ions traveled was 1.3 cm in diameter. It is important to note that the range of an alpha particle at this altitude (6600 ft) is greater than 3.5 cm. Because of Bragg peaking, most of the alpha energy was embedded in the wall. Even with these handicaps the two airflow detectors measured between 10% and 15% of the optimum signal, an increase large enough to have warranted further investigation if this reading had taken place in the field.

The airflow monitor and the combination monitor work about equally well for the metal tube case shown in Fig. 4. The airflow helps facilitate ion removal out of the tube so that the ions could be transported to the ion detection grid. In each case, the signal was evident within seconds. The source was placed in the middle of the tube and with the 9-cm drift space between the source and the detector the ions recombined in air or with the wall of the tube before reaching the detector. The electrostatic detector did not monitor any ions. This configuration had a surface activity of 300 dpm/100 cm2 at the release level, and both airflow monitors detected the activity.

The electronics module is large enough for the alpha particles to completely ionize, however, many of the ions recombine on the interior of the electronics module as they exit through the screen. The results are shown in Fig. 5. This object had an activity of 197 dpm/100 cm2 or about 66% of the release limit. Both airflow monitors detected the activity. The combination monitor measured a signal rise of about 40 fA over background. This detector normally has a background level of about 12 fA. A rise of 40 fA would be an acutely noticeable signal increase that would be noticed within 8 to 10 s. To compare, monitoring the exterior of this item with a hand probe would take at least 4 to 5 s per side, or 25-30 s total, and there would still be no information concerning the interior of the module. The LRAD-based detectors were able to detect a point source on the interior of this item whose activity was below the release level.

MONITORING UNCONTAMINATED ITEMS

We are in the process of monitoring potentially contaminated items at the Chemical Metallurgy Research (CMR) facility at LANL. The first objects that we monitored were a pile of scrap metal and other debris that had been cleared for release. Although we found no contaminated items in the pile, we did obtain valuable data concerning the speed of item monitoring using LRAD-based detectors. Using the POM, we monitored 70 different items in 7 batches of 10. Our primary goal was verification of noncontamination, and our secondary consideration was fast, timely monitoring of the items. Some of the different items/materials we monitored were copper piping, metal scraps, wire pieces, carbon rods, cardboard boxes, phenolic, brass scraps, Plexiglas, graphite tubing, hunks of concrete, and various aluminum pieces. None of the items we monitored had any detectable activity. We found that we could monitor the metal pieces at a rate of one item every 10 or 15 s. The more nonconductive items, such as the cardboard or Plexiglas, sometimes took longer to monitor because the static charge took a while to discharge and come to equilibrium inside the detector. One person both monitoring and documenting was comfortably monitoring each batch of 10 in about 30 minutes. We determined that two people could consistently monitor and document 10 items in under 10 minutes (6).

SWEATER MONITORING

At the CMR building we monitored a sweater that had become contaminated with an alpha point source of about 200-250 dpm, as seen by a hand-held probe.

LRAD detectors have had difficulty detecting alpha activity when the activity is located on non-conductive surfaces. Recently LRAD researchers have begun experimenting with "ion flooding" techniques. This involves saturating the object of interest with ions to neutralize the static charge that is created on the nonconductive object. Once neutralized, the object will not attract any of the ionization created by the alpha activity. We used three Static Masters, a static removal product, each containing 500 mCi of 210Po. Polonium 210 is an alpha emitter that decays to stable 208Pb. Taking decay into account, the three Static Masters had a combined activity of about 4.2 x 106 Bq. We used these as our ion saturation devices.

Our first monitoring attempt on the sweater proved unsuccessful. A radiation technician remonitored the sweater with the hand probe and found the contaminated spot to be on the front of the sweater near the bottom fringe area. It was determined that we had covered up that area on the earlier monitoring attempts. We decided to remonitor the sweater in the detector, exposing the contaminated area. We took three measurements of the sweater in the detector: two measurements using the ion saturation technique and one measurement without using the ion saturation technique. The measurement that did not employ the ion saturation yielded a no contamination response. The two measurements that did use ion saturation yielded measurements of 190 dpm and 252 dpm. These readings corresponded to the measurements of the hand probe (7).

CONCLUSIONS

The airflow monitors have a unique advantage over any other type of detector available today. If there is an airflow, these detectors have the ability to detect contamination in a three-dimensional configuration. The open top of the pig without the cap provided an easy exit point for ions and alphas to drift out of the object. Once the ions or alphas were liberated they could be detected with both the electrostatic monitor and a traditional hand probe. Each of the monitors detected about 25% -30% of the optimum signal, and this was the only configuration in which the hand probe was able to detect any activity. The other three graphs show that the electrostatic monitor did not detect much, if any, of the ionization. This was because the last three configurations required the ions to be swept out of the medium to be detected. A pure electrostatic monitor cannot do this, the air flow monitors can.

LRAD-based detector systems are proving that they possess capabilities that, if utilized properly, make them a valuable waste management tool. These monitors are automated, sensitive, and fast. The laboratory tests begin to prove that detectors of this design can detect alpha activity that could not otherwise be detected using conventional methods. Often, when confronted with objects that may be contaminated or if there are numerous items, it becomes easier and cheaper to just dispose of the material as if contaminated. LRAD-type detectors provide an expedient and sensitive method for monitoring these types of items, ultimately saving landfill space,

improving labor costs, saving natural resources, and improving site accountability.
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27-4
A REAL-TIME ALPHA MONITORING SYSTEM FOR RADIOACTIVE LIQUID WASTE
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ABSTRACT
A real-time monitor for radioactive liquid waste is being developed at Los Alamos National Laboratory (LANL). This detector system is designed to monitor for large changes in alpha activity in a liquid waste stream. The detector is unique in that it monitors the ionization created by alpha interactions with air, using long-range alpha detector (LRAD) technology (1). This is a non-intrusive method that can be used in various complicated geometries. This technique only allows for the monitoring of the surface of the liquid, however, preliminary tests indicate that the technique should allow for real-time, sensitive monitoring of a well-mixed liquid waste stream. This technique will require further research and development, but the final system should be an inexpensive detector system for segregating and monitoring radioactive liquids at national laboratories, processing facilities, and nuclear power plants.
INTRODUCTION
A prototype radioactive liquid waste monitor is being developed for the group CST-13 at Los Alamos National Laboratory. This monitor will be used in the radioactive liquid waste treatment facility (RLWTF) so that the operators know when there is a large change in the activity of the influent waste stream. These spikes in activity will be 10 to 100 times above the average influent activity. The average influent typically has an activity of tens of nano-Curies. Measurements were made in 1994, and the detector was able to monitor liquid below the 100pCi/L level. Current regulations mandate that both influent and effluent be monitored to determine the overall effectiveness of the treatment process. Traditional methods require up to a day and a half for analysis. This monitor will be able to monitor the gross alpha activity from the surface of a liquid stream in real time, giving the operators a chance to better segregate the liquid as well. This technique will save both time and money and may eventually be applied to effluent monitoring before discharge into the environment.
DESIGN
Traditional alpha detectors monitor the alpha particles directly, requiring that they be close to the object being monitored. It is difficult to design a detector to do this because of the changing levels of the liquid stream, humidity, and other engineering concerns. The radioactive liquid waste monitor being developed by LANL is not subject to these constraints because it monitors the airborne ionization created by the surface alpha particles. Tests at LANL have shown that this

ionization can be transported via an electrostatic field or by airflow up to distances in excess of several meters, depending on the application. Each 5-MeV alpha particle can create approximately 150,000 ion pairs, therefore this technology has proven to be highly sensitive for monitoring alpha contamination.

This prototype design is based on the electrostatic long-range alpha detector (LRAD). As shown in Fig. 1, the influent is placed in a stainless steel sink. The top portion is removable and consists of two stainless steel signal planes and two electrometers. A current, at the fA level, is measured on each signal plane. These signals are then converted to mV and output to a computer for storage and analysis. The bottom plane measures the ionization of the volume beneath it, which includes the background ionization and the ionization from the sample. The top plate measures only the background ionization in the volume above it. By taking the difference of these two signals, one gets a background-compensated signal that is representative of the alpha activity from the sample surface (2). By calibrating the detector, it is possible to directly convert this signal into a measure of activity. Figure 2 shows the standard calibration using a standard set of National Institute of Standards and Technology (NIST) traceable Pu-239 sources. The source strengths range from 100 to 1100 dpm-alpha. For this prototype, the conversion for dry samples is 1 mV = 58.98 dpm. This detector is clearly able to discriminate dry samples with less than 100 dpm-alpha activity. Initial tests using liquid samples show the prototype to be sensitive to samples of 100 pCi/L. The efficiency of such an electrostatic design is roughly 40%.

Fig. 1. Schematic drawing of the LRAD radioactive liquid waste monitor prototype. The top volume only measures background ionization, therefore the difference of the two signal planes gives only the contribution to the signal from the surface alpha contamination.

Fig. 2. A linear curve fit of the LRAD radioactive liquid waste monitor prototype using a NIST traceable set of Pu-239 sources. For these sources, the conversion was 1 mV = 59 dpm.

It should be noted that tests will also be made using the airflow LRAD design. In this design, a fan will be used to draw the ions to a detector grid. This design would be useful in monitoring liquid inside large sewer pipes. This type of detector is more subject to noise and has an inherently lower efficiency than the electrostatic design, however it might provide a better indication of the surface activity since it would be able to draw air from a larger volume. All the air drawn off would then be recycled back into the volume.

PRELIMINARY RESULTS

Tests have been performed at the radioactive liquid waste treatment facility at LANL using actual radioactive influent samples. These preliminary tests have shown that this prototype is able to monitor low-levels of alpha contamination in liquids. These tests have consisted of putting both radioactive samples and distilled water in the detector for lengths of time more than three days. As shown in Fig. 3, these tests show that this design is able to adequately compensate for changes in background radiation levels. In Fig. 3, the signals from both the top and bottom signal planes are shown. Both signals track each other well, and the difference of these signals is a smooth line corresponding to the activity of the sample only. Figure 4 shows the results of running with a radioactive liquid sample over a 3-day period at the RLWTF. The mean signal for this period was 23.5 1.4 mV. This prototype has shown no problems with humidity, despite the fact that a considerable amount of condensation builds up inside the detector. Additional testing has been performed that shows that this detector is also fairly immune to mechanical noise.

Fig. 3. The top detector plane reads only the background ionization. The result of taking the difference of these two signals is a smooth signal corresponding to the activity of the sample being monitored.

Fig. 4. This plot shows the response of the LRAD radioactive liquid waste monitor over a 3-day period using background compensation. The mean signal was 23.5 1.4 mV.

All of these tests will be repeated and a comparison to laboratory analysis using traditional alpha monitoring will be made. This will allow us to determine a conversion to alpha activity per unit volume. These tests, as well as testing with an airflow LRAD, will be made in February 1995, after which a new prototype will be built for inline testing at the treatment facility. A working real-time monitor for spikes in activity should be installed at the treatment facility and operational in the first half of 1995.

CONCLUSION

This detector has been shown in preliminary tests with radioactive liquid samples to be a sensitive alpha monitor. It is able to compensate for changes in background radiation and it operates in real time. Because the design of this system does not require costly parts and labor, it is inexpensive compared to other real-time alpha monitoring systems.

The preliminary tests carried out at LANL indicate that this detector can be used to monitor for large changes in alpha activity in liquid streams. Further funding will allow further development of this system so that it can be used as a sensitive assay device as well. This type of detector can be used for monitoring before, during, and after processing liquid waste. It can prove valuable to treatment facilities by monitoring processed liquids before release into the environment, thus ensuring regulatory compliance. It could also prove beneficial to the commercial power industry. We are currently seeking an industrial partner to assist in carrying out this development.

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27-6

INNOVATIONS IN SOIL WASHING AT THE DEPARTMENT OF ENERGY HANFORD SITE

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ABSTRACT

This paper presents the background, procedures, and results of a soil washing treatability test at the 300-FF Operable Unit (OU) of the Department of Energy (DOE) Hanford Site, Richland, Washington. The treatability test was designed to demonstrate on a pilot scale that physical/chemical treatment techniques could provide very significant volume reductions in the remediation of target soils contaminated with uranium and cobalt-60.

BACKGROUND

Physical separation processes are used extensively in the mining and mineral processing industries to assist in the recovery of valuable constituents. These physical processes have been demonstrated by the United States Environmental Protection Agency (EPA) Superfund Innovative Technology Evaluation Program for hazardous waste remediation and the Defense Nuclear Agency for the cleanup of radiologically contaminated sands.

Many physical separation systems are commercially available. A schematic of the physical/chemical soil washing system used at Hanford is shown in Fig. 1. The system consists of a feed hopper, a vibrating screen, high pressure spray bars, a hydrocyclone for the separation of sands and fines, a dewatering screen, a sludge settling tank, and a process water tank.

Fig. 1. Soil washing process flow diagram.

For the pilot study, the DOE and the Westinghouse Hanford Company (WHC) decided to conduct specific investigations into processes that could be utilized to perform required separations. For the study, Alternative Remedial Technologies, Inc. (ART) was competitively selected. The study area was the North Process Pond in the 300-FF OU of the Hanford Site. The 300-FF OU is a CERCLA NPL site and the North Pond is within the 300-FF OU.

The North Pond was constructed in 1948 to receive process wastewaters from fuel fabrication operations, cooling water blowdown, steam condensate, and a wide variety of waste liquids from laboratory drains throughout the 300-FF OU. The ponds were deactivated in 1975 and currently do not contain any liquids. The soils in the 300-FF OU of the Hanford Site are predominantly coarse granitic sands and gravels, with less than 5% silts and clays, and are therefore well suited for physical separation processes. From earlier work in the area, it appeared that the highest

concentrations of contaminants in the 300-FF OU are in the form of coatings or particulates residing on or within the soil particle size fraction less than 100 microns. Lower contaminant concentrations were found on larger soil particles mostly in the form of coatings. Additionally, the North Pond Area contained a "green" sludge material that appeared to be a copper-uranium carbonate. Both the coating/particulate and the "green" material were a feed material for the treatability test.

TREATABILITY TEST OBJECTIVES

The objectives of the treatability study were to:

1. Achieve a 90% or greater weight reduction of contaminated feed soils, while;
2. Meeting the treatment standards for the>90% clean fraction as shown in Table I; and
3. Developing practical engineering information that can be used for the remediation the North Pond Area.

TEST ORGANIZATION AND PROCEDURES

The treatability study was managed by WHC with ART acting as the prime contractor for the work. WHC provided the management team and technical support while ART provided the required equipment, personnel, and operating expertise to conduct the test, develop the required data, and report on the findings. WHC provided the required interface with DOE and the State of Washington Department of Ecology and Environmental Protection Agency.

Because the test was intended to provide scale-up information, it was necessary to have a continuous process facility in the North Pond as opposed to a bench-scale unit that could be operated on a batch basis in the laboratory. ART, working with its parent company Heidemij of The Netherlands, mobilized the 10 ton per hour (tph) soil washing pilot plant from Holland to the Hanford site. The plant was staged near the North Pond work area, inspected by WHC from a safety and health physics standpoint, and then erected in the work area.

During the period March 7, 1994 - April 1, 1994, the target soils were pre-screened, staged, and fed into the treatment plant. Four fractions, a gross oversize, a normal oversize, sand, and fines, were produced. The individual fractions were weighed and analyzed to confirm the attainment of the treatment standards. The information from this study and the methodology employed is presented in this paper.

CONCLUSION

The pilot study, believed to be the first separation-based soil treatment study, was very successful. During the study, 380 tons of contaminated soil were treated, resulting in an average volume reduction of greater than 93% while significantly improving upon the treatment standards with an average activity of 28 pCi/gm based upon Uranium-238. Based upon the work performed in this assignment, several key lessons were learned that may be helpful to other remedial contractors or complex site cognizant managers.

1. The soil matrix/contaminant relationship. The Hanford soils are extremely coarse, consisting generally of 55% oversize (greater than 2mm), 40% sand, and 5% fines (less than 0.075 mm). The contaminants encountered, principally uranium, cobalt-60, lead, and copper, are predominantly concentrated in the fines fraction. ART found that physical separation, classification, attritioning, and concentration allow a simple, proven method for attaining very high volume reductions. The ability to quantify the location, concentrations, and volume of contaminated soil is difficult, and ART believes this must be finally determined during the remedial process. In some cases visual indicators are helpful; at this site, a "green" material, consisting of a copper-uranium carbonate sludge could be identified in the field. ART also found, particularly in the case of uranium, that x-ray fluorescence (XRF) can be a very dependable field analytical tool.
2. Procedures and Planning. Working at any DOE facility can be very complicated, and thus time consuming and expensive. With proper planning and some creativity, many barriers can be overcome. Probably more than any other type of project, DOE projects demand that you be proactive as a contractor. It is essential that upon selection, focused efforts are undertaken to work very closely with the assigned DOE point of contact. Operational plans should be developed to anticipate problems, particularly in the area of sampling, analyses, product measurements, and decontamination. Pay particular attention to the Federal Acquisition Regulations (FAR) as it pertains to work stoppages, delays, and the methods whereby compensation is determined for equipment that cannot be "free released" from the work area. A very important player

in this process will be the site health physics technicians (HPTs). Generally, and certainly in the ART work at Hanford, the HPTs are always present whenever there is entry into a radiologically controlled area. During that period, the HPTs have a unilateral responsibility to control and, if necessary, stop work based upon potential exposure to, or release of, radiological contaminants.
3. Plant Description. A pilot study, by definition, consists of facilities and capabilities required to test and confirm that a particular arrangement of equipment can achieve a specified, full-scale outcome. The ART pilot plant, as described earlier, had a process throughput capacity of approximately 15 tons per hour. The plant required an on-site generator to provide power since commercial power could not be extended into the radiologically controlled area (RCA). Water for the process needed to be stored and trucked in. All of the real-time analytical equipment needed to be conveniently located, but not in the RCA. Wherever possible, try to keep supporting facilities outside the RCA where they will not require HPT control, nor will they need to be decontaminated at the completion of the project. The plant itself needs to have the flexibility to treat the anticipated feeds, while also providing the ability to add unit operations that become necessary during the performance of the project. Often, ART has found that adequate excavation, loading, and screening equipment does not exist at DOE facilities. If possible, we recommend that contractors try to provide all of the required equipment under the contract and do not rely on "site available" equipment. One particular equipment detail can become very complicated ... decontamination. It is very difficult to estimate, prior to the performance of a project, whether your particular plant will be able to be released from the site. If it can be, how do you manage the process? If it can't be released, how is the owner compensated? Of course, the ability to decontaminate will be based upon the concentrations/activity of the feeds, the type of soil, and the intricacies of the equipment itself. Many precautions can be taken on the plant even before it enters the RCA. Taping, welding, intermediate barriers, and so on, may limit component exposure even in the area. Further, it is very important to have a written, and approved decontamination survey and release plan with the HPTs. For equipment that cannot be decontaminated, the FAR will prevail on value determinations. The FAR recognizes only a documented book value for equipment, so be prepared with good records!
4. Plant Operations. The testing/operations period can easily be disrupted by weather, high winds, union problems (most DOE sites are unionized), equipment availability, and so on. Strong working relationships and a good understanding of the interests of involved parties cannot be beat! On most DOE sites, overtime labor often is not authorized. But, because of the demands of the study and the priority under which they must be performed, longer hours are often required. These additional demands require arrangements to be made for additional labor support. At the Hanford site, the management team supported the additional demands and was extremely and constructively supportive in providing the required resources. These projects, based upon the ART Hanford experience, require very close management, and a strong contractor project manager is essential.
5. Use of the Pilot Study Findings for Full-Scale Implementation. The ART treatability study was an important component of the final feasibility study for the 300-FF OU at the Hanford Site. The study was included in the final report submitted to the USEPA and the State of Washington Department of the Environment for consideration in the development of the Record of Decision (ROD). Three alternatives were submitted: excavation and removal of the contaminated soils to an "Environmental Restoration Disposal Facility" (ERDF) which is an on-site landfill currently being constructed, capping of the entire site, and soil washing. The final decision will be based upon the CERCLA selection criteria and will be highly dependent upon the accepted risk level. The ROD is expected to be issued in the first quarter 1995 and will set an extremely interesting precedent for similar DOE facilities.

27-7
BATCH TEST ANALYSES OF SORPTION OF SELECTED ACTINIDES AND FISSION/ACTIVATION PRODUCTS ON SUBSURFACE MATERIAL FROM THE SNAKE RIVER PLAIN
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ABSTRACT
Laboratory batch tests were performed to study the sorption of 60Co, 90Sr, 137Cs, 233U, 239Pu, and 241Am in basalt and sedimentary interbed from the Snake River Plain at the Idaho National Engineering Laboratory. The tests were performed with a model solution simulating regional groundwater. Sorption of the radionuclides to the solid phase was inferred from changes in aqueous phase concentrations. The data were analyzed in two ways . In the first, distribution coefficients were determined from steady-state concentrations. In the second, distribution coefficients and rate constants were determined by least squares fits of the data to two time-dependent sorption models. In one of the models sorption is approximated as a single first-order process. In the other, it is approximated as two first-order processes, one fast and one slow.
For all but 233U, sorption to basalt was greater than to sedimentary interbed. Distribution coefficients obtained were: for basalt, 60Co(30-54 mL g-1), 90Sr(9-13 mL g-1), 137Cs(38-43 mL g-1), and 233U(4-5 mL g-1); for sedimentary interbed, 60Co(1148-3912 mL g-1), 90Sr(42-63 mL g- 1), 137Cs(2228-3255 mL g-1), and 233U(3-6 mL g-1). In the initial experiments, loss of 239Pu and 241Am to container walls masked sorption by the soils. Subsequent testing with revised experimental procedures yielded 241Am distribution coefficients of 45-210 mL g-1 for basalt and 450-1100 mL g-1 for sedimentary interbed. Follow-up tests with 239Pu are currently underway. Generally, sorption was faster with sedimentary interbed than with basalt. With the exception of 60Co in basalt and 233U in interbed, both of the time dependent models produced good fits to the data.
INTRODUCTION
The development of a remediation strategy for Pit 9 at the Radioactive Waste Management Complex (RWMC) at the Idaho National Engineering Laboratory (INEL) requires reliable models of contaminant fate and transport in the hydrogeologic system of the eastern Snake River Plain subsurface. The subsurface in this region is dominated by a layered sequence of fractured volcanic rocks (primarily basalt) and sedimentary interbeds. However, the system is highly heterogeneous and complex, and there is significant uncertainty in both the hydrological and the geochemical parameters that are needed for subsurface flow and contaminant transport modeling. A combined field and laboratory program (known as the Integrated Field-Scale Aquifer Pumping and Infiltration Tests) was initiated in 1993 to improve the availability of hydrological and geochemical data for the site. To expand upon the existing base of relevant data (1-4), laboratory column tests and batch tests are being performed. Results of column tests have been reported previously (5). Presented here are preliminary results of batch tests to determine distribution coefficients and sorption rate constants.
THEORY
In the batch tests, simulated groundwater spiked with radioactivity was added to containers with either basalt or interbed material. The tests were designed to yield a detectable decrease in the aqueous phase concentration, which was measured as a function of contact time. The aqueous phase concentrations typically decreased initially and then either reached steady state or appeared to be approaching steady state level. These data were analyzed in two ways. In the first, the distribution coefficients were determined from the apparent steady-state concentrations. In the second, distribution coefficients and rate constants were determined from least squares fits of the data to two time dependent sorption models. These methods are

described below.
The equilibrium distribution coefficient, Kd, is defined as
Eq. (1)
where Cs,e is equilibrium solid phase concentration (Bq g-1) and Ca,e is equilibrium aqueous phase concentration (Bq mL-1). Under the assumption that all of the aqueous phase activity goes to the solid phase (i.e. there are no losses to the walls or the head space), solid phase concentration, Cs(t), can be inferred from aqueous concentration by
Eq. (2)
where Ca(0) is initial aqueous phase concentration, Ca(t) is aqueous phase concentration at time t, V is volume of the aqueous phase (mL), and m is solid phase mass (g). Thus, Eq. (1) can be expressed solely in terms of aqueous phase concentrations,
Eq. (3)
In one of the time dependent models sorption is approximated as a first-order process, and the rate of change of solid phase concentration is given by
Eq. (4)
where is a first-order rate constant (s-1). As above, if losses to the container can be neglected, Eq. (4) can be rewritten in terms of aqueous phase concentration and solved to yield
Eq. (5)
Presented in Fig. 1 are predictions based on Eq. (5). Aqueous phase concentration is plotted as a function of time for Kd = 0, 1, 10, and 1000 mL g-1, Ca(0) = 10 Bq mL-1, m/V = 0.033 g mL-1, and k = 0.0378 hr-1.
The second time dependent model was based on Wilczak's (6) modification of Dzombak and Morel's (7) "two-box" model in which sorption is approximated as having two independent first-order components, one rapid with rate constant l1 and one slow with rate constant l2. Wilczak's modification is basically a method for applying the model to experimental data in which equilibrium may not be reached. It includes two additional parameters, pc, the percent deviation of the measured aqueous phase concentration from equilibrium and t, the time at which pc is determined. The equation for aqueous phase concentration based on this model is
Eq. (6)
METHODS
Experiments were performed with basalt and interbed material from the INEL site. The basalt was from a boulder removed at one of the burial pits in the RWMC. The sample was crushed and sieved, and the fraction larger than 250 mm was used in the experiments. The basalt was composed primarily of the minerals pyroxene and plagioclase, and it had a rather low cation exchange capacity (6.05 meq 100 g-1) (8). Sedimentary interbed was obtained from an uncontaminated region adjacent to the RWMC. The material was composited from well cores collected at five different depths ranging from 50 to 120 mm and sieved to the size fraction smaller than 250 mm. The interbed was composed primarily of silt and sand particles with varying amounts of clay materials. It had a higher cation exchange capacity (17.39 meq 100 g-1) (8) than the basalt. The physical and chemical characteristics of the two subsurface materials are given in Table I.
TABLE I
Groundwater in the area is typically alkaline (pH8) with elevated bicarbonate concentrations (50-500 mg L-1) (9). A synthetic groundwater (SGW) was formulated which approximated the actual major ion concentrations as determined from analysis of field samples (9). The compositions of typical field samples and the synthetic groundwater are given in Table II.
TABLE II
The groundwater was spiked with the radionuclides by evaporating an aliquot from a standard and redissolving the activity in the synthetic groundwater. This solution was filtered through a 0.1 mm polysulfone filter and allowed to equilibrate overnight. The aqueous phase concentrations ranged between 1.5 and 18 Bq mL-1.
The testing procedure was adapted from ASTM method D4319 (10). Using values of distribution coefficients in the literature for similar soil, the mass of soil required to reduce the aqueous phase concentration by a factor of 10 was calculated. This ranged from 0.05 to 1.0 mg. This amount of soil (either basalt or interbed) was added to tared 15 mL polystyrene screw cap centrifuge tubes and weighed. Synthetic groundwater (5-15 mL) was added to each tube and tumbled at 10 rpm for two

consecutive 12 hour equilibration periods. At the end of each 12 hour equilibration the synthetic groundwater was discarded and replaced. The final weight of the tube was taken prior to spike addition to determine the exact amount of synthetic groundwater present.
One mL of the spiked groundwater was added to 28 centrifuge tubes containing equilibrated basalt or interbed. Aqueous phase concentration was measured in duplicate samples at the following times: 5, 10, 20 and 45 minutes, 1, 2, 4, 6, 12, 24, 50, 100, 200, and 400 hours. An additional 14 tube blanks (no solid present) were analyzed at the same times. Three solid blanks were analyzed at 1, 50, and 400 hours to check for radioactivity that might be associated with the soils. At the designated sampling time the supernatant was removed from the tube and filtered through a 0.2 mm cellulose acetate filter. One mL aliquots of the filtrate were analyzed with a liquid scintillation counter (Wallac Model 1415)
Distribution coefficients were calculated from Eq. (3) using measured concentrations at large times as approximations of equilibrium concentrations. Distribution coefficients and rate constants were determined from Equations (5) and (6) through least squares fits. In the fits to Eq. (6), pc was set at 1%.
RESULTS AND DISCUSSION
Batch test data are presented in Figs. 2 and 3 for basalt and interbed, respectively. In each figure, aqueous phase concentration is plotted as a function of contact time for 60Co, 90Sr, 137Cs, and 233U. For these radionuclides, the concentrations in the blanks remained approximately the same as the initial concentration throughout the testing period. The time dependent curve fits are also shown in Figs. 2 and 3. With the exception of 60Co and 233U in interbed, both models provided good fits to the data. Generally, sorption was more rapid in interbed than basalt. Strontium-90 sorbed relatively slowly in both basalt and interbed. A possible explanation for this is that natural strontium may have been in equilibrium prior to spike addition due to large amounts in the basalt (11). Uranium-233 exhibited an initial rapid decrease of aqueous phase activity that was followed by a rapid increase to steady state. Possible explanations for this behavior include a speciation change from a uranyl nitrate in the spike solution to a uranyl carbonate in the synthetic groundwater or a change in surface conditions of the solid material.
FIG. 2
FIG. 3
Initial tests with 239Pu and 241Am were unsuccessful due to large decreases in aqueous phase concentrations in the blanks, presumably due to loss to the walls of the tubes. Corrections for these losses could not be made because the aqueous phase concentrations in the blanks were actually lower than those in the tubes with solid present. Tests performed in glass tubes and in silanized glass tubes resulted in similar losses. Subsequent tests performed with 241Am in polycarbonate tubes yielded better results . The decrease in aqueous phase concentrations in the blanks were less than 10 percent in these tests. These subsequent tests were performed at a single contact time of 120 hours, which should have been sufficiently long for equilibrium to be reached based on the initial batch tests.
Results of the data analyses are given in Table III. Although the distribution coefficients as determined by the three different techniques showed some differences in magnitude, the trends were consistent. The distribution coefficients were ordered, from largest to smallest, as follows: 241Am>60Co/137Cs>90Sr>233U in basalt and 60Co/137Cs>241Am>90Sr>233U in interbed. Except for uranium, distribution coefficients were larger in interbed than in basalt. For 90Sr and 241Am, the difference was approximately a factor of eight; for 60Co and 137Cs, the differences were almost two orders of magnitude. For uranium, the basalt and interbed values were within about 30 percent of one another.
TABLE III
CONCLUSIONS
1. Except for 233U, distribution coefficients were larger in interbed than in basalt. For 60Co, distribution coefficients were approximately equal in the two models.
2. Distribution coefficients were ordered, from largest to smallest, as follows: 241Am > 60Co/137Cs > 90Sr > 233U in basalt and 60Co/137Cs > 241Am > 90Sr > 233U in interbed.
3. With the exception of 60Co, sorption was more rapid in interbed than in basalt.

In interbed steady state was reached in less than 100 hours. In basalt, steady state was reached in less than 400 hours.
4. With the exception of 60Co and 233U in interbed, both of the time dependent models provided good fits to the data.

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27-8

PHYSICAL TREATMENT OF CESIUM CONTAMINATED SOILS

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ABSTRACT
Treatability studies were conducted at the Idaho National Engineering Laboratory (INEL) to evaluate the most promising means of volume reduction for cesium contaminated soils. Earlier testing for remediation of an infiltration pond at the INEL led to the conclusion that extractive soil washing could not meet the cleanup criteria designated in the Record of Decision for the pond sediments. Preliminary data also indicated that even gravels and cobbles were contaminated to levels greater than the risk-based limits. Due to concerns over generation of secondary wastes by chemical decontamination methods, the treatability studies reported here were limited to primarily physical treatment methods including screening, flotation, low-energy attrition (tumbling), and attrition scrubbing. Brine extraction of fines was also evaluated to determine if silts and clays could be decontaminated.
Four soil samples were chosen representing sites with significant areal extent and various levels of contamination. These studies provided the data to evaluate segregation of cesium contaminated particles by density, particle size, spallation of surface coatings, and simple ion exchange. Partitioning to organic detritus such as grasses and sage was evaluated by flotation: both skimming off large material that floats on water, and froth flotation using hydrophobic agents and injected air to remove finer material while minimizing carryover of inorganic silt and clay particles. Inorganic material greater than 4 mesh was separated by screening and tumbled to cause some surface abrasion to determine if removal of lightly adherent

surface deposits could yield gravels and cobbles suitable for return to the excavation. Sands were scrubbed by high-energy attrition to remove surface coatings of transition metal oxides which have significant capacity for binding ionic contaminants. Finally, material smaller than 100 mesh were contacted with a potassium nitrate brine to evaluate the potential for cesium removal by simple ionic displacement.

INTRODUCTION

A large volume of radionuclide-contaminated soil exists at the Idaho National Engineering Laboratory (INEL). After four decades of reactor research, fuel reprocessing and waste management activities, surficial soil contaminants include many fission products, heavy metals, and some organics. The treatability studies reported here were focused on cesium-137 because risk assessment calculations indicate this isotope is the primary contributor to future cancer risks for residential use of the lands included within the INEL (1).

Studies done in 1992-93 to evaluate chemical extraction for decontamination of sediments in an infiltration basin concluded that the criteria specified in the Record of Decision for that site could not be met (2). Estimates based on the data from those studies indicated that even if the extent of decontamination was acceptable, the chemical handling hazards and the amount of secondary wastes generated would make chemical treatment a questionable choice. Additionally, the data indicated significant contamination in the gravel and cobble-sized material which would require further treatment prior to free release for residential use. Therefore, treatability studies were designed to evaluate physical processes applicable to all soil size fractions for reducing the volume of radionuclide-contaminated soil at the INEL.

INITIAL CHARACTERIZATION

Surface soil samples were collected from locations where increased levels of radioactivity were known to be present from past releases. The four selected contaminated samples provided a considerable range in observed radionuclide distribution to give sufficient representation of conditions expected to be encountered during operation of a full-scale system. In addition, a non-radioactive (below background) sample was used as a control sample. Surface soils ranging in depth from 0 to 0.3 m were sampled from each of the sites described below.

1. Sample CPP-03 was collected from a contaminated equipment lay-down area of approximately 75,000 ft2. The average 137Cs activity of the CPP-03 sample was 213 pCi/g.
2. Sample CPP-22 was collected from a site of approximately 130,000 ft2 contaminated by an air release of solid particles. The average 137Cs activity of the CPP-22 soil sample was 66.8 pCi/g.
3. The West SL-1 sample was collected from a site with an area greater than 150,000 ft2 contaminated by wind blown particles. The average 137Cs activity of the West SL-1 sample was 224 pCi/g.
4. The SL-1 Soil Box sample was collected from a more contaminated area of the same site as the West SL-1 sample. The SL-1 Soil Box sample had an average 137Cs activity of 1,855 pCi/g.
5. The CPP-1604 control sample was collected from an uncontaminated site on the northwest side of the Idaho Chemical Processing Plant (ICPP), in the lawn area near an office building. Field screening was performed at collection time and indicated that the sample showed no radioactive contamination above background.

Duplicate samples were wet sieved into +4, -4+10, -10+40, -40+100, -100+200, -200+400, and -400 mesh (fines) size fractions and the cesium distribution was measured by gamma spectroscopy. One sample was wet sieved in its as-received condition, and the other was pre-treated using a vigorous washing procedure supplied by the Environmental Protection Agency (EPA). This procedure involves shaking a slurry of soil which induces a mild abrasion to enhance the separation of fine particulate from the more coarse grains. Data for all samples showed the expected 10-100 fold increase in activity over the range of decreasing particle size, though not all samples showed a monotonic increase, with maximum activity levels in mid-range sand fractions. No significant difference was observed with the agitated slurry pretreatment versus simple wet sieving. This data confirmed the need for treatment of most or all of the soil fractions to achieve a significant reduction in volumes to be managed.

EVALUATION CRITERIA

Acceptable residual levels of radioactivity have not yet been codified or negotiated in a consent order. However, estimates of additional cancer risk for residential use of lands within the INEL indicate approximately 2.4 pCi of 137Cs per gram of soil would result in one additional cancer in a population of 10,000 (1). For scoping purposes, the decontamination effectiveness of all treatments was evaluated versus a residual level of 10 pCi/g, and weight percent (wt%) of treated soil meeting this criteria was reported. No other criteria such as cost or implementability were evaluated.

TREATMENT

The treatability studies were conducted at the bench scale using initial samples of one kilogram split from five gallon sample buckets. The random splitting provided a rough homogenization of the entire five gallon sample. This step is shown in box 1 at the top of Fig. 1. Major steps in the treatment train are described below:

Primary Separation

The sample material was subjected to a rough flotation (box 2) to remove low density organic matter such as grasses and sage that might interfere with sieving and physical treatment of individual particle size fractions. The primarily inorganic soil matrix was then sieved at 4 mesh (box 4) to separate a +4 mesh fraction for treatment by tumbling, from a -4 mesh fraction to be floated using chemical additives and air injection to remove remaining fine organic material (box 10). The grasses and debris were rinsed on a 100-mesh screen (box 9) to wash off fine organic and inorganic material entrained in the rough flotation. The rinsate was then added to the -4 mesh fraction prior to air flotation.

The combined -4 mesh fine material was treated in a Denver Equipment D-12 froth flotation unit using 302 pine oil to aid in attracting air bubbles to the organic matter. Low density fines removed in the final flotation were added to the organic fraction (box 11) for radiological determination. These partitioning steps were conducted for triplicate aliquots from each of the five samples described above. The products from the primary separation steps were: 1) an organic fraction which could be analyzed to determine how much cesium could be removed by simple density separation, 2) an essentially inorganic +4 mesh fraction which could be processed by low-energy attrition (tumbling), and 3) an essentially inorganic -4 mesh fraction which could be homogenized and attrition scrubbed.

Low Energy Attrition

The low-energy attrition (tumbling) treatment of the +4 mesh material (box 6) was to evaluate the benefit derived from partial surface spalling of the gravel and cobbles. This material is too large for an attrition mill containing water to impart significant energy to it, but tumbling could conceptually be accomplished with extended contact time in a rotating screen (trommel) or on a vibrating screen. With adequate contact time, the autogenous grinding caused by particle collisions could potentially remove enough contamination from low-surface-area material to meet release criteria. The +4 mesh fraction was tumbled air dry at 24 rpm for 4 hours. These conditions were not optimized in any way. The experiment was only meant to fracture a few weight percent of the material to determine if significant decontamination could be accomplished without generating an unacceptable quantity of contaminated fines. As a result of the scrubbing of the surfaces, a mixture of +4 mesh and finer material was produced. The tumbled product was split on a 4-mesh screen (box 7) to determine the decontamination effectiveness for the +4 mesh fraction, and the amount of fines (-4 mesh) produced by the treatment.

Homogenization

The triplicate -4 mesh inorganic samples from box 10 were then composited and homogenized using a multiaxial Turbula mixer (box 12). New triplicates were then prepared for radionuclide analysis. As necessary, samples were rehomogenized until the 137Cs values of the triplicates confirmed a precision of +20%. The homogenized samples were then split using a 100-mesh screen to generate a -4+100 mesh fraction for attrition scrubbing, and a -100 mesh fraction for brine extraction.

Attrition Scrubbing

The -4+100 mesh material was then attrition scrubbed at 800-850 rpm for 20 minutes to aggressively scrub the surfaces of the sand fraction (box 15). During attrition scrubbing, particles are forced to collide in an intensely mixed zone between two opposed impellers which fractures and removes some of the particles' contaminated surface exposing the underlying core. In practice, particles as large as 10 mm (3/8") and as small as 200 mesh (76m or .003") can probably be effectively attrition

scrubbed in water, but to maximize the effectiveness of this treatment for evaluation purposes, a narrower size range was used. As a result of the scrubbing of the surfaces, a mixture of +100 mesh and some finer material is produced. The scrubbed product was split on a 100-mesh screen (box 16) to determine the decontamination effectiveness for the +100 mesh fraction, and the amount of fines (-100 mesh) produced by the treatment.

Brine Extraction

The final test in this study was to treat the silt and clay fractions which cannot be decontaminated by any practical physical processes. Cesium is not believed to form any naturally occurring insoluble salts, but is known to readily sorb at ion-exchange sites. For this test, the soil fractions passing a 100 mesh sieve were exposed to 0.1 M potassium nitrate brines to determine the extent of decontamination achievable by displacing cesium with its chemical analog, potassium. The -100 mesh fractions were weighed out in 250 gram samples and contacted with 1 L of solution for two hours in a rolled 2 L sample bottle.

RESULTS

Cesium Distribution

The initial separations typically split the samples into 20-40 wt% greater than 4 mesh, 60-75 wt% smaller than 4 mesh, and less than 1.5% organic material. Weight and activity distribution data for all samples are shown in Table I. The mass balance for this separation was very good, with no losses over 5%. The +4 mesh materials ranged from essentially zero activity for the control site, up to almost 40 pCi/g for West SL-1. Based on the 10 pCi/g goal, the material from the CPP sites could potentially be released without further treatment, but the West SL-1 samples would still require decontamination by at least 70%. The SL-1 Soil Box coarse material was measured in the nanocurie range, 100 times over the risk-based value.

The -4 mesh fractions were 10-100 times more contaminated than the +4 mesh material, which could be largely explained by surface area. Even the fraction from the control (CPP-1604) sample exceeded the 10 pCi/g level with the larger material removed. Though some of the organic material was considerably more contaminated than the total soil matrix or either inorganic fraction, removal of the small fraction had no significant impact on the total sample activity. Of particular interest is the organic material from the control site which shows significant activity (273 pCi/g) after concentration by a factor of 1000 (removal of 99.9% of the sample as inorganic). The anomalous organic material from the CPP-22 sample, makes up only 0.02% of the matrix, and is only about 3% as contaminated as the inorganic -4 mesh fraction. This datum may simply result from the extremely small amount of organic media from this sample available for measurement.

The activity balances for the four contaminated samples were 30% rather than the established goal of 20%. This result is reasonable based on the amount of mixing required to bring some of the samples to within the 20% precision envelope. The data shown for the initial activity of the as-received soil are probably most suspect because the data were from a separate split taken in parallel with the material actually processed. The only mixing conducted on the as-received material was due to the splitting process.

The anomalous data for the CPP-1604 control sample yield activity balances well in excess of 100%. This may be due to the removal of the cobble and gravel size material and the subsequent concentration of trace activity into a very small amount of fine material which is completely exposed to the surface of the gamma radiation detector. The error in the measurement of trace activity in the heterogeneous mixture again makes the initial measurement highly suspect.

Low-Energy Attrition

Low-Energy Attrition (tumbling) reduced 1-2 wt% of the initially +4 mesh samples to finer material, and, as shown by the data summarized in Table II, lowered the contamination of the gravels and cobbles by 7-25% (shown as the percent decontamination factor or %DF). Extrapolation of the data suggests tumbling might reduce contamination in the +4 mesh fraction by 20-30% (equivalent to 10-15 years of radioactive decay for cesium) while generating up to 5 wt% fines. Thus tumbling may decontaminate cobbles and gravels containing activity marginally greater than release limits, but this treatment does not appear to contribute significantly to the overall process. The mass balance for this treatment was essentially 100% and the activity balance was within about 25% for all data with the exception of the CPP-1604 "control sample." Though the initial +4 mesh fraction from CPP-1604 counted

at only 0.1 pCi/g, the tumbling created 1.2 wt% fines which measured almost 40 pCi/g. By eliminating the intrinsic problems caused by self shielding and the unusual geometry of the large gravel and cobbles through tumbling, which concentrates surface contaminants in a small fine grained sample, the activity of the fines was much more exposed to the detector surface, and the measured data was much greater than that predicted from the original heterogeneous matrix.

Attrition Scrubbing

As shown in Table III, attrition scrubbing typically generated about 2-3 wt% additional -100 mesh material while removing up to 20% of the activity on the remaining -4+100 mesh particles. The SL-1 Soil Box sample was scrubbed long enough to generate 7 wt% waste fines (24 minutes), which reduced activity by over half, but due to the high initial activity of the -4+100 mesh material, the product was still over 10,000 pCi/g. The mass balance for all samples was within 2%, and the activity balance was within about 20%. The worst activity balance was for the control sample at 123%. This may be due to the variability in the measurement of the extremely low activities (<10 pCi/g) for the +100 mesh fractions both before and after treatment. Extrapolation of the data suggests attrition scrubbing might be refined to reduce contamination in the -4+100 mesh fractions by up to a factor of 2 to 3 (equivalent to 30-48 years of radioactive decay for cesium), while reducing 10-15 wt% of the material to additional fines. Thus, soils similar to the samples from CPP-03 and CPP-22, with very low initial activities, might be partitioned to recover 20-40 wt% in the 10-20 pCi/g range by sieving, and with enough contact time, an additional 30-40% by attrition scrubbing.

Brine Extraction

Up to 17% of the activity in the fine fractions was removed by this treatment, but due to the high levels of activity found in the fine fractions initially, none were reduced to below 100 pCi/g (Table IV). The mass balances for this treatment were within 2%.

CONCLUSIONS

Simple sieving provided the most effective partitioning of low activity material from relatively high-activity material. Scalping the soils at 4 mesh separates 20-40 wt% of the soils as a gravel/cobble mixture containing only 2-10% as much activity on a mass basis as the original soils. For soils within a factor of 10 of the cleanup criteria, the total soil under control could potentially be reduced by as much as 40 wt% by sieving. Separation of organic material may remove the most highly contaminated particles, but because of the small quantity of this material in the samples tested, the overall impact on the matrix is inconsequential. Tumbling does not appear to provide significant decontamination. Data from attrition scrubbing of the mid-size fractions indicates some benefit, and may be extrapolated to an estimated decontamination by a factor of 2 to 3 while converting 10-15 wt% of these fractions to waste fines. Ion-exchange by brine extraction displaces up to 21% of the cesium contamination, but starting with the fine fraction which is the most heavily contaminated, the decontamination is not significant when compared to risk-based residual contamination levels. Thus, for low activity soils with contamination within a factor of 10 of the risk-based value of 10 pCi/g (137Cs), a total of potentially 50-70 wt% of the soil could be salvaged by a combination of sieving (scalping the lowest activity fraction) and attrition scrubbing (of the mid-activity fraction), with the remainder requiring additional management. For more highly contaminated soils, treatment may be limited to recovery of some fraction of low surface area material by scalping at a size greater than 4 mesh.

RECOMMENDATIONS

Although the risk-based treatment criterion (10 pCi/g for 137Cs) was generally not achieved, sieving and attrition scrubbing are potentially useful treatment methods for low activity soils. If they are to be employed, sieving and attrition scrubbing will require additional consideration to evaluate cost-effectiveness of volume reduction versus mitigating factors such as projected costs for alternative forms of residuals management and worker risks due to activities such as excavation, additional handling, etc. Flotation of organic material, low-energy tumbling, and brine extraction can probably be eliminated from further consideration based on process complexity, ineffectiveness, and waste generation.

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URANIUM PARTITIONING UNDER ACIDIC CONDITIONS IN A SANDY SOIL AQUIFER

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ABSTRACT
The partitioning of uranium in an aquifer down gradient of two large mixed waste sites was examined with respect to the solution and soil chemistry (e.g., pH, redox potential and contaminant concentration) and aqueous-phase chemical speciation. This involved generation of field-derived, batch sorption, and reactive mineral surface sorption data. Field-derived distribution coefficients for uranium at these waste sites were found to vary between 0.40 and 15,000. Based on thermodynamic speciation modeling and a comparison of field and laboratory data, gibbsite is a potential reactive mineral surface present in modified soils at the sites. Uranium partitioning data are presented from field samples and laboratory studies of background soil and the mineral surface gibbsite. Mechanistic and empirical sorption models fit to the field-derived uranium partitioning data show an improvement of over two orders of magnitude, as measured by the normalized sum of errors squared, when compared with the single Kd model used in previous risk work. Models fit to batch sorption data provided a better fit of sorbed uranium than do models fit to the field-derived data.

INTRODUCTION

For 33 years, low-activity liquid wastes from the chemical separation areas at the U. S. Department of Energy's Savannah River Site (SRS) were disposed of in unlined seepage basins. These basins were designed to allow the natural processes of evaporation and infiltration to dispose of the polluted effluent streams. These operations have resulted in groundwater in the vicinity of these basins with lowered pH values and elevated levels of metals, radionuclides and nitrate. Additionally, it is believed that acidic waste streams associated with basin operations have significantly altered the mineral surfaces on the soils downgradient of these basins through accelerated acid weathering. On November 7, 1988, discharge to the basins was terminated in accordance with requirements of the Resource Conservation and Recovery Act. Before a multi-layered cap was placed over the basins, the liquid was removed and each was filled with a gravel bed and topped with layers of calcium carbonate and blast furnace slag (1).

Attempts to select an appropriate remediation technology for groundwater at this site have been hampered by an inability to adequately predict the transport of pollutants through, and the amount of contaminants in, the subsurface environment. The models previously utilized for risk assessment employed a single literature reported or "best guess" distribution coefficient (Kd) and predict that certain contaminants should be sorbed by the soil and not substantially transported (2,3).

Groundwater monitoring of the waste site, however, shows significant groundwater transport of some contaminants not predicted to be mobile (4,5).
Kd values (i.e. ratio of contaminant concentration sorbed to concentration in the aqueous phase) are generally valid only for the mineralogical and geochemical conditions under which they were determined. The inherent heterogeneity and geochemical variations of real aquifer systems, reduce the applicability of a single Kd value to natural systems (6,7,8). Additionally, equilibrium conditions may not be established if the groundwater velocity is fast relative to the sorption rate. These variables impact the partitioning of contaminants in natural systems; consequently, field systems are usually best represented as a function of these factors rather than a single constant Kd.
The objectives of this research were to: (1) develop a more representative model of uranium sorption at this SRS waste site which accounts for pH, ionic strength, and competitive ion sorption effects; (2) compare the uranium sorption data from field-derived porewater/soil sample sets to batch sorption studies with background soils and an aluminum hydroxide reactive mineral phase; and (3) evaluate the effects of using single Kd and non-linear isotherm approaches to predict remediation effectiveness for a simple system.

METHODS

Uranium partitioning was examined in sample sets collected at the waste site using an in-situ method and in the laboratory by performing batch sorption studies on an uncontaminated background soil and a gibbsite [Al(OH)3] mineral surface.

Field-Derived Partitioning Data

Fifty-three sets of soil and porewater samples were collected under inert conditions from the saturated zone downgradient of the waste sites using an electric friction-cone penetrometer system (9). Both the soil and porewater samples in each set were collected from the same depth, at the same location. Sampling locations and depths were selected such that they spanned a range of groundwater pH (pH 3-7), and had previously shown groundwater contaminant concentrations above the analytical detection limits. The temperature, pH, redox potential, and conductivity of the porewater samples were measured immediately upon sampling.
Porewater samples were filtered through 0.45 m filters and screened for 3H using liquid scintillation counting. Cl-, F-, NO3-, PO43-, and SO42- concentrations were determined using ion chromatography. Total organic and inorganic carbon concentrations were measured using an automated carbon analyzer. The filtered porewater samples were analyzed quantitatively using inductively coupled plasma (argon) mass spectrometry (ICP-MS) for 43 isotopes representing 28 elements. Additionally, a semi-quantitative ICP-MS scan was conducted for the isotopes of Hg and the actinides (mass less than 246 AMU).
Total digestion soil extracts were prepared in triplicate by digesting approximately 200 mg of air-dried soil in 1 ml of ultra-pure aqua regia and 10 ml of ultra-pure HF heated at 100C for 3 hours in Parr acid digestion bombs. The extracts were then filtered through 0.45 m Teflon filters, and diluted to 50 ml with deionized water. A seven step sequential extraction procedure modified from Miller (10) was performed on each soil sample to selectively remove uranium sorbed to particular soil surface coatings. This procedure was performed in triplicate using 0.75 g of air-dried soil in a 50-ml HDPE Oak Ridge centrifuge tube for each individual sample. Contaminants associated with the soil residue present after the final extraction step were HF acid digested as described above for the total digestions.
Like the porewater samples, isotopic concentrations of trace contaminants in the total digestion and sequential extracts were determined using ICP-MS. 235U and 238U concentrations in each extract were converted to soil concentrations. Concentrations of individual replicates were averaged using a weighted average method (11). In this work, it is assumed that any uranium remaining in the residual phase (after the seventh sequential extraction step) was associated with the soil matrix and not able to participate in equilibrium reactions with the uranium in the aqueous phase. The sorbed uranium concentration was assumed to be equal to the difference between the concentrations in the total digestion and residual digestion analyses. This definition of sorbed uranium probably represents the maximum amount of uranium that can participate in sorption reactions (i.e., exchangeable uranium). An alternative method to account for nonexchangeable uranium is to assume that the total concentration measured in unimpacted soil is equal to the nonexchangeable fraction. This approach was not applied because soil at the site was affected by an acidic

plume and some natural uranium has presumably been leached. Uranium distribution coefficients for each sample were calculated by dividing the sorbed uranium concentration (in g kg-1) by the uranium concentration of the associated porewater sample (in g l-1).

In addition to contaminant concentration, other physical and chemical characteristics of each soil sample were also examined. The particle size distribution of each sample was determined in duplicate using the procedure presented by Gee (12). The cation exchange capacity (CEC) of each soil sample, also performed in duplicate, was measured using the barium-magnesium exchange technique described by Rhoades (13). Soil pH was determined using a procedure derived from ASTM procedure D4972-89 (14) and EPA method 9045 (15) in deionized water and 0.01 M CaCl2. Net soil charge and the point of zero salt effect (pzse) were measured by an acid/base titration of the soil in 0.0, 0.002, 0.01, 0.1 and 1.0 M CaCl2 backing electrolyte solutions (16).

Background Soil Batch Sorption Studies

Laboratory studies on the sorption of uranium to an uncontaminated soil collected from the same lithological unit upgradient of the waste site have been performed. In this study, 1.00 g of soil was allowed to react with 42 ml of 10-5 M UO2(NO3)2 solution in a 50 ml HDPE Oak Ridge tube. Carbon dioxide was controlled in these experiments using a glove bag and CO2 scrubbed air and reagents. Before adding the solution to each tube, the pH was adjusted to an initial value of between 3 and 10 using HNO3 or NaOH. Two samples and one matrix blank at each initial pH were equilibrated overnight in a shaking water bath controlled at 25 C. Previous kinetics studies have shown that this amount of time was sufficient to achieve equilibrium. Samples were then centrifuged at 10,000 rpm (16,000 g) for 10 minutes, after which the supernate was decanted and filtered through a 0.45 m cellulose nitrate membrane. After measuring the final pH, samples were acidified to 1% v/v with 70% ultra-pure HNO3. The aqueous uranium concentration of each sample was measured via laser-induced fluoroscopy of the UO22+ ion. Sorbed uranium was calculated as the difference in concentration of the matrix blank solution and the final sample solution.

Gibbsite Batch Sorption Studies

Because thermodynamic models predict that the mineral gibbsite [Al(OH)3] will precipitate from the groundwater at the waste site and, therefore, could be a reactive surface, a laboratory study to determine stability constants for uranyl ion sorption to gibbsite was performed. In this study, fresh gibbsite was prepared by precipitation from an Al(NO3)3 solution after pH adjustment with 50% w/w sodium hydroxide (17). The gibbsite was purified by dialysis to remove excess sodium nitrate salts and then lyophilized for storage as a dry powder. The gibbsite was characterized using x-ray diffraction and thermogravimetric analysis. The surface area of the mineral was determined to be 178 m2 g-1 by a N2 BET surface area measurement. Sorption experiments were similar to the above described in the background soil sorption experiments. A gibbsite to solution ratio of 1 g l-1 was used. Solutions had NaClO4 backing electrolyte concentrations of 0.0, 0.05 or 0.1 M and were pH adjusted using either HClO4 or NaOH. A 4 hour equilibrium time was used with total uranium concentrations of 10-4, 10-5 and 10-6 M.

RESULTS

The field-collected porewater samples spanned a wide range of geochemical conditions. Sample pH varied from 3.1 to 7.1, while Eh values were between +41 and +442 mV. All samples typically had high nitrate and 3H concentrations. Nitrate levels varied from 3.9 to 1600 mg l-1. 3H levels ranged from 180 to 1.12x106 Bq l-1. The major cations in the samples included Al3+, Fe3+, K+, Mg2+, and Na+. Although not specifically analyzed for, other groundwater monitoring data from the waste site show that H4SiO4 is also present at elevated levels (4,5). The major ion chemistry is consistent with the dissolution of clay minerals (e.g. kaolinite) and existing surface mineral coatings (e.g., iron (oxy)hydroxides) resulting from the addition of large amounts of nitric acid to the system. Estimates of sample ionic strength based on major cation concentrations ranged from 1x10-4 to 4x10-2 M, with a mean value of 8x10-3 M. Conductivities varied widely and were inversely correlated with pH. At low pH values, where the effects of the contaminant plume were most obvious, sample conductivities ranged between 0.05 and 0.35 S m-1. Aqueous samples with minimal impact had typical conductivities of less than 0.03 S m-1. Aqueous inorganic carbon content was found to be < 1 mg l-1 in all samples.

As a result of basin operations, elevated levels of uranium have been observed in groundwater downgradient from the seepage basins. 238U concentrations were above the detection limit in 43 of 54 porewater samples collected for this study. Concentrations of these samples varied from 0.082 g l-1 to 3.2x103 g l-1. 235U concentrations were measurable in 26 of 54 aqueous samples with values varying between 0.040 g l-1 and 14 g l-1.

Equilibrium uranium speciation was calculated over the observed range of geochemical conditions using MINTEQA2 (18). The result was very similar to the theoretical speciation of a 10-6 M U/10-2 atm CO2 system reported by Langmuir (19). Based on these calculations, the dominant uranium species in this study are UO2+, UO22+, UO2OH+, and UO2CO30. At pH values less than 5.0, UO22+ is predicted to be the dominant aqueous species, between pH 5.0 and pH 5.7, UO2OH+ is dominant, while at pH values greater than 5.7, UO2CO30 is predicted to be the major aqueous uranium species present. The charge of the aqueous phase uranium species will have a large influence on binding to soil surfaces with the neutral and negatively charged species being generally more mobile. For example, groundwater with higher pH and inorganic carbon values is predicted to have a larger percentage of mobile uranium as the neutral UO2CO30 species is formed. For nearly all samples for which Kd values are reported in this work, groundwater chemistry is in the pH-Eh range for which UO22+ is the dominate form of uranium in the aqueous phase.

73% of the collected soils were determined to be sand or loamy sand; 25% sandy loam or sandy-clay loam, with the remainder being sandy clay. The mean CEC of the soils was 9.66.3 meq kg-1. The total carbon content was determined for 27 of the soil samples and all were found to contain less than 0.05% w/w carbon. The net surface charge of all soils measured was found to be very low, less than 0.5 eq g-1, with a typical pzse of 3.8.

238U concentrations were above the detection limit in all soil samples and ranged from 490 to 19,000 g kg-1. The average soil concentration of 238U downgradient of the waste sites was 3500 2800 g kg-1 (1.20.9 pCi g-1). This compares with previously reported value of 1.050.75 pCi g-1 for unimpacted shallow sediments at the SRS (28). 235U was detected in 79 of 86 samples with a range of 4.23 to 115 g kg-1, with an average concentrations of 2712 g kg-1.

Based on the difference between uranium concentration in the total soil and that of the residual phase remaining after the sequential extraction, an estimate of the uranium concentration potentially available for exchange with the groundwater was made. The concentration of sorbed 238U in the 78 soil samples examined varied from 0 to 17,000 g kg-1, with an average value of 2700g kg-1. Expressed as a percentage of the total 238U concentration, the soil phase exchangeable fraction ranged from 0 to 98% of the total soil concentration with a mean value of 70%. In the 50 soil samples in which 235U concentrations were above detection limits in both the total digestion and residual phases, exchangeable 235U represented between 0 and 98% (average of 46%) of the total concentration. This corresponds to a concentration range of 0 to 99 g kg-1 with an average value of 14 g kg-1

The 38 238U and 16 235U samples for which the porewater, total soil, and residual soil concentrations were greater than the detection limit, had field-derived uranium distribution coefficients between 0.4 and 15,000 l kg-1. Figure 1 shows that a marked increase in the fraction of uranium sorbed occurs above pH 4.0. Differences in the fraction of uranium sorbed at this waste site can be explained primarily by changes in aqueous pH, and, presumably, the associated change in soil surface charge. The partitioning behavior can not be explained in terms of the physical properties of the soils (i.e., particle size distribution, or CEC).

Fig. 1. Fraction of Uranium Sorbed as a Function of pH from Field-Derived Data and Laboratory Experiments of U(VI) Sorption onto Gibbsite and a Background Soil.

Laboratory data on uranyl sorption to kaolinite (20) and gibbsite are very similar to the field-derived data. Iron (oxy)hydroxides are mineral surface coatings that have also been shown to control the sorption of metal ions in many natural aquatic systems (21, 22). Iron surface coatings, however, were excluded from consideration in our model because the sorption edge of uranium in the field-derived partitioning curve occurs at hydrogen ion concentrations two orders of magnitude greater than previously reported laboratory data for uranium sorption to goethite (23). Recent data on the sorption of uranyl to ferrihydrite shows a sorption edge at a pH of about 4.5 (24). This is also consistent with the field-derived sorption data and will require further evaluation. Because sorption is not well correlated with the

soil clay content or cation exchange capacity, a reactive mineral coating (e.g., gibbsite or ferrihydrite), rather than kaolinite, is thought to be the reactive phase present in the modified soils at the waste site.

MODELING

Seven types of models were fit to field-observed uranium partitioning data. Partitioning data from laboratory batch sorption experiments were fit to four models. (See Table I.) The goodness of fit of each model was evaluated by the normalized sum of error squared (NSES) for each set of model fitting parameters. For this purpose, NSES was defined as:

(Eq. 1)

where n is the number of observations and concentrations are in M.

The sorption model most commonly applied in risk assessment is the single Kd model (Eq. 2a). In this isotherm, the concentration of contaminant sorbed (S) onto the soil surface is a linear function of the contaminant concentration in solution (C). Another isotherm commonly used to describe contaminant sorption is the Langmuir isotherm (Eq. 2b). The Langmuir isotherm is a linear model at low aqueous concentrations, but accounts for the finite number of sites that exist on a surface by limiting the sorbed concentration at higher aqueous concentrations. The Freundlich isotherm (Eq. 2c) may be used to model contaminant sorption at low aqueous concentrations. It has been shown that it may be considered as the log-normal distribution of the Langmuir terms (25) and is well suited for uses in heterogeneous media.

Models fit to the field-derived data showed an improvement of over two orders of magnitude, as measured by the NSES, when compared with the single Kd (Kd = 40) model used in previous risk assessments (Table II). In fitting the Langmuir isotherm to field data, two methods were used to determine the number of available binding sites (parameter 'b' in Eq. 2b). In the first method, the maximum concentration of sites for each sample was determined based on the CEC of the sample. The second method assumed an equal concentration of sites for all samples, but allowed it to be an adjustable fitting parameter.

The fourth model examined in this work is an equilibrium model. Assuming the simplified case where a uranyl ion is bound exclusively to a negatively charged soil binding site (XO-), the reaction may be described by the equation:

(Eq. 3)

where XOUO+2 represents a soil sorbed uranyl ion. This equilibrium reaction has a stability constant (K) defined in terms of solid concentrations ([ ]) and aqueous activities ({}) as:

(Eq. 4)

This equilibrium model was implemented using the nonlinear fitting program FITEQL 3.1 (26). It was assumed that the total concentration of available binding sites (i.e. sum of [XOH],[XO-], {XOH2+] and [XOUO2+]) was equal to the CEC of each sample. All soil samples were assumed to have a bulk density of 1200 kg m-3, a porosity of 0.3, and a surface area of 1000 m2 kg-1. Based on the combined soil titration data of 17 soil samples, the average protonation (log K+) and deprotonation (log K-) equilibrium constants of the soil samples were determined to be 2.00 and -5.44, respectively. Using these constants, functions relating the fraction of sites deprotonated (XO-) and protonated (XOH2+) to -log[H+] were developed. Using these relationships, and the estimated total binding site concentration of each sample, K values for the reaction in Eq. 3 were determined for both field and laboratory uranium partitioning data.

The constant capacitance and diffuse layer electrostatic models were also fit in this work. Both of these models account for the electrostatic effects between the charged surface and ions in solution. The difference in the models is the way in which the surface potential (y, units of V) is determined. In the constant capacitance model (Eqs. 2e and 2f), the surface potential is directly related to the charge density of the surface (s, units of C m-2). This condition is applicable under conditions of constant ionic strength and low surface potential (18). The diffuse layer model (Eqs. 2e and 2g) accounts for electrostatic effects of all charged ions in solution by relating surface potential to ionic strength.

In implementing electrostatic models, the total charge on each surface was allowed to vary with aqueous pH. As in the simple equilibrium model, the concentration of protonated and deprotonated sites for each sample was assumed equal to the maximum concentration of sites, based on the CEC, multiplied by the -log[H+] dependent site

protonation and deprotonation functions developed in the simple equilibrium model. The total charge density was then calculated by the equation
(Eq. 5)
where D is a conversion factor relating charge density in C m-2 to site concentration in molarity. Unlike the simple equilibrium model where uranium was assumed to only bind to the negatively charged sites, sorption was allowed on all sites in the electrostatic models.
In applying the constant capacitance model, Hayes (27) reports that fitting with a capacitance (C1) outside of the range of 0.1 to 2.0 F m-2 is an indication the constant capacitance model is inappropriate. In modeling the batch uranium sorption data, a capacitance of 2.0 provided the best fit of the data. The constant capacitance model failed to provide an adequate fit of field-observed uranium partitioning data. The NSES continued to decrease as capacitance increased well beyond a capacitance of 2.0 F m-2. This may be due to the varying ionic strength conditions present under natural conditions.
Fits of the diffuse layer model to uranium partitioning data did not provide as good of a prediction of sorbed uranium concentration as the constant capacitance model. This may be because the diffuse layer model only has three adjustable parameters (K, K+, K-) , whereas the constant capacitance model had four adjustable parameters (K, K+, K-, C1).
In addition to applying mechanistic models to the data, an empirical rise function was fit to the fraction sorbed versus aqueous pH of the field and soil batch study data using Eq. 2h. While this model results in the smallest NSES, it is purely empirical and, hence, can not be expected to fit the uranium sorption characteristics at other waste sites or under other experimental conditions.
Fitting parameters from sorption models were implemented in MINTEQA2 (18). Using typical field aqueous conditions, the fraction of uranium sorbed was predicted for varying pH and Eh conditions. Both the effects of uranyl ion adsorption and precipitation were accounted for in this modeling and reported as the fraction sorbed. Not surprisingly, the most empirical models (i.e. Kd, Langmuir, and Freundlich) were unable to account for the changes in the fraction of uranium sorbed with changing pH. The model that best represented field and laboratory partitioning data in these MINTEQA2 runs was the diffuse layer model. The thermodynamic modeling results using this model predicted that the fraction of uranium sorbed varied not only with pH, but also with Eh. Under slightly reducing conditions (Eh=-100), the sorption edge was at pH 2.9 and increases under more oxidizing conditions (+400 mV) to a pH of about 4.2. The constant capacitance model produced sorption edges at lower pH values, from 2.2 to 3.0, with no discernible trend with Eh.
To illustrate the potential impacts of the choice of sorption model and fitting parameters on the remediation of an actual waste site, a simple system was modeled using a single Kd model with values of 40 (a value used in previous risk assessment efforts) and 2.2 (the best-fit value from field data for the site) and the empirical rise function fit to field data. The simple system was assumed to have an initial pH of 3.0. The pH response of the system was calculated by assuming that pH 5.6 rainwater mixed with existing porewater and that one-half of the hydrogen ions were retained by the system due to soil buffering. All the aqueous-phase uranium was assumed to be removed from the system as a new porevolume of water was added to the system. From this, the fraction of remaining uranium in the system as a function of porevolume was calculated for the three models and is shown graphically in Fig. 2. Modeling of a simple system suggests that a technology such as pump-and-treat would be more effective in removing uranium than previously predicted using literature Kd values. The use of the field data best fit Kd value of 2.2, however, will likely underestimate the time required for remediation because the aqueous fraction is expected to decrease as the pH of the system returns to more neutral conditions. Therefore, a model which accounts for the pH-dependence of uranium sorption is expected to provide a much better estimate of remedial effectiveness.
Fig. 2. Estimation of the Fraction of Uranium Remaining as a Function of Porevolume.
CONCLUSIONS AND IMPLICATIONS
The partitioning of a contaminant between the aqueous and solid phases in a natural system can not be described by a single linear isotherm in situations where the surface or aqueous-phase chemistry exhibit large variations. A simple method of obtaining the actual distribution of contaminants in a polluted aquifer is to analyze sets of porewater and soil samples obtained at the same location in the

aquifer, at the same time. The actual distribution coefficients of an SRS waste site determined using this method vary several orders of magnitude.
In this study, uranium sorption can be explained largely in terms of aqueous sample pH. The sorption isotherm for uranium obtained laboratory data shows a significant increase in sorption above pH 4.0. The sorption isotherm for the field data has a sorption edge at a lower pH value than either the 10-6 M uranium on gibbsite or the 10-5 M uranium on background soils. This may be because the sorbed uranium phase, as defined by the sequential extraction, includes some nonexchangeable uranium. Further analysis of the sequential extracts is necessary to refine the uranium concentration in the sorbed phase.
This work demonstrates that the uranium in the low pH plume of the Water Table Aquifer at this waste site is more mobile then had been predicted in previous modeling efforts. The data also suggests that removal of aqueous-phase uranium will become less effective over time as the pH of the waste site aquifers returns to more neutral conditions. This is due to the gradual change in groundwater chemistry (i.e. increasing pH) due to the discontinuation of use and closure of the seepage basins. Since the inflection point of the uranium sorption isotherm in this study occurs at about pH 4.0, and a large fraction of the uranium at the site is currently associated with the low-pH portion of the plume, aqueous-phase uranium is expected to be more easily removed while the pH of the system is lower.
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27-11
EVALUATION OF IN SITU PHYTOREMEDIATION OF URANIUM-CONTAMINATED SOILS IN OHIO AND MONTANA

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ABSTRACT
The U.S. Department of Energy (U.S. DOE) is investigating the use of plants to remove heavy metals and radionuclides from contaminated soils (phytoremediation). As a first-step in the development of phytoremediation for uranium-contaminated soils, test plots were established in the summer of 1994 at the Fernald, Ohio, DOE site incinerator area and at a uranium waste rock dump in southwestern Montana. The test plots were designed to develop experimental protocols as well as carry out a preliminary screening of uranium accumulation by several crop plants (mustards, legumes, rapeseed), and indigenous plants. At Fernald, soils and plant biomass were sampled during and at the end of the growing season. Only modest uptake of uranium was observed, from 1 to 10 ppm in the leaf and stalk biomass (mg/kg dry weight biomass), compared to about 50 ppm in the soil. Similar uranium levels were observed in aboveground biomass from the mine site, although uranium concentrations averaged 340 ppm in waste rock "soils". Nevertheless, the methodologies applied are judged sufficient for verification of contaminant hyperaccumulating plant species in a field setting. So-called "hyperaccumulator plants" are known for some heavy metals, such as Zn and Ni, that concentrate these metals to over several thousand ppm. Plants species and strains that accumulate U to at least 100 ppm will be required

for phytoremediation. A screening program for uranium accumulating plants is underway at the USDA-ARS Laboratory at Cornell University.

INTRODUCTION

At many Department of Energy (DOE) sites, surface soils are contaminated with heavy metals and radionuclides (1). In many instances, levels of contamination exceed regulatory standards by relatively small amounts. Remediating these soils with conventional technologies is costly and produces significant secondary environmental impacts. Phytoremediation, in which plants are used to remove toxic elements from soils into harvestable biomass, could be a cost-effective alternative. Growing and harvesting plants is relatively inexpensive, and can be accomplished with minimal soil disturbance, compared to conventional soil remediation technologies. The harvested biomass could be thermally treated to reduce the volume requiring ultimate disposal by an order of magnitude.

For phytoremediation to clean up a site in a reasonable period of time, i.e, 10 to 20 years, the concentration factor (dry biomass to dry soil contaminant ratio) must be about 40, and the annual plant productivity be at least 10 metric tons/hectare (t/ha). These factors will vary depending on depth of contamination and degree of contaminant reduction desired (2). Successful phytoremediation will require: 1) identification of the appropriate plant species able to accumulate uranium in sufficient amounts; and 2) agricultural management to optimize biomass production as well as enhance bioavailability and uptake of the contaminant(s) of concern from soil.

From a review of the literature (3), dry plant biomass typically contains less than one ppm (mg/kg) of uranium, when growing on uncontaminated soils. Some plant families, the mustards and legumes, exhibit higher uranium concentrations, 10 to 20 ppm in LSB when grown in soils containing elevated levels (i.e., >5 ppm) of uranium. Occasional, but thus far not verified, reports of even higher uranium contents in isolated field samples are found in the literature.

The ultimate program goal is to develop phytoremediation technologies applicable to site remediation problems of the DOE complex. One problem is extensive surface contamination of soils with uranium, at relatively low levels, but still above regulatory limits. Low cost technologies for the remediation of such sites are required. The immediate objective was development and demonstration of protocols for field evaluation of potential plant species useful in phytoremediation of uranium-contaminated soils. For the initial test, examples of mustards, legumes, crop plants, and indigenous plants, were used. Two field studies were carried out, one at the Fernald, Ohio, DOE site, and the other at a uranium waste rock dump site in southwestern Montana.

EXPERIMENTAL DESIGN AND METHODS

At the Fernald, Ohio, DOE site, the experimental plot was located at the Incinerator area, immediately east of the main production facility. The test plot consisted of three replicate blocks with ten 1x2 m cells per block. Five plant species were tested on two randomly selected cells per block: from local seed sources, alfalfa (Medicago sativa cv. WAMPR) and red clover (Trifolium repens cv. "medium red"); wild mustard (Brassica kaber) and rapeseed (Brassica napus) from a California seed supplier; and tall fescue (Festuca arundinacea), an introduced grass growing naturally on this site.

Prior to seeding the test plot, the existing vegetation was removed by spraying with glyphosate and rototilling; the individual cells were delineated with plastic piping. A composite soil sample was collected from each of the test cells and analyzed for total uranium. The individual cells were then seeded without the addition of soil amendments. Midway through the growing season one cell of each of the species was harvested. In September the remaining cells were harvested (both roots and shoots). Phytomass and soil samples from each cell were analyzed again for total uranium content.

A second field experiment was established on a waste rock dump at the W. Wilson mine, an abandoned uranium mine, about 15 kilometers southwest of Helena, Montana. At this site, two blocks of 12 cells of 1 x 0.5 m each were established, each block was used to test six different plants, each represented by two provenances. Plant species acquired from Big Sky Seed (Shelby, Montana) included: tansy mustard (Descurainia pinnata), fringed sage (Artemisia frigida), hard fescue (Festuca longifolia cv. Durar), redtop (Agrostis alba), creeping red fescue (Festuca rubra cv. Boreal) and birdsfoot trefoil (Lotus corniculatus cv. Empire). The following

species were purchased from Granite Seed (Lehi, Utah): fringed sage, hard fescue (cv Aurora), redtop, creeping red fescue (no variety specified) and birdsfoot trefoil (variety not specified). Tansy mustard was also purchased from Valley Seed Service, Fresno, California. The test blocks were prepared by rototilling the top 25 cm of soil followed by removal and stockpiling of the top 5 cm of soil. Each cell was treated with 1.3 g of (NH4)2HPO4 (reagent grade) and the excavated soil replaced. A soil sample was collected from the center of each cell.

Environmental (e.g., soil, plant, irrigation water) and quality control (e.g., duplicate samples, equipment rinsate) samples from Fernald were shipped by overnight mail to Accu-Labs Research, Inc. (Golden, Colorado) for determination of those parameters requested in the test plan (4). Agricultural parameters for soils included pH, Eh, organic matter, soil texture, etc.; agricultural analyses on plant tissues included NPK levels, percent ash, etc. Radiochemical analyses focused on determination of total uranium by Kinetic Phosphorescence Analysis (KPA; ASTM D5174-91) and isotopic uranium by USEPA Method 908.0. All root samples were washed in tap water, followed by sonicator bath in deionized (DI) water and rinsed with DI water before drying.

The respective analytical parameters for environmental and QC samples taken from the Montana uranium site are described in the test plan (5). Notable differences from the Fernald analyses are: 1) determination of strong acid (HNO3) and weak acid (ammonium acetate, pH 5.5) - extractable levels of select "heavy metals" in mine soils and "total" metals in leaf and stalk biomass, using; 2) inductively coupled argon plasma emission spectroscopy (ICPAES) at the MSE Laboratory (Butte, Montana).

RESULTS

Table I summarizes the results from the Fernald site. The average U content of the soils at this site was approximately 36 pCi/g (equivalent to 54 mg/kg of soil), similar to the anticipated (though not yet finalized) soil clean-up level (about 35 pCi or 51.8 mg/kg of soil). Thus, this specific plot may not require remediation, although other areas at this site exhibit higher average levels, and will likely need soil remediation. It is in such marginally contaminated soils, where uranium need only be reduced by only a few tens of ppm, to achieve regulatory compliance, that phytoremediation would be most applicable.

Inspection of the "grand mean" uranium levels in soils implies an apparent decrease of 7 mg U/kg dry soil over the course of the growing season; a two-tailed Student's t test rejected the null hypothesis (i.e., no difference in June vs. September U mean levels) at p 0.01. However, for reasons discussed below, MSE does not attribute this change to U uptake by plants. The alternative hypothesis is that such difference is due to heterogeneous distribution of uranium oxide particles (i.e., fallout from the adjacent incinerator stack) within a given volume of soil. Although cell-specific uranium levels generally declined between the June and September sampling episodes, in a few instances U levels appeared to increase over time (e.g., from 49 to 68 mg U/kg soil in Block 3, cell1 and from 48 to 55 mg U/kg soil in Block 3, cell 10). Thus, sampling error is more likely an explanation than is analytical error, given the 15 percent differences in replicate sample results as well as close agreement between certified and reported results for the laboratory control standards.

The results for the Fernald site (Table I) indicate very little accumulation of uranium in any of the plants. The highest uranium content in any LSB sample was for a red clover at 10 ppm (from Block 3, cell 6), but there was a large variability between different plots and cells for the various plant species. Roots contained considerably more uranium, ranging up to 43 ppm in red clover from Block 2, cell 5. However, total U levels in the species evaluated are still too low to allow significant site clean-up, as supported by the mass balance data presented in Table II. Division of each species' average U content (in shoots and roots) by the respective mean U levels in baseline soils indicates annual plant U uptake rates ranging from 0.024 percent year -1 for alfalfa to 0.045 percent year -1 for wild mustard. At such rates of U removal, cleanup of the incinerator area soils would require well over a century to remediate to regulatory acceptable levels.

Table IIIa,b summarizes the results from the Montana site. In this case both acid extractable and bioavailable metal levels were determined. Plant growth was poor at this site, presumably due to the poor soil fertility (e.g., 0.01 percent N, 18 ppm NH4F-extractable P and 64 ppm NH4OAc-extractable K), strongly acid pH (5.4 avg.) and potentially phytotoxic levels of Cu (and U?) in the soils. Thus, for many samples,

the biomass from the two replicate cells was composited to produce sufficient sample biomass for chemical analysis. Note that U levels in LSB are of similar magnitude to those observed at Fernald, despite nearly 10-fold higher total U levels in "soils" at the mine site. For example, the maximum U concentration in LSB was approximately 10 ppm in tansy mustard. The presence of 119 mgAl/gLSB (vs. 3318 mg/g in the other species) in this mustard indicates the potential for soil contamination; however, the other heavy metal levels are not unreasonable in this species. Anecdotal evidence for root U concentrations exceeding those in LSB is indicated by the respective 7.8 and 2.7 ppm levels observed in spotted knapweed (Centaurea maculosa) plants growing in waste rock material immediately west of the test blocks. Background levels of Cd and Zn rarely exceed 3.0 ppm and 150 ppm, respectively, in aboveground biomass from nonmineralized soils in southwestern Montana; thus, the concentrations of these two elements in LSB from the (mineralized) mine soils are greater than typically observed in the area. This observation is particularly true for metals levels seen in the tansy mustard. Nevertheless, none of the heavy metals data indicates the presence of hyperaccumulator species at this site. Finally, as the birdsfoot trefoil germinated well and survived (albeit with stunted growth) under the harsh site conditions, it should be considered as a viable species for revegetating this abandoned mine site's waste rock dumps.

DISCUSSION

The primary objective of this investigation was demonstration of useful field and laboratory protocols required for evaluation of this innovative technology, not site remediation per se; the study was initiated using cultivated and indigenous plant species believed to accumulate (based on literature review) elevated levels of uranium from soils. Although many of the species tested did this much, the above initial field trials did not identify any plants that exhibited sufficient U uptake to meet the minimal requirements for phytoremediation. For example, to reduce soil concentrations by 10 ppm to a depth of 30 cm (about 4,000 metric tons/hectare, t/ha, of soil), would require cultivation for 20 seasons of a plant that accumulates 100 ppm of uranium with a productivity of 20 t/ha/yr. Obviously, even greater than 100 ppm levels in the harvestable biomass (both shoots, and to the extent feasible, roots) would be desirable, and even required in most cases. Although some literature data suggests the possibility of such plants, this has yet to be confirmed. However, by analogy with other metals and known hyperaccumulator plants (USDOE, 1994), some of which exhibit many thousands of ppm of toxic elements in the plants, it is likely that such plants may already exist or could be created by genetic selection.

The major challenge in phytoremediation of these sites is to find uranium hyperaccumulating plants. Productivity is, of course, also a major requirement. Present results of field trials at Fernald and Montana were not encouraging in either respect. However, they have demonstrated some basic requirements to testing phytoremediation in the field. These results also demonstrate that commonly available plants are unlikely to exhibit sufficient U accumulation. Nevertheless, the U levels in mustard plants, being an order of magnitude higher than many plants, suggests significant potential for metals accumulation, likely under genetic and environmental controls.

Work is planned for further field tests of plants known to hyperaccumulate heavy metals such as Zn, Cd and Ni, and for plants that exhibit uranium accumulation in larger laboratory screening experiments, currently being carried out in the laboratory of Dr. Leon Kochian (USDA Plant Nutrition Laboratory, Cornell University, Ithaca, NY). Even if phytoremediation of uranium contaminated sites proves not to be practical, other types of radionuclides (e.g., 90Sr) and toxic elements (e.g., Cd) are more amenable to site clean-up with this emerging technology. Such field tests are being planned for implementation in the summer of 1995.

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CONCEPTUAL RETROFIT DESIGN FOR A PILOT PLANT TO TREAT RADIONUCLIDE CONTAMINATED SOILS

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ABSTRACT
The VORCE (volume reduction and chemical extraction) Pilot-Plant (VPP) was developed to remediate radioactively contaminated soils at the Montclair/Glen Ridge, New Jersey Superfund sites. The development, design, fabrication, and testing was under the direction of the U.S. Environmental Protection Agency (EPA), Office of Radiation and Indoor Air (ORIA) and performed under contract by Sanford Cohen & Associates, Inc. (SC&A). A majority of the preliminary design work and the testing of the plant were conducted at the National Air and Radiation Environmental Laboratory (NAREL). The plant was designed to separate the contaminated soil fraction (-200 mesh) from the bulk soil using various physical separation techniques. The pilot-plant was fabricated primarily from off-the-shelf components proven in applications related to mining and mineral processing. The present design is based on the Phase II plant, which resulted from modifications of a Phase I plant.
The Phase II plant was completed and successfully tested on clean and contaminated Montclair soil in September 1992. The conceptual retrofit was initiated to allow the pilot-plant to more effectively treat Department of Energy (DOE) Superfund and FUSRAP (Formerly Utilized Site Remedial Action Program) sites, to improve the plant feed and materials conveyance systems and the initial liberation process, and to allow a feed input rate of from 2 to 5 tph.
Each modification is based on the analyses from the prior test results and computer modeling performed by SC&A, Carpco, and Virginia Polytechnic Institute and State University. These modifications produce a conceptual retrofit design that could recover up to 65 percent of the feed soil with a radium-226 and radium-228 concentration of less than 5 pCi/g. The conceptual retrofit design is an innovative remediation technology that could reduce large volumes of contaminated soil being disposed as radioactive waste. Although the retrofit is considered to be innovative, it is based on separation technology that has been used extensively in the mineral processing industry. However, the retrofit is based on characterization studies of the samples from the DOE Superfund sites, it could also be successful at other contaminated sites where the majority of the radionuclide contaminants are concentrated in the fine-size particles. Many radioactively contaminated sites are in that category, including some DOE sites.

INTRODUCTION
The Department of Energy's (DOE) Formally Utilized Sites Remedial Action Program (FUSRAP) is proposing to demonstrate the feasibility and cost effectiveness of treating soils contaminated with uranium and thorium and their decay products to reduce the volume requiring costly transportation and disposal. The demonstration is

planned at two DOE Superfund sites. The platform for the volume reduction treatment pilot-plant is the VORCE (volume reduction and chemical extraction) Phase II Pilot-Plant (VPPII). The existing pilot-plant requires modifications to correct operational and performance deficiencies found during testing, and to account for characteristic differences between the DOE Superfund soils and those for which the VPPII was originally designed to remediate. The reconfigured pilot-plant is hereafter called the Radioactive Soil Washing and Particle Separation (RSWAPS) pilot-plant.

The VPP was developed to remediate radioactively contaminated soils at the Montclair/Glen Ridge, New Jersey Superfund sites. The development, design, fabrication, and testing was under the direction of the U.S. Environmental Protection Agency (EPA), Office of Radiation and Indoor Air (ORIA) and was performed under contract by Sanford Cohen & Associates, Inc. A majority of the preliminary design work and the testing of the plant (Phases I and II) were conducted at the National Air and Radiation Environmental Laboratory (NAREL). As extensive characterization had indicated, the plant was designed to separate the contaminated soil fraction (-200 mesh) from the bulk soil using various physical separation techniques. The pilot-plant was fabricated primarily from off-the-shelf components proven in applications related to mining and mineral processing. The present plant, Phase II, is the result of extensive modifications of a Phase I Plant, which was completed in 1990.

The VPPII was completed and successfully tested on clean and contaminated Montclair soil in September 1992. After completing these test runs, it was recognized that design modifications would be necessary to allow the plant to operate more efficiently.

A preliminary assessment of the current plant design has been completed by SC&A and Carpco. Data considered in the assessment were the preliminary characterization studies on a limited number of samples from the DOE Superfund sites (1, 2), the results of radioactive and clean soil testing of VPPII, and available mineral processing information. Based on those analyses and a somewhat limited characterization of the test soils, recommendations for modifications to the existing plant have been made. By incorporating one of these design modifications, the RSWAPS plant will be able to consistently achieve the desired level of remediation at rates of either 2 or 5 tph.

PROPOSED MODIFICATIONS

Based on the initial VPPII data and operating experience during testing, several deficiencies were identified with the existing system. These problems presented operating difficulties and limited the maximum plant efficiency. The major operational problems and recommended solutions are discussed below.

Feed Conveyance System

The introduction of raw soil into the plant presents the single greatest design challenge. The physical characteristics of raw soil can vary widely during an application, and the necessity to handle relative large soil particles at the design, low input feed rates require compromises in the selection of a raw-soil feed conveyance system of a pilot-size plant.

The VPPII feed conveyance system proved to be inadequate for input rates greater than 800 - 1000 lb/hr. The system consisted of a steep-pitch feed hopper covered by a 2 x 3.5-inch openings grizzly screen. The feed hopper discharged onto an inclined screw conveyer which discharged directly into the trommel. It was intermittently filled using a "Bobcat" loader.

This system presented three problems during operation. This first problem was blockage of the feed hopper because material bridged above the screw conveyor. The short-term solution to this problem was assignment of an operator at the feed hopper to manually dislodge blockages. This arrangement, however, is not practical for long-term operation of the plant because of manpower requirements and potential radiation exposure (the feed hopper is the highest risk area for exposure to respirable dust).

The second problem encountered was binding of the screw conveyor by oversize material. Although these larger particles (2 to 4 inches) passed through the grizzly screen, they would subsequently become lodged between the conveyor flights and the hanger bearings that support the screw above the trough. The only solution to this problem was to shutdown the feed system and manually remove the blockages. In addition to limiting the plant capacity, this latter problem was detrimental to the

operating efficiency because of intermittent feeding.
The third problem was controlling the volume of material conveyed by the screw. The tapered screw did not control the rate of material flow. The result was overloading of downstream sections of the screw conveyer. The solution to this problem during testing was to limit the rate at which soil was fed into the hopper. This solution, of course, limited the overall plant feed rate.
Two different feed system configurations have been employed on the pilot-plant. A belt conveyor was used on VPPI. As stated above, a screw conveyor was used on VPPII. Both have proven to be unsatisfactory due to their inability to deliver soil to the scrubber in a consistent manner with no spillage. The only remaining options for feeding the plant were:

- bucket elevator
- slurry pump
- direct feeding via hopper and vibrating pan feeder
- apron conveyor
- drag conveyor

It is possible to eliminate the first three from consideration for the following reasons: with very low feed rates and soil characteristics, Bucket Elevator problems are anticipated with discharging the buckets from the Bucket Elevator; with very low feed rate and the existence of large particles, pumping with a Slurry Pump is not possible; and Direct Feeding would require construction of a highly elevated feed ramp to make the hopper accessible by a loader.
The remaining two options (apron conveyor and drag conveyor) function in a similar manner. The primary difference is that the apron conveyor provides a surface to carry the material, vs. sliding. This feature requires a more complicated design for the apron conveyor. It is constructed of an endless apron of overlapping pans attached to two strands of chain. The chain is equipped with chain rollers (wheels) riding on tracks. In contrast, the drag conveyor merely slides the conveyed material across the surface of a stationary conveying pan. Since this conveying pan can also serve as the main structural element, construction is simplified. The primary disadvantage of the apron conveyor is, therefore, one of increased cost and maintenance.
Another disadvantage concerns the method of feeding. The apron conveyor requires some method for controlling bed depth on the pans. The apron pans will inherently carry all material deposited on them and are limited only by their structural strength and the depth of any skirtboards provided by the design. It is therefore necessary to provide a shearing device at the discharge throat of the feed hopper. This device produces a "strike-off point" above the pan surface that limits the depth of material on the pan. This strike-off point can create flow problems if the unit is not properly designed. In contrast, the bed depth is essentially self-limiting in the drag conveyor because of the drag chain design. The chain essentially shears material to be conveyed from the feed hopper. Factors influencing the bed depth include: 1) height of the chain, 2) conveyor incline, 3) particle size, and 4) material flowability.
Based on the above considerations, a drag conveyor is the choice for feeding the RSWAPS. In the modified design, the existing screw conveyor is replaced by a drag conveyor mounted to the existing chassis. The existing variable speed drive is retrofitted to the new conveyor. The existing feed hopper is replaced. The new feed hopper is smaller in design for better accessibility by a small skid-steer loader. The loader approaches from the rear of the conveyor. In addition, the grizzly passes a larger particle size (4-in vs. 2-in minimum openings). The increase to 4 x 4 in grizzly opening size is made to improve material flow into the hopper.
Additional changes in the feed system include the removal of the feed scale and a redesign of the scrubber feed chute. The purpose of the scale in the VPPII configuration was to provide a means of controlling the plant feed rate. The scale produced a signal, proportional to feed rate, which was used by the PLC to control the speed of the feed screw. Actual operation of the plant revealed that feed rate could be easily controlled visually. Removing the feed scale simplifies operation of the plant and reduces flow problems encountered between the feed system and the trommel. The feed chute redesign is necessary with change of conveyor type.
Trommel Screen/Washer
The trommel screen was originally placed in the circuit to scrub and remove coarse material (+1/4-in) from the feed. According to the data obtained from the initial

test runs, the coarse material was adequately rejected. However, both test data and visual observations have shown that this material would benefit from additional washing/scrubbing to remove the small amount of fine particles still clinging to the +1/4-material. It was noted that the trommel was insufficiently loaded during operation such that the residence time of the material (+1/4 in) was inadequate. This indicates that either the plant feed rate was too low, the trommel screen was oversized, or that design of the trommel did not provide adequate residence time for the larger material. Since the plant feed rate is limited downstream by the throughput of the screw classifiers, increasing the plant feed rate is not recommended. Increasing the trommel screen opening would introduce material to the attrition mills too large to effectively pass through them. Therefore, to provide additional scrubbing action through particle contacting by increasing the material residence time, modifications to the original trommel design are required.

Based on designs from typical mineral processing applications, a tapered discharge added to the end of the trommel screen restricts the flow of material discharging the mill resulting in a hold-up of the coarser material. The increased working load in the mill provides better particle-particle contact and should result in improved scrubbing of the coarse material. To further ensure sufficient particle contacting, the trommel is retrofitted with an electronic variable speed drive, which allows the operator to fine tune the rotational speed of the unit to maximize its performance. Specifically, the unit should be designed and operated so that the material in the drum exhibits a cascading action rather than tumbling. This approach maximizes abrasion while minimizing particle breakage. Finally, the taper is fitted with screw-type lifter bars to remove the coarse particles from the end of the mill.

Water carry-over with the coarse product also presented a problem in the VPPII trommel. This was simply a result of leakage through the open-end design. The addition of the tapered discharge with appropriately located water sprays eliminates this problem.

Transfer Conveyors

The transfer screw conveyors from the attrition scrubbers to the screw classifiers also had problems with occasional overflow. This was most apparent when the scrubber pulp density dropped, resulting in a surge in the attrition mill discharge rate. With the nature of the system, it is assumed that this problem will always persist. Two possible solutions to this problem are considered. The simplest approach is to totally enclose the transfer conveyors and increase the speed to minimize this problem. The second, and selected, approach is to replace the screw conveyors with a viscous slurry pumping system. These types of systems are routinely used for conveying concrete over long distance and should work quite well in this system.

Launders

The launder that transferred material from the secondary screw classifier overflow to the cyclone feed sump experienced problems with sanding of material that was a result of insufficient incline between the feed and discharge end of the launders. The short-term solution on site was to add additional water to the launder to minimize sanding.

It is recommended that the launder be replaced with a vertical sand pump that directly conveys the material to the cyclone feed sump. This approach is typically used in particulate processing operations to handle potentially intermittent flows. These pumps are ideal for this situation since they can run empty for extended periods of time.

Dewatering System

After solving a few start-up problems, it appeared that the water clarification system worked very well. However, dewatering of the fine waste fraction using the filter press presented some operational problems. In addition to the operational problems, the capacity of the filter press is also a serious concern. According to the documentation provided, the rated capacity of this unit is 1100 lb/hr. Based on the feed particle-size distribution, the weight percent of one of the test soils passing 200 mesh is approximately 30 percent. At a feed rate of 2 tph, the filtration system would be required to handle 1200 lb/hr. Under realistic conditions, the solids feed rate to the filtration system would approach 1500 lb/hr. For the other test soil, where the weight of the material finer than 200 mesh is approximately 45 percent of the total soil, the solids feed rate could approach 2200 lb/hr. Thus, based on the current design, the system cannot be operated continuously. A shift-type approach would be required to operate the press while the

wash system is idle, thus allowing the dewatering system to catch-up with the remainder of the plant.

Considering the constraint placed on the plant feed rate by the dewatering system, the current filtration system must be replaced for the 5 tph option. The possible choices are to either install a dewatering centrifuge or a belt-type filter press with a capacity in excess of 4-5 tph. A centrifuge appears to be a better choice since it is more reliable and less operator sensitive. The latter is particularly important considering the nature of the material. At this point in the process, there is nearly a four-fold increase in the radioactivity of the material.

Stacking Conveyors

In the VPPII system, clean product streams were diverted to 55-gal drums placed on roller conveyors. When the drum filled, it was manually moved and replaced with an empty drum. Actual testing has proven this method to be very cumbersome and labor intensive. Spillage of material was also a problem. Therefore, the addition of stacking conveyors at the clean product stream discharge points is recommended. We believe this is absolutely necessary on the 5-tph plant. These conveyors will discharge to stockpiles. Material stockpiled will be moved using a skid-steer loader.

Stacking conveyors for the clean sand and clean fines streams will be fabricated using available screw conveyor components from the present 12-in feed screw. Two complete screw conveyor drives to be purchased and supports to be fabricated.

A new drag conveyor, similar in design to the feed conveyor, will be used to stockpile the clean gravel. This design has been chosen because of large particles in the stream. These particles would most likely cause binding in a screw conveyor. They would also tend to roll back on a belt conveyor. Spray nozzles will be used to provide an additional rinse step for this stream. Rinse water will flow to a containment sump. The rinse water will then be pumped to the cyclone feed sump.

SIMULATED PLANT PERFORMANCE

Based on the information available from the previous test runs, a series of simulations were conducted for the redesigned plant. These simulations provide an indication of plant performance and are based on the data collected during the initial test runs with Montclair soil.

Particle-Size Separation

Since the VPPII was performing only a size separation on the raw feed, simulations can be based on the partition curve for each unit operation. A partition curve represents the size-by-size mass recovery for a particular unit operation. Mathematically, it is expressed as follows:

Eq. (1)

where Xi represents the weight percent of material in size class (i) present in either the feed (f) or the product (p) streams and M represents the solids mass flow rate in the stream of interest. The partition curve is simply the size-by-size recovery plotted as a function of particle size. A perfect separation is represented by a step function which increases from 0 to 1 at the particle size of separation. Based on the available data, partition curves were constructed for the screw classifiers and the classifying cyclone (see Fig. 1). These two unit operations are the two main separating units used in the VPPII plant. These results illustrate that the units provided a reasonably efficient separation as indicated by the steep slope of the partition curve. However, in both cases, the partition curves have a significant y-axis offset. This well-known phenomenon represents short-circuiting of fine feed material to the oversize product. In the case of the classifying cyclone, 20 percent of the material finer than approximately 20 microns reported non-selectively to the cyclone underflow. Likewise, approximately 5 percent of the material finer than 100 microns fed to the secondary screw classifier reported with the oversized product. This represents significant remediated product contamination considering that essentially all radioactive contamination is present in the fines fraction (<75 microns).

Short circuiting can be avoided by eliminating the potential for fines recovery with the coarse product through multiple stages of separation (e.g., three cyclones in a series configuration) or by incorporating a unit operation that reduces or eliminates short-circuiting of material. As shown by the partition curve for the secondary screw classifier, multiple processing stages reduces the impact of fines carryover. In this case, the primary screw classifier had a fines carryover rate approaching that of the cyclone. However, multiple processing stages tend to become

quite cumbersome, particularly with cyclones. In this application at least three cyclone stages would be required to achieve the required separation. This approach could create operational problems regarding feed solids density, flow rate, etc. The best solution is to incorporate a unit operation that eliminates the problems associated with fines recovery. In this application, a hydraulic classifier, such as the Floatex unit, would be the best choice to ensure a high quality washed product. The partition curve for a Floatex classifier is also shown in Fig. 1. The Floatex classifer is capable of providing the same classification as other unit operations; however, the carry-over of fine material is largely eliminated. The Floatex Density Separator is a hydraulic classifier designed to achieve extremely sharp particle-size separations. Unlike a vibrating screen, the Floatex separator classifies material based on the principal of hindered settling. Since the separation occurs in an open tank, mechanical problems associated with screening (i.e., blinding and tearing of the screen cloth) are eliminated. In addition, by settling against a clear stream of counter-current water, inefficiencies resulting from fines carrying-over with the process water (typical of classifying cyclones) are eliminated. As a result, the Floatex separator provides a more efficient and cost effective separation.

The product quality depends on the composition of the feed material and the desired separation size. The Floatex is generally utilized for size separations on particles finer than 1/4 in. Efficient separations are achieved when the desired cut-point (i.e., separation size, d50) ranges from 14 to 200 mesh (1 mm to .075 mm).

To illustrate the advantage of the Floatex separator, simulations were conducted using this device as a final cleaning step on both the 1/4-in x 65 mesh and 65 x 200 mesh products. The results from this simulation are shown in Fig. 2 that illustrates the comparison of a Floatex unit separation to an ideal separation (laboratory wet screening) and to these results from the previous VPPII tests. As shown, the circuit including the Floatex separators provided an extremely efficient separation, approaching the results obtained by screen analysis (ideal). In this case, less than 3 percent of the material finer than 200 mesh would be recovered with the coarse product from the Floatex unit. In comparison, approximately 20 percent of the material finer than 200 mesh reported to the coarse product using the existing VPPII system. The relatively large amount of misplaced fines makes it difficult for the existing VPPII system to maintain the required product grade. These results clearly illustrate the advantage of a final clarified water washing stage such as that provided by the Floatex separator.

Flowsheet Recommendations

Based on the simulation results presented, several modifications to the original flowsheet are recommended. These modifications are illustrated in the process flowsheets shown in Fig. 3 and Fig. 4. In all cases, the mechanical modifications to the feed system, trommel screen, transfer conveyors, and launders are assumed.

Figure 3 illustrates the modifications to the circuit configuration to make the plant completely operational at a feed rate of 2 tph. In this case, two Floatex units have been added. The first Floatex unit has been installed to provide the primary remediation step after attrition scrubbing. The Floatex underflow would be equipped with a screw auger to remove the coarse material so that the washed product is also dewatered prior to discharge. This modification is greatly needed since there is only one remediation step after attrition scrubbing. Depending on the nature of soil being treated, the stream may contain excessive amounts of fine material generated during scrubbing. A second Floatex classifier has been installed on the cyclone underflow stream since, in the current configuration, this material has the largest detrimental effect on the quality of the overall plant product. The arrangement will ensure that the 65 x 200 mesh fraction is a high quality, thoroughly remediated product.

Figure 4 illustrates the modifications that would be required to convert VPPII into a 5-tph facility. The secondary screw classifier would be relocated to operate in parallel with the existing primary screw classifier. Relocation is essential to handle the tonnage present in the trommel underflow. In addition, all four attrition scrubbers would be used to handle the increased capacity. As described above, the product from the scrubbers would be remediated in a Floatex separator.

To handle the increased capacity in the fines circuit, a second classifying cyclone would be installed in a parallel arrangement. The cyclone feed system would be arranged to allow either one or both units to operate. The configuration would

depend upon the nature of the soil (i.e., loading in the fines circuit). The cyclone underflow will report to a second Floatex unit, as in the original 2 tph circuit. In this case, however, the size of the unit would be increased to handle the additional material.

Testing the VPPII revealed that the Montclair soil finer than 200 mesh collected from the filter press contained 35 percent water (65 percent solids). Modifications shown in Fig. 3, retaining the batch filter press to collect the material finer than 200 mesh, should produce a product with the same moisture content. The modification shown in Fig. 4 that replaces the filter press with a bowl centrifuge will collect the material finer than 200 mesh at 55 percent solids. Each of these modifications can include a hollow-tube dryer that will dry the product to 90 percent solids. The results of an economic analysis considering the capital investment for a dryer versus the savings realized from the transporting for disposal the drier product would be primarily affected by the amount of soil to be treated. Beyond some minimum amount of soil to be treated, the dryer would pay for itself and then reduce the overall cost of remediation or the expense of transporting the drier material.

It should be noted that the increase from 2 to 5 tph requires only a minimal increase in equipment, unlike that for dewatering. According to the above analysis, all the existing equipment (with the exception of the filter press) would be used in the final design. This represents a significant cost savings in comparison to the construction of a completely new facility.

CONCLUSIONS

A preliminary assessment of the current plant design was completed by SC&A and Carpco, Inc. Considered in the assessment were the preliminary soil characterization studies samples from the DOE Superfund sites, the results of radioactive and clean soil testing of the Phase II plant, and available mineral processing information. Based on those analyses, recommendations for modifications to the existing plant were made. The following conceptual design modifications to the Phase II plant are: 1) Replace the existing feed screw conveyance system with a drag conveyance system to achieve a desired remediation feed rate of 2 to 5 tons per hour (tph); 2) Modify existing trommel screen/washer is modified by the addition of a tapered discharge added to the outlet end of the trommel screen to increase the residence time of the coarse (+1/4-in) material and to allow adequate removal of fine particles; 3) Replace the existing transfer screw conveyors from the attrition scrubbers to the screw classifiers with a viscous slurry pumping system to handle the occasional problem of material overflowing in the existing transfer screw conveyors; 4) Replace the launder that transfers material from the secondary screw classifier overflow to the cyclone feed sump with a vertical sand pump which directly conveys the material to the cyclone feed sump to prevent problems with sanding of material; 5) Install a dewatering centrifuge, replacing the belt-type filter press that cannot match the throughput capacity of the remaining components of the plant; 6) Add stacking conveyors at the clean product stream discharge points, replacing roller conveyors that proved to be very cumbersome and labor intensive; 7) Add a hydraulic classifier designed to achieve sharp particle-size separations to ensure a high quality clean product by eliminating displacement of up to 20 percent of the fines (-200 mesh, -75 microns) to the remediated product by several of the separation units.

Each of the aforementioned modifications is based on the analyses from the testing of the existing plant, preliminary characterization studies on DOE Superfund site samples, and computer modeling performed by SC&A, Carpco, and Virginia Polytechnic Institute and State University. These modifications produce a conceptual retrofit design that could recover up to 65 percent of the feed soil with a radium-226 and radium-228 concentration of less than 5 pCi/g. The conceptual retrofit design is an innovative remediation technology that could reduce large volumes of contaminated soil being disposed as radioactive waste. Although the retrofit is considered to be an innovative remediation technology, it is based on separation technology that has been used extensively in the mineral processing industry. The retrofit is based on characterization studies of the samples from the DOE Superfund sites, however, the retrofit could also be successful at other contaminated sites where the majority of the radionuclide contaminants are concentrated in the fine-size particles. Many radioactively contaminated sites are in that category, including numerous Superfund, DOE, NRC licensee, and DOD sites. Many are being remediated by a process that consists of excavating large volumes of soil and transporting them to an approved landfill. This is a very costly remediation process and volume reductions of up to

65 percent represent significant cost savings while meeting site remediation criteria and goals.

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PERMEABLE, SUBSURFACE SORBENT BARRIER FOR 90Sr: LABORATORY STUDIES OF NATURAL AND SYNTHETIC MATERIALS

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ABSTRACT

Laboratory studies were conducted to support the design and construction of a full-scale, in-situ sorbent barrier for intercepting a 90Sr plume within a surficial water-bearing sand and gravel layer. Evaluation of a set of materials for use as a passive sorbent wall was first approached through a series of batch sorption tests, using 85Sr tracer, on a set of natural zeolites and metal oxides to determine their effectiveness for sorbing 90Sr. These candidate materials were selected on the basis of commercial availability, cost, and anticipated performance. Selected materials, based on their performance in the screening experiments, were further tested to assess hydraulic performance and sorptive characteristics under dynamic flow conditions. Test data were incorporated into a one-dimensional finite difference model and estimates were made of in-situ performance over extended time frames. The data developed during this study may be applicable to other DOE facilities.

INTRODUCTION

In this paper we present a study combining laboratory experiments and modeling intended to support the design and construction of a sorbent barrier (a passive filter wall) for retarding the migration of 90Sr within a water-bearing surficial sand and gravel layer. Evaluation of a set of materials for possible use as a passive sorbent wall to intercept and remediate the plume was first approached through a series of batch sorption tests, using 85Sr tracer, on a number of materials for screening purposes. A few materials showing the greatest amount of sorption were subsequently selected for column testing. Column tests were performed in order to evaluate the performance of short-listed materials for removing 90Sr from groundwater under dynamic flow conditions. A one-dimensional finite-difference model (1) was used to simulate the observed laboratory data so that predictions could be made regarding 90Sr behavior on a larger scale and over longer time frames. This model accounted for advection, dispersion, diffusion, radioactive decay, and linear adsorption.

Recent studies have promoted the potential use of in situ reactive barriers for remediation of contaminated groundwater. The emplacement of in situ reactive barriers may enhance the removal of groundwater contaminants through alteration of pH or Eh, precipitation, sorption, or nutrient supply for biodegradation.(2) For trace contaminants, the barriers need to preferentially adsorb contaminants as groundwater flows through the material. Candidate materials for passive barriers must be tested under site specific chemical conditions to determine if materials perform as expected in the presence of competing ions. Two major classes of materials for this application are natural zeolites and metal oxides and oxyhydroxides. The sorption properties of both materials have been studied in great

detail. (3,4,5)

METHODS

In these tests we used site specific groundwater and attempted to maintain site specific conditions, such as flow rates. The methods are described below.

Material Screening Experiments

A set of screening tests were run to provide an initial assessment of the effectiveness of a broad range of materials to sorb low concentrations of 90Sr from groundwater. These tests were simple batch sorption experiments in which 0.5 grams of the granular sorbent materials (as received) were exposed to 5 mL of filtered groundwater containing from 11 to 23 nanocuries per mL of 85Sr tracer. The specific activity of the tracer was 15.54 nCi/ng of Sr. After several days the liquid was sampled with plastic disposable syringes, filtered through 0.45 micron syringe filters and analyzed on a high purity germanium gamma-detector. All results were corrected for the mass of the counting sample as well as the mass of solid and the total liquid in each experiment. Experiments were conducted in duplicate for 20 different materials.

Column Studies

Two sizes of plexiglass columns were used depending on the grain size of the material they were to contain. Columns used for two coarse grained materials had average internal dimensions of 3.17 cm in diameter and 3.8 cm in length. For the fine grained materials the diameter was 1.52 cm and the length averaged 6.4 cm. Each column was set-up vertically with the inlet at the bottom.

Groundwater was prepared for these experiments by means of a soil column (with a cross-sectional area of 15.5 cm2) that contained site-specific soil from the geological unit of interest. Filtered site groundwater was pumped through the column at the rate of about 400 mL per day. The effluent from this column had a slight yellow color and a pH around 8.1. Turbidity measurements using method # 750 of the Hach Kit indicated very little turbidity; 2 Formazin Turbidity Units (FTUs). To this water the 85Sr tracer was added so that the count rate of the groundwater, at the beginning of this set of experiments, was about 1.7 nCi/g (~100 cpm/mL). For one of the clinoptilolite columns (CH-A) a different liquid was used; filtered groundwater plus tracer.

The prepared groundwater was pumped through the columns with a Gilson Minipuls low-speed peristaltic pump. Adjusting flow to the porosity of the material, attempts were made to maintain flow-rates that are typical of a rapidly moving groundwater (~40 cm/day). For the small diameter columns this is equivalent to 25 mL/day and for the larger diameter columns, it is about 120 mL/day. Effluent was collected daily in pre-weighed polyethylene bottles. These were weighed to determine the daily flow-rate. Aliquots were taken for radionuclide counting.

85Sr on the Solid Phase of the Columns

After periods of time ranging from 28 to 46 days, flow through the columns was stopped. Each column was then sectioned for analysis of the solid phase for 85Sr activity. This was done by extruding the material from the column in small increments that were wiped into pre-weighed counting bottles. Attempts were made to obtain as small and spatially discrete samples as possible, with distance measurements being made every few samplings. Sampling and measurement were particularly difficult for the two coarse grained materials since the material was not cohesive (the fine grained materials held together quite well) and the larger grains necessitated larger sampling intervals. Each sample was weighed and analyzed for 85Sr on a high purity Ge gamma-detector. For the fine grained materials, essentially all of the material was recovered during the sampling, allowing a relationship to be determined between sample weight and thickness of the sampling interval. For the coarse material, this was much more difficult because some grains were inevitably lost during sample collection. Since the grains at the bottom of each column (the inlet) contained the most activity and this activity changed rapidly with distance, loss of a few grains could be significant. Nevertheless, the profiles of activity as a function of distance, even for the coarse grained columns, are quite reasonable.

Desorption Treatment of Barrier Materials

Two zeolites, mordenite and clinoptilolite, were selected as possible barrier materials based on the column tests. These materials, after having served as permeable barriers, will contain appreciable activities of 90Sr. While the final disposition of the barrier materials has not been decided, it may be necessary to

remove as much activity as possible prior to disposal. It may also be desirable to regenerate the barrier. Experiments were performed to determine what solutions would be optimum for regenerating the materials. Two solutions were selected as potential regenerants; NaCl solution and HCl. Each was tried at 1, 2 and 4 molar concentrations. Six sorption experiments were set up for each of the two materials. After allowing them to sorb 85Sr, the liquid was decanted and counted. Test tubes containing the wet solids were reweighed to determine the quantity of water remaining; then the treatment solutions were added to the tubes. After standing overnight the solutions were sampled, filtered, and counted to determine the activity of 85Sr returned to solution.

RESULTS

Screening Studies

Results of the screening study to determine the materials most effective at removing 90Sr from the groundwater are summarized in Table I. This table shows only the better performing materials that were tested. The most effective were natural zeolites and metal oxides, particularly activated alumina and MnO.

Selection of Materials for Column Tests

Based on results of the screening tests, five flow-through column experiments were performed for four materials:

1. clinoptilolite (CH 20 x 50) - 2 columns; CH-A and CH-B
2. clinoptilolite (CH 5 x 20)
3. mordenite (8 x 14)
4. activated alumina (14 x 48)

The clinoptilolite and mordenite are natural zeolites; the activated alumina is a synthetic product. The numbers in parentheses beside each material refer to grain sizes. For example, an 8 x 14 product refers to particles that pass a No. 8 sieve and are retained on a No. 14 sieve.

The MnO was dropped from consideration since it contains trace amounts of Cu, As, Pb and Cd. All of the chabazites were eliminated based on material costs. Depending on the supplier, prices for this type of zeolite can exceed $ 1400/m3. In comparison, the price for clinoptilolite (CH) is about $208/m3. Results from a column test on "as received" mordenite (14 x 40) indicated that this material contained large quantities of fines that could clog pore spaces within and downstream of the barrier. Because of this result, the coarser grained mordenite (8 x 14), with less fines, was selected for column testing.

Column Studies

Results of the column studies are summarized in Table II. Only alumina actually attained breakthrough. This took place after 37 days; consequently no further consideration was given to this material. Both coarse grained zeolites were about to breakthrough, as indicated by analysis of the solid phase of the column.

Desorption Treatment

Results of the desorption treatment are shown in Table III. The 4 Molar NaCl solution was effective in desorbing 85Sr from the mordenite while a 1 to 2 Molar HCl solution was more effective for the clinoptilolite. This appears to be an advantage in the use of the clinoptilolite. Other solutions may be more efficient at removing Sr; this has yet to be determined.

MODEL SIMULATIONS OF LABORATORY COLUMN DATA

The laboratory column experiments provided a distribution of concentration or activity of 85Sr on the sorbent media at discrete intervals across the columns. Measuring concentration on the solid phase was necessary since breakthrough in the effluent was not observed during the zeolite column runs. Knowing the length of time for each column test and the average column discharge rate (along with other parameters), an analytical model was fitted to the laboratory data by adjusting the distribution coefficient (Kd) within the model until an approximate concentration match was observed on the solid phase, near the source end of the modeled column. When this was achieved, the dispersivity within the model was adjusted until a similar concentration profile was observed over distance. In this way, a unique Kd was derived for each material under dynamic flow conditions, similar to what would be experienced in the surficial sand and gravel.

Figure 1 presents two examples of the laboratory column data versus results generated by the calibrated analytical model. Significant model input parameters, such as Darcy velocity (Vd) and dispersivity, are listed in Table IV. In all cases, the model fit with the lab data is considered good. It is pointed out, that some

experimental error cannot be avoided with this type of laboratory experiment, i.e., counting on discrete sections of material extracted from the columns. As a result, a slight adjustment in experimental positioning data, to compensate for possible counting-distance errors, provides a better fit of the model. Derived Kd's from the model simulations are given in Table IV. It is noted, that the Kd's derived from model simulations of the column test data are significantly different (i.e., lower) compared to those determined in batch, "closed-system" laboratory tests. It is believed that the column tests, coupled with the analytical model simulations, provide the most useful information regarding sorptive behavior of materials under dynamic, "open-system" conditions.

The mordenite surprisingly provided the lowest Kd, which was approximately 25 times less than average values obtained from standard laboratory batch tests (on a fine grained mordenite (14 x 40)). It is apparent that the coarse grain-sized products will have less sorptive capacity under dynamic, open-system conditions. On the other hand, the fine-grained CH material (i.e., 20 x 50) yielded Kd's only slightly lower than those determined in standard batch tests (~760 mL/g). Considering the complex nature of the testing, the similar results obtained for Tests A and B provide a certain level of reliability and verification of the test methodology and the model simulations as well.

MODEL SIMULATIONS OF BARRIER PERFORMANCE

Figure 2 is an example of a model simulation which illustrates 90Sr loading on the CH 20 x 50 material as a function of time and distance. Model input parameters for porosity, density, and Kd, are listed in Table IV. A longitudinal dispersivity of 5% was used for the media in all of the simulations. This value equates to 6 cm in reference to a total barrier thickness of 120 cm. A seepage velocity of 4.6 x 10-4 cm/sec was used for all cases, which approximated in-situ field conditions. (Seepage velocity equals Darcy velocity [Vd] divided by the material porosity.) A barrier thickness of 120 cm was evaluated over a 12 year maximum time frame.

An initial 90Sr activity in the groundwater of 5.0 x 10-12 Ci/mL (5000 pCi/L) was used in all of the simulations. This was based on current groundwater sampling data. In reviewing gross beta increases in monitoring wells from 1990 through most of 1994, an average yearly increase by a factor of 1.3 was postulated. This multiplier accounts for a 90Sr migration rate of approximately 14.5 m/yr. Based on these analyses, it was estimated that it would take approximately 11 years before 100,000 pCi/L (90Sr) would be observed in groundwater at the proposed barrier wall location. Table V shows times to breakthrough of 90Sr at the downstream end of a 120 cm thick barrier. Breakthrough times for 10 pCi/L and 100 pCi/L were evaluated for four materials. The CH 20 x 50 provides the greatest attenuation followed by a "blended" CH 5 x 50 product. Table VI shows 90Sr activity on the barrier material for 7 and 10 year operating periods. For each period, an average activity was calculated for the entire 120 cm as well as a smaller thickness pertaining to the first 30 cm at the upstream end (higher activities will undoubtedly be present within this area as indicated by the model simulations). The activities were converted from Ci/g to Ci/cc for comparison to 90Sr limits as listed in 10CFR61. This conversion is accomplished by multiplying by the density of the material in question. The Class A limits for 90Sr as listed in 10CFR61 are 0.04 Ci/cu.m (4.0 x 10-8 Ci/cc). The data on Table VI indicate that for the CH 20 x 50 material, depending on how the barrier is evaluated, activity levels in the upstream portion of the filter media could exceed Class A limits within 10 years.

SUMMARY AND RECOMMENDATIONS

The summary information presented in Tables V and VI indicates that the clinoptilolite (CH 20 x 50) material will attenuate 90Sr for longer periods of time compared to the other materials that were evaluated. The estimated performance of the clinoptilolite (CH 5 x 50) is also viewed as being adequate to good. In spite of acceptable levels of 90Sr activity that could be allowed in the groundwater downstream of the barrier wall, it appears that the maximum life expectancy for the material (i.e., prior to removal with possible regeneration) will be approximately 10 years based on allowable limits of 90Sr loading on the in-place barrier. This time period is a function of site-specific conditions and assumptions regarding: increases of 90Sr in the groundwater over time, flow velocities and dispersivity. As a result, even if materials with higher sorptive capacities for 90Sr were used (which would be more expensive), they too would need to be replaced after the same amount of time.

The clinoptilolite (CH) material is presently being pursued for use within the proposed barrier wall. Based on model simulations, a minimum barrier thickness of 1.2 meters was recommended. The selection of grain size (i.e., 20 x 50 vs. 5 x 50) will be based on final design considerations and further limited laboratory testing to address concerns pertaining to in-situ hydraulic performance.

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ION CHROMATOGRAPHY COUPLED WITH ON-LINE AND OFF-LINE RADIATION DETECTION FOR THE RAPID ANALYSIS OF RADIONUCLIDES IN REACTOR COOLANT WATER

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ABSTRACT

A study was performed to evaluate the application of ion chromatography and radiation detection to characterize the radionuclides in nuclear reactor coolant water. In this approach, cationic radionuclides in an aqueous sample are loaded onto an ion exchange column and subsequently separated by an appropriate set of chemical eluents. The resulting effluent passes sequentially through an on-line scintillation flow-cell for quantitative measurement. Further identification/analysis was performed on the collected fractions by gamma-ray spectroscopy with a high-purity germanium detector and electron/beta spectroscopy with a liquid scintillation counter. These off-line techniques assist in the identification and quantification of radionuclides that are either below the detection limit of the on-line detection system and/or not present in the calibration standard. The ion chromatography technique was found useful in identifying common non-gamma emitting (55Fe, 63Ni, 89Sr, 90Sr and 90Y) as well as gamma-ray emitting (58Co, 60Co and 140La) fission and activation products found in nuclear reactor coolant water. Preliminary quantitative results for some of the radionuclides are also presented.

INTRODUCTION

Currently, most commercial nuclear facilities find compliance with regulations that require characterization of non-gamma emitting radionuclides in reactor related samples expensive and problematic. The regulatory requirements for radiological characterization of wastes from commercial nuclear facilities is typically comprehensive in that it must account for fission products, activation products, and transuranics with requisite sensitivities on the order of 10-1 - 102 MBq m-3 (10-5 - 10-2 mCi mL-1) (1). High resolution gamma-ray spectroscopy produces efficient and accurate results for radionuclides that emit measurable gamma rays. However, other important radionuclides either do not emit gamma rays or have emission energies or frequencies that are too low for gamma-ray spectroscopy to be practical. These are operationally defined here as "non-gamma" emitters and include pure beta emitters such as 63Ni, 89Sr, 90Sr and 99Tc, electron capture radionuclides such as 55Fe, and transuranics such as 239Pu and 242Cm. Radiochemical separation techniques for the

non-gamma emitters typically involve long, complex procedures which are costly and which tend to be less accurate due to the low and variable product yields. There is interest in the development of rapid analytical techniques for the analysis of samples containing activation/fission products as well as actinides. One system that shows potential for characterization of "non-gamma" activation and fission products is an ion chromatography system sold under the trade name of ANABET (ANalyis of Alpha Beta and Electron capture Technology (Bradtec, United Kingdom)). The ANABET system was originally developed specifically for anionic (e.g. 14C, 99Tc, 129I) and cationic (e.g. 55Fe, 60Co, 63Ni, 89,90Sr, 90Y, and 134,137Cs) activation/fission products (2). This ion chromatography system was further characterized and used to quantify activities of cations and anions in aqueous effluents (3). In addition, an elution scheme was developed to separate Np, Am, Cm, Th, Pu, and U from a relatively high-purity, high-activity solution (4). The actinide elution system was used to analyze the above six actinides that were spiked into an synthetic natural water (distilled water containing humic acid, dissolved ions and radium). The resulting chromatogram under these conditions resulted in poor resolution of the elements and poor quantification (5). Reboul and Fjeld (5) indicate that significant pretreatment of samples is required for samples that are not of high-purity. Reported previously (6) was the implementation of the ion chromatography system and the evaluation of its performance for characterizing synthetic reactor coolant water. With this scoping study complete, this paper presents the results of the ion chromatography system on actual reactor coolant water from a pressurized water reactor.

EXPERIMENTAL METHODS

Apparatus

The ANABET system used in this study has been described previously (2,3). Ionic radionuclides in an aqueous solution are concentrated on an ion exchange pre-concentration column and subsequently removed to a separation column with chemical eluents. Through the appropriate selection and sequencing of the eluents, elemental separation of the radioactive constituents is achieved. The effluent stream then passes through a flow-cell scintillation cell packed with polycrystalline calcium fluoride which detects the radioactive constituents. Peaks in the chromatogram occur as each "packet" of radioactivity passes through the detector and elemental identification is based on elution time. For background reduction, the scintillation cell is located between two photomultiplier tubes operating in coincidence. The active volume of the scintillation cell is 0.4 mL with a background count rate of approximately 1.5 counts per second. The eluent flow rate is 1 mL min-1 and the effective counting time is 24 seconds. Data collection and reduction is accomplished through an external personal computer. Count rates are determined by integrating the net areas (i.e. background subtracted) and dividing by the count time.

In the work presented here, off-line analysis was performed on some samples so that isotopic identification of peaks in the chromatogram could be made by gamma spectroscopy (using a high purity germanium detector) and electron/beta spectroscopy (using a liquid scintillation counter). For the off-line analysis, a fraction collector (Model Retriever IV, ISCO Inc., Lincoln, NE) was connected to the waste stream to collect fractions at 1 minute intervals. Fractions were analyzed by liquid scintillation counting (Model 1415, Wallac Oy, Finland) and high resolution gamma-ray spectroscopy (Model IGC 17, Princeton Gamma Tech., New Jersey) for qualitative identification of the radionuclides present in the fractions.

Interferences

The principal potential chemical interferences in the reactor coolant were boric acid, lithium hydroxide and hydrazine. These chemicals have a very minor effect on the radionuclide elution time (7). Cesium is known to interfere with the elution time and detection of radionuclides by masking the presence of radionuclides such as iron (7). The preconcentration column was washed, after sample loading, with 0.05 M nitric acid in an attempt to remove monovalent cations (i.e. cesium) Two wash times were used 10 minutes (1 mL/min) for sample 2 and 20 minutes (1 mL/min) for sample 2.

Analysis of Reactor Coolant Water

Two 0.5 liter aliquots of coolant water, one from Reactor Unit 1 and the other from Unit 2, were obtained from Duke Power Company's Oconee Nuclear Station. Due to limited sample volume and relatively high minimum detectable activity of the ion chromatography system, the coolant water from units 1 and 2 were combined to form a

composite for the three analyses. Analytical methods for pretreatment of the samples are summarized in Table I.
The first sample (RCS040894) was a 750-mL aliquot that was reduced to 70 mL by evaporation and then filtered through a 0.45 mm filter to remove suspended solids which could block the IC columns. After loading the sample onto the preconcentration column, the column was washed with 0.05M nitric acid for 10 minutes using a flow rate of 1 mL/min in an attempt to eliminate monovalent cations (e.g. cesium).
The second sample (RCS041294), a 50-mL aliquot, was loaded onto the preconcentration column without sample pretreatment and the column was subsequently washed with 0.05 M nitric acid for 10 minutes (1 mL/min). Because of the smaller sample and the need to identify some of the unknown elution peaks that were observed in the sample 1 chromatogram, both on-line and off-line analyses were performed.
The third sample (RCS041994), an 84.5 mL aliquot, was filtered through a 0.45 mm filter prior to being loaded onto the preconcentration column. Again the column was washed with 0.05 M nitric acid, but this time for 20 minutes. As with sample 2, the waste stream was collected with a fraction collector for further analysis.

RESULTS AND DISCUSSION

Analysis of Reactor Coolant Water

Figure 1 displays the chromatogram of sample 1 which had six well resolved peaks at approximately 5, 9, 10, 12, 33, and 43 minutes and a small broad peak at 53 minutes; these peaks are labeled peaks 1 through 7 respectively. Identities of the peaks 1, 3, 5 and 6 were not immediately known because the elution time was not consistent with any cations in our standard solution. Peaks 2, 4 and 7 are due to 63Ni, 55Fe, and 89Sr respectively, based on previous laboratory studies (7). Peak 3 could be due to either cesium or cobalt. To identify the elements of these unknown elution times, subsequent samples were fraction collected for off-line analysis.
Samples 2 and 3 were loaded and fractions were collected to resolve the identity of the peaks as well as the effects of sample pretreatment. As noted in Experimental Methods, sample 2 was washed with 0.05 M nitric acid for 10 minutes and sample 3 was washed for 20 minutes in an attempt to remove cesium. It was determined from sample 2 that a 10 minute wash was inadequate and therefore increased to 20 minutes for sample 3.
The identify of peak 1 (approximate elution time of 5 minutes) is unknown since it failed to appear in either the chromatograms from samples 2 or 3. It is assumed that its concentration was sufficiently low to be undetected during the on-line and off-line.
Off-line analysis using liquid scintillation yielded the spectrum for peak 2 presented as Fig. 2. The spectrum is consistent with a beta emitter with an endpoint energy in the 0.06 to 0.08 MeV range. This analysis confirms the assumption that the fraction contains 63Ni, which emits a beta particle with an endpoint energy of 0.067 MeV.
Sample 3, which received the 20 minute wash, revealed peak 3 to be cobalt. The analysis of the fraction was performed using the gamma-ray spectrometry to reveal both 58Co and 60Co. The fraction from sample 2, which received a 10 minute wash, showed considerable cesium (134,137 Cs) activity which may have masked any cobalt. Cesium was also detected in peak 4 of sample 2.
The liquid scintillation spectrum for peak 4 (Fig. 3) appeared to have two components. There was a large peak at 0.01 MeV that was likely due to low energy monoenergetic x-rays from 55Fe. In addition, there was a beta spectrum with an endpoint energy in the 0.4 to 0.5 MeV energy range and a smaller peak in the 0.5 to 0.6 MeV range. This signature suggested 137Cs which emits a beta particle with an endpoint energy of 0.514 MeV and conversion electrons of 0.620 and 0.656 MeV.
Similar analyses of the liquid scintillation spectrum for peak 5 suggested a beta emitter with an endpoint energy in the 1.3 to 1.5 MeV range. This is consistent with 140La, which emits a beta particle with an endpoint energy of 1.4 MeV (79%). The gamma-ray spectrum of this fraction confirmed the presence of 140La. The spectrum also revealed 134,137Cs to be in this fraction. Surprisingly the gamma-ray spectrum of the peak 6 also indicated the presence of 140La. Since the fractions were analyzed a few days after they were collected, elution of 140Ba (th = 12.8 d) at either peaks 5 or 6 would explain the presence 140La in both the peaks. For 140La to be present, it must be in secular equilibrium with 140Ba since the half-life of 140La (th = 40 h) is too short to be present 60 days after withdrawal from the reactor. Furthermore, since 140Ba and 140La are both cations and since

ion-chromatographic separations are based on chemical differences between elements, one could speculate that one of the peaks may be due to 140Ba. However, gamma rays from 140Ba were not seen in either of the fractions. One possible explanation is that the low energy gamma rays of 140Ba may have been masked by the cesium gamma rays. The decay rate of the fractions (peaks 5 and 6 in Fig. 1) were quantified by gamma counting them at five to six days intervals. The half life determined by this method was approximately 50 hours, indicating the presence of only 140La.
Liquid scintillation analysis of peak 7 (Fig. 4) of samples 2 and 3 indicated the presence of 89Sr (Emax = 1.480 MeV). This confirms the elution time characteristics of strontium.

Quantification of Non-Gamma Emitting Radionuclides

Non-gamma emitting radionuclide concentrations were calculated by integrating the area under the peaks from the on-line chromatograms and using the efficiencies determined in earlier studies (7). These analyses were compared with those of an independent contractor, Rust Federal Inc.-Clemson Technical Center. The 63Ni concentration (0.11 MBq m-3) determined by ion chromatography was within 50% of the independent analysis. The 89Sr concentration (0.78 MBq m-3) was within 1% of the independent analysis. These results are encouraging, but further test and more elements are warranted.

CONCLUSIONS

Based on these results ion chromatography and on-line scintillation counting as implemented has considerable potential for the radiological characterization of non-gamma emitting radionuclides in primary reactor coolant water. The advantages of this technique over traditional methods are minimal sample pre-treatment, analysis speed, process automation, on-line analyses and collection of fractions for isotopic identification. The analysis speed (60 minutes for cations) is the principal advantage, reducing analysis time by as much as factor of 100 over traditional methods. The limitations of this method include interference of cesium isotopes and low detection sensitivity due to a short counting time (24 s). Further investigation is required to develop a method for removal of cesium from reactor coolant water and to decrease the minimum detectable activity.

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AN INDOOR RADIOLOGICAL CHARACTERIZATION SURVEY USING AUTOMATED POSITIONING TECHNIQUES

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ABSTRACT
Radiological surveys are a time consuming component of the total decommissioning process. Manual gridding is the common and accepted method used by survey teams to give spatial significance to the measured levels of radiation found on site-specific surveys. However, the gridding process requires substantial man-hours and is not conducive to real-time data analysis and assessment. In addition, several forms, pertaining to the results attained during the survey, are required to be completed manually as a part of the final report necessary to complete the decommissioning process.
An automated, computer-based system of performing radiological surveys has been developed and tested at an indoor site. The system was used to determine spatial data automatically for a characterization survey at a formerly utilized radiochemistry lab. Thus, the labor-intensive gridding technique was eliminated from the survey/decommissioning process in total.
Two methods of obtaining positioning data were piloted. The first method involved the use of an ultrasonic range finder interfaced through a PCMCIA data acquisition board to a notebook computer. The second method utilized a relative positioning technique called "mouse-traverse" to resolve sampling locations and subsequent measurements. In both cases, the Ludlum 2350 datalogger/ratemeter/scaler, coupled with the appropriate detector, was used to measure the activity at the sampled locations. As a control, the radiochemistry lab was characterized using traditional instrumentation and manual gridding.
The three methodologies (i.e., ultrasonic ranging, mouse traverse, and gridding) were then compared on merit. The methods were compared on such attributes as time allocation, costs, accuracy, precision, resolution, etc. It was found that, when compared to the manual gridding method, both of the automated methods reduced the time necessary to perform the radiological survey several-fold. However, the initial costs, instrument lack of ruggedness, and range limitations are the major drawbacks to the automated approach.
In conclusion, a technique for providing automated positioning to the radiological survey process was elucidated. The integrated system, whether using either the mouse-traverse or the ultrasonic ranging positioning method, reduced the time to perform the radiological survey. In addition, the data handling, control, and management capabilities of the notebook computer made it possible to manipulate and report survey results in a more timely fashion. However, system modifications need to be made to increase user friendliness, ruggedness, and ranging capabilities.
INTRODUCTION
The objective of the indoor radiological survey is to identify the levels of radiation in a potentially contaminated room. The levels are then compared with the existing regulatory criteria and an assessment is made on whether or not the site will require further remediation before being classified as uncontrolled (1,2,3,4,5). However, surveys are currently performed manually with somewhat primitive recording and spatial measurement techniques (6).
The purpose of this paper is to discuss the progress of an automated radiological survey technique developed for performing indoor site surveys. An experimental design was devised that included a characterization survey study of a potentially contaminated building. The rooms within the building varied in size from approximately 6 m2 in area up to 100 m2 in area. The building was previously used as an instrumentation laboratory. The expected contaminants were alpha emitters from the U-238 decay chain (i.e., U-238, U-234, Th-230, etc.).
Specifically, the main objective of performing the surveys was to test two automated positioning techniques (i.e., ultrasonic positioning and mouse-traverse positioning) for field viability. The experimental design included statistical measures of comparing the automated methods to the traditional, manual methods of performing the indoor survey. The purpose of the rest of this paper is to elucidate the methods and procedures utilized during the evaluation, to present the results on the accuracy

and repeatability of the two automated techniques, and to provide a real-time data output generated by the total automated system.

METHODS

Figure 1 shows the total automated system as designed and piloted. Figure 2 gives a detailed illustration of the necessary system interfaces. The main components of the survey apparatus included a NEC 486 notebook computer with a configured National Instruments DAQ-700TM PCMCIA data acquisition card, a Ludlum Model 2350 Ratemeter/DataloggerTM, a Ludlum Model 43-89 Alpha Scintillation DetectorTM, and two spatial positioning assemblies (i.e., ultrasonic positioning and mouse-traverse positioning).

The ultrasonic assembly included a mechanism for accurate positioning as well as a Piezo-type transducer/circuit board. The mouse-traverse technique employed a computer trackball that rolled along the surface of the floor (7). Thus, this technique is a relative positioning methodology that must be zeroed at point (0,0). The surveys were limited to floor locations in the various rooms.

The necessary computer protocol modifications were made based on the site data and the techniques previously employed on the site survey. While the normal process of performing indoor site surveys was to take five samples from each 1 m2 survey unit, the system protocol was written to include nine sample points per survey unit. This not only would increase data confidence, but would also provide a more energy efficient means of collecting the data. In addition, the nine points could help to serve as justification for not needing to sample in the difficult to get to corner and wall areas (i.e., only six points are taken in the one meter square area defined by one or more walls/surfaces).

The Environmental Survey software, written in Microsoft Visual BasicTM, was developed to include a spreadsheet that identified spatial coordinates, cpm/100m2, dpm/100m2, survey unit average, and a 95% confidence comparison value. In addition, the software included algorithms for detector efficiency, minimum detectable activity (MDA), the survey unit mean, the survey unit standard deviation, and the 95% confidence value. Code was written that would not allow a value of less than the calculated MDA to be logged. Computer forms (screens) were also created that provided for survey background counts and detector/ratemeter configuration. Since the WindowsTM software environment was used, dynamic links between the various software packages made it easy to move data to and from various software programs (e.g., Visual Basic to Microsoft AccessTM, Access to Stanford GraphicsTM, etc..)

The algorithm entered for determining the MDA was as follows:

Eq. (1)

where

MDA	=	activity level in disintegrations/minute/100 cm2
B	=	background rate in counts per minute
t	=	counting time in minutes
E	=	detector efficiency in counts per disintegration
A	=	active probe area in cm2

Since the counting time is usually set for one minute and the active probe area of the Model 43-89 detector is 100 cm2, the MDA is primarily a function of the site background rate.

The equation used to determine the 95% confidence comparison value was as follows:

Eq. (2)

where

95% CCV	=	95% confidence comparison value (positive side)
1.86	=	the t-value for sample size equal to 9
s	=	the standard deviation for the survey unit
n	=	the number of samples per survey unit

For this survey, the value for n is equal to nine and the calculation for the mean and standard deviation is dependent upon the sample measurements for each survey unit. The standard statistical equations used to determine the sample mean and the sample standard deviation are included in the computer code to determine the appropriate 95% confidence comparison value.

Prior to leaving for the site, a calibration curve was generated by using a check source of thorium-230. In order to determine the ideal high voltage value to use for the survey, an optimum threshold value must be first decided upon. A good "rule-of-thumb" on determining the threshold is to use a value that, with several subsequent measurements, a background rate of not greater than 3 dpm is found. The

threshold value that met this criteria for the thorium source was 30 mV. At this threshold, the calibration curve resulted in a plateau midpoint of 850 volts. This value would be used as the survey high voltage and would be entered during the Ludlum 2350 configuration routine.

Upon arrival at the facility, it was resolved that alpha characterization surveys were to be performed on floor areas of 6 m2, 12 m2, 25 m2, and 42 m2. Due to the security and time constraints at the facility, the manual method was performed on-site by the health physics technicians and observations were made for comparison. In addition, the mouse-traverse technique and the ultrasonics positioning method were both utilized to survey the 6 m2 room. However, compatible "clean" rooms off-site were used to obtain the data for the other three areas.

The operational check-out and calibration procedures were completed prior to beginning the first survey. A 20,000 dpm calibrated check source of thorium-230 was used to determine the efficiency of the detector. The procedure included taking the mean of five subsequent one minute counts from the detector and dividing it by the calibrated value. Then, by multiplying this resultant by 100, the survey detector efficiency was attained. This value was found to be 19.6%.

A survey background rate was determined both manually and automatically. Nine points were taken from adjacent, unaffected areas. The mean of these nine samples was then used to set a baseline for determining net sample counts. In addition, the background mean is an important parameter used in the calculation of the detector MDA. The time required to take the background readings first manually and then automatically was essentially the same.

After a detector efficiency was determined from a calibrated thorium-230 check source, two survey technicians proceeded with a calibrated field survey meter (with attached alpha detector) to perform a quick alpha scan of the rooms to determine potentially affected areas. The technicians then began to grid the rooms into survey units of approximately one square meter. The technicians used a calibrated field measuring tape and duct tape to grid the room. Five sample points were taken (i.e., the four corners and the midpoint) to define each survey unit. In addition, one sample location was repeated for every two survey units completed. This is in compliance with the 10% quality assurance criteria recommended for the detector. Since one of the primary contaminants was thorium-230, which has a rather stringent release criteria (i.e., 100 dpm/100 cm2 average), the count time was one minute for each sample. One technician operated the instrument while the other manually documented the readings. The detector was moved around the sampling area in a "snake-like" fashion for the duration of the one minute sampling. The manual surveys were observed for time requirements and technique. After a survey was complete, the technicians calculated the required values.

After the background rate and detector efficiency was determined, each of the automated characterization surveys began with an initial tape measurement of room dimensions. The procedures for operational check-out, site calibration, and standard operating procedures are detailed in another document and are beyond the scope of this paper. A special effort was made to determine values for repeatability (both spatial and magnitude) and spatial measurement error/accuracy. In order to do so, some of the sampling locations had to be specified as either control or repeat sampling locations. The control data points are defined as those points used as a means of showing spatial repeatability and accuracy. The control points are marked with a pencil or a piece of tape, and the surveyor returns to this location, during the course of the survey traverse, to determine the level of positioning repeatability. In addition, these control points are compared to actual tape measurements for accuracy.

A repeat sample point is one in which the surveyors just repeat the count with the detector/ratemeter. The repeat points are necessary as a quality control measure, required to meet the 10% repeat point criteria recommended by ORISE. The repeat points are usually taken at the midpoint location on the survey lines of adjacent survey units.

Specifically, the experimental design followed for the automated techniques involved data gathering methodologies aimed at gaining an incite on the actual process viability. As was hinted to in the previous paragraph, spatial accuracy, temporal efficiency, and measurement repeatability are critical indicators of process field viability. In order to provide some indication of the viability of the automated positioning techniques for field use, the values for spatial accuracy and

repeatability for the three meter by four meter room were taken. The survey traverse for the 12 m2 area (i.e., 3 meter by 4 meter room) is shown pictorially in Fig. 3. Figure 3 defines the twelve survey units, with specific identification made to the sampling locations that were used as control points (CP) and repeat points (RP).
A viability and time efficiency comparison, between the traditional manual method and the two automated techniques, was performed. This was accomplished by defining a man-hour comparison coefficient, the dependent variable, and plotting it versus the room floor area, the independent variable. Thus, an equation can be determined from the data taken from the smaller rooms to help describe the time component for surveys of larger areas. The man-hour comparison coefficient, k, was defined as follows:
Eq. (3)
where
k = man-hour comparison coefficient
ta = total man-hours for automated survey
tm = total man-hours for manual survey
In addition, to show the real-time data analysis capabilities of the system, a real-time profile of spatial and magnitude components for a 2 meter by 3 meter room was generated on-site. The data was taken from the one minute alpha counts at each floor location and plotted three-dimensionally by linking the database with Stanford Graphics. Finally, a total area assessment was made by determining the population mean and standard deviation for the 6 m2 area.
DATA AND RESULTS
The main purpose of this research was to show that automated positioning could provide an effective and efficient means of replacing the time consuming efforts involved in the manual gridding of indoor survey sites. Calculations for spatial accuracy and spatial repeatability provide two objective measures for the evaluation.
The mean accuracy for the ultrasonic positioning technique was 99.6% in the x-direction and 98.5% in the y-direction. The standard deviation was calculated to be 1.7% in the x-direction and 1.5% in the y-direction. Thus, at the 95% confidence level, the accuracy for the ultrasonic positioning technique can be stated as 99.6% +/- 0.8% in the x-direction and 98.5% +/- 0.6% in the y-direction. The 95 % confidence levels for these and the following spatial parameters were calculated by using the following equation:
Eq. (4)
where
P95 = the 95% confidence parameter desired
X-bar = the mean of the data
s = standard deviation
n = the number of samples
The mean calculated for the relative percent difference (RPD) between the control points was 0.3% in the x-direction and 0.1% in the y-direction. The standard deviation was found to be 0.5% in the x-direction and 0.5% in the y-direction. Thus, at the 95% confidence level, the RPD for the ultrasonic positioning technique can be given as 0.3% +/- 0.2% in the x-direction and 0.1% +/- 0.2% in the y-direction.
For the mouse-traverse technique, the mean accuracy calculated for the x-direction at the 95% confidence level was 100.1% +/- 1.4% while the 95% confidence level mean in the y-direction was found to be 100.7% +/- 2.0%. The 95% confidence level RPD for the mouse-traverse technique in the x-direction was calculated to be 1.7% +/- 0.7%. The corresponding 95% level for the RPD in the y-direction was determined to be 2.7% +/- 1.2%.
The positioning parameters for both the ultrasonic positioning technique and the mouse-traverse technique are summarized in Table I.
Survey Time Comparison
In order to provide some indication of the time efficiency of the automated techniques, a test was needed that would allow for the comparison of the temporal components of both of the automated survey techniques against the traditional, manual survey. In order to do so, it was necessary to define a man-hour comparison coefficient. In essence, this is just the man-hours required to perform an automated survey of a particular divided by the time required to complete the survey with the manual technique.
Survey areas of 6 m2, 12 m2, 25 m2, and 42 m2 were chosen to perform controlled

surveys utilizing each of the three of the techniques (i.e., manual, ultrasonic, and mouse-traverse). The time component for each of the surveys was recorded and the comparison coefficient was plotted as a function of survey area. An extrapolation was made and an equation was estimated that would help to describe the time efficiency of the automated techniques for very large rooms. Since the relative time component required for gridding and manual calculations would be greater with increasing survey area, it was expected that the plot would follow a negative exponential path.

By extrapolating the plot, it appears that both curves are approaching approximately 0.40. This would indicate that there is a finite limit to the time savings that could be attained by the automated techniques. In essence, the data indicates that the automated processes can result in nearly twice the sampling points in forty percent of the time.

The following equation closely emulates the relationship between the coefficient and the room size:

$$y = 0.9 x -0.16 \quad \text{Eq. (5)}$$

where

y = the man-hours comparison coefficient
x = the survey area (square meters)

From the equation it can be discerned that the optimum man-hours coefficient is approximately 0.40 for the largest of survey areas. Thus, by further evaluation of the man-hours comparison coefficient, it can be determined that the system can facilitate the survey of large rooms at about two and one-half times the rate of that of the manual survey. In addition, the automated survey procedure includes nearly twice as many data points for statistical analyses and site assessments.

The site mean and standard deviation, calculated at the 95% confidence level, was 69 dpm/100 cm2 and 7 dpm/100 cm2, respectively, for the 6 m2 area surveyed using ultrasonic positioning. For the 6 m2 area surveyed using the mouse-traverse technique, the site mean was determined to be 64 dpm/100 cm2 and the standard deviation was calculated to be equal to 10 dpm/100 cm2. These values are less than the 100 dpm/100cm2 release criteria established for thorium-230. All of the unit means were within the criteria.

A three-dimensional profile, generated from the spatial and magnitude data sampled during the survey, is given in Fig. 4. Figure 4 shows the three-dimensional plot of the data from the mouse-traverse survey of the 6 m2 area. It is possible to generate this type of drawing anytime during the survey process, thus, allowing for real-time, analytical assessments.

CONCLUSIONS

In summary, the two automated positioning techniques, ultrasonic positioning and mouse-traverse positioning, provided viable alternatives to the manual survey methods currently employed at indoor radiological sites. The total automated system made it possible to make accurate assessments, of both space and magnitude, in real-time. In essence, the system, as designed, eliminated the need for manual gridding and manual calculations. Thus, since the time requirements for gridding and performing site calculations are a major component of the survey process, the system made it possible to complete each survey more efficiently.

Some of the system problems identified during the surveys include:

1. Durability limitations to the trackball/mouse design
2 .The range capabilities of the ultrasonic device was limited to 10 meters.
3. It was difficult to get the system apparatus into "tight" places.
4. The learning curve for the effective use of the system software.
5. Due to having only one serial port, a data selector was necessary.
6. The positioning techniques required substantial QC/QA measures.
7. Rooms with obstacles or "rough" floors present positioning problems.

A list of the major conclusions drawn from the field surveys were as follows:

1. Both automated techniques provide an effective and efficient means for facilitating the indoor radiological survey process.
2. Even for smaller areas, the automated techniques are viable.
3. For larger area rooms, both automated techniques provide the capability of sampling nearly twice as many points as that of the manual technique in 40% of the time.
4. The temporal characteristics of both of the automated techniques are essentially the same.

5. An equation can be developed that describes the temporal characteristics of the automated methods for larger area rooms.
6. The spatial accuracy characteristics of both automated techniques are comparable.
7. The spatial repeatability of the ultrasonic technique exceeds that of the mouse-traverse technique.
8. Data profiles can be generated in real-time for field analytical assessments.
9. The characterization survey for thorium-230 resulted in criteria compliance for the area.
10. The RPD analysis uncovered three out of fifteen points beyond the 20% criteria.
11. Future research should be directed at extending the range and flexibility of automated positioning survey system.
12. On-site changes to the computer program were sometimes necessary but could be performed in a matter of minutes.

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INNOVATIVE DOWNHOLE VOC SAMPLING

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ABSTRACT

Two innovative downhole Volatile Organic Compound (VOC) sampling devices were recently deployed at the U.S. Department of Energy (DOE) Hanford Site in southeast Washington State. The Multisorbent Arrayed Sampler developed by Oak Ridge National Laboratory and the Portable Acoustic Wave Sensor (PAWS) system developed by Sandia National Laboratories were deployed at a VOC-contaminated site in the Hanford 200 West Area. Soil vapor extraction is currently being performed at the site, and these two sampling devices were used to further characterize nearby wells.
The two technologies met well field characterization needs for (1) precise quantitative analysis of VOCs in wells without adequate sample data, and (2) sensors to provide real-time in situ analysis of higher VOC concentrations in wells previously characterized. The Multisorbent Arrayed Sampler collected VOCs on sorbent traps that when analyzed provided precise quantitative analysis of VOCs from the low parts per trillion (ppt) range to thousands of parts per million (ppm). PAWS, available in both a downhole and surface-based system, gave real-time analysis of VOCs up to 100,000 ppm.
These two technologies were selected for field demonstration by the U.S. Department of Energy Office of Technology Development. The demonstration was performed as part of the VOC Arid-ID program. The Arid-ID is designed to support the acquisition, development, demonstration, and deployment of technologies for evaluation and cleanup of VOCs and associated contaminants in soils and groundwater at arid DOE

sites.
The Hanford Site Environmental Restoration Contractor Team is responsible for VOC-Arid ID technology demonstrations at Hanford. When possible, VOC-Arid ID innovative technology demonstrations are conducted as systems and in coordination with an operational environmental restoration project. This allows comparison of two or more technologies under actual field conditions and in conjunction with operating baseline technologies. Such was the case with the two subject technologies.

BACKGROUND

Previous wellfield characterization primarily involved lowering a sampling tube downhole and withdrawing a soil gas sample to the surface. At the surface, the sample would be screened with a hand-held photo-ionization detector, and a portion would be collected in a syringe or bag for analysis with a gas chromatograph (GC). One problem with this method is that wells screened in low-porosity soils may not be capable of replenishing VOCs at the screened interval through diffusion at a rate comparable to the pumping volume. As a result, the subsurface equilibrium may be disturbed, resulting in samples withdrawn to the surface unrepresentative of true soil conditions due to extraction of larger volumes. Additionally, significant losses of VOCs occur through permeable tubing, and absorption and subsequent release of VOCs occur with non-porous tubing. Longer tubing required for deeper wells multiplies this problem.
Field deployments of the Multisorbent Arrayed Sampler and the PAWS System occurred at a VOC contaminated site in the Hanford 200 West Area. Site characterization activities have identified a broad VOC plume in the groundwater and vadose zone. The principal contaminants are carbon tetrachloride and chloroform. These downhole sampling devices were used to help define the nature and extent of VOC contamination.

MULTISORBENT ARRAYED SAMPLER

The Multisorbent Arrayed Sampler is a downhole sampler designed to collect soil gas samples on multisorbent traps while the sampler is deployed at selected downhole intervals (1,2). The sampler body (Fig. 1) consists of a cylindrical stainless steel assembly housing an array of 13 micro-miniature solenoid valves in a mounting structure to accept 6 multisorbent traps, which are connected to the system via Cajon Ultra-Torr fittings. As mounted in the body, solenoid valves are connected to each trap for isolation or sampling as desired. The outlet of each trap is connected (via a solenoid valve) to a common sample conduction manifold. A pump at the surface draws accurately metered sample air through any selected trap via the manifold. An additional solenoid valve in the sampler head provides for rapid direct evacuation of the residual air near the sampler (if appropriate) before sampling through a trap. This valve may also be used to draw soil gas samples directly to the surface. Each end of the sampler body is fitted with an inflatable pneumatic packer to isolate the sampler in a selected well interval. A third detachable dangling packer can be attached to the bottom of the sampler to isolate an extended zone of interest. Pressure and vacuum gauges, a 0-200 ml/min mass flowmeter, power supplies, and all necessary switches and valves are contained in a 43x53x20 cm portable case. The control assembly requires 120 V electric power or a 600 watt portable generator, a source of low-pressure compressed air or inert gas, and a source of moderate vacuum. The vacuum is provided by a small, portable diaphragm pump.
Fig. 1. Multisorbent Arrayed Sampler Diagram.
To operate the system, the sampler is loaded with six fresh multisorbent traps and lowered into a monitoring well to a predetermined depth. The packers are then inflated. To acquire a sample, the operator throws a switch controlling the solenoid valves for one of the six traps and moves the vacuum source control valve to the sampling position. Then the operator records the flow rate through the trap and time of flow. When an adequate volume has collected on the trap, the solenoid valves are closed. Up to six separate samples on separate traps can be collected before the sampler must be returned to the surface where fresh traps are loaded. Downhole times can range from 20 minutes to 4 (or more) hours, depending on data needs.
The multisorbent traps are specially made by sequentially loading beds of Carbotrap C, Carbotrap, and Carbosieve SIII (Supelco, Inc.) in a stainless steel tube (3). These carbonaceous sorbents can retain a variety of VOCs. Prior to shipment to a site for sampling, each trap is spiked with a bolus of vapor containing known quantities of three surrogate standard species. Quantities of these surrogates remaining on the traps after analysis provide an indication of the stability of the

VOCs retained on the traps following collection, sample transport, and storage prior to analysis. The sample traps are returned to the laboratory for thermal desorption followed by GC or gas chromatography/mass spectrometry analysis (4).
Although the sampler may be used in a number of scenarios or concentration ranges to obtain high-quality chronologically and spatially resolved samples, it is intended for use in at least two specific applications. In the first, where relatively large concentrations of contaminants exist (ppm levels), the sampler can be employed to collect samples so small (10-20 mL) that there is essentially no disturbance of the subsurface equilibrium. This is especially important in strata of low porosity. In the second scenario, where porosities are higher, the sampler can be used to concentrate very large samples of soil gas. Typically, such a capability would be important when strict regulatory levels have been imposed. For example, if 1 ng of carbon tetrachloride (CCl4) can be sorbed on a trap and accurately quantified, the collection of soil gas at the rate of 200 ml/min for eight hours on a single trap would permit the observation of CCl4 concentrations of 2 ppt.

HANFORD DEPLOYMENT OF MULTISORBENT ARRAYED SAMPLER

The Multisorbent Arrayed Sampler was deployed at Hanford in May and August of 1994. A total of 20 separate events were sampled, and 129 multisorbent traps were used over a 6-day period. Sample results indicated concentrations of CCl4 ranged from non-detects to a lower value of 42 to a maximum of 94,000 ppbv. Chloroform concentrations ranged from 6 to 1,740 ppbv. Table I is a summary of selected trap analyses. Other previously unidentified compounds retained on the multisorbent trap included a-methyl styrene, 2-phenyl-2-propanol, butanedoic acid-diethyl ester, and 4-methyl-2,6-di-tert-butyl phenol.
Detection of these compounds would not have been possible with other sampling systems, since sorbent trap sampling provides a way of concentrating species present at low pptv levels. Figure 2 is a chromatogram of GC/Flame Ionization Detection analysis of trap #313 from well 299-W15-219.
The multisorbent traps were fortified with three surrogates before deployment to Hanford. The average recoveries of surrogates Hexafluorobenzene, d8-Toluene, and d5-Bromobenzene were 98%, 92%, and 97%, respectively.
Fig. 2. Chromatogram of Trap 313 from Well 299-W15-219 (4th day, 263 mL sample)

PORTABLE ACOUSTIC WAVE SENSOR SYSTEM

Central to the PAWS system are two polymer-coated surface acoustic wave (SAW) sensors, one acting as a reference and the other as the active sensor. Figure 3 shows the PAWS system schematically. SAW devices are extremely sensitive to changes in the mass or other physical properties of a thin film cast onto the device surface (5,6). Mass changes of tens of picograms can be detected. Using thin films to absorb the chemical species of interest, this sensitivity can be used to detect and monitor a wide range of VOCs (7,8). Current PAWS systems are capable of minimum detections of 0.1 to 10 ppm for typical VOCs.
Fig. 3. Schematic of the portable acoustic wave sensor (PAWS) system.
Diffusional properties of films made with elastomeric polymers such as polyisobutylene result in rapid (few seconds) and reversible responses, enabling real-time and continuous monitoring. Oscillator electronics drive sensors and provide two independent outputs: a mixed frequency that tracks changes in wave velocity and a power detector voltage that tracks changes in wave attenuation. The ratio of these two signals provides limited discrimination of isolated chemical species, while the more sensitive frequency response can be used to quantify isolated species (6,9). This discrimination is made possible by the unique combination of the mass response and the response due to changes in the polymer viscoelastic properties based on the plasticizing action of the absorbed chemical (6). Viscoelastic properties also depend on temperature (6), resulting in a temperature dependence of the device response. Temperature induced drift is minimized in the PAWS system based on active temperature control of the sensor test fixture (10).
The system is configured as a down-hole probe with associated packers for isolating a selected section of a well (11). A cable provides both power and data communication to a notebook computer at the surface, and a stainless steel or teflon sampling line is used to compare the in situ measurements with surface analyses. In addition to the SAW sensors and associated electronics, the probe contains data acquisition and system control electronics consisting of a frequency counter, eight 16-bit A-to-D channels for measuring voltages, eight digital outputs for system

control, a microprocessor, and an RS-232/RS-422 serial interface.
Environmental sampling hardware consists of a pump to pull in a sample and a three-way valve to periodically pass air from the surface across the sensor to establish sensor baseline. (This air is pulled from the line used to inflate the packers and is scrubbed with activated carbon to remove any VOCs.) This in situ determination of baseline enables long-term continuous monitoring. A Nafion membrane on the sample line maintains a constant humidity between sample and baseline gases and prevents errors in baseline determination resulting from the slight humidity sensitivity of the SAW sensor (10).

HANFORD DEPLOYMENT OF PORTABLE ACOUSTIC WAVE SENSOR SYSTEM

As shown in Fig. 4, field demonstrations of the PAWS downhole probe for detection of CCl4 at the Hanford Site have demonstrated accurate in situ analysis from 10 ppmv to over 25,000 ppmv, indicating the wide dynamic range of this sensor. Because the sampling rate is low (<100 ml/min), the system can be used to determine the concentration in the unperturbed well bore. In addition, using either a downhole pump or a large Teflon line connected to a pump at the surface, the system can be used to monitor in real-time the changes in concentrations upon purging of a well. Tests of this type demonstrated that large purge volumes can be required for relatively porous soils typical at the Hanford CCl4 site. For example, during high atmospheric pressure, no CCl4 was observed even after purging 150 well volumes from a well, although more than 1,000 ppmv had been observed during a previous test during low atmospheric pressure. The ability to place the PAWS probe in a well and leave it for long periods with the packers inflated should allow the unperturbed concentration to be determined and tracked as a function of time.

Fig. 4. CCl4 concentration vs. time.

During one field demonstration, the in situ measurements were compared with an above-ground PAWS system and a photoacoustic infrared analyzer (Brel & Kjaer Instruments, Inc., Marlborough, Massachusetts). As shown in Fig. 4, agreement between the above-ground PAWS and the commercial instrument was within 2% from 100 ppmv to over 20,000 ppmv, demonstrating the accuracy of the PAWS system. However, differences were observed between the above-ground and downhole systems. At high concentrations, the downhole readings were higher (by about 20%), while at low concentrations in subsequent wells, the above-ground values were higher. These differences were determined to result from perturbations in the sample being brought to the surface through the Teflon sampling line. The loss of chemical is shown by the data in the main figure in Fig. 4. In contrast, the data in the inset figure were taken prior to turning on the sample pump to the well. In this case, both the above-ground PAWS system and the Brel & Kjaer instrument were detecting about 100 ppmv of CCl4 added to the ambient air being pulled through the sampling line, which had been previously exposed to high CCl4 concentrations.

In the next field test, a stainless steel sampling line was used to minimize these sample perturbations. The only calibrated comparison was with an above-ground PAWS system. The results of this test showed some loss was still apparent at high concentrations; however, no erroneous high readings in the above-ground instrument were observed after purging the sample line. The loss of chemical at high concentrations was determined to result mainly from the piston pump used to pull the sample to the surface. Stainless steel tubing has the disadvantage of being more difficult to deploy into wells than Teflon, which is more flexible. These results demonstrate the difficulty in obtaining a representative sample and the advantage of an in situ monitoring system.

CONCLUSIONS

The Multisorbent Arrayed Sampler allows collecting a high-quality in situ soil-gas sample on a sorbent trap. Compared to existing methods, it can collect and concentrate (1) samples of a wider variety of VOCs and (2) samples having a wider range of concentrations. In some applications, the sampler can acquire samples for analysis in the ppt range (below the detection limits of existing monitoring technology).

The PAWS is a rugged VOC monitoring system capable of providing real-time, continuous, and in situ monitoring of isolated species. The system has been shown to be accurate and have a wide dynamic range, making it well suited for sites having high contaminant concentrations. The probe is easy to deploy, and the data acquisition and analysis is performed with user friendly software that provides real-time display of contaminant concentrations. The ability to establish sensor

baseline in situ permits continuous long-term monitoring of contaminant concentrations during a remediation effort. Results comparing in situ PAWS measurements with analyses of samples pumped to the surface demonstrate the advantage of a downhole system.

STATUS

Both the Multisorbent Arrayed Sampler and the PAWS system were successfully deployed at Hanford and other DOE sites. Current work on Multisorbent Arrayed Sampler technology includes deploying the sampler concept in a cone penetrometer point and deployment of a simplified arrayed sampler in mixed waste underground storage tanks for headspace sample collection. A commercial partner is needed to take the Multisorbent Arrayed Sampler from the development/deployment stage to actual commercial production.

Current work on PAWS is focused on extending the capabilities to analyze mixtures of VOCs using arrays of SAW sensors, to lower detection levels using adsorbent-based preconcentrators, and to analyze water using specially designed sampling modules. Some of this development is being done in conjunction with two industrial partners interested in developing and manufacturing more advanced sensing systems based on the technology demonstrated in the PAWS systems. Additional industrial partners are involved in demonstrating the utility of the PAWS technology for selected applications.

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CONTAMINANT DETECTION THRESHOLDS FOR THE MULTISPECTRAL BOREHOLE LOGGING SYSTEM

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ABSTRACT

Geophysical borehole logging techniques are used for the in situ determination of subsurface physical, chemical, geological, and hydrological parameters. In a project funded by the Department of Energy Office of Technology Development we addressed the adaptation of one nuclear borehole measurement technique, neutron-induced gamma-ray spectroscopy ("multispectral") logging, to map environmental contaminants along boreholes. It has been known for some time that multispectral logging is capable of identifying many elements, but earlier equipment was not very sensitive. To demonstrate that the technique can be made much more sensitive, we have estimated detection thresholds for a number of contaminants for a state-of-the-art borehole instrument.

Experiments in a specially built test facility constructed at the DOE Grand Junction Projects Office with an experimental prototype multispectral borehole instrument developed for this project allowed us to estimate detection thresholds for mercury, cadmium, chlorine, gadolinium, and samarium. Because our experimental prototype instrument was more expedient than elegant, we extrapolated from the experimental threshold values to what we believe a "production" system could achieve using existing hardware and data processing technology. These results indicate that the multispectral borehole logging technique is sufficiently sensitive to be useful in screening for these contaminants; its value is greatly enhanced by the fact that the technique analyzes some three orders of magnitude more material than physical sampling at each depth where a measurement is made. In monitoring applications, this in situ technique can reanalyze the same borehole at any desired time interval for as many years as required.

INTRODUCTION

Some nuclear borehole logging techniques are capable of identifying and mapping specific nuclides in the rock or soil through which a borehole passes. Such techniques can be used to characterize contaminated sites and monitor those sites as long as necessary following either cleanup or stabilization. Nuclear borehole logging techniques have advantages and disadvantages that tend to be complementary to those of physical sampling and these in situ measurements can help address the drawbacks of physical sampling. Drawbacks addressed by nuclear logging include high costs, lengthy delays in obtaining results of analyses from laboratories, under-sampling, sample-handling problems, and ambiguity in long-term monitoring.

One potentially useful borehole logging technique for contaminant mapping is neutron-induced spectral gamma-ray or multispectral logging. Multispectral logging records gamma-ray energy spectra during and/or after irradiation of the borehole environment by neutrons, similar to laboratory techniques such as neutron activation analysis. Neutrons interact with nuclei in a variety of ways to produce gamma rays that enable those specific nuclides to be identified. Detectable contaminants

include chlorine, a component of many organic contaminants, as well as heavy metals and other nuclides. The physics behind this technique is generally well understood and it is known that the technique can identify many radioactive and nonradioactive nuclides. The key question is what is the detection threshold for those nuclides in a borehole environment? We are using a combination of experiments with the experimental prototype, along with computer simulations and analytical calculations, to estimate detection thresholds for a number of contaminants; initial results are presented here.

THE VARIABLE CONTAMINANT TEST MODEL

For this project, we constructed a variable-contaminant test model, a full-scale physical model that allows insertion of various contaminants under controlled conditions. This model, constructed at the DOE Grand Junction Projects Office, provides a known contaminated test environment. Experiments performed in the variable-contaminant model helped determine detection thresholds for the multispectral borehole instrument or tool for several contaminants of concern and provided a final benchmark of the computer simulation program that is being used to investigate other contaminants. To investigate several contaminants, remain within budget, and meet environmental regulations, we chose a design using 15-cm-thick disks of contaminant-doped concrete sandwiched between thicker layers of uncontaminated concrete (Fig. 1). Two contaminated disks were manufactured containing different concentrations of chlorine along with four disks each containing a small quantity of one of the following: mercury, cadmium, samarium and gadolinium. The latter two elements were included to help us understand the instrument response, not because they are considered major contaminants.

The variable-contaminant model includes a gravel-and-water-filled tank surrounding the concrete cylinders to extend the neutron and gamma-ray transport medium radially and provide shielding for personnel during operation of the neutron source. The borehole has a nominal diameter of 15 cm and is air filled. The outside diameter of the contaminated and uncontaminated cylinders, 75 cm, was chosen to provide most of the response that an infinite medium would exhibit while minimizing the amount of contaminant required to build the 15-cm-thick zones. Although the composition of the gravel/water tank is not the same as the concrete, it should be sufficiently similar for these experiments considering its distance from the borehole. The 15-cm-thick zones are not effectively infinitely thick, so the infinite zone response must be constructed from the thin zone response by superposition (1).

EXPERIMENTS IN THE VARIABLE-CONTAMINANT MODEL

Measurements in the variable-contaminant model provided estimates of detection thresholds, the concentration at which one can say with some specified level of certainty that a given contaminant is present, for several contaminant elements.

Equipment Description

A contractor participating in the experiments supplied the self-contained neutron generator package for the logging instrument; the generator package is manufactured by a commercial vendor. The spectroscopy system used in these experiments is a commercially available system with ultra-high-countrate capability. (2,3) The germanium detector is an N-type crystal with an efficiency of 22.5% cooled by liquid nitrogen in a dewar with an outside diameter of about 7.5 cm. The detector and dewar were housed in a brass sleeve that was mated with the neutron generator section of the commercial tool. The preamplifier, contained in the probe, is a transistor-reset type. The detector and its associated instrumentation and software were tested extensively before the experiments began. The detector performed remarkably well, successfully handling count rates of 2,000,000 counts per second while maintaining good spectral performance.

Experimental Procedure

We acquired a 16,000-channel spectrum at each of 13 depth positions in the variable-contaminant model for each of the 6 removable contaminant disks. Repeat measurements were made in some cases to verify reproducibility of results. In addition, spectra were acquired in other Grand Junction calibration models to determine the effects of hole size and formation nuclear properties on the instrument response. Each spectrum was acquired using a timing window selected to ensure that the spectrum was the result only of prompt thermal-neutron activation. Each spectrum was acquired for 1000 s of live time, corresponding to about 1200 s or 20 min of real time in these experiments.

In addition to the capture spectra, inelastic and activation spectra were acquired

for some positions in the variable-contaminant test model. Like the capture spectra, inelastic spectra were accumulated for 1000 s of live time each while activation spectra were accumulated for various times following generator shutdown because of the mix of short- and long-lived decay products. A preliminary examination of the activation and inelastic spectra suggests that they will be of little use in identifying most contaminants, although these other spectra provide important information related to lithology and water content of the material around the borehole and may provide other useful information as well. This report focuses on the capture spectra only.

Results of the Contaminant Tests

Figure 2 shows a portion of the spectrum taken at the position of maximum response with the detector positioned near the mercury contaminant disk. The 368-keV peak from mercury is indicated on the spectrum, and numerous other peaks are present including other mercury peaks. The energy width of each of the 16,000 channels is nominally 685 eV, so the recorded spectrum extends beyond 11 MeV. For the lower energies where most contaminant peaks occur, the energy resolution is about 3.5 keV full-width at half-maximum, corresponding to about five channels of data. Energy resolution is an important system parameter because improved resolution decreases the background count rate associated with each photopeak and improves detection threshold, all else being equal. Also apparent is the continuum background resulting from Compton scattering of gamma rays in the variable-contaminant model and the tool materials, including the detector. The continuum represents a fundamental limitation to the detection limits that can be achieved for a given contaminant for a given counting time.A few activation reactions contribute to the capture spectra because of their relatively short half-lives. Major contributors to all capture spectra are hydrogen, silicon, calcium, and iron (largely from the concrete models) and copper, chromium, and germanium (largely from the logging tool). The atypical peak at 478 keV is a result of Doppler-broadening of the gamma-ray energy line from an (n,a) reaction on 10B.

Detection Threshold Estimates

The working definitions for the detection limits used in this analysis are based on earlier work by Currie (4) and others. Currie showed that the detection limit for radioactivity measurements of this type should be based on the variability in the background. For this preliminary assessment, we have assumed that there was no interference with the signal of interest from anything other than the uncertainty in the background.

Estimated detection thresholds for the contaminants studied in the Grand Junction experiments are given in Table I, along with the energy of the spectral peak used for each contaminant. The contaminant concentrations in the column labeled minimum detectable concentration correspond to the critical level of Currie (4). These are concentrations at which the net peak area is 2.33 times the standard deviation of the background, a level at which one can say with 95% confidence that a particular contaminant is present. The contaminant concentrations in the column labeled quantitative determination level are concentrations at which the net peak area is 14 times the standard deviation of the background. According to Currie, that is the level at which the contaminant concentration can be estimated to a relative accuracy of 10%. Minimum detectable concentration represents a reasonable minimum screening level while the quantitative determination level represents concentrations at which the multispectral logging system could provide fairly accurate concentration estimates.

The estimated detection thresholds given in Table I are based on a thick (say, 1 m) contaminated zone and a spectral accumulation time of 1000 s of live time. Longer counting times would improve the thresholds while shorter counting times would degrade the thresholds. Similarly, the logging system is less sensitive to contaminated zones substantially thinner than the 1-m-thick zone mentioned above; special processing is required to avoid under-estimates of concentrations in the case of thin zones.

Detection thresholds are given in Table I both for the experimental prototype and for a proposed state-of-the-art instrument. Due to budget and time constraints, expediency played a major role in the construction of the experimental prototype. While the detection thresholds listed under "Experimental Prototype" in Table I represent a substantial improvement over previous generation multispectral logging systems, better detection thresholds can be achieved with available technology. Such

enhancements as a larger and more efficient detector, a more intense neutron source, lower-noise electronics, and advanced data processing are currently available, along with increased counting times.

We estimate that the detection thresholds could be improved by as much as an order of magnitude or more for some contaminants. Improving the detection thresholds by a factor of 8 would give the values listed in Table I under "State-of-the-Art Instrument." These estimates are based on a simple analysis that does not address the complicated interactions of the proposed improvements or the energy-or nuclide-dependency of some of those effects. However, the estimates tend to be substantiated by the results of computer simulations, which suggest a detection threshold for mercury of 1 ppm and for cadmium of 0.02 ppm for a perfect detector, where the energy of every gamma ray that enters the detector is determined accurately.

COMPUTER SIMULATIONS AND ANALYTICAL CALCULATIONS

Physical models tend to be costly, especially considering the strict safety and environmental regulations that exist today. Furthermore, if the behavior of the instrument is to be determined for typical field applications, there are many factors that must be studied over a range of borehole and formation conditions, such as borehole diameter, casing, rock chemistry and density, instrument geometry, and specific contaminants. Some of these factors interact, and many physical models would be needed to sort out the various combinations of conditions. In addition, we do not have the kind of detailed and precise control over the properties of physical models that would be required to explore all of the factors that could be affecting the data. Thus, while this approach provides important basic understanding of instrument behavior it is not suitable for studying the details.

Computer codes that simulate nuclear processes are widely used in nuclear engineering applications such as radiation shielding design and criticality studies. Computer simulations have also been used by a number of organizations around the world to help design and understand many nuclear systems, including nuclear borehole logging instruments. The theory behind radiation transport simulations is well understood, as is the process of applying the theory numerically. In this project, we used Monte Carlo computer simulations to help design the prototype instrument and to estimate detection thresholds. Because computer simulations can be very time consuming, we also used analytical or theoretical calculations to extend the detection threshold estimates to other elements.

The Computer Simulation Code

We used the radiation transport code MCNP (Monte Carlo N-Particle) for the computer simulations. MCNP is written and maintained at Los Alamos National Laboratory (5,6). Although MCNP is widely used for simulations of this type, it is designed primarily for other applications such as reactor shielding design and criticality calculations. When we started the simulations we found that MCNP was only marginally suitable for simulating neutron-induced gamma-ray spectroscopy problems where high-purity germanium (HPGe) detectors were used. This is primarily due to the poor treatment of photon production in the ENDF cross-section evaluations, and therefore the MCNP data libraries were not adequate for many nuclides of interest. This also holds true for many other evaluated data libraries, and therefore other radiation transport codes as well. At best, the use of MCNP for simulating gamma-ray spectra for HPGe detectors required a number of compromises and workarounds.

MCNP was primarily limited by the availability and quality of neutron data libraries. Some data libraries were unavailable due to lack of experimental data, something we could not address. Other data libraries were unavailable because existing data have not been processed into the MCNP format, a problem we were able to remedy for some important nuclides with assistance from the Nuclear Theory and Applications group at Los Alamos. That group also improved the accuracy of some of the photon production spectra in the existing evaluations and processed new data libraries for those nuclides. That was needed because, with the exception of some lighter elements, most evaluations represent photon production with wide energy bins (50 keV in the case of iron, 250 keV in most other cases) resulting in spectra that bear little resemblance to experimental data.

Other improvements were made to MCNP which helped to alleviate some of the problems discussed above and to reduce the amount of computer time needed to perform these types of simulations. These improvements included second-order perturbation, a technique for testing the sensitivity of the simulated system to small changes in

the environment, and the ability to generate separate spectra for each element or isotope of interest as well as the total spectrum.

The unavailability of germanium photon-production data made the actual detector response difficult to simulate. The efficiency of the HPGe detector cannot be reproduced; there is no generation of the escape peaks found in HPGe spectra and the gamma rays from Ge(n,) reactions seen experimentally below about 1.5 MeV are also not produced. The MCNP detector response is therefore the response that would be obtained by a perfect detector, accurately detecting all gamma rays entering its space and generating no escape peaks.

Instrument Design Simulations

We used computer simulations to help design the prototype borehole instrument used in the Grand Junction experiments. The simulations indicated that changes in the material composition of the instrument would enhance its detection capabilities. We adopted a brass housing for the detector package because eliminating the steel housing greatly reduced the iron signal present in the experimental data. Simulations also indicated that the detector could be placed as close to the neutron source as possible to increase count rates without causing other problems. Also, 6Li-loaded polyethylene was used for shielding instead of the more common borated-polyethylene or steel, greatly reducing the number of gamma rays from the shielding materials.

Detection Threshold Simulations and Analytical Estimates

We used computer simulations and analytical calculations to estimate detection thresholds for a number of contaminants in addition to those investigated in the Grand Junction experiments. We were unable to study those additional contaminants experimentally because of high cost and environmental issues associated with eventual disposal of the contaminated models. We also used computer simulations to help estimate the improvement in detection thresholds and sensitivity to various contaminants that could be achieved beyond the experimental prototype, as described earlier.

Analytical or theoretical estimates of detection threshold were performed to extrapolate from the detection thresholds determined in the experiments for the five test elements. Key to the analytical estimates is a factor based on the experimental results that includes such quantities as detector efficiency and borehole geometry. This factor and nuclear cross-section data for photon production in the experimental contaminants and additional elements were used to obtain rough estimates of detection threshold for the additional elements. These calculations were approximate with a number of simplifying assumptions. For example, the neutron spectrum was approximated as being monoenergetic and we considered only (n,g) capture reactions, ignoring other, possibly higher yield, reactions. The extrapolated detection threshold results are currently being validated and will be presented in a future report.

CONCLUSIONS

The results presented in this paper indicate that the multispectral borehole logging technique is sufficiently sensitive to be useful in screening for at least the contaminants used in the Grand Junction experiments. The value of the technique is greatly enhanced by the fact that the technique analyzes some three orders of magnitude more material than physical sampling at each depth where a measurement is made. In monitoring applications, this in situ technique can reanalyze the same borehole at any desired time interval for as many years as required.

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27-20
TRANSPORT PROCESSES INVESTIGATION: A NECESSARY FIRST STEP IN SITE SCALE CHARACTERIZATION PLANS

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ABSTRACT

We propose an approach, which we call the Transport Processes Investigation or TPI, to identify and verify site-scale transport processes and their controls. The TPI aids in the formulation of an accurate conceptual model of flow and transport, an essential first step in the development of a cost effective site characterization strategy. The TPI is demonstrated in the highly complex vadose zone in glacial tills underlying the Fernald Environmental Remediation Project (FEMP) in Fernald, Ohio. As a result of the TPI, we identify and verify the pertinent flow processes and their controls, such as extensive macropore and fracture flow through layered clays, which must be included in an accurate conceptual model of site-scale contaminant transport. We are able to conclude that the classical modeling and sampling methods commonly employed in most site characterization programs will be insufficient to characterize contaminant concentrations or distributions at contaminated or hazardous waste facilities sited in such media.

INTRODUCTION

Fundamental to any effective compliance or remediation strategy at a contaminated site is the formulation of an accurate predictive model of contaminant transport and groundwater flow. All successful predictive models must in turn be based upon a valid conceptual model. In practice, conceptual models are often based on general simplifying assumptions of standard saturated and unsaturated flow rather than site specific transport processes and their physical, chemical, geologic, and biologic controls. Predictive models based on these simplistic, generic conceptual models have consistently failed to correctly predict contaminant distribution, even when apparently conservative constraints and parameters have been incorporated (1). As a result, compliance and remediation strategies can fail with associated large remediation costs.

To avoid such failure, we propose that any site characterization or remediation strategy begin with an investigation of the transport processes and their controls active at a site. The TPI integrates information from a sequence of site specific activities in order to maximize the validity of conceptual models, yielding enhanced efficiency and cost effectiveness in compliance and remediation strategies. The TPI incorporates three main steps:

An exploratory phase identifies probable hydrostratigraphic units and their degree of connectivity by integrating a search of the literature and existing site data with small scale field tests and exploratory surveys of local outcrops. Site specific transport processes and associated controls that must be investigated in order to yield a minimal scientific understanding of contaminant movement are hypothesized, and an uncontaminated analog site to test these hypotheses is located.

Testing of hypothesized transport processes and controls is carried out by conducting appropriately designed and sited transport processes characterization tests (TPCTs). As necessary, the TPCTs employ techniques such as pump tests, infiltration and tracer studies, dye pulses and excavation, and detailed soil/rock

sampling in order to thoroughly investigative the field-scale transport processes active at the site.

Analysis of TPCT hypotheses testing confirms or refutes the efficacy of the hypothesized transport processes and determines whether the active transport processes at the site are adequately understood or if additional investigation is required.

Information gained from the TPI must be integrated with information from elsewhere at the site to develop a valid site wide conceptual model of contaminant transport used to guide site characterization activities. This is accomplished by mapping of local geology with an emphasis on the relation of physical characteristics to confirmed transport processes and controls in order to provide a data set adequate for geostatistical and hydrologic transport simulations. These simulations, conditioned on the results of the TPI and other studies, will properly reflect the active site transport processes and their controls. Without this initial study of the physical processes controlling contaminant transport, characterization and remediation efforts may proceed in an expensive and undirected mode governed only by previously discovered contamination and compliance mandates.

The near surface vadose zone of the Fernald site was chosen for the TPI demonstration because of its location within a glacial till typical of deposits covering much of the North American continent (2). Glacial till vadose zones contain complex media and transport processes that provide a challenging conceptual model test. In addition, soluble species of uranium have been identified during surface based testing (3) and have been found at depth (4). Current conceptual models do not provide any reasonable mechanisms for the presence of uranium at depth, implying that unanticipated transport processes are active. In this paper, we present our demonstration of the TPI conducted at Fernald, Ohio as part of the DOE Technology Development Landfill Waste Area program.

EXPLORATORY PHASE

The exploratory phase integrated information gained from a literature search, several small preliminary tests (5), and a survey of local outcrops. The literature search identified transport processes and controls common in glacial tills. In this type of environment, macropore or fracture flow may define fast transport pathways (6, 7, 8, 9, 10, and 11). Media heterogeneity may also create zones of preferential flow (12, 13, and 14). Transport through the matrix of massive clays is likely controlled by diffusive mechanisms (2).

Paddys Run Creek, an intermittent stream crossing the Fernald site, afforded 2.5 km of stream cut bank for a survey of local outcrops. Stratigraphic units at these outcrops were investigated and mapped. Active seeps were noted in structural features (thrust faults) and at topographically low contacts at clay units with sand and gravel channels. Small scale dye infiltration tests were conducted in several outcrops that identified transport processes such as fracture and macropore flow. This exploratory phase suggested a connection among transport processes identified by the small scale dye transport tests, media heterogeniety, and larger scale geologic structural features.

Hydrostratigraphy and Hypothesized Flow Processes and Controls

Uranium is the most important contaminant at the site. Uranium transport may occur as soluble forms or as insoluble forms adsorbed to colloids. In either of these cases, aqueous subsurface transport processes are critical, and lithologic units were consequently defined on the basis of their hydrologic, rather than geologic, characteristics. Figure 1 presents six hydrostratigraphic units that were identified, each distinguished by unique hypothesized flow and transport processes and controls.

Unit 5 is the uppermost unit and contains the topsoil and silty loess comprising the bulk of the rooted zone. It was hypothesized that flow through this unit would be primarily vertical and controlled by plant material, root holes, and wormholes. Unit 4 consists of blocky structured silty clays and sandy silts apparently derived from paleosols. Flow in this unit was thought to be primarily vertical along fracture networks and through the matrix. Unit 3 is composed of bedded and fractured oxidized clays and silty clays. Both horizontal and vertical flow were expected to be possible in this unit, horizontally along bedding planes and laminae, and vertically downward through fractures. Unit 2 is composed of layered and bedded unoxidized clays. It was anticipated that horizontal and vertical fractures and bedding planes provide potential avenues for fast transport and preferential flow. Unit 1, the

lowermost unit investigated, is composed of massive unoxidized clay with frequent and irregular gravels, pebbles, and cobbles. This "gray clay" has often been modeled as a barrier to vertical flow in the Fernald area.
Unit 6 is composed of sands and gravels. In general, the matrix is loosely compacted and was thought to provide the most transmissive avenue for contaminant transport of any unit. The relation of Unit 6 to the other units was varied and its sand stringers or channels were noted penetrating all other hydrostratigraphic units with the exception of the topsoils (Unit 5). Sand and gravel channels, as identified in outcrop, were hypothesized to provide possible avenues for transport through the massive gray clay (Unit 1).

Analog Site Selection

To efficiently investigate the anticipated flow processes at the contaminated site, an uncontaminated test site must be located that will provide an analog with respect to the active transport processes and controls that occur at the contaminated site. The analog site should fulfill the following criteria. It should be uncontaminated so that investigations may be conducted without causing contaminant migration or increased procedural burden. It should be nearby so that proximity provides an adequate analog (climate, weather, geology, etc.). Nearby outcrops for geologic mapping and interpretation are necessary, and ownership is desirable. Implementation of these selection criteria for our demonstration yielded the target selection of FEMP owned land along the west side of Paddys Run creek. This approximately 90 hectare uncontaminated field, used for dairy cow pasture, is within 500 meters of the main processing facilities and within 250 meters of several known locations of uranium contamination. All of these locations are on the East side of Paddys Run. Dissection of the local glacial till to depths of 6 meters by Paddys Run and the high relative topographic elevation of the field suggested hydrologic separation of the near surface from contaminated sites. Dissection also afforded approximately 2.5 km of stream cut outcrops ranging in elevation from 1 to 8 meters for geologic mapping.

TESTING OF HYPOTHESIZED FLOW PROCESSES AND CONTROLS

Design of the Subsurface Transport Processes Characterization Test (TPCT)

Based on the information gained from the exploratory phase, we anticipated the dominant active transport processes would be macropore flow through insect, worm, and root holes; flow along bedding planes and through fractures; and preferential flow due to media heterogeneity. The most important heterogeneity was anticipated to be highly transmissive sand and gravel stringers and channels penetrating more restrictive layers.
To confirm or refute our hypotheses we designed TPCTs that would investigate the presumed transport processes and controls. Because saturated conditions were encountered during an early spring site survey and these conditions can exist for a three or four month period, and in order to provide the worst case conditions for colloidal or dissolved contaminant transport, the test was designed to simulate saturated conditions. A 2.2 meter diameter circular ponded surface infiltrometer was used to produce a steady-state saturated flow field for a fifty to one hundred day period. The pressure field was monitored in time and space throughout the test to allow definition of the overall flow field and to detect any preferential flow. To aid in identifying transport pathways, Cl- tracer pulses were released into the flow system through the infiltrometers after saturated, steady-state conditions had been obtained. Near the end of the test and while still under steady-state saturated conditions, a pulse of FD&C Red #3 dye was introduced through the infiltrometer at one TPCT. The site was then excavated to identify dyed transport pathways and zones of preferential flow. The excavation also allowed inspection of hydrostratigraphic units and their local connections unaffected by weathering or erosional dynamics such as those present along Paddys Run. Site excavation allows us to confirm or refute interpretations derived from sensor and sampling analysis of the Cl- tracer pulses and to visually relate transport pathways to the site geologic conditions and flow processes.

TPCT Site Selection

After appropriate TPCTs have been designed, specific sites suitable for implementation must be located. Within the analog area west of Paddys Run, several candidate sites for the TPCTs were chosen during a preliminary survey in the early spring. An archeological survey was conducted at these locations and in late May a number of exploratory holes were augured. Based on samples from the exploratory

holes and the hydrostratigraphic units and transport controls identified in the exploratory phase, two sites were chosen for TPCTs. The two sites contained distinct geologic sequences that together represented the major hydrostratigraphic units and transport process controls found in outcrop during the exploratory phase, allowing investigation of all hypothesized transport processes and their controls.

The Sand Site (Fig. 1) was composed of topsoils and loess (Unit 5) overlying blocky structured clays (Unit 4). Beneath these units the only geologic unit encountered in any significant amount was a large, unconsolidated body of sand and gravels (Unit 6). This underlying sand and gravel unit could be penetrated only to a depth of approximately 2.5 meters before the borehole walls collapsed. Bedded clays (Units 2 and 3) were encountered at the Sand Site only in isolated samples above the sand and gravel unit.

The Clay Site (Fig. 1) contains a hydrostratigraphic sequence including all of the units at the Sand Site plus horizontal beds of oxidized and unoxidzed clays (Units 3 and 2) extending across the site and overlying the massive unoxidized gray clays (Unit 1). The sands and gravels of Unit 6 were evidenced only in stringers and one sand channel, not in a large sand body as at the Sand Site.

Fig. 1. Hydrostratigraphic units at the "Clay" and "Sand" Sites. A cross section of the instrument array at the Sand Site is also shown. The instrument array at the Clay Site is identical except the upper and lower level instrument packs are approximately 1 m deeper, with the lower packs just above Unit 1.

Instrumentation

Infiltration rates, the pressure field, Cl- tracer concentration fields, and rainfall were monitored by instrument and sampling arrays throughout the tests. The 10 meter wide, bi-level radial design of the array (Fig. 1 and Fig. 2) acquires both three dimensional pressure and solute concentration fields, allowing a sensor derived interpretation of the flow field emanating from the infiltrometer and identification of transport pathways.

The size and design of the instrument arrays and sampling points were intended to allow spatially defined measurements and still ensure interception of field-scale transport pathways. Each instrument array contained forty-four sampling points with each sampling point containing a tensiometer, soil water samplers, and a Time Domain Reflectometry (TDR) probe in a sand filled instrument pack. The dimensions of the sampling point instrument packs (50 cm length by 10 cm diameter) were intended to afford interception of the anticipated macropore and fracture network without requiring integration of results over such large volumes as to render spatial definition of flow paths impossible. The use of sand in the instrument packs provided a high conductivity medium to facilitate interception of macropores. The elevation of the upper and lower sampling levels was determined by the stratigraphy encountered during sampling at each site. At the Clay Site, the upper levels were contained in the layered clays (Units 2 or 3) approximately 2 meters below ground surface, with the lower levels located at the upper surface of the massive unoxidized clay (Unit 1) at approximately 3 meters depth. At the Sand Site, the upper level instrument packs were located in the blocky structured silty clays (Unit 4) at approximately 1.1 meters depth, while the lower level packs were contained in the underlying sands and gravels (Unit 6) at approximately 2.1 meters depth.

Fig. 2. Cl- tracer concentrations distribution at the lower (3 m) level of the Clay Site 119 hours after start of infiltration. White dots are operational sampling points. The 2.2 m diameter infiltrometer is centered in the figure.

The pressure field was monitored by transducer equipped tensiometers at each of the forty-four sampling points at each site. Measurements were collected with a datalogger controlled, automatic acquisition system at fifteen minute intervals throughout the test. Transducers monitored by this system also supplied rain gage and infiltration rate measurements, also at fifteen minute intervals.

Tracer concentration fields were monitored by both passive and active means. Soil water samples for laboratory Cl- tracer analysis were collected on one per day to three per day sampling schedules. TDR probes provided passive monitoring of solute concentrations at fifteen minute to one hour intervals over a multiplexed automated data acquisition system. Soil moisture was not of prime importance under the saturated conditions generated by the ponded infiltration, and TDR soil moisture calculations were made only to determine if saturated conditions did indeed exist in the sampling point instrument packs.

Infiltration and Chloride Tracer Application

Infiltration was initiated at both the Clay and Sand Sites in mid June by pouring water into the circular infiltrometers and simultaneously opening the water supply valves, creating an essentially instantaneous ponded head of 5 cm. Infiltration was terminated at both sites in late August, 66 days after the start of the TPCTs.
Six days after the start of infiltration, an approximately 0.1M $CaCl_2$ tracer pulse was applied through the Clay Site infiltrometer and changing salinity levels were monitored by passive TDR probes and active soil water sampling at the forty-four locations at this TPCT site. On the following day, the same procedure was followed for the Sand Site. The Cl^- tracer pulse at the Clay Site was terminated after infiltration of 1,534 liters in 48 hours. The total Cl^- pulse at the Sand Site was 1,209 liters in 53 hours. The lower volume infiltrated at the Sand Site was due to a lower infiltration rate through the Sand Site infiltrometer surface, which at the time was less conductive than at the Clay Site.

Dye Pulse and Excavation of the Clay Site

After 61 days of infiltration, FD&C Red #3 dye was added to the Clay Site infiltrometer supply tanks. Four days later, after infiltration of 400 gallons of dye, the dye tracer pulse was terminated. At the same time, clean water infiltration was halted at the Sand Site.
Excavation of the Clay Site down to and including the lowermost massive gray clay (Unit 1) was accomplished using both a large back hoe and hand tools. Use of the back hoe allowed rapid removal of one foot to two foot layers. The subsequent hand tool examination ensured that complete and detailed transport features would be revealed.

ANALYSIS OF TPCT HYPOTHESES TESTING

Results from each type of measurement obtained during the TPCTs confirmed the validity of our hypothesized transport processes and yielded a much improved understanding of their active controls at the Fernald site. We are not able to present all results in detail in this paper.

Infiltration

The long time, steady-state infiltration rate should approximate the saturated hydraulic conductivity of the least conductive media being infiltrated (15). At the Sand Site (Fig. 1), the limiting media would be silts and loess with saturated conductivities ranging from 10^{-5} cm/s to 10^{-7} cm/s (16). At the Clay Site, the limiting media would be the massive clay (Fig. 1) with conductivities ranging from 10^{-7} cm/s to 10^{-11} cm/s (16). Instead, infiltration rates at both sites approached 10^{-3} cm/s, exceeding the limiting conductivities by two to eight orders of magnitude and suggesting that infiltration was controlled by the capacity of the conducting macropore system through and below the infiltrometer surface.
Over the sixty-six day infiltration period, 61,000 liters of water were infiltrated through the system at the Clay Site and 78,000 liters of water at the Sand Site. Neither site became saturated at the surface anywhere outside the infiltrometer, also implying an interconnected subsurface transport-pathway system capable of significant transport.

Pressure Head Field

The pressure head field across the 3-D instrument array showed a general pattern of mounding beneath the infiltrometer, with local head perturbations of -10 to -5 cm distinguishing areas of preferential flow. Most sampling points showed rapid response to rainfall events (e.g., approximately 30 minutes at 3 meters depth), indicating that the majority of near subsurface media was well connected to the surface by a macropore system. The rise in head was typically almost ten times the amount of precipitation. However, some sampling points displayed neither rapid nor large rises in head, suggesting that the macropore and fracture system did not connect the entire investigated volume.

Tracer Concentration Field

Based on soil water samples and TDR measurement, overall tracer movement at both sites was initially vertically downward below the infiltrometer, then horizontally. Horizontal distribution of tracer concentrations began sooner at the Clay Site than at the Sand Site and dissipated almost ten days sooner. The distribution of tracer concentrations within the generally vertical and radial flow field was highly heterogeneous at the Clay Site and flow appeared to follow definite pathways (Fig. 2). Arrival times of peak concentrations took less than one hour at some sampling points in the Clay Site and affirmed our hypothesis of rapid transport pathways. Arrival times of peak concentrations exceeded those calculated solely on the basis

of bulk media properties by four to eight orders of magnitude. Contributions of multiple transport pathways at some sampling points were observed as multiple peaks in breakthrough curves (Fig. 3).
Fig. 3. Repeated peaks in Cl- tracer breakthrough curves due to contributions from multiple flowpaths. The sampling point is located 2.5 m East of the infiltrometer at a depth of 3 m (see Fig. 2).
Dye Pulse and Excavation
The dye pulse and subsequent excavation of the Clay Site presented striking visualization of flow processes and transport pathways, and confirmed data from the instrument array (Fig. 4). Dye capture was prevalent in the uppermost layer of topsoils and loess (hydrostratigraphic Unit 5), while vertical root, worm, and insect macropores and fractures transported the dye tracer through the underlying blocky structured clays (Unit 4). The matrix of this unit was stained only in the immediate vicinity of macropore flow and transmissive fractures. Macropores were noticeably missing from sand and gravel media, likely because the unconsolidated matrix does not support pore walls just as it limited instrument borehole depth (see TPCT site selection above). As hypothesized, horizontal dye movement along horizontal fractures and bedding planes was noted both in the oxidized bedded clays (Unit 3) and in the unoxidized bedded clays (Unit 2).
Fig. 4. Plan view of major macropores, fractures, and zones of preferential flow found during excavation at the Clay Site. Noted features are integrated over the depth of the excavation.
In addition to these expected features we found tubular pathways approximately 2 cm by 3 cm in cross sectional area meandering from just beneath the topsoil down to depths well below the upper surface of perched water tables as defined by standing water in boreholes during installation in late May. These large macropores were later identified as crayfish burrows that are endemic to the area. Crayfish burrows and burrow complexes filled with dye connected the topsoils to fracture networks in the bedded clays. These pathways are each capable of transmitting more than the maximum flux observed passing through the infiltrometers. The large incidence of dye stained crayfish complexes in the southern portion of the site (Fig. 4), none intercepted by sampling points, may explain the lack of tracer concentrations sensed in that area by the instrument array. At the Clay Site, these and all other bioturbated fast transport pathways terminated in the unoxidized layered clays (Unit 2).
Sand and gravel stringers and channels were observed at all depths and penetrated even the massive gray clays (Unit 1). Dye transport through numerous silty sand zones was discovered in the layered clays. Unfortunately, at the depths where Unit 1 was encountered (3 to 5 meters), the dye pulse was diluted to an extent that no confident association could be made between the infiltrated dye pulse and flow through sand stringers within the massive gray clay.
CONCLUSIONS
Transport Processes, Controls and Connectivity
The Transport Process Investigation was effective in identifying pertinent flow processes and controls. We can conclude that a conceptual model for the near surface vadose zone at the Fernald site must incorporate the following processes and controls. The dominant flow and transport pathways through the topsoil to the bedded clays are vertical bioturbated macropores and fractures. These pathways are capable of rapid and abundant transport during most of the year, when essentially saturated conditions exist or when rainfall events are of sufficient magnitude to enter the macropore system. The bedded clays support a pervasive network of transmissive horizontal and vertical fractures and are connected to the surface by vertical fractures and macropores. Some crayfish burrows are continuous from the surface to the bedded clays and intersect the fracture network in the clays. Crayfish burrows may provide the dominant transport pathways in some areas, allowing contaminants to bypass extensive volumes of the near surface media.
Channels and stringers of sands and gravels provide localized zones of preferential flow and are well interconnected by fractures and macropores. In this extensively reworked and deformed glacial depositional environment, highly transmissive sand and gravel channels and shear zone fractures penetrate portions of the otherwise impervious massive gray clays, calling into doubt its effectiveness as a barrier to flow and transport. The seeps and preferential flow features noted in the exploratory geologic survey may be representative of field-scale transport pathways.

Trends in infiltration rates and changes in the pressure field indicate that the system is well drained, with a significant potential for contaminant transport.

Further Studies

The connectivity of transport pathways from the surface to depth has been implied at the field scale by the well drained nature of the flow system and should be fully investigated by larger scale tracer or pump tests. Larger scale tests will require that the hydrologic separation of contaminated and uncontaminated sites be conclusively demonstrated. Tests also must be conducted to conclusively determine if the deposits of massive gray clay are effectively breached by discontinuities and penetrations by highly transmissive sand and gravel channels.

Implications for Site Characterization Strategies

Through implementation of the Transport Processes Investigation we were able to distinguish the governing transport processes and controls active in this highly complex near surface glacial till and identify those that must be included in a defensible conceptual model for contaminant transport at the site. Site characterization plans formulated by classical statistics and sampling strategies, and predicated on generic conceptual models of subsurface flow and transport, will likely not recognize the potentially rapid transport processes and controls identified by the TPI. Consequently, such generic conceptual models will characterize contaminant concentrations and distributions only by chance or through prohibitively extensive and expensive classical sampling plans.

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27-21
DETAIL RECONNAISSANCE SURVEYS FACILITATE AN EXPEDITED SITE CHARACTERIZATION
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ABSTRACT
An expedited investigation requires extensive planning in order to accomplish the objectives of the program in a single field season. At the Palos Forest Site a significant portion of the sample planning was facilitated through the use of four reconnaissance surveys and a civil survey. The civil survey established a 100 foot grid across the entire forty acre site. This grid forms the basis for establishing on-site locations for the other surveys and subsequent sampling events.
The reconnaissance surveys included a historic air photo analysis that provided insight to the layout of the original facility and the locations of key features. The historic photos were digitally scanned and converted to image files. The files were manipulated on a VAX workstation where they were scale corrected to the civil survey grid. This scale corrected image allowed locations of key features to be located on the work station with a high degree of accuracy.
A detail surface features map was constructed for the main portion of the facility. This map was constructed on a scale of 1 inch = twenty feet tied into the site civil survey grid. The map showed the location of objects as small as a few inches in area.
A radiological walkover survey was performed using global positioning equipment linked to the radiological metering equipment that was carried over the site by field personnel. This survey produced approximately 150,000 measurements over the site that were linked to discrete coordinate locations.
A geophysical survey was conducted that collected electromagnetic and magnetic data on ten foot centers across the majority of the site and on five foot centers in three areas of interest. The geophysical surveys located shallow buried objects including, water supply lines, utility corridors, underground storage tanks, drain lines and other key features critical to the full characterization of the site.
Integrating the data from the reconnaissance surveys facilitates an efficient targeted sampling strategy. Sample locations can be easily located in the field using the site civil survey. Analytical results are integrated into the reconnaissance survey maps to focus on areas of significant contamination. Using these types of surveys as a presampling strategy saves time and money in the field. The amount of savings realized is dependent upon the complexity of the site and selection of the proper presampling surveys. Time and cost savings can be as high as 30 percent of total investigation costs if significant areas can be removed from further consideration. In almost all situations, reconnaissance surveys provide an excellent basis from which to attach a quick and thorough investigation.
SITE DESCRIPTION AND HISTORY
Following the first successful sustained nuclear reaction attained by Enrico Fermi and his staff at the University of Chicago in the early 1940s a more secluded and secure site was established in the 'Palos Forest preserve southwest of Chicago. Site A was leased from the Cook County Forest Preserve district to support the continuing research associated with the development of the first production nuclear reactors. Two reactors (CP-2 and CP-3) were constructed and operated at this facility between 1943 and 1954 for research purposes. Program research included reactor physics, plutonium production, fission product separation, tritium recovery and laboratory animal experimentation.
Beginning in the late 1940's, facilities at Site A were closed and research operations were relocated. All buildings and equipment were decommissioned and demolished, ongoing operations were relocated by 1956 to Argonne National Laboratory (ANL), approximately three miles west of Site A. As programs were relocated empty buildings were surveyed for radiological contamination, decontaminated as necessary, and razed. The emptied CP-3 reactor concrete bioshield was partially filled with small contaminated items such as hardware and piping, filled with concrete and

buried on-site, in an excavation approximately forty five feet in depth. Remnants of the CP-2 reactor, principally, graphite blocks, and building rubble were entombed in the excavation above the CP-3 bioshield and the entire mass of debris was buried. Subsurface structures such as building basements (partially filled with building rubble and debris), drain lines, utility tunnels, and water supply lines were left undisturbed.

In 1956 Site A was returned to the Cook County Forest Preserve District. Since that time, several investigations of the site have been conducted. ANL performed limited investigations of the site between 1956 and 1978 and has conducted annual surveillance of groundwater, surface water and sediments since 1978. Illinois Department of Nuclear Safety also conducted periodic walkover surveillance of the site and the surrounding acreage. In 1990 during a routine walkover a small piece of processed natural uranium metal was found west of the main facility in a purported former waste disposal area. Following this discovery DOE- Chicago requested Oak Ridge Associated Universities conduct shallow soil sampling, a radiological walkover survey, and a limited geophysical survey (ground penetrating radar) (Berger 1993). With technical management from DOE-Chicago, Bechtel Environmental Inc. (BEI) was contracted to perform follow-up activities in 1993 that included: securing the site by constructing a perimeter fence, supplying electrical power to the site, establishing temporary working facilities, and clearing and grubbing the site in preparation for a detailed site characterization that was conducted in 1994.

Fig.1 Location of the Palos Forest Site

OBJECTIVE AND PLANNING OF THE FIELD PROGRAM

Previous investigations had been preformed at Site A however the focus had been primarily to determine if. radiological contaminants were present and to what extent. The scope of work to be performed under the phase two investigation involved a full site characterization with emphasis on metals and chemical contamination as well as further definition of the extent of radiological contaminants. The field program was designed to reduce the amount of investigation derived waste and was to be expedited to complete all investigative work in one field season. Limited historical information regarding site operations was available prior to initiating field work. Additional historic information was made available early in the field program that served to help verify information gained from the reconnaissance surveys. As part of the planning phase of the field work , four reconnaissance surveys and a site wide civil survey were conducted. This initial effort served to focus the subsequent field sampling program in areas having a high potential of contamination and cleared large portions of the site from further in-depth sample investigation.

RECONNAISSANCE SURVEYS

Following clearing of the site a civil survey was performed to stake grid corner locations at 100 foot intervals over the entire controlled area. The site grid was established from an arbitrary origin near the center of the site with grid lines oriented north-south and east to west. The grid was not tied into state plane coordinates, which did result in some difficulties establishing the basis for the subsequent radiological walkover survey. Having this grid established early and clearly staked greatly facilitated definition of locations for all the following field activities.

AIR PHOTO ANALYSIS

Historic aerial photographs of the site were available from DOE-Chicago. However, due to the condition of the site most of the cultural features where either altered or obscured by fill dirt and soil that had been used to cover foundations and facilities during and following decommissioning of the facility. Coordinate locations were obtained for the few identifiable cultural features. These grid locations were used to scale correct and rectify the photo imagery to the civil survey grid. This scale correction was accomplished by digitally scanning the photographs at a resolution of three hundred dots per inch (dpi). The digital images were manipulated on an VAX arch/info workstation where the survey points were assigned to the appropriate locations on the photograph, and the images where then rectified to the site coordinate grid. The resulting image was then examined to identify areas that were being impacted during the operation of the facility. Areas of barren ground, small buildings, above-ground piping and dumping areas were targeted. The civil survey grid coordinates were read directly from the digital image, with the corresponding area being located on the ground. These areas were

targeted for emphasis in the following recon surveys and in the initial rounds of sampling. This work greatly assisted the early planning of the field program while providing a satisfactory level of confidence in the locations of the targeted areas.

RADIOLOGICAL WALKOVER

The radiological walkover survey conducted at Site A combined several emerging technologies to achieve a cost-effective means of accurately mapping approximately 23 acres of the controlled area. The technique used integrates radiation measuring instrumentation with the Global Positioning System (GPS). The integrated mapping and radiation survey system (MARSS) developed by J.S. Bland Associates, Inc. simultaneously recorded direct radiation, exposure rate, and position data. The survey covered approximately two acres per day, including background determinations, source checks, calibrations, and downloading of position and radiological measurements. In the field data were automatically recorded and stored electronically each second while the technician traversed specified grid blocks. Real-time differentiation of position data was performed via direct radio link between the roving and the GPS base receivers.

Data were electronically transferred daily to Bland's office in Maryland and also were downloaded at the field office in Chicago. Track maps were viewed on site through Golden Software's SURFER program, version 5.0. At Bland's office, MARSS produced color track maps showing ranges of radiation readings in relation to location. These maps were evaluated in the field and used to verify survey coverage and definition of points of interest.

With this radiological data collection system, the field crew was able to survey 23 acres in approximately 14 working days. Use of this integrated system provided adequate spatial coverage of the site and detected and mapped areas of varying radiation levels. These data were compared with the results of a previously performed, more traditional survey to confirm areas of elevated radiation. The 1994 survey defined in far greater detail the boundaries of areas identified by the previous survey and located and mapped the presence of several other small areas of elevated radioactivity.

Over 150,000 data points were collected on-site during the survey. Bland provided a final composite map that included all of the color track maps in a single display. The high density of data allowed patterns to be observed throughout the site.

The results of this reconnaissance survey were used to select sampling locations specifically targeting areas of elevated radiological readings. This survey was also used in conjunction with the detail surface features map identifying areas of elevated radiological readings that corresponded to
coke slag deposits. In these areas sampling was directed at characterizing the coke deposits which could then be used to characterize large portions of the site highlighted by the radiological survey.

Fig. 2 Radiological Walkover Map Site A

SURFACE FEATURES MAP

While the radiological survey was the application of state-of-the-art technologies, the surface features map was the traditional observational field map created using brute force and a lead pencil. The surface features map was constructed based on the established civil survey grid. Baselines 100 foot survey tape measures were laid out along the north and south boundaries of a grid block. A third tape was laid out north to south between the two baselines and the location of surface features were estimated and sketched on grid paper. The location of any and all surface objects were noted on the field sketches including, concrete rubble, debris, metal objects, pipe, fencing, asphalt debris and paving, road material, building foundations footers and slabs, transite shingles, telephone pole bases, trees, slope breaks, surface drainages, depressions, sidewalks, and any other identifiable objects. No determination had been made prior to the start of mapping as to what features were important and what were unimportant. The mapping work required two field personnel eighteen days to cover the central 20 acres of the facility. Field sketches were composited into a 1"= 20' scale map. This map served as the primary planning reference for subsequent sampling activities. Features identified by surface mapping were targeted for sampling and in many cases served to explain anomalies that were defined by the other reconnaissance surveys (i.e. the coke slag correlation to the elevated radiological signatures).

Fig. 3 Portion of the Surface Features Map

GEOPHYSICAL SURVEY

Information from historic documents and drawings and interviews with personnel who had worked at the site during its operation indicated the possibility of the existence of subsurface features. These features included; underground tunnels, building basements, pipeline conduits, and underground storage tanks. A limited Ground Penetrating Radar survey had been performed in selected areas during a previous investigation. This survey had identified anomalous areas that were interpreted as burial areas, and as a number of underground tanks. The results of the GPR survey were highly subjective due to the nature of the subsurface soils (conductive clays) and the preponderance of scattered debris that would produce point sources on a GPR survey. Also the central portion of the facility where the majority of the subsurface features were expected to be located had not been surveyed. From Applied Technologies was contracted to perform a geophysical survey of the entire central portion of the site and two peripheral waste burial areas. The survey included electromagnetics, magnetics and three limited test areas where GPR was performed. GPR data was collected in conjunction with the other two methods in areas where subsurface conditions were known. While the GPR data collected was very similar to the previously acquired GPR data the results were highly interpretive. The other two methods provided more reliable less interpretive information thus, the GPR survey was deleted from the scope of work. In excess of thirty thousand measurements were taken with the EM and mag surveys site wide. Three areas were surveyed at a five foot station offset the remainder of the central portion of the site was surveyed at ten foot offsets. The high density surveys were conducted in areas where underground tanks existed, and in known waste burial areas. The geophysical surveys identified six anomalies that appeared to be underground storage tanks. Of these anomalies three were found to be tanks. The other three were buried building rubble, and a metals disposal pit. The magnetic survey located two buried building basements and the bioshield burial pit that had been filled with building rubble and trash. The EM survey located a series of underground water supply and drain lines, a utility tunnel, buried steam lines, and water supply feed and return lines that connected to a cooling tower foundation. The geophysics was not capable of locating buried terra cotta drain lines and concrete and clay brick storm sewer lines. Due to the nature of the clay rich soils trenches that were dug to place these lines were indistinguishable from the undisturbed materials.

Fig. 4. Portion of the Electromagnetics survey

INTEGRATION OF RECONNAISSANCE SURVEY INFORMATION

Each of the recon surveys provided a specific insight to the facility. Target areas were identified from each survey for further investigation. Integrating the information from all four surveys allowed large portions of the site to be designated as "suspected uncontaminated areas". These areas were sampled using a broadly spaced systematic grid to substantiate the uncontaminated designation. Areas that were targeted on one or more of the surveys as possibly contaminated were initially sampled using four spot composite surface soil samples. A short list of contaminants was identified from these initial screening samples and subsequent bias sampling was performed to define the nature and extent of contamination in the targeted areas. Recon data integration also served to explain anomalies defined on one survey with data from another survey such as metal objects mapped on the surface features map that exhibited a strong response on the geophysics survey, or the correlation between elevated rad levels and coke slag deposits. These correlations further allowed portions of the site to be cleared from all but confirmatory sampling.

Using these four recon surveys focused the sampling efforts into a few small manageable areas where contamination was present. By clearing large areas of the site from further investigation a well focused sampling effort was easily capable of gathering sufficient data to define the nature and extent of all significant contamination in a singe field season.

SUMMARY AND CONCLUSIONS

Five reconnaissance surveys were performed at Site A prior to the initiation of field sampling activities. These surveys included an historic aerial photo analysis, detailed surface features mapping, radiological walkover survey, and geophysical EM and magnetic surveys. This early work facilitated a focused approach to the field sampling portion of the program by removing large areas of the site from extensive sampling efforts. Time and cost savings are difficult to quantify however, Site A

was adequately characterized through an expedited approach, work that would commonly have required two field seasons to complete. Savings estimates from the expedited characterization of the site are on the order of 30 % reduction in costs, with presampling reconnaissance survey playing a key role in completing this effort in one field season.

RECOMMENDATIONS

Establish a clearly identifiable semi-permanent grid system over the entire site.

Tie the grid to state plain coordinates.

Orient the grid north-south east-west even if grid lines are skewed to property boundaries.

Use available aerial photography from state, federal, and private sources as a first look at former site conditions.

Rectify imagery to the established grid for identifying possible target areas.

Select reconnaissance survey that target the type of contaminants expected on the facility.

Run small scale tests of the selected surveys to determine if they will provide the expected information.

Use more than one type of reconnaissance survey on complex sites.

Integrate and evaluate the data prior to planning the initial rounds of sampling.

Emphasize clearing portions of the site from further detail sample consideration.

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INITIAL CHARACTERIZATION OF SOIL SAMPLES FROM A DOE SUPERFUND SITE

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ABSTRACT

Soil samples from a storage pile at a DOE Superfund site were sent to the National Air and Environmental Radiation Laboratory (NAREL) for initial characterization prior to consideration of a full-scale treatability study. The primary objective was to determine if particle-size separation could be used for volume reduction of the contaminated soil. Since the predominate radioactivity in many contaminated soils is found in the small soil particles (<0.074 mm), partitioning of these fractions by particle-size separation techniques represents an alternate, cost-effective remediation method that might reduce the volume of soil that would, otherwise, require costly transportation and disposal. An initial examination of the soil to determine the particle-size and radionuclide distribution is, therefore, a reasonable study to perform before beginning a full treatability study. It may also suggest a treatment solution with minimum expenditure of funds or provide information to select an alternate approach to remediation, or, at a minimum reduce the scope of the treatability study.

Six samples from within the pile, designated Pile (P) samples, were examined initially. Particle-size separation by wet sieving and hydroclassification was performed after the samples were subjected to particle liberation by vigorous washing. Fractions were analyzed for radionuclides, and select fractions were subjected to petrographic analysis after density separation at 2.89 g/cc. Five additional samples from under the pile, designated Under-Pile (UP) samples, were also wet sieved or hydroclassified and analyzed for radionuclides to provide additional characterization of the site.

The following conclusions were drawn from these studies: a) The particle-size distributions of three Under-Pile samples are very similar to each other and to the six samples from within the pile; each contain approximately 48 percent -200-mesh (-0.074 mm) material. One Under-Pile sample contains approximately 60 percent -200-mesh material, while another contains 40 percent. b) The radionuclide concentrations in the Under-Pile samples are 10 to 50 times those of the Pile

samples. The predominate radionuclides are thorium-232 and -228 and radium-228. The maximum radium-228 concentration is 1700 pCi/g in the Under-Pile samples but only 35 pCi/g in the Pile samples. Uranium-238, thorium-230, and radium-226 are also present in significant but lesser concentrations, with radium-226 up to 330 pCi/g and 15 pCi/g in the Under-Pile and Pile samples, respectively. c) The concentration of each radionuclide in the size fractions generally increases as particle size decreases. In all cases, the -200-mesh (-0.074-mm) or the -400-mesh (-0.037-mm) fractions contain much higher concentrations than the whole soil. d) Wet sieving and hydroclassification produced similar results. Although similar in particle-size distribution, the radionuclide concentrations of the Under-Pile size fractions are much higher than those of the Pile samples and are above most acceptable site clean-up criteria. Therefore, volume reduction of the Under-Pile samples by particle-size separation, alone, is not a viable option for remediation. In contrast, either 53 or 65 percent of the soil of the Pile samples can be recovered with radionuclide-contaminant concentration of less than 5 or 15 pCi/g, respectively, by separation at either 200 mesh (0.074 mm) or 325 mesh (0.045 mm). Analysis of the process water indicates that the radionuclides are insoluble in water. e) The primary source of thorium activity in the Pile samples is monazite. Zircon also contributes a small amount to the activity. These radiominerals are concentrated in the fine-sand to upper silt-size range, -100/+400-mesh (-0.149/+0.0387-mm). The specific gravities of monazite and zircon are about two units above the average specific gravity of the host material. Therefore, density separation might be used to provide additional volume reduction of these fractions. The magnetic susceptibility of monazite is in a unique range offering another potential separation method to reduce the volume of contaminated soil.

INTRODUCTION

Soil samples from a DOE Superfund site storage pile were sent to the NAREL for partial characterization prior to consideration of a treatability study. The primary objective of this study was to determine if particle-size separation could be used for volume reduction of the contaminated soil. The predominate radioactivity in many radionuclide-contaminated soils is found in the small soil particles, typically the silt-size and clay-size fractions. Partition of these fractions by particle-size separation techniques routinely used by the mining industry can represent an alternative, cost-effective remediation method to reduce the volume of soil requiring costly transportation and disposal. An initial examination of the soil to determine the particle-size and radionuclide distribution is, therefore, a reasonable study to perform before beginning a full treatability study. It offers the possibility of suggesting a treatment solution with minimum expenditure of funds or providing an indication of another treatment approach, or, at a minimum, reduce the scope of the treatability study.

Six borehole samples from within the pile, designated Pile 1 (P1) through Pile 6, were examined initially. The specific objectives of the initial study were to assess the homogeneity of the storage pile to determine if the radionuclide contamination is similar throughout the pile, to determine if particle-size separation techniques might be effective in reducing the volume of contaminated soil, to determine the physical form of the contamination, and identify other physical and/or chemical properties of these materials and contaminants that might be exploited for volume reduction.

Five additional samples from under the pile, designated Under-Pile 1 (UP1) through Under-Pile 5, were subsequently examined to provide additional characterization of the site. This paper presents the results of both characterizations.

EXPERIMENTAL

Sample Preparation

Five 3-kg, Under-Pile soil samples were received for study, UP1 through UP5. The samples were screened for gross beta/gamma activity using a Geiger/Muller survey instrument. After screening, each sample was thoroughly mixed and split into 400-mL aliquots using a riffler. Each aliquot was weighed, dried at 60C, and reweighed. Three aliquots from each whole soil were analyzed by gamma spectrometry (see below); one aliquot was also analyzed for uranium and thorium by alpha spectrometry (see below). Six 1-kg, Pile samples were also received, P1 through P6. The samples were prepared and analyzed for radionuclides as described above, except the 400-mL aliquots were split by hand producing two aliquots for each Pile sample.

Vigorous Wash

The vigorous washing process liberates small particles from large particles without generating excessive fines. After sample preparation, each aliquot of all samples was vigorously washed in water for 30 min in an orbital shaker at a rotational velocity of 350 rpm with a liquid-to-solid ratio of 4 mL/1g (1). The aliquots were then fractionated according to particle size using either wet sieving or vertical-column hydroclassification as described below.

Wet Sieving

After vigorous washing, an aliquot of each sample was fractionated by size using wet sieving with ASTM standard sieves (2). The aliquots were separated into size fractions at 6.35 mm (in), 2.38 mm (8 mesh), 1.19 mm (16 mesh), 0.590 mm (30 mesh), 0.297 mm (50 mesh), 0.250 mm (60 mesh), 0.149 mm (100 mesh), 0.074 mm (200 mesh), 0.053 mm (270 mesh), and 0.037 mm (400 mesh). The resulting fractions were dried at 60C, weighed, and analyzed for radionuclide content by alpha and gamma spectrometry. One aliquot of each Pile sample was fractionated by wet sieving at 6.35 mm (in), 0.297 mm (50 mesh), 0.149 mm (100 mesh), and 0.074 mm (200 mesh). The fractions were analyzed by gamma spectrometry. An aliquot of each whole soil and each fraction from sieving sample P1 were also analyzed by alpha spectrometry. A 100-g dried portion of the -0.074-mm (-200-mesh) fraction from wet sieving an aliquot of sample P1 was crushed and further separated using a 0.045-mm (325-mesh) sieve to determine the size distribution of soil particles larger and smaller than 0.045 mm.

Vertical-Column Hydroclassification

Vertical-column hydroclassification is a method for separating soil by settling velocity, which is a function of particle size and density. This procedure closely simulates the process used by full-size hydroclassification equipment. Since hydroclassification is not practical with particles larger than 0.297 mm (50 mesh), after vigorous washing, an aliquot of each Under-Pile sample was first fractionated according to particle size by wet sieving at 6.35 mm (mesh), 2.38 mm (8 mesh), 1.19 mm (16 mesh), 0.59 mm (30 mesh), and 0.297 mm (50 mesh). The particles that passed through the 0.297 mm (50 mesh) sieve were separated at 0.25 mm (60 mesh), 0.149 mm (100 mesh), and 0.074 mm (200 mesh) by hydroclassification (3). The resulting fractions were dried at 60C , weighed, and analyzed for radionuclides by gamma and alpha spectrometry. One aliquot of each Pile sample was separated at 0.149 mm (100 mesh) and 0.074 mm (200 mesh). The resulting fractions were dried at 60C, weighed, and analyzed for radionuclides by gamma spectrometry. A 100-g dried portion of the -0.074-mm (-200-mesh) fraction from the hydroclassification of the aliquot of sample P1 was crushed and further separated using a 0.045-mm (325-mesh) sieve to determine the size distribution of the -0.074/+0.045-mm (-200/+325-mesh) and -0.045-mm fractions.

Process Water

Water from the vigorous wash and separation procedures for each sample was collected and a Percol 788N flocculent was added to settle suspended material. The water was then filtered through a 0.025-mm pore paper filter to separate suspended solids. An aliquot of filtered process water from each sample was analyzed by gamma spectrometry.

Gamma Spectrometry

Aliquots of samples, each particle-size fraction, and process waters were analyzed for gamma-emitting radionuclides using high-purity germanium detectors (4). The water samples were counted for 1000 min and the soil samples were counted for 10, 100, or 1000 min, according to the levels of radioactivity.

Alpha Spectrometry

Analyses for uranium and thorium radionuclides were performed by alpha spectrometry (5). Uranium was extracted from the mixture, coprecipitated with lanthanum fluoride carrier, and analyzed by alpha spectrometry. Thorium was separated by ion-exchange chromatography, coprecipitated with lanthanum fluoride carrier, and analyzed by alpha spectrometry. One whole soil aliquot and each particle-size fraction from UP1 and UP2 were analyzed for uranium and thorium. One whole sample aliquot and each particle-size fraction from P1 were analyzed for uranium and thorium.

Petrographic Analysis

Petrographic examination was performed on the six Pile samples. The samples were separated by size as described above. The heavy (more dense) minerals in the -0.297/+0.149-mm (-50/+100-mesh) and the -0.149/+0.0074-mm (-100/+200-mesh) fractions for each of the samples were separated by the sink-float method using tetrabromoethane (specific gravity of 2.89). The composition of the gravel-sized

(+6.35-mm; +-in) material was determined by megascopic examination. The composition of the sand-sized material was determined by microscopic examination using both binocular and polarizing petrographic microscopes. Sedimented slides of the silt- and clay-sized (-0.074-mm; -200-mesh) material were analyzed by X-ray diffraction at the Auburn University Soil Laboratory (Auburn, Alabama). The X-ray diffractograms were analyzed as part of the petrographic examination.

RESULTS AND DISCUSSION

Particle-Size Distribution

All samples were characterized by particle size after vigorous washing using both wet sieving and vertical-column hydroclassification. Aliquots of Under-Pile samples separated by hydroclassification were first sieved to collect the +0.297-mm (+50-mesh) fractions. The -0.297-mm (-50-mesh) material was then separated by hydroclassification, since hydroclassification is not effective for particles larger than 0.250-mm (60-mesh) size. The Pile samples were not sieved prior to hydroclassification since the first fraction collected was 0.149-mm (100-mesh) size. The particle-size distributions of UP2, UP3, and UP4 are very similar whether wet sieving or hydroclassification is used for separation. They are illustrated by UP3 (Fig. 1). Each of these samples contains approximately 48 percent -0.074-mm (-200-mesh) material. Although UP1 is similar in particle-size distribution to UP2, UP3, and UP4, it contains approximately 60 percent -0.074-mm (-200-mesh) material. In contrast, UP5 contains only 40 percent -0.074-mm (-200-mesh) material.

Each of the five Under-Pile samples reveals an unusual difference in behavior when results from wet sieving are compared to those of hydroclassification. The -0.297/+0.250-mm (-50/+60-mesh) fraction of each sample after wet sieving contains from 1.5 to 4.2 percent of the total soil fractions recovered. By comparison, this hydroclassification fraction contains from 0.3 to 0.4 percent material, an average of about 85 percent less material. When comparing wet sieving to hydroclassification, there are always slight differences in the amount of material collected in a given size fraction produced by the two separation methods. It is exceptional, however, to observe such a large difference in the amount of material contained in the same particle-size fraction when one method is compared to another. This change in amount of material produced by hydroclassification as compared to the amount of the corresponding sieved fraction must be the result of larger-size particles, as determined by sieving, reporting to smaller-size hydroclassified fraction. Vertical-column hydroclassification is based on Stoke's Law that states that the settling velocity of particles in a fluid is proportional to the second power of effective particle diameters and the first power of the difference in density between the particle and the fluid. Thus soil particles with a certain effective diameter that behave in hydroclassification as smaller particles, when compared to other particles of the same size, are likely less dense than those particles. An examination of the particle-size content of the -0.250-mm (-60-mesh) fractions of all soil samples is consistent with this behavior during hydroclassification. In each sample, the -0.250-mm (-60-mesh) fractions contain slightly more particles than those produced by wet sieving (Fig. 1).

The particle-size distributions of the Pile samples, P1-P6, are very similar to each other, whether wet sieving or hydroclassification is used for separation, as exemplified by P4 (Fig. 1). Each Pile sample contains approximately 47 percent -0.074-mm (-200-mesh) material. On sieving, sample P5 contains a slightly higher percentage of -0.074-mm (-200-mesh) material, 55 percent, but less particles in the -6.35/+0.297-mm (--in/+50-mesh) fraction, which balances the higher percentage of -0.074-mm (-200-mesh) material. This difference in particle-size distribution exhibited by P5 from the other five samples, is not observed when hydroclassification is used to separate the sample.

Sample P1 was also examined by additional wet sieving with a 0.045-mm (325-mesh) sieve in an attempt to obtain an estimate of the particle-size distribution of the sample below 0.074-mm (200-mesh) size (Table I). An examination of the radionuclide distribution of the fractions initially produced (see next section) indicates that a favorable particle separation for volume reduction might be made at less than a 0.074-mm (200-mesh) size. The -0.074-mm (-200-mesh) fractions, produced during initial sieving and hydroclassification, had already dried to a hard cake and was crushed for gamma spectrometric analysis before making the decision to further examine the fraction. Crushing the fractions may have altered the distribution of particles that would typically be found in the sample below 0.074-mm (200-mesh)

size. However, the data presented in Table I indicate that the -0.074-mm (-200-mesh) fraction originally produced by sieving or hydroclassification contains more than 30 percent -0.045-mm (-325-mesh) material.

The particle-size distributions of the six Pile samples are very similar to UP2, UP3, and UP4. Each of these soil samples also contains approximately 47 percent -0.074-mm (-200-mesh) material. Thus, nine of the 11 samples contain approximately 53 percent +0.074-mm (+200-mesh) material that might be recovered in a remediation process based on particle-size separation. As indicated in the previous paragraph, there is indication that over 65 percent would be recovered if sample P1 is representative of the site.

Radionuclide Distribution

The results of radionuclide analysis for select uranium, thorium, and radium isotopes in Under-Pile and Pile samples and their fractions reveal that the Under-Pile samples contain from 5 to 100 times the concentration of radionuclides in the Pile samples. Table II illustrates the concentrations of uranium-238 and thorium-232 and select decay products in both series in UP1, a typical Under-Pile sample. The data indicate that the predominant contaminants are thorium-232 and its progeny, radium-228 and thorium-228. Their maximum concentrations in this sample are 760 pCi/g thorium-232, 1020 pCi/g radium-228, and 792 pCi/g thorium-228. Data collected from UP1 and UP2 indicate that thorium-232 and its progeny are present in approximately equal concentrations, indicating that these radionuclides are in secular equilibrium. The data also indicate that uranium-238 and its decay products, thorium-230 and radium-226, are also present in elevated concentrations. The maximum concentrations in UP1 are 51 pCi/g uranium-238, 120 pCi/g thorium-230, and 331 pCi/g radium-226. However, the data reveal that uranium-238 and its progeny are not present in equal concentrations, and, therefore, are not in secular equilibrium. This may result from the presence of tailings from mineral processing operations that selectively concentrated radionuclides of this series in the soils. Although radionuclides in the thorium-232 and uranium-238 series were not determined for UP3, UP4, and UP5, one would expect concentrations of the radionuclides in these soil samples to be similar to those of UP1 and UP2. The expectation is that the other radionuclides in the thorium-232 series appear to be in equilibrium and that the concentrations of these radionuclides are approximately equal to those of radium-228. The radionuclides in the uranium-238 series are not in secular equilibrium in UP1 and UP2 and are not expected to be in equilibrium in UP3, UP4, and UP5. By comparison to the concentration of the series in UP1 and UP2, one would expect radium-226 and thorium-230 to be the predominant radionuclides present from this series.

The data from separation by both wet sieving and hydroclassification of Under-Pile samples, exemplified by the data for UP1 (Table II and Fig. 2), illustrate that the concentration of each radionuclide generally increases as particle size decreases. In each case, the -0.037-mm (-400-mesh) fraction from wet sieving and the -0.074-mm (-200-mesh) fraction from hydroclassification contain considerably higher concentrations of all radionuclides reported. For example, the concentration of thorium-228 in UP1 is 792 pCi/g while the -0.037-mm (-400-mesh) fraction from wet sieving contains 2300 pCi/g. Note that the preceding size fraction, -0.053/+0.0037-mm (-270/+400-mesh), contains less thorium-228 than the whole soil itself, 549 pCi/g. This behavior is likely the result of radioactive cations absorbed to smaller silt and clay particles and the presence of radiominerals of smaller size.

One soil fraction from hydroclassification of both UP1 and UP2, the -0.297/+0.250-mm (-50/+60-mesh) fraction, contains an unexpectedly high concentration of each radionuclide reported. As noted above, this size fraction contains less material than the corresponding fraction from wet sieving, likely because less-dense particles of the -0.297/+0.250-mm (-50/+60-mesh) fraction are reporting to smaller-sized fractions during hydroclassification. Apparently, these less dense particles are natural soil particles, and those particles remaining in the -0.297/+0.250-mm (-50/+60-mesh) fraction after hydroclassification are more highly radioactive particles consisting of the radiominerals that constitute part of the contamination.

The process water from the vigorous wash and separation procedures on the Under-Pile samples contained less than the MDC for radium-226, 102 pCi/L, and less than or equal to 107 pCi/L radium-228, indicating that the contaminants are insoluble.

The results of radionuclide analysis for uranium, thorium, and radium for the Pile samples are presented in Table III. The data indicate that the predominate contaminants are from the thorium-232 series (thorium-232, thorium-228, and radium-228), although they are not in equal concentrations in the samples. The thorium-232 and thorium-228 concentrations range from approximately 3 pCi/g to 27 pCi/g, while the radium-228 ranges from approximately 3 to 35 pCi/g. The data also indicate that these radionuclides are in equilibrium for all but two Pile samples, P4 and P5. Radionuclides from the uranium-238 series (uranium-238, thorium-230, and radium-226) are also present in lesser amounts with concentrations ranging from approximately 2 to 10 pCi/g for uranium-238, 1 to 5 pCi/g for thorium-230, and 2 to 10 pCi/g for radium-226. The data indicate that these radionuclides are not in equilibrium since the thorium-230 concentrations are considerable less than those of uranium-238 and radium-226.

The results of analysis for select radium, thorium, and uranium isotopes for Pile whole-soil samples and their particle-size distributions produced by wet sieving and hydroclassification are illustrated in Table IV and Fig. 2 for P1 which is representative of all five Pile samples. The data from separation by wet sieving and hydroclassification illustrate that the concentrations of radium-226 and -228 generally increase as the particle size decreases. All fractions larger than 0.074*7-mesh) size contain less than 5 pCi/g radium-226 and -228. The majority of the radium contamination is in the -0.074-mm (-200-mesh) fractions ranging from 3 to 22 pCi/g radium-226 with an average of 14 pCi/g, and 6 to 63 pCi/g radium-228 with an average of 37 pCi/g. Radioanalyses of sample P1 for uranium-238 and thorium-232 and -230 (Table IV) also indicate that these radionuclides are concentrated in the -0.074-mm (-200-mesh) fraction.

Additional sieving analyses of the -0.074-mm (-200-mesh) fractions initially produced by both wet sieving and hydroclassification (see Table IV) indicate that the -0.045-mm (-325-mesh) fraction contains a major portion of the radium. Figure 3 illustrates that approximately 65 percent of sample P1 can be recovered with a radium-226 concentration of approximately 2 pCi/g and radium-228 of 4 pCi/g. The separation results of sample P1 at both 0.074-mm (200-mesh) and 0.045-mm (325-mesh) size by wet sieving and hydroclassification. Both methods produce a +0.045-mm (+325-mesh) product with less than 5 pCi/g radium-226 or radium-228 that represents approximately two-thirds of the sample. Crushing the -0.074-mm (-200-mesh) fractions before sieving at 0.045-mm (325-mesh) size (see above) may have generated additional fine particles in the samples thus altering the size distribution that would typically be found in the sample below 0.074 mm (200 mesh). Even if the particle-size distribution of the -0.074-mm (-200-mesh) fractions is not as precise as one produced from a whole Pile sample, the data do indicate that separation at 0.045-mm (325-mesh) size is a reasonable remediation candidate. This conclusion is based on the premise that the separation behavior of P1 is representative of the remaining five Pile samples, a reasonable premise since the particle-size and radionuclide distributions of all six samples are very similar.

The process water from the vigorous wash and separation procedures on the Pile samples contained less than the MDC for radium-226, 41 pCi/L, and less than or equal to 21 pCi/L radium-228.

The five Under-Pile samples contain considerably greater concentrations of radionuclides than the six Pile samples. The Pile samples contain approximately 17 pCi/g of both thorium-232 and radium-228. Radium-226, thorium-230, and uranium-238 are also present in lesser concentrations. Both wet sieving and hydroclassification of the Pile samples separated a -0.074-mm (-200-mesh) fraction that represents 47 percent of the whole sample, producing a +0.074-mm (+200-mesh) product that, in turn, represents 53 percent of the soil with an average radium-228 concentration of 1.6 pCi/g and an average radium-228 concentration of 1.0 pCi/g. Although the Under-Pile samples and Pile samples have similar particle-size distributions and are amenable to remediation from this standpoint, the radionuclide concentrations of the Pile samples from this study are considerably higher, and the concentration of radionuclides in the size fractions may well be above clean-up criteria to be established for this site (see Table II). Therefore, volume reduction by particle-size separation alone is not likely a viable option for remediating the material from under the pile represented by UP1 through UP5 in this study. In sharp contrast, soil represented by the Pile samples is a viable candidate for volume reduction by particle-size separation as illustrated in Fig. 3, with a potential

recovery of 53 percent of the soil above 0.074-mm (200-mesh) size or 65 percent above 0.045-mm (325-mesh) size containing a radium-228 concentration of less than 5 pCi/g and 15 pCi/g, respectively. No samples adjacent to the pile or adjacent to the site were provided for examination, and no conclusions about treatability of these areas can be drawn from the results of this study.

Petrographic Analysis

Petrographic examination was performed on the six Pile samples. The purpose of this examination was to identify the waste forms of the radioactivity and their distribution within the various size fractions. The mineral composition of the six samples is presented in Table V. An inspection of the composition of the soils reveals a variety of materials common to the unconsolidated surface formation of the area. The granitic rocks average 10 percent of the soil samples and are the predominant materials in the medium-to-coarse sand and the gravel-size fractions (+0.297-mm or +50-mesh). Granitic rocks contain 10-20 ppm thorium (6) and 0.15-210 ppm uranium (7) and constitute the bulk of low, and virtually equal, concentrations of the radium-226 and radium-228 reported in the medium and coarse sand and gravel-size fractions of the soil samples.

The concentrations of radionuclides increase markedly in the -0.074-mm (-200-mesh) fractions. The increase corresponds to the quantity of heavy minerals, which averages 4 percent of the soil but is restricted to this fraction. The heavy minerals range from 6 to 10 percent of the -0.074-mm material. The radioactive minerals in the heavy mineral suite are monazite, (Ce, La, Y, Th)PO4, and zircon, ZrSiO4. Analysis of the heavy minerals reveals that monazite, the chief ore mineral of rare earth elements and thorium, comprises 10 percent of the mixture. Its clean, smooth, subrounded-to-rounded appearance reflects a placer deposit origin as do many of the other heavy minerals that are commonly called gangue minerals. Zircon constitutes 7 percent of the heavy-mineral mixture.

Monazite, the principal ore mineral of thorium (primarily thorium-232), contains from 3 to 10 percent by weight thorium oxide and from 0.2 to 0.6 percent uranium-238; zircon can contain up to 4 percent thorium or uranium substituted for zirconium (8). The predominance of thorium-232 and radium-228 over uranium-238 and its progeny, thorium-230 and radium-226, in the -0.074-mm material suggests that thorium-232 and radium-228 are primarily from monazite. The thorium-232 and radium-228 are essentially in equilibrium as are uranium-238 and its progeny, indicating that no isotope has been enriched or depleted by artificial or natural means. Thus, the insoluble monazite appears to be responsible for most of the thorium-232 activity in the Pile samples with very minor amounts from insoluble zircon and possibly absorbed cations. Both minerals are from the placer ore material introduced into the host soil. The absence of specific materials typically found associated with industrial sites that chemically extract rare-earth elements or thorium is also apparent in the Pile samples from the Site. In recent investigations of another Superfund site (9, 10), one location had anthropogenic materials that reflected these industrial processes. Finger-print materials include porous, low-density calcite, gypsum, anhydrite, and calcium (thorium) orthophosphate. It is concluded, therefore, that the thorium contamination at the site is primarily the result of insoluble monazite, which comprises less than one percent of the soil by weight (4 percent of the soil is heavy mineral; of which approximately 10 percent is monazite). Zircon makes a very minor contribution.

The sample host material is generally comprised of fresh to moderately weathered, rough surfaced, subangular-to-angular, rock and mineral particles with a specific gravity averaging about 2.7. Less-dense wood particles and hollow plant stems make up less than one percent of this material. The heavy minerals (approximately 4 percent of the material) consist of fresh, smooth surfaced, subangular to subrounded particles with a grain specific gravity ranging from 3.0 to 5.2. Monazite and zircon, with grain specific gravities of 4.5 to 5.5 and 3.9 to 4.8, respectively, are the most dense of the heavy minerals and are smooth, subrounded materials that could separate well from the host material by settling in a water medium. About one-third of the heavy minerals are tabular-elongate in shape with specific gravities ranging from 3.0 to 3.4 that, because of their shape, would tend to behave more like a less-dense particle; these minerals include hornblende, hypersthene, kyanite, sillimanite, and tourmaline. Although a complete analysis was not performed on the composition of the fine silt (-0.045/+0.002 mm) and clay-size materials (less than 0.002 mm), the -0.045-mm material is believed to be predominantly clay minerals

of flat shape and light-weight specific gravity. A density separation of host materials from heavy minerals in the -0.074-mm (-200 mesh) fraction might be a candidate for removal of large amounts of the radionuclide activity from the fraction to enhance the recovery of soil material with acceptable radionuclide concentrations.

Monazite also has a magnetic susceptibility that might permit its separation on a Frantz Isodynamic Separator. The magnetic susceptibilities of the heavy minerals occurring in the Pile range from 0.0 to greater than 1.7 (11). The monazite has an intermediate magnetic susceptibility with virtually no other mineral except tourmaline, present in minuscule quantities, in the best extraction range on the Frantz Isodynamic Separator. The zircon has a magnetic susceptibility common to several other heavy minerals as well as the abundant light mineral quartz. For that reason, magnetic separation would not be a viable candidate for isolation of this contaminant.

Chemical extraction with select reagents might be exploited to solubilize the monazite and zircon in the -0.074-mm fractions (12). Radiocations absorbed on the surface of silt and clay particles might also be released by this process.

CONCLUSIONS

Soil samples from a storage pile at a DOE Superfund site were partially characterized prior to consideration of a full-scale treatability study. The primary objective was to determine if particle-size separation could be used for volume reduction of the contaminated soil. Six samples from within the pile, P1 through P6, were examined initially to assess the homogeneity of the storage pile to determine if the radionuclide contamination is similar throughout the pile, to determine if particle-size separation techniques might be effective in reducing the volume of contaminated soil, and to determine the physical form and properties of the contamination in order to evaluate other physical and/or chemical properties of these materials that might be exploited for volume reduction. Five additional samples from under the pile, UP1 through UP5, were subsequently examined to provide additional characterization of the site. This section presents the conclusions of these studies. No samples adjacent to the pile or adjacent to the site were provided for examination, however, and no conclusions about treatability of these areas can be drawn from the results of this study.

The particle-size distributions of UP2, UP3, and UP4 are very similar to each other and to the six Pile samples, P1 through P6. Each sample contains approximately 48 percent -0.074-mm (-200-mesh) material. The Pile samples may contain up to 35 percent -0.045-mm (-325-mesh) material, as indicated by additional sieving of sample P1 whose particle-size distribution is very similar to the other five Pile samples. Although UP1 is similar in particle-size distribution to UP2, UP3, and UP4, UP1 contains approximately 60 percent -0.074-mm (-200-mesh) material. Sample UP5 contains only 40 percent -0.074-mm (-200-mesh) material with more particles reporting to the fine sand-size fraction.

While the relative particle-size distributions are similar, the radionuclide concentrations in the Under-Pile samples are approximately 10 to 50 times those of the samples from within the storage pile. The predominate radionuclide contaminants in the Under-Pile samples are thorium-232 and its decay products, thorium-228 and radium-228, with concentrations of radium-228 up to 1700 pCi/g. Uranium-238 and its progeny, thorium-230 and radium-226, are also present in significant but lesser concentrations, with radium-226 concentrations up to 330 pCi/g. The major contaminants in the Pile samples are radionuclides in the thorium-232 series, with concentrations of radium-228 up to only 35 pCi/g. Uranium-238 and its progeny, thorium-230 and radium-226, are present in lesser concentrations with radium-226 concentrations less than 15 pCi/g.

The concentration of each radionuclide in the size fractions generally increases as particle size decreases. In all cases, the -0.037-mm (-400-mesh) fractions contain a much higher concentration of each radionuclide than the whole sample from which it was separated; sample UP3 contains 810 pCi/g radium-226 in the -0.037-mm fraction compared to an average of 231 pCi/g in the whole sample. Radionuclides in the -0.074-mm (-200-mesh) fraction of the Pile samples and the -0.045-mm (-325-mesh) fraction of P1 represent much of the contaminant content in these samples. Radionuclide analyses of the whole sample and size fractions from P1, for example, reveal that the whole sample contains 6 pCi/g radium-226 while the fractions larger than 0.074-mm (200-mesh) contain 3 pCi/g or less radium-226. The -0.074-mm

(-200-mesh) fraction contains approximately 12 pCi/g and the -0.045-mm (-325-mesh) fraction contains 14 pCi/g.

Wet sieving and hydroclassification of the Under-Pile samples produced similar results except for one size fraction, -0.297/+0.25-mm (-50/+60-mesh). This fraction produced by hydroclassification contained approximately 85 percent less material than that produced by wet sieving, likely as a result of hydrodynamic behavior of less dense particles from the -0.297/+0.250-mm (-50/+60-mesh) fraction. Wet sieving and hydroclassification of the Pile samples produced almost identical results. The process water from the separation procedures on the Under-Pile samples contained less than the MDC of radium-226, 102 pCi/L, and less than or equal to 107 pCi/L radium-228. The water from the separation procedures on the Pile samples contained less than MDC of radium-226, 41 pCi/L, and less than or equal to 21 pCi/L radium-228. These results indicate that the radionuclides are essentially insoluble in water.

Radionuclides in the thorium-232 series are in secular equilibrium in the whole Under-Pile samples, while those in the uranium-238 series are not. In the Pile samples, the thorium-232 series is in secular equilibrium, except for samples P4 and P5. The uranium-238 series is not in equilibrium in any of the Pile samples since the thorium-230 concentrations are significantly less than those of uranium-238 and radium-226.

Although the particle-size distributions of the Under-Pile samples indicate the samples from under the pile are conducive to separation by particle-size, the radionuclide concentrations of all radionuclides reported are higher than those of the six Pile samples analyzed from the storage pile, and the concentrations throughout the soils are above acceptable clean-up criteria for most sites. Therefore, volume reduction by particle-size separation, alone, of the Under-Pile samples examined in this study is not a viable option for remediation, although other physical properties of the contaminants, such as density, might be used to achieve this goal. In contrast, the particle-size distribution of the Pile samples indicates that 53 to 65 percent of the samples could be recovered by particle-size separation with radionuclide-contaminant concentration of less than 5 or 15 pCi/g, respectively. The data indicate that these results could be accomplished by separation at 0.074 mm (200 mesh) or 0.045 mm (325 mesh), respectively.

The primary source of the thorium radioactivity in the Pile samples is monazite. Zircon is also present and contributes a small amount of thorium to the sample. The monazite and zircon are concentrated in the fine-sand to upper silt-size range, -0.149/+0.037-mm (-100/+400-mesh). The specific gravities of the monazite and zircon are about two units above the average specific gravity of the host material. Density separation might be used to provide additional volume reduction of these fractions. The magnetic susceptibility of monazite is in a unique range, when compared to other magnetically susceptible minerals, this might also offer a separation method to reduce the volume.

Chemical extraction might be employed to solubilize the monazite and zircon in the -0.074-mm (-200-mesh) fractions. Any radiocations absorbed to the surface of these particles might also be released during these procedures.

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TANGENTIAL SCANNING SYSTEM FOR WASTE DRUM INSPECTION

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ABSTRACT

We have developed an advanced x-ray Tangential CT Scanner System to image and automatically detect flaws in rocket motors, warheads and missiles under a SBIR contract from the Naval Surface Warfare Center. The prototype scanner system uses a linear array of 160 solid state x-ray detectors and a 320 kV x-ray tube. During the scan, the rocket motor rapidly rotates around it's own axis and the detector array slowly translates, slowly varying the diameter of the tangential path in the rocket motor. In priciple, this scanning method is analogous to unrolling of a roll of paper towel. The x-ray scanning is continued until the entire radius (for 180oC scans) or the diameter (for 360o scans) of the rocket motor is covered by the tangential path Each individual detector scans a single CT slice of the rocket motor. The 160 detector array simultaneously collects raw data for 160 individual CT slices and covers the entire length of a 10" long rocket motor. Thus such a scanner system is a true volume CT imaging system. In addition to the hardware, we have also developed a windows based unique data analysis and display software with this system. Our software package displays the entire volume of collected data from the tangential scanner in three different modes. In first, mode the data is displayed in the form of Tangential views at various radius of the object. The second mode displays the data in the form of sinogram for each detecor of the detector array. The third mode displays the data in the form of many digital x-ray radiographs of the object from various directions.
Tangential scanning scheme has a great potential for low and high level Nuclear Waste Drum and other similar objects. The tangential scanner system naturally collects hundreds of digital radiographic images at various rotational angles in very short period. For comparison, the recently purchased digital radiographic system by the WRAP-1 at Hanford is expected to collect only two digital radiographs of each waste drum. The tangential views from such a scanner system are very sensitive to the defects in the wall of the drums. Our initial experiments with a simulated waste drums clearly show that quantitative measurements of the actual wall thickness and wall material loss due to corrosion can be easily measured with the tangential scanner system.
At present, we are searching for funds to develop and fabricate a prototype tangential scanner system suitable for actual production applications. Our proposed system will be able to collect individual Digital Radiographs in <5 seconds and complete tangential scans in about 2.5 minutes

INTRODUCTION

We have developed an advanced x-ray Tangential CT Scanner System to image and automatically detect flaws in rocket motors, warheads and missiles under a SBIR contract from the Naval Surface Warfare Center. The prototype scanner system uses a linear array of 160 solid state x-ray detectors which is placed parallel to the axis of the rocket motor. A 320 kV x-ray tube is located on one side of the rocket motor and the detector array on the other side. The x-ray tube and the detector array are

mounted on a C-arm with a pivot point at the x-ray focal spot. The rocket motor sits on top of two rollers. One of the rollers rotates the rocket motor while the other containing an encoder tracks the azimuthal location of the rocket motor. An indexing motor rotates the detector array slowly around the pivot point i.e. the x-ray focal spot. The x-ray beams from the focal spot to the detector array form tangential paths in the rocket motor. Figure 1 show the top and side views of this system. During the scan, the rocket motor is rapidly rotated around it's own axis and the detector array is slowly rotated around the x-ray focal spot (pivot point) to slowly vary the diameter of the tangential path in the rocket motor. In principle, this scanning method is analogous to unrolling of a roll of paper towel. The x-ray scanning is continued until the entire radius (for 180o scans) or the diameter (for 360o scans) of the rocket motor is covered by the tangential path. The detectors measure the x- ray intensity through all possible tangents through the object. Each individual detector scans a single CT slice of the rocket motor. The 160 detector array covers the entire length of a 10" long rocket motor. Thus the scanner system is a true volume CT imaging system.

The prototype system uses a 320 kV Pantak high frequency constant potential x-ray system. The system uses a high performance x-ray detector system specifically developed for this system by Omega. Though the prototype scanner system only uses 320 kV x-ray source, the detector system is designed for up to 450 kV x-ray energy. The hardware design of this scanner system has several unique and substantial advantages over the presently available CT imaging systems. Few of the major advantages are that this system design leads to significantly faster scanning time, better image contrast and simultaneous volume scanning. This system is particularly suitable and extremely sensitive to the unbond type defects in rocket motors.

The tangential imaging is known to be very sensitive for the detection of unbonds and other defects which are parallel to the wall of the object. At present, the motor manufacturers use x-ray tangential film radiography to detect unbond defects. The film radiography requires x-ray exposures from many directions to cover the entire rocket motor. Even with many exposures, often small unbond defects can be missed. The tangential scanner collects data from all directions and thus even small defects are naturally detected with the present system.

In addition to the hardware, we have also developed a windows based unique data analysis and display software with the system. Our software package displays the entire volume of collected data from the tangential scanner in three different modes. All three modes of data display are simultaneously available in real-time. The actual data collection concept and three modes of display are shown at the top of Fig. 2.

In one mode the data is displayed in the form of Tangential views at various radius of the object. This data display is equivalent to unfolding a roll of paper towel one layer at time. This mode displays the unrolled view of each layer at a time. Any and all individual layers of the object can be viewed in this mode. The data display in this mode is very sensitive to the unbond defects.

The second mode displays the data for each detector of the detector array. The individual detector data is a sinogram for an individual slice of the rocket motor. The individual Sinogram from each detector is a complete set of raw data which can be used to reconstruct individual CT slices. Thus the Sinograms from all detectors together can be used to reconstruct all CT slices of the entire object i.e. volume CT image of the object. Data from each detector is contained in one layer of the detector data set. Since our scanner has 160 detectors, there are 160 layers of sinogram data available to the operator.

The third mode displays the data in the form of digital X-ray radiographs of the object. The total data set is organized in layers where each layer contains one radiographic image of the object. The successive layer in this mode are x-ray radiographs after a slight rotation of the object. This mode is similar to radiographic images of a rotating object in a real-time radiography (RTR) system. Operator can roam through various layers of the data set and actually view successive radiographic images as the object rotates. In a typical x-ray data set, there may be several hundred to several thousand such digital radiographs available. For comparison, in a typical digital radiography system, only one or two such digital x-ray radiographs are collected for any object.

The tangential CT scanning system has a great potential for inspecting small and large Rocket Motors, Warheads, Missiles. The data collected from a rocket motor

phantom is displayed at the bottom of Fig. 2. This phantom is made of a 4" diameter cylindrical pipe filled with simulated rocket fuel. Another 1.5" diameter tube is located in the middle of this phantom. This 1.5" diameter tube extends from one end to only the middle of the phantom along it's length. This particular phantom contains two unbond type defects between the simulated rocket fuel and the external 4" wall. It contains two unbond types defects between the simulated fuel and internal 1.5" wall. It also contains two notches cut out at the inside of the internal 1.5" wall along it's full length.

In Fig. 2 all three views i.e. tangential, detector sinogram and radiographic views are shown in that order for the navy phantom. The left side image of this figure shows the tangential view. In this view we see several bright sections in the image indicating larger x-ray signal or loss of material due to unbond defects. Due to 360o scanning, each defect shows up two times in the tangential data, once when the defect is towards the x-ray tube and second time when it is towards the detector side. We also see indication of two notches which extend from one end of the image to the middle of the image.

The middle image of this figure shows the detector sinogram view. In this view we see two faint sine waves in the middle which extend only to the inside wall of the internal 1.5" tube. This is a signal from the two notches towards the inside of the internal tube. These two sine waves are also out of phase from each other, indicating that these notches are 180o. from each other. We also see a fairly bright sine wave which extends to the outside wall of the internal tube, indicating a unbond defect on the outside of the internal 1.5" tube. We also see a faint indication of another sine wave which extends to the outside wall, indicating a unbond at the outer wall. Since the defects are located at different locations along the length of the phantom, all defects do not show up in this particular detector. Other unbond defects are indicated in other detector Sinograms.

The right side image of this figure shows the radiographic view of the phantom. In this image we see the presence of the two notches and several unbond features. When we view the successive layer of the radiographic view images, we view the rotation of the notches and unbond features. Again, the notches extend only to the middle of the phantom along it's length. Two of the unbond features travel to the outside tube while the other two travel to the inside tube of the phantom. Thus each of these three image views contains sufficient information to exactly pinpoint the extent and location of a defect. The three views together make it significantly easier to find the location and amplitude of the problem.

In addition to the rocket motors, the system has a great potential for low and high level Nuclear Waste Drum and other similar objects. The tangential scanner system naturally collects hundreds of digital radiographic images at various rotational angles in very short period. For comparison, the recently purchased digital radiographic system by the Westinghouse Hanford group is expected to collect only two digital radiographs of each waste drum. If desired, the data collected from the tangential system can also be used to reconstruct a volume cross sectional CT image of the waste drum.

The tangential views from the tangential scanner system are very sensitive to the defects in the wall of the drums. Our initial experiments with a simulated waste drums clearly show that quantitative measurements of the actual wall thickness and wall material loss due to corrosion can be easily measured with the tangential scanner system. The wall measurements can be made independent of the complexities and the contents of the drum. This makes such a system concept extremely suitable for quantitative measurement of the drum wall integrity.

We had constructed a small phantom to simulate the waste drum filled with unknown waste. Since our prototype tangential scanner is designed for objects up to 14" diameter only, we can not use an actual waste drum for experimentation. Also, our prototype scanner is designed to scan the objects with their axis in horizontal direction only. To construct the simulation phantom, we used a six gallon water bottle. From the top of the bottle, we mounted three objects to simulate random waste. These three objects are an empty glass bottle, an empty aluminum pop can and a 3/8" diameter threaded steel rod. These objects were mounted on a " thick wood disk and hung from the top of the six gallon plastic bottle. To simulate the steel wall of the drum and it's corrosion, we surrounded the plastic bottle with a 0.031" thick steel sheet. At few places we placed additional pieces of 0.031" thick steel sheets. This gave us the simulated wall thickness of 0.031", 0.062" and 0.093" at

different places of the drum phantom. Also, we had drilled six small holes to simulate pinholes in the drum wall due to corrosion. The diameter of these six holes ranged from 0.037" to 0.127" and they were drilled in region where the wall thickness was only 0.031" i.e. single sheet of 0.031" steel. The photographs of this simulation phantom as constructed and during actual scan in our prototype gantry are shown in Fig. 3. In this figure, the bottom left photograph shows the inside components of the phantom before final assembly. The top left photograph shows the steel band with area of 0.031", 0.062" and 0.093" thicknesses and six holes ranging in diameter from 0.037" to 0.127".

In the right side of Fig. 3, we present part of the data from the simulation phantom. This figure shows the three views of the data display as discussed before. The left side view shows the tangential views of the data. The data shown in this view is the 63rd tangential layer of the data. As discussed before, it is a 360 degree circular layer data unfolded. In this tangential view, one can easily see sections with different thickness of steel sheets. We also see the region with six holes in the steel sheet. As discussed before, all of these features show up twice in the data, once when they are towards the x-ray tube and second time when they are towards the detector array. The data also shows the presence of the glass bottle, empty pop can and other features inside the plastic bottle phantom. In this figure, the middle view shows the detector sinogram data. The data shown in this view is for detector number 80. The data clearly shows the sine waves due to various thickness regions of the steel sheets. When a detector is lined up with one of the holes, data also shows the sine waves due to holes as well. In addition to the steel sheets, the data also shows the sine waves due to other objects inside the plastic bottle phantom. The right side view in this figure, shows the radiographic image of the data. The data shown in this view is the 181th radiographic view. The data in this clearly shows the presence of several thickness of the steel sheet and six holes. It also shows the presence of the glass bottle and empty pop can among other things. If we try to view the successive radiographic views, we find images similar to images from a real time radiography system with the object rotation.

This limited experimentation with the simulation phantom clearly shows that the tangential scanning system has capabilities far beyond any presently available technologies for characterization of waste drums. Specifically, the wall of the drum can be analyzed with the tangential system without interference from the content of the drum. This is an unique and powerful method to examine the condition of the drum wall without an advance need to know what is inside the drum. Figure 4 shows a system concept for scanning a waste drum in production application. The drum is located vertically on a rotary motion stage. The rotary stage is mounted on a linear conveyor belt or some other type of linear motion system. The solid state detector array is located on one side of the waste drum and the x-ray tube on the other side. At present, we are searching for funds from DOE to develop and fabricate a prototype tangential scanner system suitable for actual production applications. In our proposed system, there will be 600 detectors and the length of the detector array will be sufficient to cover the entire drum. The energy of the x-ray tube will be 450 kV and is sufficient to penetrate the drum and it's contents. The drum rotates around it's axis at maximum speed of about 30 RPM and maximum linear speed of the conveyor belt will be about 60 cm per second.

Following are the scanning time estimates for various scan modes for an entire waste drum using the proposed prototype designs.

Digital Radiography for initial mode	< 5 seconds
High Resolution Digital Radiography	< 5 seconds
Regular Tangential scan	2.5 minutes
High resolution tangential scan	10 minutes

Following are the tentative design specifications of our proposed system

1) Detector and data acquisition system

Total number of x-ray detectors	600
Detector active area	3.3 mm x 3.3 mm
Total length of detector array	198 cm
Detector system dynamic range	16 Bits (65535 levels)
Maximum data rate	300,000 readings/second

The detector array will be placed in the vertical direction on an arc and will view the entire height of the waste drum.

2) System geometry

Focal spot-to-detector distance	272 cm
X-ray beam fan angle	40.0o
Focal spot-to-drum distance	132 cm

3) X-ray system

Maximum x-ray energy	450 kV
Maximum mA	10 mA

4) Motion system for drum

The x-ray tube and the detector array will mostly remain stationary in the tangential scanner system. Mostly the waste drum will carry out all required motions to collect the tangential scan data. The drum will sit on a rotary motion stage. The rotary motion stage will be located on the top of a linear motion system. Thus the drum will provide linear and rotary motions. For high resolution digital radiography imaging, the x-ray tube system will provide a very small added up-down linear motion. During the high resolution digital radiography scan, after the first pass of data collection, the x-ray tube will move by 1.6 mm in upward direction to collect the data for the second pass.

Maximum linear speed of drum	60 cm / second
Maximum rotation speed of drum	30 rotations per minute

5) Scanning schemes

For digital radiography for initial sorting, the drum will traverse (without rotation) through the x-ray fan beam at a speed of about 60 cm per second to collect such scans. This will collect a digital radiographic image of the entire drum within < 1 second.

For high resolution digital radiography, the drum will traverse (without rotation) through the x-ray fan beam at a speed of 30 cm per second to collect first pass of the digital radiographic data. After the first pass, the x-ray tube will rise vertically by 1.6 mm and the drum will traverse backwards through the new x-ray fan beam to collect the second pass of the radiographic data. The data sets collected during these two passes will be interleaved to reconstruct a high resolution radiographic image.

For regular tangential scan, the drum will simultaneously make linear and rotary Motion to collect tangential scan data. For 180o scans, the drum will travel by a distance which is equal to it's radius (about 30 cm) while also rotating. For 360o scans, the drum will travel by a distance which is equal to it's diameter (about 60 cm) while also rotating. The linear speed of the drum will be about 4 cm per minute, and the rotation speed of the drum will be about 2.4 seconds per rotation (or 25 RPM).

For high resolution tangential scan, the drum will simultaneously make linear and rotary motion to collect high resolution tangential scan data similar to case before. The only difference between the regular and high resolution tangential scans is the speed of translation and rotation. In high resolution scan, the rotation speed is and the linear speed is 1/4 of the regular scan. Thus the system collects 2 times more data points for each rotation and there are 2 times more rotations in the high resolution data set.

6) Spatial resolution expected in the middle of a 55 gallon drum

Digital Radiography for initial mode	1.6 mm x 1.6 mm
High Resolution Digital Radiography	0.8 mm x 0.8 mm
Regular Tangential scan	1.6 mm x 1.6 mm x 1.6 mm
High resolution tangential scan	1.6 mm x 0.8 mm x 0.8 mm

7) Volume CT reconstruction time estimates for the tangential data

Detectors added together for CT slices	5
Reconstructed CT slice width with 5 detectors	8 mm
Number of CT slices reconstructed	120
Reconstruction time for regular scan	5 minutes
Reconstruction time for High resolution scan	20 minutes

At present, we are fabricating an experimental system which will use fewer detectors and slower data collection system for demonstration to potential users even before DOE funding can be found.

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this paper.

27-24
REMOTE TECHNOLOGIES FOR VENTING, PURGING AND OPENING DRUMS OF TRANSURANIC WASTE
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INTRODUCTION
During four decades of operation the DOE complex has produced hundreds of thousands of drums of transuranic (Tru) waste. These drums were packed and have been stored with the best available technology at the time of their creation. Many of these drums are not packed in a manner that is suitable for long term storage by today's standards. Many are in earth covered storage. These drums will be retrieved from their current storage, repackaged and put into a long term storage facility such as the Waste Isolation Pilot Plant (WIPP).
In the summer of 1991 the Equipment Engineering Section (EES) of the Savannah River Technology Center (SRTC) was given the task of developing methods to sample and vent head space gasses in unearthed drums and opening and removing the waste from these drums. EES began this program by investigating the existing technology to accomplish these tasks that already existed within the DOE complex. In meetings with engineers from other DOE sites many concepts were evaluated. During FY92, FY93, and FY94 EES designed, fabricated and tested equipment based on these evaluations. The following report is a summary of this work.
DRUM VENTING AND PURGING
One of the problems of dealing with Transuranic waste is how easily the contamination is spread. Even today much of the waste generated is bagged out of contaminated areas in plastic before being placed into waste containers. Radiolysis of these plastics produces hydrogen. In addition, volatile organics were used in many cleaning processes in contaminated areas and these also were placed into waste containers. Modern waste containers have vents with filters that allows these gasses to vent preventing a build up of dangerous concentrations. During the era when transuranic waste was buried in drums, these drums were sealed with gaskets and had no filter vents installed. During retrieval the head space gasses of these drums must be sampled and analyzed to determine if flammable mixtures of hydrogen or volatile organic vapors are present. If dangerous concentrations are found the vapors will have to be purged from the drum before they can be transported. At that time a filter vent will be installed to prevent further build up.
To sample the head space gasses the drum must be penetrated. This penetration creates a potential path for the spread of contamination. To solve this problem a chamber with a seal at the spindle penetration and a seal where it attaches to the drum are used. This ensures the containment of any contamination. In addition this allows a vacuum to be drawn in the chamber before the drum is penetrated. In this way an undiluted sample of the headspace vapors can be drawn from the drum for analyses. Two other potential causes of releasing contamination are removing the penetration tool and removing the chamber to install the filter. So a filter with a penetration tool attached was designed. In this way the penetration can be made, the gasses sampled and the filter vent installed while the vacuum chamber is installed. Because of solid objects that may be near the top of a filled drum the penetration tool should be designed as short as possible. The penetration tool also has to be designed to minimize the chance of creating an ignition source. 3/8" drills with titanium carbide coatings have been tested on 1/4" plate steel and have not created temperatures that exceed the auto ignition temperature of hydrogen.
At the time the sealed drums of transuranic waste are unearthed they will have to be handled as if the headspace gasses are flammable until the chemistry of the gasses can be analyzed and shown to be safe. Although analyses has shown that detonations will not occur, if a deflagration were to occur contamination would be spread. Workers will have to be protected from the potential hazards. For this reason a system that can be easily transported to the retrieval site is highly desirable.
The potential spread of contamination from a breached drum does not pose a threat to employees not at the immediate work site or to off site exposure. But if the threat of breaching a drum exists the workers in the immediate area must be protected. If this is done with personal protective equipment several operational considerations become factors. Waste in the form of protective clothing is generated every day. In addition there is a loss of productivity when working in protective equipment along

with the time of dressing out several times each shift. By placing the drum in a containment or confinement system the operators are protected from possible contamination if a breach occurs.
Testing was conducted with a variety of penetration tools. Initial tests were performed with slow and high speed punches of various configurations. These produced rough holes that were difficult to seal the filters on. 1" drills were tried but required high torque. Even step drills would stall out a one horsepower spindle when the large diameter broke through the sheet metal drum lid. Hole saws required less torque but had difficulty starting on the sloped drum lid. Hole saws with pilot drills were tried with success. Clean holes were produced consistently without overloading the motor. After producing a series of holes, filter installation was attempted. High starting torques were required for the large filters. This would often cause the threads to strip. Reducing the torque would cause the unit to stall before the filter was completely seated.
Next a series of tests were conducted using filters with 7/16 inch diameter threads. U, V and W drills were tried, all successfully drilled clean holes. Seating of the filters was more difficult. Both the U and V drills would require high torque sometimes stalling before the filter was seated. Other times they would strip the threads off the filter. The W drill provided holes that the filters could be reliably seated in. Several holders for the filters were designed and tried. Ease of loading and concentric holding to prevent drill point wobble were the criteria. Holders using the hexagonal shaped filters proved troublesome and complicated. By using a round filter body with groves for a key way drive a simple filter holder was designed. The vacuum chamber was tested and holds vacuums to less than 1/20th of an atmosphere.
DRUM OPENING
After the drums have been retrieved and sampled they will have to be opened and the contents processed and repackaged for long term storage. This could be performed hands-on because of the low dose rates. Some of the advantages of doing this remotely are the reduction in personnel exposure and the avoidance of new waste generation, in the form of protective clothing. The condition of the drums will be highly variable, some will be dented and/or corroded. The contents will also be inconsistent. Some of the drums will have containers of liquids, glass, heavy objects, aerosol cans, containers of fine powders, etc. The opening equipment will need to minimize the disturbance of the waste and minimize the spread of contamination.
To minimize the spread of contamination it was decided to open the drum and remove the contents rather than to try and pour contents from the drum. In this way the contents could be handled individually without the risk of heavy objects crashing down on soft bags or fragile containers that could cause excessive contamination inside the cell. The equipment was designed to cut the drum away from the liner then cut off the liner top. Attempting to remove the lids remotely introduced several technical problems that would have made the equipment complicated. In addition rotating tools around the drum would add complexity. The design configuration selected has a turntable to rotate the drum and tools mounted to vertical tool posts, basically a vertical turning lathe. Tools were tested to determine which were most effective in remote operations. The main criteria were repeatable remote operation, minimum penetration into the drum and minimum production of secondary waste from the cutting operation. Since the drums will vary in condition a variety of cutters will be required. Tests were conducted on the drum with a rigid pipe cutter, a pneumatically driven compliant pipe cutter, a parting tool, a pneumatic circular saw and a router. Tests on the liner were conducted with a rigid knife, a heated rigid knife, a pneumatic saw and a pipe wheel cutter.
The rigid pipe cutter was tested using a constant RPM on the turntable and a constant feed rate on the cutter. The drum rotation was initiated, then the cutter advanced until it contacted the drum. At that time the constant feed rate was set. Feed rates of .002 to .005 in/rev were tested at turntable speeds of 20 to 70 RPM. When the higher feed rates were tried the surface of the drum would roll into the liner during cutting, producing a large burr and causing it to grip the liner. This made removing the drum from the liner difficult. Turntable rates above 50 RPM caused excessive vibration because the drums are out of round. This also directed us to using a large pipe cutter wheel, 2.2 in. diameter, which helped keep the tool holder from running into the drum side wall. Turntable speeds of 50 RPM with feed rates of

.003 in/rev produced clean cuts. Cycle time at these speeds were between 3 and 4 minutes.
The compliant pipe cutter was tested using a rigid slide driven by a 2 inch pneumatic cylinder, RPM and air pressure were held constant during each test. The cutter would track the surface of the drum as it rotated. Two stand off wheels were mounted above and below the cutter to prevent deep penetration into the drum when it broke through. Tests were conducted between 25 and 75 RPM and with pneumatic pressures of 30 to 70 psig. Again clean cuts were obtained most of the time. Running at 40 RPM and 70 psig the cycle times were between 2 and 3 minutes. Problems observed with this system were a jack hammering effect that would sometimes occur. When this happened the drum would end up with a series of dents that would make removing the lid difficult.
The rigid parting tool consisted of a standard 1/8 in. thick lathe cutoff tool. The parting tool was fed off a rigid arm at a constant feed rate. Speeds tested were 20 RPM and .0015 to .003 in/rev feed. The cuts began nicely but as soon as one section cut through it would snag the drum and lock up the system. Cutting thin wall members with a cutoff tool is difficult because even though the tool may be rigidly held, the wall will flex during the cut and spring out when the first break through occurs. This results in too high a feed to the cutting edge just past the break through.
The pneumatic circular saw was a one horsepower saw with a 4 in. 36 tooth metal cutting blade. The saw was to be slowly brought in and then the drum slowly rotated. Feed rates for the saw were .01 to .001 in. per second. The saw blade stalled each time a cut was attempted.
The router tests were conducted in partnership with Merrick Engineers from Los Alamos who are developing a system for Idaho National Engineering Lab. The router consisted of a one horsepower high speed spindle. Tests were conducted between 2,000 and 12,000 RPM and feed rates 4 to 18 in/min. It was discovered early that very light tooth loading, around .0003 in. to .0004 in. worked the best. Carbide cutters were used that held up to the heat. Best results were with a 1/4 in., 2 fluted end mill run at 6,000 RPM with a 4 in/min feed rate. At this speed it would take nineteen minutes to cut the circumference of the drum. This is a long cycle time in addition to the chips produced. The advantage of this system is it has great flexibility and can be used on seriously damaged or dented drums.
The rigid knife tool on the liner consisted of a modified lathe cutoff tool with a sharpened edge. It was fed at a constant rate into the side of the liner being rotated at a constant rate. Tests were conducted between 10 and 20 RPM at feed rates of .030 to .060 in/rev. This tool produced clean cuts with no chop generation at cycle times of 3 to 4 minutes. However as the tool began to break through, torque generated by the cutter on the uncut edge would cause the top portion to twist and violently whip about. In addition the edge would usually grab at some point and cause the liner to spin inside the rotating drum preventing the cut from being completed. The knife was then electrically heated to 350 degrees F. Cuts were performed in 2 minutes using a feed of .005 in/sec at 10 RPM. When the heated knife broke through the there was none of the violent action seen with the unheated knife.
A pneumatic circular saw was tested in a similar manner as with the drum cut. A 4 in. diameter 20 tooth carbide tipped plastic saw blade was used. It was advanced into the liner and then the liner slowly rotated. Turntable speeds of between .12 and .45 Rpm were tried. Cutting times were between 2.2 and 5 minutes. The saw cut the liners but produced many chips as was expected. In addition the liners had a wide variation of wall thickness, at points being 1/4 in. thick. At these extreme thicknesses the saw blade would stall.
A rigid cutter wheel, similar to the pipe cutter was tested. A wheel made for plastic was used. Feed rates of .003 in/rev with a 30 RPM turntable speed were used resulting in 7 minute cycle times. The cuts were of good quality but had a larger cycle time than other techniques.
SUMMARY
By integrating the penetration tool with the drum filter, drums of transuranic waste can be sampled, vented and have filters installed in a single, sealed operation.
A variety of tools will need to be available for remote opening of drums and liners. The rigid pipe wheel cutter produced good results on the metal drum with good cycle times while producing no chips. The problem with this cutter is that it will not perform on badly dented drums. For this reason another cutter like the router must

be available since some damaged drums are to be expected. A vacuum system will be needed to control chips. The heated knife was a reliable performer as a liner cutter and also produced no chips. This cutter though, can also have some problems particularly if there are large solid objects pressing out from the inside of the liner. Here too the router, running at a lower RPM, will prove effective.

27-26
ENVIRONMENTAL RADIOACTIVITY MONITORING, USING HELICOPTER-BORNE GAMMA MAPPING EQUIPMENT RESULTS AND RECENT ADVANCES
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ABSTRACT
The French Commissariat for Atomic Energy (CEA) has been developing for some 10 years, at VALDUC Centre, a new system of helicopter-borne gamma mapping. This system enables to map, in a few hours, the radioactivity of large areas spreading over several km up to several hundreds km, by identifying the radioelements involved, with a sensitivity ranging from the natural radioactivity level, up to those man-made radioactive levels due to an accident. In the event of nuclear accidents, such a system may be rushed to the scene, all over the country. Fitting out the helicopter, surveying a 5-10 km2 area and plotting the first contamination maps take approx. 4 hours only.
By developing new methods for analyzing helicopter-borne gamma mapping measurements, provision is made for detecting the major natural and/or man-made radioelements at ground-level, with sensitivities in the order of the natural radioactivity background level (Detection limit for Cs-137=2kBq/m).
The use of digitized 3-D maps in support to isoactivity maps improves the readability and increases the bulk of information available. Furthermore, this mapping method is applicable to environmental monitoring purposes, for the surveillance of industrial and agricultural pollutions.
FOREWORD
In any site restoration operation, the initial step is site characterization. From various characterization methods, the helicopter-borne gamma mapping is a high-grade technique, as it allows to plot, in a few hours only, the whole radioactivity map of extensive areas (from a few to up to several hundreds km), with no need for allocating ground resources, and without exposing any operator to high potential risks.
With the recent improvements brought to this technique, especially in regard of sensitivity, it is possible as from now, to apply it to monitoring the restoration work required for bringing back any contaminated area, to a radioactivity status close to natural background.
The CEA has available, for the 10 past years, a fully equipped system called HELINUC (1,2), which is described hereunder with relevant examples, and information on current developments for improving both sensitivities and map layout.
PRINCIPLE AND DESCRIPTION
Introduction to the Phenomena
The various natural or man-made gamma-emitting radioelements, liable to be detected via helicopter-borne gamma spectrometry range within 0 to 3 MeV. Hence, the spectrum detected after absorption or diffusion in the atmosphere features summing of two components :
  a series of full-absorption peaks centering on the radioelement-emitted energy, and with their FWHM as bound to detector resolution,
  a noise distributed over the whole energy range, denoting the energy distribution of Compton scattering in both the atmosphere and detector, of the cosmic radiation, of the noise originated by the equipment...
Experimental Device
The helicopter-borne device comprises a NaI detector with a capacity of 16 liters, connected to a spectrometer for signal analysis with 256 or 512 channels, ranging from 30 to 3200 keV.
Positioning of the helicopter is ensured by either a Trident transponder or a Global

Positioning System (GPS).
The Trident transponder queries the ground-level beacons set up at known geodetic points, computes the distance between the helicopter and such beacons, inferring the helicopter position with an accuracy within a few meters.
The Global Positioning System (GPS) is a satellite positioning system used in many applications, which has already been fully described elsewhere. The GPS will be soon replaced by a differential GPS featuring an accuracy equivalent to the Trident system.
The flight altitude is accurately measured by altimetric radar. The spectrum and position data management is ensured by a computer that stores the data on a Bernoulli magnetic unit.
Data Collection
The detailed site analysis provides for scanning over the area to be surveyed, according to the following set parameters : pattern size, altitude, speed, and time integration.
Fig. 1. Data collection - block diagram.
Every 2 or 3 seconds, the system records a full spectrum within the 30-3200 keV energy range, together with the helicopter X, Y, Z coordinates.
Mapping
The data collected and stored on magnetic support is processed by a dedicated ground-based computer. On the basis of the flight data record, colour-maps are processed and overprinted on topographic maps. Processing starts with the trajectory checks. Once the potential aberrant points are corrected, then starts reading of the spectra related with every measured point, and also identification of the radioelements involved. Selecting a detection window for each detected radioelement allows further computation and mapping of the standard variation of same at a reference altitude.
The isoactivity map for cesium-137 in Koslodoui N.P.P. (Bulgaria) is shown in Fig. 2.
Such maps provide, in a very short space of time, the position of the contaminated areas, which could be, in the event of any nuclear incident, usefully handed over to competent authorities.
RESULTS AND APPLICATIONS
Equivalent-Surface Activity
Any emission of radioelement detected by the data acquisition unit comprises both :
  the emission of the radioelement at the ground surface,
  the emission of such radioelement migrating into the soil.
The attenuation of the detected signal is all the more significant as such radioelement is deep in the soil and radiation energy is low. The equivalent surface activity refers to the activity arising from the remaining surface deposition of such radioelement and the (non absorbed) detectable portion of same as attenuated by burying.
In practice, with respect to cesium-137, only the initial 5 centimeters contribute to the measurement.
System Detection Limits
Under normal operating conditions :
Integration time : 3 sec.
Flight altitude : 50 meters
Flight speed : 70 km/hour
The detection limits are as follows :
Insert a
Application to Incidental or Accidental Situations
4 people are required to operate the equipment. In France, it can be operational within 12 hours, and moved to any point of the country, in an air-transport vehicle. Fitting out the helicopter, surveying a 5-10 km2 area and plotting the first contamination maps take approximately 4 hours only.
Application to Environmental Surveys
The airborne gamma mapping system offers multifarious applications. In France, the functions assigned to this system cover the following :
  systematic radiological surveillance of military sites,
  radiological surveillance and monitoring of nuclear sites, e.g. environmental surveillance of nuclear power plants,
  environment radiological surveillance of industrial sites ; refer to the

surveillance of the Danube river banks, on behalf of IAEA, in 1992 (3,4).

CURRENT DEVELOPMENTS

Mathematical Processing of Spectra(5)

The gamma spectrum analysis consists in extracting the spectrum full-absorption peaks, as only these peaks depict the nature and activity of the radioelements at ground-level. The method for peak positioning and quantification has to consider the three cardinal constraints in airborne spectrometry, namely :

the scintillation detector (NaI) features resolution varying from 8 to 10%, which makes the nuclide identification a rather tricky point, especially when they are numerous, or when their energy values are close (convoluted picks),

the counting time, necessarily low to keep up with a satisfactory spatial resolution, often results in poor counting statistics,

the distance between the source and the detector attenuates significantly the signal-to-noise ratio.

A genuine mathematical spectrum processing has been developed to analyze airborne gamma spectra. This consists in searching for the absorption peaks over the whole energy range, without making any assumption as to the nature of the radioelements involved. The relevant algorithm includes the following steps :

Use of filter algorithms adapted to the spectrum characteristics enabling to break away from the high statistical fluctuations without changing the spectrum shape.

Searching for the absorption peaks by means of one or two spectrum derivations, with prior filtering. Hence, low-intensity peaks may be located in terms of background noise.

Modelling and subtraction of the Compton scattering background enables to compute the full-absorption peaks of various radioelements.

The absorption peaks derived from processing are then subjected to validity checks for FWMH and from the statistical standpoint.

The advantages of such a process are highlighted in a typical background noise spectrum, as shown in Fig. 3 and Fig. 4. Its purpose is twofold :

systematic detection of those radioelements with absorption peak intensities higher than the detection threshold ;

provision of a sensitivity gain varying by a factor of 2 to 5 in terms of signal-to-noise ratio.

Fig. 3. Background noise spectrum and modelized Compton scattering background.

Fig. 4. Full-sbsorption peaks derived from the above background noise spectrum.

Implementation of this process enables to detect the major natural or man-made radioelements, with sensitivities approximating the natural background activity levels.

Three-dimensional Mapping

The conventional methods used in contamination maps consist in superimposing the isoactivity map and the topographic map of a given site, through scanning of a printed map to the scale of 1: 25 000. This technique is fast and low-cost; however, the plotting quality of the topographic maps is rather poor. Therefore, we have investigated a high-tech method based on the Digital Earth Model (referred to as DEM). Such models performed by ISTAR company are derived from pairs of stereoscopic pictures taken by the SPOT observation satellites.

The DEMs allow computing relief perspective views, and achieving 3-D views by 'matching' with the orthoimago.

This raised relief map will be further used in supporting the airborne gamma spectrometry data, this resulting in very realistic and accurate maps, providing a comprehensive bulk of information (relief, city-planning, etc) as a useful back-up to decision making in many respects.

CONCLUSION

The up-to-date helicopter-borne mapping equipment is a valuable tool for plotting, just in a few hours, accurate radiological-status maps regarding any contaminated area spread over several square kilometers. The detection limits (2 kBq/m for Cs-137), the flexibility in use, the quantity and quality of the measurements, make this method a highly valuable choice for monitoring any remediation work, and for characterizing the reversal to a normal radiological status of the site concerned. Nevertheless, airborne gamma mapping is not limited to this above specific domain, as it is also a requisite method used in geophysical detection as well as in environmental surveillance of nuclear sites.

Recent and forthcoming advances, such as spectrum mathematical processing, use of

digitized geographic maps and of higher-resolution detectors (HPGe) are intended to enlarge and improve the capacity, and the fields of application of such a system.

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27-27

GROUNDWATER PLUME AND SOURCE CHARACTERIZATION ON THE NORTH PLATEAU USING THE GEOPROBE SAMPLING SYSTEM

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ABSTRACT

A Strontium-90 (Sr-90) groundwater plume at the West Valley Demonstration Project in West Valley, New York, was laterally and vertically characterized in alluvial sediments using a van-mounted Geoprobe sampling system. The Geoprobe system hydraulically advances a 0.025 m diameter probe for discrete-depth sampling. Groundwater and subsurface soil samples were obtained from the Geoprobe system down to depths of 12 m. The heterogenous nature of on-site alluvial sediments have created a complex groundwater flow situation, where Sr-90 migration is controlled by local differences in hydraulic conductivity. Relative flow rates during groundwater sampling, and the gross beta activity in groundwater were used to determine that the Sr-90 groundwater plume, north of the Process Building, funnels down to a narrow flow path. Narrowing of the plume was not obvious from monitoring well sample data alone.

Groundwater and subsurface soil samples were collected in the Process Building with a modified Geoprobe sampling system, due to accessibility limitations. The sampler was driven by an air-powered jackhammer suspended from scaffolding. Groundwater and soil sampling data beneath the building were used for contaminant source identification.

The Geoprobe sampling system was found to have many advantages over conventional groundwater and subsurface soil sampling methodologies (i.e., monitoring wells and soil borings). These advantages include:

- Multiple discrete-depth groundwater samples at each probing location
- Proposed location and number of probing sites able to be modified, based on screening results on groundwater samples
- Significant waste minimization
- Accessible to difficult sampling areas
- Sample collection significantly faster

When combined with monitoring well sampling results, Geoprobe data can provide a better characterization of groundwater flow and contaminant migration in shallow, unconsolidated materials. This data can ultimately be used to develop cost effective remedial solutions, or to locate additional monitoring wells more effectively.

INTRODUCTION

Site History

The Western New York Nuclear Service Center (WNYNSC), located near West Valley, New York, about 50 km south of Buffalo, was the site of the only commercial nuclear fuel reprocessing facility ever operated in the United States.

The New York State Energy Research and Development Authority (NYSERDA) owns the site and facilities. Nuclear Fuel Services, Inc. (NFS), the commercial operator, operated the plant and the WNYNSC under a Nuclear Regulatory Commission (NRC) operating

license from 1966 to 1972 on land leased by NFS from New York State, processing about 640 metric tons of spent reactor fuel. The reprocessing operation generated approximately 2.3 million liters of high-level radioactive waste that was transferred into underground tanks for storage. Reprocessing operations ceased in 1972.

In 1980 Congress passed the West Valley Demonstration Project (WVDP) Act to demonstrate that high-level waste can be solidified in a form suitable for transportation and disposal; suitable containers can be developed and the waste transported to a federal repository for permanent disposal; low-level and transuranic waste from the project in accordance with applicable licensing requirements can be safely disposed and the tanks, facilities, and any material and hardware used in connection with the project can be decontaminated and decommissioned in accordance with NRC requirements. In September 1981, the NRC amended the operating license to transfer possession of the facilities to the Department of Energy (DOE) for conduct of the WVDP (NYSERDA continues to be licensed as the owner). On February 25, 1982, DOE assumed operational control of approximately 89 hectares of the 1336-hectare WNYNSC in order to conduct the WVDP. The DOE through its contractor, West Valley Nuclear Services Company, Inc. (WVNS), a wholly-owned subsidiary of Westinghouse Electric Corporation, is currently performing operations as stipulated in the WVDP Act, the Memorandum of Understanding between DOE and NRC, and the Cooperative Agreement between DOE and NYSERDA.

Scope of Groundwater Sampling Program

In the summer of 1994, WVNS implemented a subsurface probing (Geoprobe) program on the North Plateau of the WVDP. The Geoprobe system collected groundwater and subsurface soil samples by hydraulically advancing a 0.025 m diameter probe to a depth of up to 12 m. Seventy-five locations were sampled outdoors and seven locations were sampled inside the Process Building. Maps of the site showing the probe sampling locations are illustrated in Figs. 1 and 2.

Analytical data from these samples were used to further characterize the vertical distribution and areal extent of the Sr-90 groundwater plume. Preferential contaminant migration pathways were also identified during data analysis. The data helped to identify and characterize potential source areas of groundwater contamination. Geoprobe sample analyses were used to expand upon the existing data obtained from monitoring wells and soil borings.

GEOLOGY AND HYDROGEOLOGY

Sand and Gravel Unit

The saturated interval of the Sand and Gravel unit is the primary zone of groundwater on the North Plateau. The Sr-90 plume shown in Fig. 3 is located within the Sand and Gravel unit. The thickness of the Sand and Gravel unit ranges from 1 to 10.7 m.

This unit can be subdivided into two intervals - an upper sequence of muddy gravels and muddy sandy gravels and a lower sequence of thin-bedded clays, silts, sands, and very fine-grained gravels. The thin-bedded sequence is located in a channel-like depression on the surface of the underlying glacial till. Portions of the thin-bedded sequence that contain sands and fine grained gravels are believed to be more permeable than the upper sequence.

Depth to groundwater varies seasonally, but is typically deepest beneath the Process Building (4.6 - 5.5 m) and becomes shallower, until it surfaces in swampy areas and in the drainage ditch north of Lag Storage Areas 3 and 4 and the Construction and Demolition Debris Landfill, (CDDL). Groundwater flow in the Sand and Gravel unit is predominantly toward the northeast. Hydraulic conductivities in the unit have an approximate range of 1 x 10-6 to 1 x 10-5 m/sec.

Lavery Till

The Lavery Till is a low permeability glacial till that lies immediately beneath the Sand and Gravel unit on the North Plateau. Lavery Till on the North Plateau is unweathered and has an approximate hydraulic conductivity range of 1 x 10-10 to 1 x 10-9 m/sec.

SAMPLING PROCEDURES

Groundwater Sampling Procedures

A van equipped with a Model 8-ML Geoprobe system was used to advance the subsurface probe to selected depths at points located outside the Process Building. A modified Geoprobe sampling system was used to collect groundwater and subsurface samples from within the Process Building, due to accessibility limitations. An air powered,

27.3-kilogram jackhammer was used to drive the subsurface probe at points located inside the Process Building and other locations that could not be accessed by the Geoprobe van. The following procedures were used to advance the probe and collect groundwater samples:

Advance the probe (fitted with 0.01 m polyethylene tubing attached to a screen point sampler) down to a depth at least 1.5 m below the water table.

Pull up the probe rods 0.6 m to expose the screen point to the groundwater.

Attach 0.01 m tubing to a peristaltic pump and purge one probe volume or to dryness, whichever occurs first. Collect all purge water in 19-liter pails and then transport to a central staging area and pour into 208-liter drums. After purging, pump the water directly into the proper sample bottles.

Remove the probe rod and tubing from the test hole. Advance clean probe rod, screen, and sample tubing 1.5 m below the first sampling interval. Repeat the procedure until the entire groundwater zone is sampled.

This procedure allowed groundwater samples to be collected from discrete elevations within the Sand and Gravel unit.

Sample preservatives were added to the bottles by the sampling team prior to sample collection. The containerized and appropriately labelled samples were radiologically surveyed with a beta/gamma frisker and placed in a controlled area prior to shipping.

All groundwater samples were analyzed for gross alpha, gross beta, tritium, and Sr-90. Selected samples were also analyzed for alpha and beta radioisotopes. Unequilibrated gross beta analysis (within 24 hours of sampling) was used as a sample screening methodology. Unequilibrated gross beta results helped in making field decisions about additional or modified sampling locations. Equilibrated gross beta activities were available 10 days after sampling.

Subsurface Soil Sampling Procedures

Subsurface soils were collected with the Geoprobe in the following manner:

Mount a 0.30 m, standard diameter soil sampler to the lead Geoprobe rod and lower to the top of the sampling interval.

Detach the probe point from the soil sampler and drive the sampler 0.30 m to obtain a soil sample.

Retract the probe rods from the hole and remove the sampler.

Extrude soil sample from the sampler and place on clean herculite. Survey soil sample for radioactivity.

Place soil into labelled sample bottles for radiological analysis.

All excess soil was placed in bags and properly disposed.

Approval for hole closure was given after all required samples were collected, logged, and placed in the designated storage area. Bentonite (1/4") pellets were poured down the test hole to the ground surface. For those holes drilled in the Process Building, bentonite was poured to the base of the concrete floor then backfilled with epoxy grout to the floor surface.

DECONTAMINATION PROCEDURES

Prior to commencement of the subsurface probing activities, between sampling locations, and prior to removal from the facility the probing equipment (i.e., probe rods) was thoroughly cleaned to remove oil, grease, mud, and radiologically contaminated material. All equipment decontamination took place in a designated area.

The procedures used for equipment decontamination were as follows:

Wash and brush equipment with nonphosphate detergent to remove contamination

Rinse with clean tap water

Lay the equipment aside on plastic sheeting to dry

Radiation Technician wipes the equipment and screens wipes for radiological contamination

SR-90 GROUNDWATER PLUME AND PREFERENTIAL ROUTES OF MIGRATION

Sr-90 Groundwater Plume

Groundwater analytical data from Geoprobe sampling helped to define the Sr-90 plume on the North Plateau. Due to the heterogeneity of the Sand and Gravel unit, the areal and vertical activity of Sr-90 in the plume does not follow a typical contaminant distribution pattern. The following factors have had a significant influence on Sr-90 levels in the plume:

Preferential Sr-90 migration within the thin-bedded lower interval of the Sand and Gravel unit. Preferential migration is believed to be limited to the areas of the

thin-bedded interval that contain sand and very fine-gravel layers. Sr-90 activity in this portion of the thin-bedded interval was generally higher than the overlying Sand and Gravel sediments.

A narrowing of the Sr-90 plume east of LSA-4. The plume migration appears to be "funnelled" into this area and does not follow the northeast direction of groundwater flow. Monitoring well data alone did not clearly identify this narrowing of the Sr-90 plume.

Geoprobe sampling results were instrumental in identifying these features, when evaluated in conjunction with previously collected geologic and groundwater quality data.

Source Identification

Another goal of the Geoprobe sampling program was to identify and characterize sources of radiological contamination in groundwater. The Process Building and Fuel Receiving and Storage (FRS) area have long been suspected as the primary sources for groundwater contamination. However, prior to the Geoprobe sampling program, it was not possible to differentiate source contribution from these two areas. Groundwater and subsurface soil samples collected within and around the Process Building and FRS enabled WVDP to evaluate the relative contribution from these two potential sources. There were two major obstacles to collecting samples in the FRS and Process Building:

All sampling equipment had to pass through standard doorways. The van-mounted Geoprobe could not be used.

The generation of contaminated soil and water had to be kept to a minimum, to reduce the potential for introducing radioactive contamination into clean work areas.

In the Process Building, a 27.3-kilogram jackhammer was used to advance the Geoprobe sampling tools. The jackhammer was suspended from scaffolding and manned by an operator and a helper. Two 13.6-kilogram "Load Balancers" were used to lift the jackhammer to the top of the sampling rods. The "Load Balancers" were disconnected during probe advancement. Prior to Geoprobe sampling, a 0.076 m core drill was used to penetrate the 0.15 to 0.46 m thick concrete floor of the Process Building. Samples collected beneath the Process Building indicated that the primary source for the Sr-90 groundwater plume was located beneath the Process Building.

COMPARISON OF GEOPROBE AND MONITORING WELL GROUNDWATER SAMPLES

Table I presents a comparison of gross beta activities in groundwater at Geoprobe locations and nearby monitoring wells. Gross beta activity at the WVDP site primarily represents the sum of activities from Sr-90 and its daughter Yttrium-90 (Y-90). In spite of the numerous factors that could produce differences in analytical results, the Geoprobe data appear to be a reasonable approximation of groundwater gross beta activities in on-site monitoring wells. Parameters that could affect monitoring well and Geoprobe sampling results include, but are not limited to: differences in sampling methodology, differences in the thickness of the sampled interval, spacial variations in gross beta concentrations, time variation, geologic heterogeneity, and general sampling variability.

DEPTH DISCRETE NATURE OF GEOPROBE GROUNDWATER SAMPLES

Geoprobe groundwater samples collected at multiple depths appeared to be representative of the sampling interval. For example, several locations had relatively high gross beta activities at one sampling depth and relatively low activities at the next lower sampling depth. This trend suggested that little or no cross contamination occurred between the two sampling intervals. Safeguards to minimize the potential for cross contamination between sampling intervals include:

An "O" ring seal on the Geoprobe drive point to prevent water from entering the sampler until the desired sampling depth has been reached.

The procedure used for groundwater sampling helped to prevent cross contamination. The probe was advanced 1.5 m below the previous sampling interval, pulled back 0.6 m, then exposed to the groundwater by pushing out the screen. There was a 0.9 m thickness of soil that separated the upper portion of the sampling interval with the previous sampling interval.

COST AND TIME ADVANTAGES USING THE GEOPROBE TECHNOLOGY

Utilization of the Geoprobe system for groundwater sampling and plume/source characterization at WVDP produced a significant cost and time savings over traditional groundwater and sampling methodologies. Table II presents a cost and time comparison of a hypothetical Geoprobe sampling location and the equivalent

monitoring well installation/sampling program. Unit costs and estimated times are based on previous work performed at WVDP. Groundwater sampling intervals for the Geoprobe scenario were assumed to be 5.5 m - 6.1 m; 7 m - 7.6 m; 8.5 m - 9.1 m; and 10 m - 10.6 m. Screened intervals in the four adjacent monitoring well scenario were assumed to be 4.6 m - 6.1 m; 6.1 m - 7.6 m; 7.6 m - 9.1 m; and 9.1 m - 10.6 m. Due to the scope of the Geoprobe sampling program at WVDP, a detailed comparison of subsurface soil sampling costs for drilling and Geoprobe sampling will not be made. A small number of soil samples were collected using the Geoprobe. The relative cost of each Geoprobe soil sample (during the WVDP program) was higher than typical costs for split-spoon samples during drilling.

WASTE MINIMIZATION ADVANTAGES USING THE GEOPROBE TECHNOLOGY

Geoprobe sampling technology significantly reduced the volume of water and soil that would typically be generated during the installation and sampling of monitoring wells. Table III presents a comparison of the waste soil and water generated by each of these methodologies.

SUMMARY

We have found the Geoprobe sampling system to be an advantageous alternative to conventional groundwater and subsurface soil sampling methodologies. The Geoprobe system gave us the opportunity to investigate the groundwater and soil under the Process Building, something that conventional drilling would not have provided. Waste generated was greatly minimized since no cuttings were produced and samples were obtained through smaller boreholes only 0.025 m in diameter. The Geoprobe system was also found to be a very useful and efficient method for defining shallow groundwater plumes, particularly in heterogenous aquifers. Additionally, the Geoprobe sampling system reduced the time and cost of the groundwater and soil sampling program.

We are continuing to explore future uses for the Geoprobe sampling system at our facility.

27-28

COAL SORBENT SYSTEM FOR RADIONUCLIDES, HEAVY METALS, AND ORGANIC COMPOUNDS EXTRACTION AND DISPOSAL

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ABSTRACT

Processes involved in current pump and treat systems have led to high treatment costs and the need for complex sludge stabilization chemistries, because they are essentially unit operations and not developed as a unified system for removal and disposal. TRW has developed a system for the removal and disposal of radionuclides, heavy metals, and organics utilizing a coal-based adsorbent, produced by the chemical leaching of ordinary coal using the DOE-sponsored Molten Caustic Leaching (MCL) process. The coal sorbent produced from the MCL process is much like activated carbon with a large internal surface area up to 1000 m2/gram. In addition to its high surface area, the coal sorbent, unlike activated carbon, has inherent carboxyl groups much like ion exchange resins that can remove heavy metals from wastewater. Uranium capacities of 30 mg/g (3%) of coal sorbent have been demonstrated. The coal sorbent could be used as a wastewater treating media to remove both organics due to high surface area and heavy metals due to the carboxyl groups present. Once the heavy metals and organics are adsorbed on the virtually sulfur and ash-free coal sorbent, the spent coal sorbent can be added as a low volume feedstock to existing DOE-developed or commercially available combustion/vitrification processes to encapsulate the metals and destroy the organics. Waste volume reductions as high as 420,000:1 have been demonstrated for uranium from simulated Fernald groundwater at an estimated cost through disposal of $0.001/gallon water treated.

INTRODUCTION

Groundwater under hazardous waste sites and sites used for the processing of nuclear fuels contain heavy metals and toxic organic compounds. Of major concern with groundwater contamination is the potential for contaminating aquifers that supply drinking water or feed waterways. Permanent cleanup of contaminated groundwater requires that the hazardous materials are removed and the resulting waste products are stabilized or detoxified.

The coal sorbent system (CSS) deals with complete end-to-end clean-up, unlike other

processes that deal with simply the removal or disposal steps. For example, ion exchange, due to high resin cost, deals only with the removal of the metal from the waste stream, but does not produce a product that is readily disposed of without additional processing of a liquid waste. In contrast, the CSS process is designed to address the total problem of removing contaminants from aqueous wastes and stabilizing them in an environmentally safe manner.

The CSS process operates as follows: 1) coal is converted to the unique sorbent medium by treatment with caustic at high temperatures utilizing TRW's Molten Caustic Leaching (MCL) Process technology in our existing U.S. Department of Energy test circuit at San Juan Capistrano, California, 2) the product coal sorbent, which is the unique sorbent, is utilized in a portable, fixed-bed processing unit for treatment of contaminated water, reducing heavy metals and radioactive contaminants to parts per billion levels, 3) the loaded sorbent is as combustible as ordinary coal and when burned in a DOE or commercial vitrifier/combustor along with a glass, such as borosilicate, can produce an encapsulated material suitable for storage. Overall waste volume reduction ratios are estimated at up to 420,000:1 while reducing, for example, uranium levels in treated water to below 5 g/L. The basic operations of the CSS process have been reduced to practice under TRW IR&D funds and a patent has been issued ("Improved Metal Ion and Organic Contaminant Disposal," Patent No. 5,169,534 December 8, 1992).

CHARACTERISTICS OF THE TRW COAL SORBENT

TRW's coal sorbent has carboxyl groups providing ion exchange sites for heavy metal adsorption and high surface area for the adsorption of organics. Single point nominal surface areas are in the range of 600 to 1,000 m2/g. TRW has conducted a significant number of laboratory tests of the proposed application using cadmium, lead, nickel and uranium as the test metals and methylene blue as the test organic compound. These removal efficiency tests were initially conducted as batch tests with the coal sorbent in excess. Both the coal sorbent (by titration) and the test solution were adjusted to the test pH (generally between 5 and 7) before they were allowed to contact. Test solutions containing approximately 7,000 mg/L $NaNO_3$ as the background were filtered and analyzed after pH adjustment to ensure that no precipitates were formed prior to contact with the coal sorbent. The filtered sample at the analyzed concentration was then contacted with a fixed amount of coal sorbent in a test tube. Table I shows that the coal sorbent is an effective absorber for solutions of nickel, uranium and cadmium ions. Target uranium discharge concentration for the DOE Fernald, Ohio site is <5 g/L.

Figures 1 and Fig. 2 are the isotherms for nickel and uranium created using a flow-through system consisting of a peristaltic pump, a 0.45 mm filter, and a glass column with a frit. The test amount of coal sorbent (typically 0.1g) is loaded as slurry into the column and is allowed to compact under a deionized water flow. Test solutions and coal were prepared and pH adjusted. The test solution containing the nickel and uranium was pumped through the filter upstream of the coal sorbent bed. A test sample was taken just after the filter to determine the exact concentration passing through the coal bed. In this manner the only removal mechanism would be via ion exchange since any precipitate would have been removed prior to reaching the absorber. Based on these experiments the maximum capacity is 31.3 mg/g (3.1%) in the absence of background divalent ions. In an experiment to determine the capacity of the coal sorbent for uranium in the presence of calcium and magnesium, a solution of calcium, magnesium and uranium at 100,000, 1,000 and 1,500 g/L, respectively, was made. When this solution was pumped through the coal sorbent bed the uranium capacity was determined to be 11.1 mg/g (1.1%). The uranium concentration in the effluent was below the detection limit (0.03 ug/L) of the ICP-MS used to analyze the test solutions.

While TRW's coal sorbent is an excellent ion exchange material, the MCL process also modifies the coal sorbent to produce a material that has many of the properties of an activated carbon. An industry standard test using methylene blue was used to compare the adsorption capacities of TRW's coal sorbent and standard activated carbon. Samples of activated carbon and TRW's coal sorbent were placed in separate test tubes with water. After agitation to thoroughly wet the solids, 0.25 mL portions of a solution of methylene blue (0.15%) were incrementally added to each tube. The tubes were agitated and the disappearance of the blue color was noted after each incremental addition. Approximately 15 minutes elapsed between additions of methylene blue. Under these conditions both the activated carbon and TRW's coal

sorbent adsorbed 93 mg of methylene blue per gram of solid before the color no longer disappeared. In a separate experiment the raw coal did not adsorb any methylene blue. Other tests with coal sorbent derived from brown coal showed higher capacities and faster adsorption rates than the activated carbon.

A direct comparison of TRW's coal sorbent (derived from both brown and black coals) with commercial ion exchange resin and activated carbon is presented in Table II. The data in this table is based on recent tests. TRW's coal sorbent compares favorably with commercial products at a much lower cost.

In summary, these tests with TRW's coal sorbent have demonstrated its effectiveness in reducing uranium levels in laboratory and simulated groundwater matrices with a capacity equivalent to current ion exchange resins used for uranium recovery. Even in the presence of calcium and magnesium at a 100-fold excess, metal ion levels were reduced below target concentrations. An added advantage for water cleanup applications is the affinity of TRW's coal sorbent for organic materials. Ion exchange resins, such as Dowex 21K, have a similar capacity for uranium but typically cost the equivalent of $16,000/ton and are not effective in the removal of organics. Activated carbon products cost from $500 to $2000/ton and, although effective for organic removal, are not effective for heavy metal removal.

COAL SORBENT SYSTEM

TRW's coal sorbent system is a process for removal of heavy metals (including uranium) and organic compounds from wastewaters and groundwater which also destroys the organic compounds and encapsulates the metals for disposal while attaining 4 to 5 orders of magnitude volume reduction. The process employs coal sorbent derived from the MCL process to remove contaminates present in groundwater and then provides a method to combust/encapsulate the coal sorbent with slagging agents to encapsulate the metals and destroy the organics. Contaminated water is contacted with MCL-processed coal sorbent in a continuous fixed bed contactor. The contaminated water is pH adjusted and pumped through the fixed bed unit until the bed reaches saturation of contaminants and breakthrough occurs. At this time the unit is taken off-line and the water flow switched to the readied second fixed bed unit. The cleaned water is filtered for removal of coal sorbent fines and is sent to disposal as nonhazardous waste or pumped back into the aquifer down gradient from the contamination site or injected up-gradient to add additional volume to flush the contaminants from the saturated zone.

To this point the coal sorbent system is similar to most ion exchange processes. However, because the metals and organics are trapped on a low cost, combustible material, a unique and safe disposal process is possible. The coal sorbent is a high energy, clean fuel (no SOx controls required) which can be combusted in a number of way. Because of the low cost of the coal sorbent, the process can be designed as a "once-through" system where the loaded coal sorbent goes directly to a vitrification or combustion plant designed to destroy the organics and encapsulate the metals in an environmentally safe and stable glass. In a proof-of-concept test, a portion of the coal sorbent loaded with uranium was mixed with 10% by weight of ground soft glass and fired in a crucible at 1000oC. Within the limits of the analysis, 100% of the uranium was found in the resulting glass after the coal sorbent had burned off indicating the potential for trapping non-volatile metals in a glassy matrix.

Examples of available vitrification/combustion technologies include:

Combustion in a rotary kiln - the reduced gas velocities will reduce the generation of fine particles and reduce particulate control/costs. A commercial system for non-radioactive waste such as the Waste Technologies System is an example of this type of disposal.

Vitrification - the spent coal sorbent is a low volume feed stream. Conditions and controls in the vitrification process would burn off the carbon matrix and capture the metals in the glass matrix. For radioactive waste, this option can be easily implemented at many of the DOE sites which will have a vitrification process.

Specialized vitrification technologies - for example, Vortec Corporation under DOE sponsorship has developed a fossil fuel fired vitrification process.

The vitrification/combustion step enables the coal sorbent system to exhibit exceptional capability to concentrate metals in a safe disposable matrix. For example, water containing 1,500 g/L uranium can be cleaned to levels below 5 g/L with the coal sorbent retaining up to 30 mg/g uranium. Under these conditions one ton of coal sorbent can be expected to treat from 34,000 cubic meters of contaminated water. This represents a waste volume reduction of 42,000:1. The waste

volume can be further reduced by the vitrification/combustion of the coal under controlled conditions. Glass mineral additives would be used during combustion as a means of vitrifying and encapsulating the waste materials in the slag. Organic materials would be destroyed by the high combustion temperatures. Assuming that adsorbed contaminants, glass mineral additives, and remaining coal mineral matter comprise 10 percent of the coal sorbent, there is a further waste volume reduction of 10:1 for the vitrification/combustion step. This represents overall waste volume reduction of up to approximately 420,000:1 for the complete process. Waste minimization is particularly important for radioactive waste because of the high cost of disposal based on volume and also because of severe limitations on storage area.

COAL SORBENT SYSTEM PROCESS ECONOMICS

Cost studies indicate that TRW coal sorbent can be produced for $330 per ton in a 2 ton per hour MCL plant and at even lower costs for larger sized MCL plants. In comparison, activated carbons for organic removal typically cost on the order of $2000/ton depending on quality and intended application. Unlike the TRW coal sorbent, activated carbon derived from coal contains sulfur which precludes burning without sulfur oxide emissions controls. Activated carbon derived from coconut shells is low in sulfur but is difficult to burn. Although efficient for organics removal, activated carbon is far less effective than TRW coal sorbent for metals (such as uranium) removal from water.

Ion exchange resins used for the removal of heavy metals and uranium from water, such as Dowex 21K, typically cost about $16,000 per ton and a single resin is not effective for the removal of both metals and organics. The high cost of ion exchange resins prohibits once-through use, therefore, ion exchange resins are periodically regenerated in batch operations to elute the contaminants into another aqueous stream which requires additional processing for recovery and disposal of wastes.

For the specific application of uranium removal from wastewater, a process (1) has been proposed based on the co-precipitation of uranyl hydroxide $[(UO2)3(OH)5]+$ with potassium ferrate. The process uses a chemical treatment step to remove most dissolved and suspended solids, radionuclides and priority pollutant metals from aqueous waste streams. Potassium ferrate and magnesium salts are then used to enhance contaminant removal. Magnesium acts as a seed crystal to ensure the formation of colloidal particles and eventual floc formation. Tests to date using the potassium ferrate treatment process allowed 0.5 hour for flocculation followed by 1 hour for the resulting floc to settle. Before the addition of potassium ferrate, the pH of the waste stream was adjusted to the range of 10.6 to 11.2 with sodium hydroxide. From 72% to 86% of the uranium in the wastewater was removed in a once through system. When the waste stream was treated a second time, more than 98% of the uranium was removed.

A comparison of the CSS to the potassium ferrate process was made on a the basis of volume generated (Table III) and the treatment costs (Table IV) for 6.2 million gallons/yr of Fernald groundwater uranium at a concentration of 490 mg/L. In this application of the CSS, uranium contaminated groundwater will be pumped to an agitated tank where it is mixed with a dilute hydrochloric acid solution. The purpose of this step is to remove the carbonate/ bicarbonate from the Fernald groundwater as carbon dioxide. This is essential because in the presence of carbonate uranium forms an anionic complex which would not be adsorbed by the carboxylic acid ion exchange groups on the TRW coal sorbent. The carbonate-free groundwater will then be pumped to a second agitated tank to adjust the pH to ~6. The water then will be pumped at a controlled feed rate to the CSS contactor where uranium and other heavy metals will be removed from the water by adsorption onto TRW coal sorbent. The coal sorbent requirement is calculated assuming a minimal capacity of 0.5% uranium. Laboratory studies have actually shown an uranium ion capacity of 1.1% in the simulated Fernald groundwater (Fig. 2). The encapsulated product quantity includes the 10 weight percent ground borosilicate glass added to the spent coal prior to combustion or vitrification. In calculating the disposal costs for the CSS process, the waste monitoring and disposal charges are assumed to be the same as for the potassium ferrate process on a volume basis. The cost of the combustion/ vitrification operation is based on information supplied by Vortec Corporation which is developing one of the waste encapsulation technologies for spent coal that will be evaluated in this proposal.

Based on chemical costs alone, treating the wastewater with potassium ferrate costs

$0.036/gal. However, when the costs for monitoring, analyzing, treating, transporting, and disposing of the resulting solids are included, the total cost of the potassium ferrate treatment process is $0.08/gal. The total cost of treating the same quantity of wastewater using the CSS process, however, is $0.001/gal. The main cost savings are a direct consequence of the reduced volume of waste generated in the CSS process: 23:1 for wet sludge:spent coal and 244:1 for dry sludge:encapsulated product when compared to the ferrate process.

The CSS process is also less expensive than ion exchange systems requiring regeneration and processing of the regeneration liquor. An estimated operating cost for an ion exchange system for removing uranium from drinking water in Colorado was given at $0.006/gal (2). However, this cost only includes the cost of trucking the regeneration brine containing the uranium to a waste water treatment plant where it is dumped into the process treatment stream. No provision or cost is included for stabilization and disposal.

OTHER APPLICATIONS FOR THE COAL SORBENT

In a broad sense, the CSS process has the unique ability to simultaneously remove both organic compounds and heavy metals from aqueous streams. This is because the TRW coal sorbent has both the high surface areas found in activated carbon and the ionic surface functionalities of ion exchange resins. Because of this dual capability, capital and operating costs of a water treatment facility would be reduced as only a single absorber system would be necessary to remove both types of contaminants. Because TRW coal sorbent is inexpensive and thus can be used on a once-through basis, it may have applications in other areas (e.g., wastewater, process water, acid mine drainage, etc.) where activated carbon or ion exchange would be considered. For example:

The ability to slurry the coal and its affinity for water would enable the TRW coal sorbent to be used as a semipermeable barrier that would retain the metals and organics yet allow the flow of water (uranium mine tailing pond control).

A hybrid system using specialized ion exchange resins coupled with the CSS as the regeneration loop to concentrate and dispose of the metals without the generation of sludge (soil washing systems).

Where favorable economics exist, non-volatile metals can be recovered by roasting the coal to remove the coal matrix and leaving the oxide of the metal. For selected metals the coal sorbent could act as a metallurgical coal and reduce the metal oxides to metal. Because of the low ash and sulfur in the coal sorbent, this can be done without additional pollution controls.

Future applications in acid mine waters will be enabled by a new variant of the coal sorbent having strong acid functional groups added to the sorbent surface. The net effect is a 3 fold or more increase in capacity with a laboratory demonstrated use range down to a pH of 2.8.

TRW has demonstrated mercury vapor capture from a gas stream at flue gas temperatures (250 and 400oF). This opens the use of the coal sorbent as a low cost air emission control material.

SUMMARY

TRW's CSS process is unique as it provides for the simultaneous removal of heavy metals, radionuclides and organics in pumped water and washed-soil leachate, with the additional benefit of providing an effective waste concentration and stabilization method for metals together with thermal destruction of hazardous organic chemicals. Waste minimization is attained by the coal sorbent system in three areas:

The coal sorbent system adsorbs both metals (radionuclides) and organics on a low cost, easy to handle material.

There is no generation of primary wastes such as alkali sludges which are hard to handle and dewater or secondary acid or brine streams derived from the regeneration and elution of ion exchange columns and which require neutralization and precipitation.

Due to the low cost and capacity, the coal sorbent is used as a once-through sorbent.

The coal sorbent has virtually no ash or sulfur and can be destroyed by current combustion/vitrification technologies without any additional environmental emission controls.

Trace metals and radionuclides are encapsulated in the vitrified glass and the coal sorbent while any absorbed organics are destroyed.

Waste volume reductions as high as 420,000:1 have been demonstrated for uranium from simulated Fernald groundwater at an estimated cost through disposal of $0.001/gallon water treated.

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27-29

REMEDIATION OPTIONS FOR A CHROMIUM CONTAMINATED LANDFILL USING CEMENTITIOUS GROUTS

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ABSTRACT

In-situ remediation of a chemical waste landfill with excessive chromium levels is being investigated as part of the Mixed Waste Landfill Integrated Demonstration. This paper is concerned with the design of advanced cementitious grouts for in-situ stabilization of chromium contaminated soil and in-situ installation of subsurface containment barriers. Grouts have been developed to improve the performance and cost effectiveness of remediation compared with conventional materials. In addition to restoration of chromium contaminated soils, the developed grouts have applications in other environmental operations where superior properties are required.

INTRODUCTION

As part of the Mixed Waste Landfill Integrated Demonstration (MWLID), the authors are developing advanced grouting materials for remediation of a chromium plume resultant from disposal of hexavalent chromium solutions. Further details of the MWLID have been given by Burford et al. (1). The site has an arid climate, a groundwater depth of approximately 160 to 170 m and alluvial type soil. The chromium is present in both the trivalent and hexavalent oxidation states, indicating that some transformation of redox state has occurred since the solutions were disposed. The remediation aims to convert hexavalent chromium into the less mobile and less toxic trivalent state and to contain the contaminated soil through use of subsurface barriers and a surface cover. The Brookhaven research is focused on 1) in-situ solidification/stabilization of the chromium contaminated soil and 2) in-situ subsurface containment barriers.

STABILIZATION OF CHROMIUM CONTAMINATED SOIL

Remediation of the chromium plume requires stabilization of both the trivalent and hexavalent states. Hexavalent chromium requires reduction to Cr(III) before stabilization as Cr(OH)3 (2). For example, Cr(VI) contaminated soil can be pretreated with a ferrous sulphate solution, followed by cement or lime stabilization. However, a concern with this approach for in-situ remediation is that reduction of Cr(VI) may be incomplete and that verification of satisfactory reduction is necessary before proceeding with further treatment. Furthermore, the economics of remediation are detrimentally affected by the need to perform two separate stages.

Research has been directed towards simultaneous reduction of Cr(VI) to Cr(III) and stabilization so that soils contaminated with Cr(VI) or a combination of Cr(VI) and Cr(III) can be remediated in a one step in-situ process. The approach is to use grouts in which ordinary Portland cement is partially replaced with ground granulated blast furnace slag. Previous work has demonstrated reduction and stabilization of Cr(VI) by slag-modified cements (3-5). The low redox potential of slag-modified cements is responsible for the reduction of Cr(VI) (6,7).

The in-situ techniques under consideration for treating the contaminated soil with grout are jet grouting (8-11) and deep soil mixing (2,8,12,13). In these techniques grout is intimately mixed with soil to form soil cement. The choice of technique will depend on ability to meet stabilization objectives and regulatory acceptance, in addition to cost effectiveness for the scope of soil treatment.

Initial research investigated the effects of slag content, water/cementitious material ratio and soil/cementitious material ratio on the leachability, permeability, wet-dry durability and compressive strength of treated soils.

Following successful stabilization of soils spiked with 200 ppm Cr(VI) and Cr(III) and of soil samples retrieved from the landfill under study, further leachability tests were conducted at a higher level of Cr. Additional details of the initial work on stabilization of Cr contaminated soils with slag-modified grouts are available (14,15).

Initial Leach Tests

The grouts consisted of Type I cement, ground granulated blast furnace slag, water, bentonite and superplasticizer. The slag used was ASTM C 989 Grade 100, supplied by Koch Minerals. Sodium montmorillonite type bentonite was added at a rate of 2% by mass of water to improve grout stability. Sodium naphthalene sulphonate formaldehyde superplasticizer with 42% solids by weight was used to enable reduction of the water/cementitious material ratio (w/c) while maintaining fluidity. Superplasticizer was added at a rate of 20 ml/kg cementitious material. Uncontaminated soil collected adjacent to the landfill of interest was spiked with known quantities of Cr and mixed with the grouts.

In the initial studies uncontaminated soil was spiked with either 200 ppm Cr(VI) or 200 ppm Cr(III) and then mixed with grout. The grout was added to soil at soil/cementitious material ratios (s/c) by weight of 1, 2 and 5. The corresponding water/cementitious material ratios were 0.48, 0.72 and 1.40. The concentrations of Cr in the specimens after dilution with grout were 80, 106 and 135 ppm for soil/cementitious material ratios of 1, 2 and 5, respectively. Slag was used at cement replacement levels of 0, 20, 40, 60 and 80% in the grouts. The treated soils were cured by burying in dry soil for 28 days. The objective of the curing was to simulate subsurface conditions under which the extent of hydration is expected to be reduced as compared with conditions where water is readily available. The cured specimens were crushed and subjected to EPA Toxicity Characteristic Leaching Procedure (TCLP) tests. The concentration of leached Cr from stabilized Cr(III) contaminated soil was less than 0.5 ppm for the range of grout slag contents and soil/cementitious material ratios. This compared with 6.3 ppm for the unstabilized soil and the EPA limit of 5.0 ppm. The amount of leached Cr(III) decreased with increasing slag replacement level.

The tests on soils spiked with 200 ppm Cr(VI) showed that the proportion of Cr(VI) to total Cr [Cr(III) plus Cr(VI)] was reduced as the slag replacement level increased. This indicates that reduction of Cr(VI) occurred. The concentrations of total Cr and Cr(VI) in the leachates increased as soil/cementitious material ratio increased. However, at slag replacement levels equal to or greater than 40% the concentrations remained below 0.5 ppm. The untreated soil spiked with 200 ppm Cr(VI) had leachate concentrations of 8.75 ppm Cr(VI) and 9.5 ppm total Cr. Hence, the slag-modified grouts were demonstrated to be potential stabilization agents on the spiked soils.

A significant finding from the TCLP tests on Cr(VI) contaminated soils stabilized with slag-modified grout was that 40% replacement of cement with slag was sufficient to cause reduction of Cr(VI). Studies by Angus and Glasser (7) on slag-modified cements showed that the redox potential was not significantly lowered until slag content reached around 85%. Curing conditions differed from those used in this work, but the curing period was approximately the same. Atkins and Glasser (16) comment that the level of slag replacement necessary for reducing behavior depends on time since more S^{2-} is released as slag continues to hydrate. Thus, the redox conditions in the soils treated with slag-modified grout can be expected to change with time and with the availability of water for hydration.

Leach Resistance with Time

A question arises as to whether the observed immobilization of Cr(VI) with slag-modified grouts is permanent. Oxidation of S^{2-} may occur if oxygen diffuses into the treated soil. Consequently, the reducing characteristics may be diminished. Diffusion of oxygen into the stabilized landfill soil is not expected to be a major concern under subsurface conditions, but requires consideration. The potential deterioration of the reductive capacity of grout treated soil was investigated to determine whether this could result in increased leachability of Cr. Soils spiked with 200 ppm Cr(VI) and treated with grouts containing 40% and 80% slag were crushed as required for the TCLP test and left in laboratory air for periods of 7 and 14 months. The specimens were from the same batches used in the initial TCLP tests and, therefore, should not vary in content or properties except for inherent heterogeneity. At the conclusion of the exposure time the specimens were subjected

to TCLP tests and the results were compared with those collected at an age of 28 days.
The soils treated with the 40% slag grout showed an increase in leached Cr(VI) and total Cr at both 7 and 14 months when the soil/cementitious material ratio equalled 5. At 14 months the material with 40% slag and a soil/cementitious material ratio of 2 had increased leachate concentrations compared to the initial and 7 month values. In contrast, the soil treated by grout with 80% slag did not demonstrate significant increased leachability at any soil content for either of the test periods. The limit of 5.0 ppm leached Cr was not exceeded for either grout and the maximum value for total Cr was 1.0 ppm. The results suggest increased potential for Cr release for soils treated with lower slag content grouts exposed to oxidizing conditions. Higher slag contents appear preferable in this respect and will continue to be monitored.
Leach Tests on 1000 ppm Cr Soil
The initial successful stabilization of spiked and landfill soils with the slag-modified grouts warranted further investigation with higher Cr contamination levels. Two grout mixes were used: 1) 100% cement, 2) 40% slag/60% cement. It is also intended to test grout with 80% slag on higher Cr concentrations for comparison, with emphasis on the long-term immobilization performance. The test specimens had a water/cementitious material ratio of 0.72 and a soil/cementitious material ratio of 2. The grouts were used to stabilize soils spiked with 1000 ppm Cr(III) and with 1000 ppm Cr(VI). Dilution by grout resulted in the Cr concentration of the stabilized soils being 532 ppm in each case.
The results for the leach tests on soil spiked with 1000 ppm Cr are presented in Figs. 1 and 2. The leachate concentrations for the Cr(III) soil illustrated in Fig. 1 were well below the limit of 5.0 ppm and showed enhancement of leach resistance due to addition of slag to the grout. Figure 2 shows that slag decreased the amount of Cr(VI) and total Cr that was leached. At 1000 ppm Cr(VI) in the original soil the grout without slag was unable to comply with EPA leachate requirements. Tests on stabilizing 1000 ppm Cr(VI) soil with higher slag content grouts will determine whether reduction of Cr(VI) to Cr(III) and leach resistance can be improved further.
Fig. 1. TCLP Results for 1000 ppm Cr(III) Soil.
Fig. 2. TCLP Results for 1000 ppm Cr(VI) Soil.
Future Work on Cr Stabilization
The laboratory prepared stabilized soils will be continually monitored for leachability to test that Cr stabilization remains acceptable. The practicality of using slag-modified grouts for in-situ stabilization of the chromium plume at the landfill will be investigated. The planned field trials will involve optimization of grout mix proportions and placement equipment parameters for the site conditions. The properties of the stabilized soil will be tested for leach resistance, durability and physical and mechanical properties. The safety, efficiency and economics of the grout stabilization process will be compared with other in-situ methods.
SUBSURFACE CONTAINMENT BARRIERS
Vertical and horizontal subsurface barriers in uncontaminated soil are required to reduce migration of any remaining leachates and to prevent penetration from external sources. Together with a surface cover, the subsurface barriers will effectively contain the landfill contents. Cementitious grouts containing admixtures have been evaluated for this purpose with the objectives of improving hydraulic and mechanical properties over conventional, high water/cement ratio grouts and producing a more cost effective and reliable solution than polymer or chemical grouts.
Jet grouting and deep soil mixing are under consideration for in-situ vertical barrier placement at the site of interest. Jet grouting can also be used to form horizontal barriers (10,17,18). Permeation grouting is not considered to have a high probability of producing continuous barriers in a controlled manner at the site with cementitious grouts due to the fine particle size and heterogeneity of the soil. Uncertainty of forming continuous barriers at the site is also a concern for claquage, or hydrofracture, grouting.
Two versions of jet grouting are of interest for the barrier construction. The first method is in-situ mixing of grout with soil to form a soil cement barrier. Replacement jet grouting is the second method and involves removal of soil by a jet of water and air, followed by placement of grout in the formed cavity. Of the two, the first method is more readily available and less expensive.
Containment barriers are required to be free from major defects that could give rise

to leakage. Examples of such defects include cracks and discontinuities between successive grout or soil cement columns. The potential for cracking can be minimized to some degree through mix design, such as using minimum water/cementitious material ratio, and through materials selection to reduce shrinkage, increase fracture resistance and decrease thermal effects.

Another concern is that leakage may develop at column joints if the bond is insufficient. Jet grouted or soil mixed columns are usually overlapped and keyed together to form an interlocked barrier. Thus, a mechanical bond is formed. The chemical bond between columns is also of interest since this will influence the hydraulic integrity of the joint, and hence, the overall performance of the subsurface barriers. Studies of the permeability of bonded grouts and soil cements were initiated to examine the interfacial properties. The permeabilities were compared with those of bulk materials.

Bond Permeability Tests

Soil cements and sanded grouts determined to be barrier contenders from earlier property studies are currently being prepared and tested for bond permeability. Soil cements have been prepared from plain and from slag-modified grouts. Sanded grouts containing supplementary cementing materials such as silica fume and slag are being investigated for comparison with soil cements. The results for one soil cement and one grout have been selected for presentation. Both materials contained slag. The same types of superplasticizer and bentonite as used in the Cr stabilization described above were used for the barrier materials. Uncontaminated site soil collected from a location adjacent to the landfill was used. The mix proportions are presented in Table I.

Table I. Mix Proportions of Selected Soil Cement and Sanded Grout used for Bond Studies

Permeability of bonded soil cements and sanded grouts was measured on cylindrical specimens cast in two halves. Wax coated cardboard cylinders 75 mm diameter and 150 mm long were fitted with a vertical insert which had a smooth polyethylene surface. The first half of each specimen was cast, covered with polyethylene sheet and allowed to sit for 24 hours. The insert was then removed and the second half of each specimen was cast. After 24 hours the cardboard mould was removed and the specimens were maintained in water until testing at 28 days age. A total of three specimens per mix were tested.

Figure 3 shows the permeabilities of the bonded materials in addition to the values for bulk materials at the same age and cured under the same conditions. Assuming that the permeabilities are normally distributed, t-tests were performed to determine whether the permeabilities of the bonded and bulk materials were significantly different. It was determined that the permeability of the bonded soil cement was significantly higher than the bulk material at a 5% level of significance. This can be attributed to shrinkage of the second half of the specimen, causing preferential flow to occur at the interface. The permeabilities of the bonded and bulk grouts were not significantly different. This suggests low shrinkage and sound bonding between successive grout batches. In each case, the permeability of the bonded material remained sufficiently low for containment purposes. It is necessary to repeat permeability measurements of column joints produced in the field with full scale in-situ placement equipment and subjected to subsurface shrinkage conditions to test that adequate performance can be achieved.

Fig. 3. Permeability of Bonded and Bulk Materials

The bonded specimens will be subjected to wet-dry cycles and re-measured for permeability to determine the susceptibility to preferential flow at the interface. The permeability under unsaturated conditions is of interest since the barriers will be placed in the vadose zone in this case. The microstructure of the bond will also be investigated to elucidate potential means of improving interfacial properties.

Proposed Field Trials for Barriers

It is proposed to conduct extensive field trials at the landfill site so that the grout proportions, grouting process and barrier performance can be optimized for both vertical and horizontal barriers. Grouting parameters for the unique site conditions require further systematic variation with corresponding evaluation of barrier quality and continuity. The minimization of water/cementitious material ratio while retaining compatibility with the placement equipment is of particular importance since this parameter has the greatest influence on final permeability.

CONCLUSIONS

Slag-modified cementitious grouts have been found to be potential stabilization agents for soil contaminated with up to 1000 ppm Cr(VI). Further tests are in progress to assess the permanence of stabilization. Grouts with enhanced hydraulic, physical and mechanical properties have also been developed and characterized for subsurface barriers. The permeability of joints between barrier materials has been measured and found to be suitably low. The program is now at the stage of demonstrating the slag-modified grouts for in-situ stabilization of chromium plumes and demonstrating superplasticized cementitious grouts for subsurface containment barriers using placement techniques such as jet grouting.

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27-30

IN SITU BIOREMEDIATION: COST EFFECTIVENESS OF A REMEDIATION TECHNOLOGY FIELD TESTED AT SAVANNAH RIVER

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ABSTRACT
In Situ Bioremediation (ISBR) is an innovative new remediation technology for the removal of chlorinated solvents from contaminated soils and groundwater. The principal contaminant at the SRID is the volatile organic compound (VOC), tricloroetylene(TCE). A 384 day test run at Savannah River, sponsored by the U.S. Department of Energy, Office of Technology Development (EM-50), furnished information about the performance and applications of ISBR.
In Situ Bioremediation, as tested, is based on two distinct processes occurring simultaneously; the physical process of in situ air stripping and the biological process of bioremediation. Both processes have the potential to remediate some amount of contamination. A quantity of VOCs, directly measured from the extracted air stream, was removed from the test area by the physical process of air stripping. The biological process is difficult to examine. However, the results of several tests performed at the SRID and independent numerical modeling determined that the biological process remediated an additional 40% above the physical process. Given this data, the cost effectiveness of this new technology can be evaluated.
INTRODUCTION
The purpose of this report is to study the cost effectiveness of In Situ Bioremediation (ISBR) with horizontal wells as tested at the Savannah River Integrated Demonstration (SRID) site in Aiken, South Carolina. ISBR is an innovative new remediation technology for the removal of chlorinated solvents from contaminated soils and groundwater. The principal contaminant at the SRID is the volatile organic compound (VOC), trichloroethylene (TCE). A 384 day test run at Savannah River, sponsored by the U.S. Department of Energy, Office of Technology Development (EM-50), furnished information about the performance and applications of ISBR.

The overall cost effectiveness of In Situ Bioremediation (ISBR) is based on the cost sensitivity of the biological component; as the biological addition increases, the cost per pound of VOCs remediated decreases.

The short-term cost of ISBR with a biological addition of 40% above the vacuum component is $21 per pound of VOCs remediated. The worse case scenario, ISBR + 0% addition costs $29/lb of VOCs remediated, and is based solely on the vacuum component.

The baseline pump and treat/soil vapor extraction system costs $31/lb in the short-term and has no possibility of a biological addition.

Life-cycle analysis shows that ISBR is more cost effective than the baseline pump and treat/soil vapor extraction system.

As demonstrated, ISBR has a possible savings of $1 million at the SRID site alone.
In Situ Bioremediation is based on two distinct processes occurring simultaneously: the physical process of in situ air stripping and the biological process of bioremediation (see Fig. 1). Both processes have the potential to remediate some amount of contamination. A quantity of VOCs, directly measured from the extracted air stream, was removed from the test area by the physical process of air stripping. The biological process is difficult to examine. However, the results of several tests performed at the SRID and independent numerical modeling determined that the biological process remediated an additional 40% above the physical process. Given this data, the cost effectiveness of this new technology can be evaluated. In addition to calculating the cost effectiveness on the ISBR demonstration at the SRID, sensitivity analysis is conducted in order to determine how the overall cost of ISBR changes in regards to the performance of the biological component. By comparing the overall cost of this system and the price per pound of VOCs remediated against a conventional pump and treat/soil vapor extraction system, we can evaluate the overall cost effectiveness of the alternative technologies.
SYSTEM CAVEATS
The ISBR demonstration at the SRID was set up to address a "hot spot" of an overall larger VOC contaminant plume. The pump and treat/soil vapor extraction system is engineer designed and presumed to perform optimally. Both pump and treat and soil vapor extraction systems have been tested at the SRID. The baseline system (a combination of pump and treat/soil vapor extraction apparatus) is integrated to avoid overlapping of equipment and materials, and is located in an area exactly like the ISBR demonstration in regards to all necessary site characteristics, including overall concentration of contaminants. By designing both the baseline and the innovative systems to handle equal flow and assuming equal vacuum extraction

performance, a level playing field for a cost comparison is created.

ANALYSIS

The data used in these analyses have a "field demonstration" level of confidence and are based on an actual field demonstration. The performance comparison consists of Plan 1, which is based on the new ISBR technology as demonstrated at the SRID, and Plan 2, which is based on "equivalent" conventional technologies, pump and treat/soil vapor extraction, necessary to remediate the contamination problems addressed by ISBR. Plan 2 is constructed so that it remediates the same conditions treated by ISBR at the SRID. In order to be fair to both technologies, equal physical process performance is forced from both Plan 1 and Plan 2. Plan 1 and Plan 2 are compared based on what it costs to operate them over equal periods of time. Performance data indicate that the vacuum component of ISBR destroyed 12,096 pounds of VOCs in 384 days, and an additional 40% above the vacuum component was destroyed by bioremediation. The vacuum component data is used in the pump and treat/soil vapor extraction system, assuming that the equal flow rates will remove the same quantity in an equal amount of time.

The ISBR system, as tested, uses two horizontal wells. The first well is an injection well, 300 ft long and 165 ft deep (about 35 ft below the water table). The second well is an extraction well, 175 ft long and 75 ft below the surface (in the vadose zone). A concentration of methane (between 1% and 4%) and any necessary chemical nutrients (nitrogen in the form of nitrous oxide and phosphorus in the form of triethyl phosphate) are blended into the injected air stream to create a biological element for remediation. The methane provides the necessary material substrate for the indigenous microorganism to produce the enzyme methane monooxygenase which, in turn, degrades the principal contaminant, trichloroethylene (TCE). For the conventional technologies used in Plan 2, four vertical SVE extraction wells are assumed to be equal in area influenced to the one horizontal extraction well of ISBR. One vertical pump and treat well is also used. Volatilized contaminants from both remediation systems are sent to a catalytic oxidation off-gas system where they are destroyed.

Economic comparisons for short-term costs are made by relying on actual field data and using cost sensitivity analysis; life-cycle costs are estimated in relation to possible time to achieve cleanup. The first economic comparison is a calculation of the short-term costs in relation to performance. Short term costs are those expenses incurred during the immediate field test demonstration of the technologies compared (generally about a year). The equipment capital costs are amortized yearly over the useful life of the equipment, which is assumed to be 10 years. All short-term equipment costs are amortized at 7%, which is the interest on the loan.

For ISBR there is a total cost of about $354,000 with total 16,934 pounds of VOCs being destroyed by the vacuum component and biological component, giving a cost per pound of VOCs remediated at about $21. The integrated pump and treat/soil vapor extraction with 4 vertical SVE wells has a total cost of about $380,000. Assuming an equal vacuum extraction performance of 12,096 pounds of VOCs removed, the integrated system has a cost per pound of VOCs remediated at about $31. A ratio of ISBR to the baseline shows that ISBR is 32% less expensive than the baseline.

Next, an analysis of life-cycle cost is conducted. A real discount rate of 2.3% is used to calculate the present value. ISBR, with its combination of vacuum component and bioremediation, costs $1 million and remediates the site in only 3 years. The baseline takes 10 years to remediate the site and costs $2 million. ISBR, therefore, saves $1 million and 7 years of remediation. Even when we assume the baseline can perform at twice the expected time and cleans the site in only 5 years, it still costs $1.4 million. ISBR still beats the baseline by $400,000 and 2 years remediation time.

Where ISBR has the potential to exceed the baseline technologies is its ability to remediate a portion of the contamination in situ, thereby eliminating the need to physically remove the contaminant and process it. Since ISBR relies heavily on the biological component to achieve greater performance, sensitivity analysis is conducted to compare the cost per pound of VOCs remediated versus the performance of the biological component. Of particular interest is ISBR + 0% addition. This is a worse case scenario based on a 0% addition from the biological component. It assumes that all the necessary materials are added to stimulate the biological addition, but no additional remediation occurs. In this situation, ISBR still costs slightly less than the baseline, $29 versus $31, respectively. By adding a percent addition of

pounds of VOCs destroyed by bioremediation in addition to that removed via the vacuum component, we can examine how the cost per pound changes with respect to the biological component. Six hypothetical percentages are used to account for the bioremediation levels: 0%, 20%, 40%, 50%, 70%, and 90%. Figure 2 shows the various hypothetical additions and the decrease in cost per pound of VOCs remediated.
The baseline technologies in Plan 2 have a constant price per pound of VOCs remediated of $31 because there is no biological component. As the biological addition of ISBR increases, the price per pound of VOCs decreases. So, even in the worse case scenario where no bioremediation occurs, ISBR breaks even with the baseline. There is, therefore, no cost risk to run ISBR over the baseline system. The savings, however, are quite substantial when the biological component is stimulated. In order for the biological component to occur, it is necessary to inject methane and nutrients into the system. Without this material, only the physical, vacuum component of ISBR is possible. Because the cost of the biological component is so inexpensive, ISBR only has to remediate an additional 1,570 lbs of VOCs over the 12,096 lbs of VOCs remediated with the vacuum component in order for the system to completely pay for the cost of the methane injection. Any additional remediation is achieved at no extra cost and increases the cost savings of ISBR over the baseline technologies.
Next, the total present value cost for operating each plan for five years, including all necessary equipment, is computed. The total equipment costs are included in the first year so that no amortization is needed. As with the short-term cost, the potential cost-savings for ISBR lie with its ability to remediate VOCs in addition to the physical process, thereby lowering the cost per pound and increasing the total amount remediated over equal time. The same hypothetical percent additions of 0%, 20%, 40%, 50%, 70%, and 90% are used. Table I shows the decrease in price per pound as bioremediation increases. The $38 per pound of VOCs remediated with the pump and treat/soil vapor extraction remains constant because there is no equivalent biological addition.

PERSPECTIVES AND COST DRIVERS

The two largest categories in regards to cost for both ISBR and the baseline system are the costs of consumables and labor. The labor and consumables are greater than 85% of the overall operating costs; therefore, if the overall remediation time of the project is shortened, the cost will drop. This is due to the nature of the labor and consumables which are incurred each day of operation. Since ISBR can significantly decrease operation time, ISBR lowers the overall cost of the remediation effort.

APPLICABILITY

ISBR can be very effective in settings where some interbedded thin and/or discontinuous clays are present. ISBR should prove even move successful than in situ air stripping alone because ISBR contains a biological component as well as the physical air stripping process. A potential concern with the use of ISBR is the possible lateral spread of the contaminant plume. If the geology constricts vertical flow, the injection process can push the dissolved contamination concentrically from the injection point. Thus, it may be advisable in heterogeneous formations to use ISBR in conjunction with a surrounding pump and treat system that provides hydraulic control at the site. Note that the limitations on applicable geologic settings described above also apply to soil vapor extraction and pump and treat systems.

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INTEGRATION OF PNEUMATIC FRACTURING AND IN SITU VITRIFICATION IN THE SOIL SUBSURFACE

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ABSTRACT

Pacific Northwest Laboratory is evaluating ways to increase the applicability of the

in situ vitrification (ISV) process at hazardous and radioactive waste sites. One innovation is the placement of a conductive material that will facilitate initiating the ISV process at a target depth.

A series of laboratory tests performed at the New Jersey Institute of Technology (NJIT) assessed the feasibility of pneumatic fracturing (PF) in the highly permeable soils of the Hanford Site. The NJIT tests included an analysis of Hanford soils, a series of PF injection tests, and a parametric analysis to determine how soil properties affect the PF process. Results suggest that the PF process can be applied to Hanford soils and that dry medium (e.g., conductive material such as graphite flake) can be injected into the fracture.

INTRODUCTION

ISV is a thermal treatment technology, developed by the Pacific Northwest Laboratory for the U.S. Department of Energy, to treat soils contaminated with transuranic elements. The process is traditionally initiated at the soil surface through the use of conductive material placed within an array of electrodes. Heat generated in the conductive material melts the soil. Molten soil conducts electricity and is the heat source for downward and outward propagation of the process. Propagation continues until a target volume has been treated. Operating temperatures in excess of 1400C destroy or pyrolyze organic and inorganic molecules. Heavy metals and radionuclides are immobilized in the final ISV product, a glass and crystalline material that has excellent leach resistance.

Initiation of the ISV process in the soil subsurface: 1) has the potential to increase the applicable treatment depth of the process (demonstrated to 6 m), 2) permits the creation of subsurface vitrified structures, and 3) permits the selective treatment of contamination located at depth. A key step for this ISV application is the placement of a conductive material in the electrode array to initiate the process at a target depth. Murphy et al. (1) identified horizontal drilling, direct injection, and subsurface fracturing technologies as methods to initiate the process underground. In laboratory experiments, Luey and Seiler (2) showed that the ISV electrode array can be connected either by linear paths of conductive material or by a plane of such material. Successful initiation by a plane of conductive material provides the basis for investigating pneumatic fracturing for placement of conductive material at a target depth (fractures typically are horizontal in the soil subsurface).

PNEUMATIC FRACTURING

Pneumatic fracturing enhances the in situ removal and treatment of contaminants in low-permeability soil and rock formations. It was developed at the Hazardous Substance Management Research Center (HSMRC) at the New Jersey Institute of Technology (NJIT). The process generally involves injecting air into a contaminated geologic formation at a pressure that exceeds the natural in situ stresses and a flow rate that exceeds the permeability of the formation. The resultant failure of the medium creates a fracture network radiating from the injection point. The established fractures increase the permeability of the formation, thereby enhancing the flow rate of vapors or liquids for more efficient removal or treatment of contaminants.

The principle objectives of pneumatic fracturing are reduction of treatment time and extension of available technologies to more difficult geologic conditions. It is designed to be integrated with other in situ treatment technologies, such as vapor extraction, bioremediation, and pump and treat. Initial applications focused on enhancing treatment of the vadose zone, but recently the technology has been extended into the saturated zone. The PF system has also been modified to deliver biological supplements (e.g., nutrients, buffers, and microorganisms) directly into the fractured formation to enhance bioremediation.

To date, pilot tests of pneumatic fracturing have been conducted at 10 sites in a variety of geologic formations, including a U.S. Environmental Protection Agency (EPA) Superfund Innovative Technology Evaluation (SITE) Demonstration at a contaminated industrial site in Hillsborough, New Jersey (3). Pisciotta et al. (4), Schuring and Chan (5) and Schuring, Valdis, and Chan (6) have published on PF applications.

Pneumatic fracturing is a patented process (7) and is commercially available from Accutech Remedial Systems of Keyport, New Jersey. Accutech offers a fully mobile, production version called Pneumatic Fracturing Extraction (PFE). PFE is an integrated remedial system for removing and treating volatile organic carbon

molecules from geologic formations of low to moderate permeability.

LABORATORY STUDIES

To support PNL's ISV technology, a series of laboratory tests were performed at NJIT to assess the feasibility of pneumatic fracturing in the highly permeable soils of the Hanford Site. The NJIT tests included an analysis of Hanford soils, a series of PF injection tests, and a parametric analysis to determine how soil properties affect the PF process.

Analysis of Hanford Soils

For a field demonstration of ISV initiated in the soil subsurface, PNL selected a test site on the Hanford Site in south-eastern Washington. The underlying geologic unit is the Hanford Formation, which consists of coarse-grained sands and gravel deposited in a matter of days during cataclysmic floods at the end of the last ice age. The rapid deposition preserved a high ratio of void space between particles. The nature of the Hanford Formation results in high porosity (>30%) and high saturated hydraulic conductivity (0.35 to 3.5 cm/s) (8,9).

Investigators at NJIT tested soil from the PNL test site to determine physical properties of the Hanford Formation. Testing included grain size analysis, Unified Soil Classification System (USCS) classification, Atterburg limit tests, organic content, specific gravity, and the standard Proctor density test. All testing was performed in accordance with the standard methods of the American Society of Testing and Materials (ASTM). The investigators then prepared a surrogate test soil at NJIT to support a series of bench-scale PF injection tests. Table I compares the physical data for the Hanford soils with the surrogate test soil; the surrogate test soil compared well with the Hanford soil and was deemed suitable for bench-scale PF injection tests.

Bench-Scale PF Injection Tests

Bench-scale PF injection tests addressed the application of an integrated PF/ISV system in Hanford-type soils. Bench-scale tests physically simulate the integrated process and thereby permit study of critical soil parameters and their effect on fracture injection behavior. Figure 1 illustrates the key components of the bench-scale system: a test tank with plexiglass walls, an air supply system, a flow-directing injector-nozzle, a venting system, and a dry-media injection system. The plexiglass walls of the test tank permit direct viewing of the results of each injection and permit real-time adjustment of injection pressures and flow rates.

Of the 38 PF injection tests performed, 28 tests were with air alone, 6 with air and silica sand, and 4 with air and ISV conductive material. Soil density for the sessions varied between 1.65 and 1.80 g/cm3; the moisture content was 0.8% to 7.8% by weight. Seven nozzle designs were used during the initial 10 fracture sessions with air alone. A nozzle previously designed for clays and other fine-grained soils (the traditional application of PF) allowed air to escape along the injection pipe. Subsequent designs using discs to isolate the nozzle from the injection pipe provided limited success but erratic results. A design using a disc that provided directional flow was the most effective and was used in most sessions.

Results from the bench-scale tests show that dry density and moisture content of the soils are critical factors in creating and controlling fractures and injecting dry media. The most important factor was dry density. Fractures at a density of 1.7 g/cm3 were successful; inconsistencies were observed at lower densities. A density of 1.7 g/cm3 represents the lower range for Hanford soils. As density increased, the pressure required to initiate fracturing tended to decrease. This effect was accentuated through the use of a surcharge on the soil surface to simulate soil overburden pressure.

Fracturing was generally successful when moisture content exceeded 2 wt%. Fracturing at lower moisture contents was successful if density exceeded 1.7 g/cm3. At densities at or below 1.65 g/cm3 and moisture content below 1wt%, discrete fractures were unattainable.

Table I

Fig. 1

This is attributed to a reduction of interstitial surface tension and the corresponding loss of apparent soil cohesion. The initial pressure required to fracture was not affected by the moisture content, contrary to expectations that the required pressure would decrease with increasing moisture content as apparent cohesion reduced the escape of air.

Results from bench-scale tests involving the injection of dry media (silica sand or

ISV conductive material) showed a significant difference in lens geometry for different materials. Excavation was performed after injection of sand or ISV conductive material. For the sand injection, the lens was nearly continuous from the nozzle throughout the plane of fracture. Injection of sand, which is denser and less viscous than the ISV material, probably enhanced displacement or "cutting" of the Hanford surrogate soils. In contrast, excavation after injection of ISV material revealed a continuous layer at the fracture level, but the lens was not traceable to the nozzle. The ISV material appeared to have traveled upward along the injector pipe to the fracture. The ISV material is less dense than sand and has self-lubricating properties, which may have allowed it to travel along the small annular space between the injector and the soil.

The four tests involving the injection of ISV material were modeled to simulate the integrated PF/ISV process, in which conductivity is measured across the filled fracture and through the graphite electrodes of the ISV process. Results show that pneumatic fracturing can establish a conductive link in the soil subsurface between electrodes. The key to successfully establishing this link will be the formation of the fracture in the Hanford soil.

Parametric Analysis

A modelling study was performed to investigate selected physical processes of the integrated PF/ISV system. This was necessary because there was no previous experience with injecting granular media into a coarse-grain soil formation. The results provided insight into the critical mechanisms of the PF/ISV system and allowed estimates to be made of key design parameters for a field-scale system.

In a radial flow model for pneumatic fracturing, the velocity of the injected air decreased rapidly. Calculations suggest that the effective radius of the PF/ISV system may be limited to several feet in Hanford soil at standard injection flow rates and pressures. This distance is significantly less than the radius of influence typically observed in fine-grained soil and rock formations. The difference is attributable to the high permeability of the Hanford soil and corresponding rapid leak-off. Results underscore the need for proper nozzle design for the field, as well as maximization of system flow rates and pressures.

The critical suspension velocities for transport of the ISV conductive material in the fractures were determined by two different methods: Shields diagram method (10) and dust transport method (11). The suspension velocity determined from these methods was generally less than 0.4 m/s. This relatively low velocity indicates that transport of the ISV conductive material should not be an issue in open fractures; rather, the limiting condition will be the ability to propagate the fractures.

The ability to transport ISV conductive material through the interstices of the Hanford soil was also analyzed with standard published filtration criteria. The results show that interstitial penetration of the conductive material will not be a significant mechanism in the Hanford soil; most of the particles are predicted to strain or cake at the fracture interface. A benefit from this caking may be the extension of the radius of the fracture as caking reduces the permeability of the upper and lower boundaries of the fracture. The reduced permeability extends the fracture by minimizing leak-off from the upper and lower boundaries.

CONCLUSION

Results of bench scale tests suggest that pneumatic fracturing and in situ vitrification can be successfully integrated to expand the applicability of both technologies. The establishment of fractures in Hanford surrogate soils shows that pneumatic fracturing is applicable in soils other than clays and fine-grained soils. Successful injection of ISV conductive material in the soil subsurface to provide a conductive link between ISV electrodes provides a means for initiating the ISV process at a target depth. The consequent increased ISV capability enhances the applicability of the technology beyond current demonstrated depths.

We are integrating the results of the laboratory results into plans for a field demonstration of pneumatic fracturing at Hanford to be performed in the first quarter of 1995 on an uncontaminated soil site. If a suitable conductive path is created, then the field demonstration will continue the ISV process initiated in the soil subsurface.

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27-33
IN SITU VITRIFICATION: RESULTS FROM THREE LARGE-SCALE COMMERCIAL APPLICATIONS INVOLVING CONTAMINATED SOIL AND DEBRIS AND THE STATUS OF AN INTERNATIONAL APPLICATION ON A MIXED-TRU BURIED WASTE SITE

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ABSTRACT

Commercial applications of the in situ vitrification (ISV) remediation process are underway. The first commercial project was completed in mid 1994, a second project was completed in October 1994, and a third project commenced in November 1994. All three projects involve the treatment of contaminated soil and buried debris. The three projects include:

the Parsons Chemical/ETM Enterprises Superfund Site in Michigan State. The project involved the treatment of 4,800 tons of soil contaminated with mercury and pesticides including chlordane, DDT, dieldrin and aldrin. Buried debris included drum lids, plastic sheeting, concrete, secondary wastes, and automobile tires.

a private Superfund Site in Washington State. The project involved the treatment of 3,100 tons of PCB-contaminated soil and debris with maximum PCB concentrations to 17,000 ppm. Debris included drums and chunks of contaminated asphalt and concrete. This project is serving as the TSCA demonstration project required to obtain a National TSCA Operating Permit for the application of ISV to PCB contaminated soil.

a private Superfund Site in Salt Lake City, Utah. The project involves the treatment of 5,600 tons of soil and sludge contaminated with various organics

including dioxins, furans, pentachlorophenol, xylenes, toluene and pesticides including DDE, DDT, 2,4-D and chlordane. Debris buried in the site includes plastic sheeting, wooden timbers, pieces of clay pipe, investigation derived wastes, and sample bottles.

In addition to these three projects, Geosafe is preparing to conduct on-site treatability tests and intermediate-scale demonstrations involving plutonium and uranium-contaminated debris buried in pits at the Maralinga Test Range in South Australia. This demonstration project is being performed for the Australian Government as part of the Maralinga Rehabilitation Program. The Maralinga Test Range became contaminated from British nuclear weapons tests conducted in the late 1950's and early 1960's. Geosafe's role in the Rehabilitation Program will involve the treatment of burial pits at the Taranaki area; the site's most contaminated area. The burial pits are believed to contain approximately 5 kg of plutonium, about 20 kg of enriched and depleted uranium, and heavy metals such as beryllium, lead, and barium along with massive amounts of debris resulting from the weapons tests.

This paper will present a brief summary of the ISV process, an overview of each of these four projects and present performance results for these commercial applications. Together, these projects demonstrate the capabilities of the ISV process in terms of treating contaminated soil that contains a wide range of contaminant types as well as significant amounts of buried debris.

ISV PROCESS

ISV is an innovative on site and in situ treatment process that involves the electric melting of contaminated soil and/or other earthen materials for purposes of permanently destroying, removing, and/or immobilizing hazardous and radioactive contaminants. ISV was invented by Battelle, Pacific Northwest Laboratories in 1980 for the U.S. Department of Energy. More than 190 developmental tests and demonstrations of the technology have been performed since that time at four scales: bench, engineering, pilot, and large-scale.

The process involves forming a melt at the surface of a treatment zone between four electrodes. The molten soil serves as the heating element of the process, wherein electrical energy is directly converted to heat as it passes between the electrodes. Continued application of energy results in the melt growing deeper and wider until the desired treatment volume has been encompassed. When electrical power is shut off, the molten mass solidifies into a vitrified monolith with unequalled physical, chemical, and weathering properties compared to alternative solidification/ stabilization technologies.

ISV melting typically involves molten soil temperatures in the range of 1600-2000C. This high temperature results in the removal of organics from the treatment volume by vaporization followed by pyrolyzation within the soil closely adjacent to the melt. No organics remain in the melt or the vitrified monolith due to the inability of organics to exist at the temperatures involved. A broad range of organic contaminant types have been successfully treated in various ISV tests and demonstrations, including volatiles (e.g., benzene), semi-volatiles (e.g., pesticides), and nonvolatile organics (e.g., PCBs, dioxin).

The predominant disposition of heavy metal oxides during ISV processing involves physical and chemical incorporation into the vitrified product, which produces a permanent immobilization result. Most species of metals remain as oxides in the melt and are incorporated into the vitrified product upon cooling. However, since ferrous metals do not have a strong affinity for oxygen in an ISV melt, they will remain in a reduced state. Therefore, ferrous metals (e.g., scrap metals, piping, drums) present in the treatment zone typically melt and sink to the bottom of the melt pool where they are encapsulated there by the vitrified product. It should be noted that ISV processes both organic and heavy metal (including radioactive) contaminants simultaneously, which is a capability largely limited to vitrification processes.

ISV is also distinguished by its ability to tolerate debris within the treatment zone. Organic debris materials behave as other organics during ISV processing ... they are destroyed primarily by pyrolysis. Inorganic debris materials are typically incorporated into the melt and the resulting vitrified product. Types of debris previously processed by ISV include: vegetation, wood, plastic, rubber, cardboard, paper, protective clothing, HEPA filters, activated carbon filters, drums, concrete, asphalt, tires, scrap metal, and general construction demolition debris.

ISV results in a 25-50% volume reduction for most soils, and even greater volume reduction for sludges and wastes that dewater and/or decompose during processing.

The volume reduction results in creation of a subsidence volume above the vitrified monolith. In most applications the subsidence volume is filled with clean soil and the monolith is left in the ground since it is no longer hazardous. Sites treated by ISV should be capable of future use without restriction associated with the vitrified monolith.

The ISV process off-gas treatment equipment used for the first three commercial projects includes a stainless steel off-gas containment hood that spans the area being treated; a quencher; a two-stage high efficiency wet scrubber that removes particulates and neutralizes acidic gases; high efficiency air filtration; and thermal oxidation as a final polishing step. The configuration of the off-gas treatment system can be modified to address site specific requirements.

PARSONS CHEMICAL/ETM ENTERPRISES SUPERFUND PROJECT

The Parsons Chemical/ETM Enterprises (Parsons) site represented the first commercial application of the ISV technology. The Parsons site is located near Lansing, MI. The site involved 4,800 tons of silty clay soil contaminated by a variety of pesticides (DDT, dieldrin, aldrin, chlordane), heavy metals (mercury, lead, arsenic), and trace amounts of dioxin. The contaminated soil was present at various locations on the site, and in a drainage ditch about 400-m from the site. To facilitate the melting process, the contaminated soil was excavated and consolidated into nine cells in a 4.9-m (16-ft) deep treatment trench located in a large open area of the site (see Fig. 1). Each cell was 7.9-m (26-ft) square. A significant amount of debris from the site (including protective clothing, roots and vegetation, wood, plastic sheeting, drum lids, and tires) was also placed in the trench with the contaminated soil. Vitrification operations commenced in July of 1993 and were completed in mid 1994. The site conditions posed many challenges that had not been previously experienced with the ISV technology at large-scale. Although these challenges increased the time needed to complete the first few melts, they were all overcome, and the ISV technology is much stronger for the experience. Some of the key challenges associated with the site included the high moisture content of the soil, the added moisture associated with perched water near the surface, and the high sulfur content of the soil which contributed to a non-hazardous sulfur-based odor associated with the off-gas.

A large number of samples were acquired during the project. Preliminary results indicate that the ISV process met cleanup requirements and related ARARs (1). The primary ARAR of concern was the State of Michigan air emissions standards set for this project. Table I presents typical off-gas performance data obtained during the project.

The EPA SITE Demonstration Program monitored the sixth melt. Their results indicate that the cleanup criteria for the contaminants were met (1). A comparison of pre-test and post-test contaminant concentrations in soil is provided in Table II. Analysis of off-gas emissions indicated compliance with Michigan State air emissions standards. The treatment resulted in about a 35% volume reduction of the contaminated soil. EPA has published a Demonstration Bulletin and a Technology Capsule on this demonstration and have also prepared a video describing the Site Program evaluation. Full details of the SITE Program data acquisition and evaluation will be presented in an Innovative Technology Evaluation Report which is presently planned for publication during the first quarter of 1995. Additional sampling of vitrified product and adjacent soil will be performed in the Spring of 1995 when the vitrified product has cooled.

The Parsons project had value for the ISV technology far beyond the meeting of site cleanup and performance objectives. This project was Geosafe's first commercial project and provided an opportunity to prove the ISV technology and Geosafe's capabilities. This was the first large-scale ISV application in a wet clay soil and several site specific challenges had to be overcome. The technology is better for the experience and several innovations implemented during the project have become a routine part of Geosafe's operations. Geosafe acknowledges the support provided by EPA during the project; and notes that such support is critical to the development, demonstration, and commercial implementation of innovative technologies.

TSCA DEMONSTRATION PROJECT

The TSCA Demonstration Project involved the treatment of 3,100 tons of PCB-contaminated soil at a private site in Region X. Geosafe initiated melting operations at the site in June, 1994, and completed the melting operations in October, 1994. In addition to remediation of 3,100 tons of contaminated soil, the

demonstration was performed as required to obtain a National TSCA Operating Permit from EPA for application of ISV to PCB-contaminated sites throughout the U.S. At this time, most of the data has been analyzed and evaluated but surrounding soil sampling activities are on-going.

The demonstration melts were performed in soil and debris containing PCBs to maximum concentrations of 17,000 ppm. The second melt setting was configured with three layers of PCB-contaminated soil with the PCB concentrations in these layers averaging 12,000 ppm with a maximum of approximately 17,000 ppm. The materials were staged for treatment to a depth of 4.9-m (16-ft). In addition to the contaminated soil, some of the melts contained significant amounts of debris, including drums, asphalt, concrete, protective clothing and other secondary waste. Figure 2 illustrates the configuration of the five melts in which the debris content was varied to allow testing of ISV under different conditions of interest.

Geosafe prefers not to treat sites containing sealed containers because the sudden release of gas or vapors from containers can disrupt the melting process. Because the TSCA demonstration site contained sealed drums, a vibratory beam technique was used to breach the drums in situ prior to treatment by ISV. The technique involves using a crane with a vibratory drive attachment to vibrate a beam into the ground in a grid pattern. The vibrating beam has an effective radius of up to several feet and efficiently breaches the drums. This quick and efficient technique effectively demonstrated a means to prepare a site containing drums for treatment by ISV.

The demonstration project was designed to show the efficacy of the ISV technology for treatment of PCB-contaminated soil and debris sites. The process and equipment operated flawlessly during the demonstration and target melt depths were easily attained. Final analytical results of stack samples indicated that the total DRE of the process met Demonstration objectives. DRE data for the second PCB layer of Melt 2 is provided in Table III. Sampling and analysis of surrounding soil samples is on-going and is expected to be finalized within the next one to two months.

WASATCH CHEMICAL SITE PROJECT

The Wasatch Chemical Superfund site is associated with the former Wasatch Chemical Company. Past practices resulted in the contamination of soil and groundwater at the approximate 18 acre site
in Salt Lake City, Utah (2). The ISV remediation involves the treatment of soils, sludges, and debris contained in a concrete evaporation pond. The evaporation pond was formerly used for evaporation of liquid process wastes. These wastes were discharged into the evaporation pond through an underground process drain system that received wastes from several buildings at the site. The pond has a concrete floor that was built approximately 30-cm (1-ft) below grade level. The pond is 38-m by 38-m by 1.2-m deep (125-ft by 125-ft by 4-ft deep). The maximum contaminant concentrations, based on sampling supporting prior ISV treatability tests, are provided in Table IV.

Additional materials from the site have been gathered and staged on top of the sludge and existing earthen materials in the evaporation pond for treatment including various types of debris associated with the on-going site investigation and clean-up. Debris buried in the pond includes plastic sheeting, wooden timbers, pieces of clay pipe, investigation derived wastes, and sample bottles.

Soil and debris staged in and above the pond result in a treatment configuration requiring only 1.8-m (6-ft) to 2.1-m (7-ft) of melt depth. Although configurations involving deeper treatment depths are more economical for the ISV process, it was decided not to restage the material in the pond. Approximately 36 melts are expected to be required to treat the pond area. The final configuration of the pond is shown in Fig. 3.

The groundwater level at the site is typically 1 to 2 feet below grade at the site depending on the season. The groundwater is contaminated with organics in certain areas. Because the water is very near the surface and, at times, can be above the bottom of the pond floor, pumping activities are on-going to temporarily lower the water table to facilitate ISV processing. Although the ISV process can accommodate saturated media as long as the recharge rate is low, the extra water requires more processing time and can result in a larger volume of secondary wastes.

Near the center of the pond area, materials contaminated with high concentrations of dioxin have been staged. This area, referred to as the dioxin cell, will be treated early in the project.

Mobilization to the site occurred in November, 1994. Thus far, three melts have been

completed and the fourth melt is underway. The remediation project is structured to allow the first four melts to be performed in a demonstration mode to so that site-specific process performance data can be obtained and assessed. Variables are being explored during the first four melts. Prior to starting the remaining melts, adjustments will be made as necessary to optimize the process.

During the second melt, it was found that melting into the standing water table was unacceptable, in this case, with the combination of the concrete floor, because it caused excessive melt agitation and overheating of the hood. The transition of the melt penetrating through the floor resulted in a significant amount of water vapor bubbling through the melt. For subsequent melts, Geosafe will limit melt depth, while dewatering will continue to lower the water table to at least one foot below the pond floor.

The third melt was completed without difficulty and the fourth melt is underway. During the third melt stack sampling was performed to verify that stack emissions were meeting ARARs. It is expected that some of the initial performance assessment data including the stack sampling results will be available for presentation at WM-95.

This project is the first to involve the use of two off-gas containment hoods. Since the required melt depths are shallow and the time for each melt is minimal (few days), two hoods are being used so as to minimize the down time between individual melts. Generally, after each melt, the melt is allowed to cool for several hours while the off-gas system is operated before the hood is moved. Then the hood is repositioned, electrical and piping systems reconnected, electrodes installed, and the process reinitiated. With a two hood operation, one hood will be positioned and readied while the other hood is being used for operation. The use of two hoods results in a more efficient and economical operation since down time between melts is minimized.

MIXED TRU-CONTAMINATED BURIED WASTE

ISV has been selected by the Commonwealth of Australia for use on TRU and mixed waste-contaminated burial pits in the Taranaki area of the Maralinga Test Range in South Australia. The burial pits contain contaminated soil and debris resulting from weapons tests conducted by the British in the late 1950's and early 1960's (3). The pits are believed to contain approximately 5 kg of plutonium, about 20-kg of enriched and depleted uranium, and various heavy metals including lead, barium, and beryllium, along with massive amounts of debris.

The ISV process was initially identified by the Commonwealth as the preferred alternative for stabilizing the Taranaki pits in the report by the Maralinga Technical Assessment Group (3). ISV was determined to be the leading candidate for this application because of the ability of the process to immobilize radionuclides in the vitrified product, the associated volume reduction, the ability of the process to destroy organic contaminants, and the improved occupational, public, and environmental safety benefits resulting from the in situ treatment.

Geosafe completed Phase 1 of the initial Maralinga ISV project in mid-1994 and is now preparing to conduct on-site tests as part of Phase 2. Phase 1 involved an initial site assessment and various tests and analyses, including two engineering-scale ISV tests using uncontaminated soil and debris from the Taranaki site. The Phase 1 work was designed to determine key properties necessary to properly apply the ISV technology to the Taranaki pits. Phase 2 is designed to gather comprehensive test information necessary for the proper scale-up of the process and to collect information to support safety assessments. Phase 2 will involve several engineering and intermediate-scale tests performed in situ and onsite at Maralinga. Some of the Phase 2 tests will involve the vitrification of plutonium and uranium contaminated soil and debris. For the intermediate-scale tests, kg-quantities of uranium oxide are expected to be used and a steel plate resulting from the original weapons tests that is contaminated with approximately 3-g of plutonium oxide will be used in the testing.

At this time, plans are being finalized and equipment readied for the on-site Phase 2 testing. The testing will initially involve several engineering-scale tests on small pits constructed to represent the expected configurations of the Taranaki Pits. The engineering-scale tests are being performed prior to intermediate-scale tests to verify immobilization efficiencies of plutonium and uranium using cerium as a surrogate. In addition, the initial tests will be used to determine various operating parameters such as flow rates, process temperatures, particulate loading,

and electrical properties of the melts that are specific to these test configurations. Following the initial engineering-scale tests, a series of intermediate-scale tests will be performed at the Taranaki area. The intermediate-scale tests are anticipated to produce melts in the size range of 3 to 5-tons achieving melt depths of approximately 1.5 to 2-m.
On-site testing should commence by mid-1995. A successful Phase 2 will be followed by equipment design and fabrication for the full-scale remedial operations. It is expected that remedial operations may commence in 1997.

SUMMARY

All of Geosafe's commercial ISV remediation projects thus far have involved the treatment of soils and/or sludges contaminated with all contaminant types (VOCs, BNAs, heavy metals) and containing significant amounts of buried or staged debris. Debris including drums, wood, tires, concrete, asphalt, and various types of investigation-derived wastes have been successfully processed with the ISV technology. As Geosafe gains more commercial experience with sites containing buried and staged debris and as Geosafe progresses with the Taranaki application involving the treatment of mixed TRU-contaminated soil and buried debris, ISV should be considered as a particularly attractive remediation technology alternative for complex sites containing mixtures of contaminant types and debris.
Use of ISV for treatment of buried wastes or staged debris are applications that hold tremendous potential for this very difficult class of remediation problems. The class of buried wastes is a particular remediation challenge because of the difficulty and high cost of retrieving, characterizing, treating and/or finally disposing such wastes. ISV holds the potential to treat the major types of buried wastes onsite and in situ at much lower costs. In some cases of direct in situ treatment of buried waste, ISV will require improved heat removal and off-gas containment and treatment capabilities beyond that of the existing equipment which is designed for contaminated soil applications. Such equipment capabilities are presently being defined and developed.
Alternatively, ISV can presently be considered for those buried waste sites that present too many challenges for direct in situ treatment and cannot be pretreated in situ (e.g., via the vibratory beam technique) or where there are other reasons for retrieval. In these cases, ISV holds promise as a technique for treating the retrieved waste by simply sorting and restaging the debris into treatment cells. In this configuration, materials not suited for direct treatment by ISV could be removed while all other contaminated debris and soil could be staged for treatment by ISV.

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INTEGRATED DESIGN PROCESS OF A RADIOACTIVE WASTE FACILITY CLOSURE UTILIZING THREE DIMENSIONAL TECHNOLOGY

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ABSTRACT

Technological advances in Computer Aided Design (CAD) have allowed engineers to improve the design of manufactured products for many years. This technology is also becoming increasingly more available to engineers concerned about the environment, when performing activities such as modifications of terrain configurations, alterations of surface water hydrology, and isolation of wastes from the biosphere. The purpose of this paper is to describe the process used in a recent low level radioactive waste facility closure design utilizing three dimensional CAD technology. To accomplish this, several aspects of the design are covered including:
existing site conditions and topography

- design considerations and criteria
- electronic surface models
- design interaction and optimization
- drawing compilation and production

This paper presents the design process used, problems and solutions encountered, and the benefits of using a three dimensional computer modeling tool to produce a radioactive waste facility closure design.

INTRODUCTION

The utilization of three dimensional modeling for civil engineering projects, like most recent technological breakthroughs, offers incredible opportunities for waste closure design. Three dimensional modeling provides overwhelming speed and flexibility in developing design alternatives; literally taking seconds to produce what used to take days. The strength of this tool comes from a combination of instant output and feedback, coupled with the ability to make changes and re-perform calculations. Engineers focus their energy on maximizing design efficiency, tweak and pinch alternatives, and produce an optimal design. In addition, powerful presentation tools result in improved client and regulator understanding (see Fig. 1).

The waste facility closure described in this paper provides an overview of the design process for a typical closure project utilizing three dimensional modeling software.

EXISTING SITE CONDITIONS AND TOPOGRAPHY

Just a few years ago, surveyors would document a topographical landscape, recording their findings into a field book, which an engineer could utilize to manually reproduce the field topography onto paper, creating a topographical map. The process, while fairly time consuming, held the advantage of personal interaction. Both the person recording into the fieldbook, and the person reducing the fieldbook onto a set of plans offered human capacity and recognition of obvious error. These positions were eliminated by the invention of a data collector; a device that electronically stores the survey information as it is shot and later transfers the entire record directly into the computer. While saving an enormous amount of time, the data collector does have its draw backs. Surveyors need to account for the lack of human interaction during survey reduction, by enhancing their methods during the field survey. Complex areas require extra shots. A straight retaining wall used to be defined by one survey shot on each corner and sketched in the survey book; the person reducing the survey would understand the wall configuration by the notes and the sketch. Data collectors remove the human interaction, causing far greater challenge to understand what is in the field.

The survey data is imported into the computer where each survey shot becomes a three dimensional point in space. Horizontal and vertical coordinates are relative to the datum established. Each point represents the actual shot taken by the surveyor in the field. Computer software then generates triangles by connecting adjacent points. The result is a three dimensional surface formed by an array of interconnecting triangles, each having a specific size, shape, and plane orientation. This "triangulated model" is the heart of three dimensional surfaces.

Once the Triangulated model is constructed, there are several means to display and analyze the data. Contours can be generated in a matter of seconds. Each triangle is made up of three - three dimensional points, each point having a specific vertical coordinate. For any two points in space, there is a difference in elevation. For example, if a triangle consists of two points at elevation 95.5 and the third at 96.5, it is easy to see that connecting these points, the two vertical segments would have a center at elevation 96.0. For this triangle, a contour at elevation 96.0 would be drawn across the face. Utilizing this principle, the computer draws contours, for any triangulated model. It is important to note that the contours generated for any model are important only for human knowledge. The computer does not require, nor process contour data. (See Fig. 2)

The area being closed was surrounded by existing features, each offering unique design hurdles. An existing road including drainage channels, fence and power poles, an existing closure cap consisting of two feet of cover soil and three feet of Kaolin clay, and an open field. A large amount of the design effort was to make the existing ground triangulated model(s) accurately represent actual field conditions. Existing features such as power poles and fence were not represented three dimensionally, as there would be little gain for a large effort to incorporate this

data. They were always represented two dimensionally (in plan view), so as not to be overlooked during the design process. Concrete drainage channels and other like features were represented three dimensionally (at least the surface of them) because these were critical to the design outcome. The existing closure cap consisted of a two foot vegetative cover and three feet of Kaolin clay.

DESIGN CONSIDERATIONS AND CRITERIA

The waste facility closure "cap" conformed to current EPA regulations for landfill covers, which set final grading of the closure area between three and five percent. The proposed closure cap consisted of (top down) a two foot vegetative layer, a geocomposite drainage layer (GDL), a flexible membrane liner (FML), a geosynthetic clay liner (GCL), and a two foot (minimum) foundation layer with a high strength geotextile reinforcement layer (HSGR) embedded (see Fig. 3). Three aspects of the cover varied from "standard" EPA landfill covers. 1) Most closure covers constructed in the past utilized a two to three foot layer of Kaolin Clay as a barrier material. As mentioned above, the combined layers of GCL and FML are being utilized here. There are many advantages of these materials, among which are resistance to failure during settlements, ease of construction, and cost. 2) There was no biodegradable waste common to municipal landfills within the waste area, and therefore, no vent layer was required. 3) In addition, a typical cover has a one foot foundation layer. This cover has a two foot minimum, which satisfies two criteria; to provide extra protection against radioactive exposure during construction, and provide additional support strength to the GCL and FML layers, i.e. large settlements within the waste area are anticipated.

For the purposes of computer aided design, the cap consisted of two primary layers, the surface of the vegetative layer, and the drainage layer two feet below. These layers impact drainage flow, which is critical for grading purposes. The other proposed layers mirror the drainage layer, and therefore are not considered during the initial grading effort; after the cap is designed these layers will be added to the construction drawings.

ELECTRONIC SURFACE MODELS

The cap design was a trial and error process, beginning from the top down. The waste boundary became the closure cap boundary (plus 1 foot for construction tolerance), defining the area which the top of the closure cap would encompass. Once cap boundary was defined, the existing ground within the boundary was copied four feet higher, and considered a design control surface. That is, since this new surface was exactly the same as the existing ground, but four feet higher, and since the cover section is four feet minimum (see above), this new surface represented the lowest possible design surface. However, this surface still did not meet the slope requirements of 3% minimum and 5% maximum.

Another surface was input into the computer, one which was shaped similar to a gable roof on a house, and met all design criteria. One of the useful design tools on the software package allowed the user to view the design in a plan view (top down) and label the depth between the proposed design surface and the original ground surface in a grid. In this way the depth of additional fill required over and above the four foot design surface was identified. It was quickly seen that in some areas there was over 10 feet of fill between surfaces. Those areas which had large fills were regraded to reduce the proposed grade. Several iterations later the top of cap surface was defined. Once the top surface was approved by the design team, it was copied downward two vertical feet and became the top of drainage layer surface.

At this point the problem of tying the proposed surface into the existing ground, was the next design hurdle. Each of the different conditions which surrounded the closure area required a unique solution. To tie into the existing Kaolin clay closure cover the as-built survey information of the clay was input into the computer, and a new surface created consisting of existing ground/existing clay. In this way one computer generated surface represented two field conditions, the existing Kaolin clay and (where there was no clay) the existing ground. The drainage layer side slope would tie-in to whichever surface it came across.

Utilizing the three dimensional software, a 4:1 side slope was draped from the edge of the drainage layer surface to the existing ground/existing clay surface. Within seconds the proposed surface tied into the existing. However, drainage must always flow downhill; the tie-in had to be checked and modified in those areas where the it formed a low spot. This proved to be fairly difficult along the southern border, where the existing closure cap was sloping downward to the north, counter to the

proposed cap which was sloping downward to the south. At the junction a subsurface swale had formed (the drainage layer is below the cap surface), which had to run longitudinally downhill to the south. This subsurface drainage was outlet into an existing concrete channel. The channel had been used to collect subsurface drainage for the Kaolin clay, so the elevation of the channel was low enough for the new subsurface drainage system to feed into.

Tie-in to the adjacent roadway drainage ditch was less complicated. The proposed side slope tied into the existing ground cleanly, with few modifications necessary. A cursory check proved that the existing ground sloped away from the cap, so that further grading of the existing ground was not required.

DESIGN INTERACTION AND OPTIMIZATION

Many aspects of the design process were enhanced by utilizing three dimensional technology. Cross sections and profiles used for design evaluation and in the final construction drawings were literally cut in seconds. A process which previously would take days was reduced to issuing a command in which several parameters were given; which layers (surface models) to display, where to take the cross section from, and where to place it in the drawing. The ease at which this information could be produced should not be under estimated. The ability to cut sections instantly, allowed the designers the flexibility of cutting them whenever needed. This proved an invaluable tool throughout the design. Temporary profiles and sections were utilized during the design process, to enhance understanding of what was being designed. In other words, in areas in which the design was complex, the ability to display certain information from any perspective allowed the designer to fully comprehend the design three dimensionally and recognize whether an area required modification.

By regulation, slopes were critical to the design. The computer software enabled views of the design, in which slope was represented by color code. Since many areas of the closure cap were actually graded by the computer, this gave an instant view of what grades the computer used in the design. In certain areas the computer deviated from the slopes which were programmed in for the design. This happened due to too many parameters specified for a particular area, which made the design of that area impossible to hold all parameters. In these situations, the computer would hold as many parameters as it could. Unfortunately, these were not always the same as what would be held if a designer were faced with the decision. By displaying the surface slope information by color code, an immediate surveillance of the entire design was visually evaluated, and areas which required attention were flagged.

Earthwork computations are another great example of the benefit reaped from three dimensional design. A process which performed manually would take weeks and have large room for error, was achieved at the push of a button. The process consisted of choosing which layers of material are to be calculated. The computer took approximately a minute, and provided a detailed report, and summary.

DRAWING COMPILATION AND PRODUCTION

The end result of this design was the production of design drawings which were to be used for the construction of the closure cap. The three dimensional software was of great aid in the development of the design but required a fair amount of polishing and embellishment to produce final construction documents. Construction notes, title blocks, legends, details, and labeling are among those items which still require manual input. "Manual input" being a relative term, since even these items are electronically produced on a CADD station, however, they are manual in the respect that they are not automatically produced by software and are created by hand. Figure 4 shows a portion of the final construction drawing plan view as compared to the three dimensional software output, and gives an indication of the type of embellishment required. The final construction drawings set consisted of; cover sheet and index, site map, plan sheets, section sheets and detail sheets. Of these, the plan and section drawings held the greatest benefit from the three dimensional software. The section sheets were almost completely output from the three dimensional software, with minimal embellishment required. Each civil detail was originated by the three dimensional software and manually enhanced.

CONCLUSION

The benefits of utilizing three dimensional software to design a closure cap include flexibility, a host of design tools, output presentation, and most importantly large cost savings. Although, a large percentage of time is required to make sure that existing data is input into the computer accurately. However, the benefit is far

greater than the cost. The process of designing a closure cap has not changed, however the tools to accomplish the design are among those within the high tech. avalanche. As these tools continue to advance, the design method and flexibility will continue to refine saving both time and money. These advancements are not without cost. Each advancement in technology translates into increased costs for hardware and software. A common mistake is to perceive a one time initial cost. In addition, each software advancement requires personal training. Only the passage of time will dictate what the future holds for civil engineering software. In the past, technologies are developed for large budget types of fields, and then adapted to other applications. If this continues to be true, then perhaps one day the civil engineering future, including radioactive waste facility closure design, will contain virtual reality technology or even computer aided construction.

27-35

A DECISION PROCESS TO MAINTAIN COMPLIANCE: POSITION DESCRIPTIONS TO TRAINING REQUIREMENTS RELATED TO HAZARDOUS WASTE MANAGEMENT

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ABSTRACT

Title 40 Code of Federal Regulations Parts 264 and 265 require a written job description for each position at a facility related to hazardous waste management. An operator of a treatment and storage facility of hazardous waste is regulated under the training requirements of Title 40 Code of Federal Regulations (CFR) Parts 264.16 and 265.16, Personnel Training.

Documents reviewed during the development of position descriptions at EG&G Rocky Flats Environmental Technology Site (RFETS) included; (1) the Colorado Department of Health Notice of Violation No. 92-06-17-01, (2) the results of the EG&G Rocky Flats Causal Factor Analysis of Environmental Compliance Violations at the Rocky Flats Plant, (3) the Colorado Hazardous Waste Regulations 6 CCR 1007-3, (4) the State R.C.R.A. Permit ID# 91-09-30-01, (5) pertinent information on the training and qualification history at the Rocky Flats Plant, (6) the Savannah River Site Environmental Training Decision Flow Diagrams, (7) research study on determining position descriptions, and (8) compliance agreements with the Colorado Department of Public Health and Environment for compliance with Title 40 Code of Federal Regulations Parts 264.16 and 265.16 (d)(1)(2)(3).

Hazardous waste worker positions to training requirements for personnel involved with hazardous waste management was reviewed and accepted by the Department of Energy and is in its first phase of implementation at the Rocky Flats Environmental and Technology Site. Facility specific position descriptions and required training for personnel involved in hazardous waste management provide a significant, manageable, and innovative approach for compliance with Resource Conservation and Recovery Act (RCRA) requirements, Title 40 Code of Federal Regulations Parts 264.16 and 265.16 (d)(1)(2)(3) for a hazardous waste facility.

INTRODUCTION

The Resource Conservation and Recovery Act (RCRA), a regulatory statute designed to provide "cradle-to-grave" control of hazardous waste, establishes the national hazardous waste management program (1). Facilities involved with hazardous waste management programs are required to comply with Title 40 Code of Federal Regulations Parts 264 and 265, Standards and Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal (TSD) Facilities.

The Colorado State administrating agency for RCRA is the Colorado Department of Public Health and Environment (CDPHE) under the Colorado Hazardous Waste Regulations 6 CCR 1007-3. Sections 264.16 and 265.16 require that personnel training requirements are identified and outline the hazardous waste facility's RCRA environmental training compliance programs (2).

Pursuant to RCRA, each facility must maintain complete training documents and records. These documents and records include "(1) The job title for each position at the facility related to hazardous waste management, and the name of the employee filling each job; (2) A written job description for each position listed .... (which) must include the requisite skill, education, or other qualifications, and duties of employees assigned to each position; (3) A written description of the type

and amount of both introductory and continuing training that will be given to each person filling a position listed under ... this section"(3)
An ongoing practice for compliance with these requirements has been to provide initial RCRA training and annual RCRA refresher to all employees, also referred to as "compliance madness," at hazardous waste generator and Treatment, Storage and Disposal (TSD) facilities. Although this method was initially accepted, this practice has been identified as inadequate for meeting the regulatory training requirements for maintaining position descriptions related to hazardous waste management.
While some facilities were "training everyone" other, due to widely varying interpretations of the RCRA training regulations, argued that generator training was not required, and therefore maintained that the absence of generator training was not a compliance issue (3). On the other hand, there are facilities where the organizational structure and the tasks and responsibilities assigned to generators dictate extensive training of thousands of personnel to assure compliance with the RCRA regulatory requirements.
Finally, it was noted that while some facilities had training programs in place, the training curricula was inconsistent with the specific needs of the trainees (4). Specifically, the content of the training courses did not include all the topics necessary to meet the tasks of the employees or to ensure a facility's compliance with the requirements of 40 CFR.
Conclusion from initial research indicates a need for a formal process to identify worker positions involved in hazardous waste management and the related training requirements for those positions.

STATEMENT OF PROBLEM

Notice of Violation

On June 17, 1992 the Colorado Department of Public Health and Environment (CDPHE), through the Hazardous Materials and Waste Management Division issued a Notice of Violation (NOV) to the Department of Energy (DOE) Rocky Flats Office (RFO).
The Notice of Violation was based on findings that Rocky Flats had violated the Colorado Hazardous Waste Act, Section 25-15-301 through 313, C.S.R. (1990) and its implementing regulations.
The violations occurred over a period of approximately twenty-two months and demonstrated "systematic problems" with regard to compliance with hazardous waste requirements. Of the fifty-six violations, twenty-eight were in violation of 6 CCR 1007-3, Section 265.16 Personnel Training (5).
Rocky Flats Plant's immediate response to the NOV included a risk assessment through a causal factor analysis and the development of corrective action plans.

Causal Factor Analysis

The causal factor analysis examined the events and conditions that led the Colorado Department of Public Health and Environment to issue a Notice of Violation to EG&G Rocky Flats on June 17, 1992. Of the fifty-six separate violations of the Colorado Hazardous Waste Act, twenty-eight were in violation of 6 CCR 1007-3, Section 265.16 Personnel Training. The CDPHE made a determination that the employees interviewed demonstrated poor knowledge of regulatory requirements in both waste management and production.
Specific to the Resource Conservation and Recovery Act (RCRA) Training and Qualification Program at Rocky Flats Environmental Technology Site (RFETS), the causal factor analysis resulted in a training related problem statement noting "an insufficient cadre of adequately trained, qualified, and accountable RCRA custodians, knowledgeable supervision and knowledgeable support personnel assigned to RCRA-regulated areas across the RFP site" (6).

Status of RCRA Training at Defense Program (DP) Sites

Prompted by the June 17, 1992 issuance of the RCRA Notice of Violation by the State of Colorado Department of Public Health and Environment, a review of the status of RCRA training at Department of Energy Defense Production sites was conducted. The Office of Self-Assessment and the Office of Engineering and Operations Support conducted on-site reviews at eight Defense Programs facilities between November 1992 and April 1993 (4).
The objective of the on-site reviews was to identify systemic weaknesses in the RCRA training programs and to identify weaknesses which leave the complex vulnerable to Federal and State RCRA violations. Findings were categorized into four areas: 1) Programmatic Deficiencies, 2) Federal and State Compliance, 3) DOE Orders and

Guidance Documents, and 4) Other Issues.
The final report identified deficient areas of which some are directly related to adequate selection of hazardous waste workers training requirements. These findings included the following:
1. There are widely varying interpretations of the RCRA training regulations, some of which exclude the training of hazardous waste generators.
2. Formal mechanisms do not exist to ensure that the process used for the identification and training of personnel is functioning.
3. The training curricula at some sites are inconsistent with the specific needs of the trainees. Often, the content of the training courses does not include all the topics necessary to ensure a facility's compliance with the requirements of 40 CFR.
4. Supervisors' RCRA training is often inadequate for them to identify the training needs of their staff and to assess staff's performance at RCRA-related activities.
5. Personnel job titles and duties, and associated training requirements related to hazardous waste management, were not available at some of the contractor sites.
6. The RCRA training records reviewed typically did not contain all the required information, such as position description and introductory and continuing training plans.
Inconsistent interpretations of the RCRA training requirements across the complex was identified as a root cause for these deficiencies.
Recommendations were made for compliance with RCRA training programs to include, as a minimum, documentation for hazardous waste management personnel by employee name, job title, job duties, and required training (introductory and continuing). This documentation is required, as a minimum, for all hazardous waste generators, RCRA Custodians, managers (supervisors), trainers, and course developers. In addition, "the process by which personnel are identified for training should be documented and a formal mechanism established to ensure this process is functioning, such as in a quality assurance plan" (4).
A decision process to determine position-to-training requirements is a critical component of hazardous waste management. Position-to-training requirements ensure appropriate training and qualification based on tasks and duties of the position. The U. S. Department of Energy concluded that a comprehensive decision process resulting in position-to-training requirements must be developed for compliance with 40 CFR and to strengthen the RCRA training program for the prevention of future Notices of Violation (4).
METHOD
Development of a decision process ultimately determined the required training for hazardous waste workers fulfilling compliance requirements of the Colorado Hazardous Waste Regulations 6 CCR 1007-3 Part 264.16 and 265.16 (d)(1)(2)(3) at EG&G Rocky Flats Environmental Technology Site.
This decision process ensured hazardous waste compliance for the facility, and ensured accurate identification of personnel for specific training requirements related to their duty area. This accurate identification also provided significant cost savings by avoiding unnecessary time attending inappropriate training.
The decision process for determining a training path for hazardous waste workers can be modified and made specific for personnel at any hazardous waste facility.
Description of Process; Major Objectives
Major objectives of this process included the following:
1. Thorough research of all applicable federal and state regulations.
2. Interviews with personnel in similar outside industries required to maintain hazardous waste compliance training programs.
3. Identification of the requirements for personnel training documents and records to be maintained at a hazardous waste facility.
4. Analysis which encompassed; (a) identification of existing regulatory requirements for training, (b) identification of personnel who have the potential to be exposed to hazardous waste and hazardous waste management while performing their job duties, and (c) identification of classroom training consisting of applicable regulatory objectives.
5. Development of Hazardous Waste Management Position to Training Requirements Matrix.
6. Pilot study on the use of the decision process and subsequent position descriptions.
RESULTS

The hazardous waste facilities that were contacted and interviewed during the initial research were cognizant of the importance of RCRA regulations and committed to complying with RCRA training requirements. These facilities have expended considerable effort in the development and implementation of their RCRA training programs and contributed invaluable information on maintaining compliance with RCRA regulated training requirements (7).

Position Description and Categories for Hazardous Waste Workers

Position descriptions were developed for positions at the Rocky Flats Environmental Technology Site (RFETS) and grouped into regulatory categories. Position categories were chosen from existing hazardous waste management positions already in place and generic positions identified within DOE guidance documentation. These positions ranged from; 1) senior managers, whose responsibility it is to manage facility programs that involve hazardous waste even though they themselves may never be exposed to hazardous waste or the risks associated with hazardous waste management, to 2) "hands-on" waste handlers, whose responsibility it is to generate, treat, store, or dispose of hazardous waste, to 3) clerical and support personnel who provide administrative support to organizations that deal with the treatment, storage, or disposal of hazardous waste, but, again, are never actually exposed to the risks posed by hazardous wastes. Final documentation identified nine position categories and examples of job tasks and responsibilities that may fall within each category.

Colorado Hazardous Waste Regulatory Matrices

Two matrices were developed for RFETS to show; (1) a relationship between the RCRA regulatory requirements for hazardous waste management and the facility position (based on the job-task-analysis of that position), and (2) a relationship between the RCRA regulatory requirements for hazardous waste management and the facility's training curricula.

Position to Training Requirements Matrix

The Position to Training Requirements Matrix for RFETS was the third matrix in the series of matrices that were developed providing the relationship between a facility position and the required training for that position. This position to training requirements matrix provides an easy reference to determine an individual's training requirements.

Hazardous Waste Worker Training Program

The RFETS official tracking system for training requirements is capable of listing the type and amount of both introductory and continuing training that is available to each person employed in a position related to hazardous waste management. This tracking system is used for identifying a person's training path as it relates to a hazardous waste position. This tracking system is one available reference tool for the immediate supervisor, the trainee, as well as for the training scheduler when determining required training. Importantly, this system ensures that the required personnel training, as indicated, is completed within the required six months (2). In addition to satisfying the requirements of Title 40 Code of Federal Regulations Part 264.16 (d)(1)(2)(3), establishing a process for determining position-to-training requirements for personnel involved with hazardous waste management results in training that is targeted to a specific audience for specific duties and reduces the cost of unnecessary and inappropriate training.

CONCLUSIONS

Development of position descriptions for personnel involved in hazardous waste management is the first step in assuring integral compliance with Resource Conservation and Recovery Act (RCRA) training requirements. A second step is to identify all environmental training requirements applicable to the facility and initiate the same process to assess and identify job duties and tasks, positions, and evaluate specific regulatory driven training requirements for those positions. Once positions, duties and tasks, and required training have been identified, a facility can develop a comprehensive and cost effective environmental training program, meet regulatory requirements and demonstrate compliance.

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27-36

DEVELOPMENT OF A "SHOULD COST" DATABASE FOR ESTIMATING ENVIRONMENTAL RESTORATION ACTIVITIES

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ABSTRACT
A new estimating philosophy has been developed at Rocky Flats Environmental Technology Site (RFETS) for estimating the cost of environmental restoration activities. The traditional cost estimating philosophy is to estimate what a task will cost based on the historical cost of similar tasks. This new approach estimates what it "should cost" to perform a task, based on a detailed analysis of the activities and the resources required to accomplish these tasks. A software system has been developed to assemble "should cost" estimates for all environmental restoration activities and generate new estimates for specific projects. Utilization of this software system and the "should cost" approach to estimating can result in significant cost reductions to DOE projects. This paper will discuss the traditional estimating philosophy, understanding the "should cost" philosophy, and how to implement this new approach to estimating.

The Traditional Estimating Philosophy

The traditional estimating philosophy is to estimate what a task will cost given a set of parameters, conditions, and historical cost information. This technique worked well when the goal of the estimate was to establish a budget that had a low probability of overrun. However, this approach often led to project estimates based on the inefficiencies of past performance. Many Department of Energy (DOE) projects are performed by contractors selected for their expertise, with little or no concern for the cost involved. Some projects, especially Environmental Restoration (ER) projects, are performed on a cost-plus fee contract rather than a competitively bid contract. The costs from these non-competitively bid contracts become the historical data used to develop new estimates. These historical costs are then escalated and contingency is added for unknowns. Many of the projects involved extensive overtime costs which also lead to increasingly higher estimates for comparable scopes of work. In short, the old way of estimating costs perpetuates the past instead of improving upon it.

This traditional estimating philosophy has been used to establish DOE budgets that exceed those experienced for similar projects in industry. The ER standdown in January 1994 provided the results of a study by Independent Project Analysis, Inc., that indicate DOE ER projects cost approximately 32% more than similar projects in industry. This 32% above industry cost is a DOE complex-wide average of many projects, with some costing much more than the 32% and others being more comparable to industry. To continue to estimate costs based on project costs that are 32% above industry standards only perpetuates this pattern of continued high costs.

The continual development of inflated estimates leads to a condition where the estimate determines the scope of work rather than the scope of work determining the estimate, e.g., projects are expanded, enhanced, or poorly executed, leading to the utilization of all available funds. These changes are typically not identified in revisions to the estimate basis. Subsequent estimates which are developed utilizing

the historical data, are not based on comparable project scopes. This leads to higher estimates and higher costs.

DOE's New Estimating Goal

Reductions in funding and the awareness that DOE projects consistently cost more than similar projects in industry have forced DOE to change its emphasis on cost estimating and project performance. The emphasis now is to improve performance and reduce costs. The goal is to improve on the past, not repeat it. With this goal in mind, EG&G Rocky Flats assembled a team of experts to improve the approach to estimating projects at the RFETS and advance the philosophy of cost estimating. This innovative new philosophy is called "should cost" estimating and will assist DOE in achieving the goal of reducing costs to a level that is comparable with industry standards. This new philosophy to cost estimating has the following advantages over traditional estimating:

"Should cost" estimating provides the sound basis needed to reduce project costs.

"Should cost" estimating is an improvement on the past, not just a repeat of history.

"Should cost" estimating provides the basis for developing credible and consistent cost estimates for all activities in the ER program, including activities where historical costs are not available or are unreliable.

"Should cost" estimating establishes the benchmarks needed to identify sources of cost reduction, quantify the expected reduction, and improve productivity.

"Should cost" estimating identifies the project cost drivers which are essential for value engineering and other cost reducing exercises.

"Should cost" estimating bases the estimate on the worth or value of an activity, not the average historical cost.

"Should cost" estimating targets project costs at the 100% productivity level. The cost, schedule, and technical risks are addressed separately.

The "should cost" database can provide a resource listing that can be used to level labor requirements.

This new philosophy of "should cost" estimating provides the opportunity for the change needed to reduce ER project costs to industry levels. Estimates can be developed on the worth or value of a task rather than what the task has historically cost. The traditional approach to cost estimating was consistent with DOE policy and procedures in place at the time. The need to improve cost estimating is driven by changing times. By utilizing the "should cost" estimating tool now available, estimators are able to prepare ER estimates that will promote improved performance and cost reduction.

Understanding the New Estimating Philosophy

There are three parts to the "should cost" estimating system at Rocky Flats: the should cost database, the software system that accesses the database and generates new estimates, and the work breakdown structure (WBS). The "should cost" estimating system is organized by the WBS. For the ER program, the WBS is called a Code of Accounts (COA). The Interagency task force, in which DOE participated, developed a complete COA for all ER activities. This COA has been adopted for the ER "should cost" estimating system at Rocky Flats. Additional levels have been added to the COA to establish the base activity level at which the detailed benchmark estimates are developed.

A supplemental COA Dictionary has also been developed which defines the scope of each COA element including scope exclusions. Since the benchmark database is based on the Interagency COA, it can be used throughout the DOE complex and other agencies, if desired. Using a standardized COA allows for a direct comparison of "should cost" benchmarks for similar tasks at RFETS and similar tasks throughout DOE. These benchmarks can then be compared with the Hazardous, Toxic, and Radioactive Waste Historical Cost Analysis System (HCAS) database, which uses the Interagency COA, to facilitate a meaningful comparison with actual historical costs.

The benchmark "should cost" database is the key to the "should cost" estimating philosophy. The benchmarks established in the database are specific for each activity. Where possible, the benchmarks were developed using an industrial engineering approach for establishing performance standards for each activity. Performance standards have been established and used successfully for production work and repetitive tasks throughout industry; however, performance standards have not previously been established for most of the activities associated with the ER program. To establish the "should cost" benchmarks for each activity in the ER

program, an estimate of the worth or value of the activity was developed by professionals experienced in performing each activity. Each benchmark was then reviewed for consistency and credibility before being adopted.
The database of benchmarks for ER activities provides the details needed to support a cost estimate if the specifics are unknown. It also provides the details of what is included, so modifications can be made if project-specific details are known to be different than the benchmarks.
Each benchmark in the database contains the following specific data:

- COA Number
- Name
- Scope of Work
- Basis of Estimate
- Assumptions
- Exclusions
- Notes

The initial set of benchmarks represents a mid-size effort for each COA. Each activity was further defined by the individual tasks required to complete the activity, as well as the specific resources needed to support these tasks. Each activity in the COA is a specifically defined benchmark supported by detailed estimates for what it "should cost" to complete the activity. The COA for developing a conceptual site model, for example, details the activities required to develop the model, i.e., evaluate resources, evaluate exposure pathways, evaluate exposure scenarios, and develop the model. Specific resources required to support these activities are an engineer, physical scientist, and risk assessor. Units required by each resource were estimated and incorporated into the benchmark database.
Additional benchmarks will be added for both small and large efforts for each activity identified in the COA. Parametric modeling can then be used to develop estimates for projects with specific details that are between the benchmarks.
The "should cost" estimating database eliminates two common problems that exist in estimating ER activities. One is understanding and quantifying the work scope of an activity. The benchmarks detail the scope that is included in each COA activity. If project-specific data is not available the benchmarks can be used without adjustment. The second is finding credible historical cost data. The benchmarks replace the need for historical cost data. In the database, the scope, assumptions, and exclusions are well defined for each activity, which greatly enhances one's ability to perform root cause analysis on project cost overruns and initiating corrective action.
The software system is written in FoxPro and provides the user access to the "should cost" database and software required to develop new cost estimates. FoxPro was selected as a cost savings measure to utilize an off-the-shelf item rather than spending funds on development. The database can be accessed by the user but can not be changed by them. The software system also allows the user to assemble information from the database into an estimate. The user has the option to use the benchmarks as they exist in the database or to adjust them to reflect the detailed work scope of a specific project.
The "should cost" estimating software system will automate the formulation, documentation and archiving of cost estimates in such a way that estimates can be created and adjusted confidently using "should cost" benchmarks for COA activities. Cost estimates developed using this "should cost" database template are streamlined, standardized and supported by detailed backup that is defensible and traceable.
The cost estimate database consists of two related although distinct database modules. The first module, the COA activity module, encompasses the entry and maintenance of the "should cost" benchmarks. This involves applying resources and associated costs for what it should cost to support each activity in the COA. The second module supports the development of new estimates based on site-specific activities. The new estimates are developed by using code of account activities as templates for project work package activities. The system allows for easy modifications based on project-specific requirements that differ from the standards defined in the database. As the project manager adds an activity, the software creates a copy of the COA activity and its associated resources into a working estimate. The project manager can then adjust resources and base units from the benchmark considering site-specific complexity and scope requirements. The information contained in both modules is detailed below.

The COA activity module includes the following data fields and outputs:

COA - The codes of account are grouped and organized by likely project activities. The COA includes scope and assumption narratives.

COA Activities - The COA is organized into activities which are basic elements of work composing each COA. Activities have a scope, base quantity (unit of measure), and a complexity factor.

COA Activity Resources - Activity resources are generic resources as they are applied to accomplish activities. The generic resource base costs are stated and the units of the resource are adjusted for the specific activity.

Generic Resources - A library of generic resources is provided with associated base unit cost from which COA activity resources are drawn. A table of standardized costs for each identified resource is included in this module.

A COA Data Sheet compiles data from the fields in the COA module and a report can be printed showing what the activity "should cost." An example of a COA Data Sheet is shown in Table I.

The new cost estimate development module includes the following data fields and outputs:

Operating Units - Operating units are the overall organization of estimate activities.

Work Packages - Work packages are the second level of organization for estimate activities.

New Cost Estimate Activities - Estimate activities are copied from COA activities and can be adjusted to fit the particular requirements of the estimate.

New Cost Estimate Activities Resources - New cost estimate activity resources are attached to the COA activity when the cost estimate activity is added. This resource can be adjusted to fit specific work on the new estimate.

New Cost Estimate Log - Compiles estimates logged by revision number, estimator, revision date, reviewer, and reviewed date.

A Cost Estimate Activity Summary compiles the data from the cost estimate development module and generates a new estimate. An example of a Cost Estimate Activity Summary is shown in Table II.

The software has been designed to offer a variety of report options, including a Summary Data Sheet Rollup (Table III), which can be provided at any level of the COA structure and a Cost Estimate Activity Detail (Table IV).

Implementing the New Estimating Philosophy

The ability to improve on the past and obtain the full effectiveness of the "should cost" estimate philosophy requires the ability to track actual costs in the same format that they were estimated. RFETS intends to enforce utilization of the COA to prepare all ER work packages, to promote a consistency and uniformity in defining the scope of activities that has not existed before. The software system will then feed the "should cost" estimates directly to a cost accountability system for comparison with the actual costs collected in the existing accounting system. Any future comparison of data truly becomes a credible comparison of similar activities with the same scope.

At RFETS, the "should cost" estimating philosophy is being implemented on all projects. Projects at RFETS are categorized into three types: construction projects, maintenance tasks and operating projects. Construction projects include line item projects, general plant projects and maintenance projects that are covered by the Davis-Bacon Act. Maintenance tasks are maintenance projects that are not covered by the Davis-Bacon Act. Operating projects are specific portions of an operating program. The Should Cost Estimating philosophy can be applied to any of these projects because it identifies specific resources to accomplish each activity and assigns a value to each resource. Associated risk is considered for each activity as well as appropriate indirect costs. This results in an all-encompassing estimate for what each activity "should cost."

The "should cost" estimating philosophy at RFETS, along with a cost accountability system that compares the estimated cost to the actual cost, has been fully implemented for all maintenance tasks. The result is exciting reductions in the cost of doing business. This philosophy is currently being implemented on all construction projects. The database of "should cost" benchmarks is nearing completion for the ER program and will be implemented along with a cost accountability system. The full database will be available for use in May 1995.

The development of this "should cost" estimating system was accomplished utilizing a

team of personnel with an extensive background in cost estimating, software design and all phases of ER support. The team consisted of personnel from EG&G Rocky Flats, Inc., Dames & Moore, Inc., The S.M. Stoller Corp. and ICF Kaiser Engineers. EG&G Rocky Flats, Inc. developed the "should cost" estimating concept and provided guidance and oversight to the team. EG&G Rocky Flats, Inc. also provides a review and acceptance of all the benchmarks within the database. After assessing the ER activity at RFETS, Dames & Moore developed a COA and associated dictionary that specifically defines the scope of each COA account. "Should cost" estimates for all RI/FS and remedial design account activities were then prepared by Dames & Moore and reviewed by the team. This process only took six months with a minimal expenditure of funds. In conjunction with this effort, Dames & Moore developed a database program in FoxPro for compiling and reporting "should cost" data. Ongoing efforts include the programming necessary to support the development and documentation of new cost estimates utilizing the "should cost" data files as a template, and accommodating site-specific cost and quantity adjustments. ICF Kaiser Engineers is developing "should costs" for the remediation activities which will be reviewed by the team.

It is obvious that the traditional way of preparing cost estimates does not reduce project costs to industry levels. The "should cost" estimating system is a new tool that, when combined with effective project management, can help project managers move forward and away from the past. The challenge for project managers is to utilize this tool and change the existing culture to attain new DOE goals for cost effectiveness.

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THE ROLE AND PROCESS OF INDEPENDENT VERIFICATION OF THE ENVIRONMENTAL RESTORATION OF NORTHWESTERN AREA DEPARTMENT OF ENERGY SITES

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ABSTRACT

This paper describes the Independent Verification program for DOE's Office of Northwestern Area Programs, which has HQ oversight for the environmental restoration activities at the Chicago, Idaho, Oakland, and Richland Operations Offices.

The purpose of the DOE independent verification (IV) program within the Office of Environmental Restoration is to provide an independent assessment of site conditions versus project plans and release criteria prior to project close-out, resulting in a validation of the accuracy and completeness of field measurements and the credibility of the procedures followed.

Both RCRA and CERCLA provide for some degree of verification, usually performed by State or EPA regulators, that planned cleanup levels at environmental restoration sites are achieved. Ultimately, this leads to a certification docket assembled at the completion of the project. However, the processes under RCRA and CERCLA vary from site to site depending on NPL status, state regulations, federal facility agreements and the type of cleanup action.

The need to inform and involve stakeholders in decisions regarding the DOE environmental restoration program has brought a renewed emphasis on the established process of DOE IV. The IV process involves a third party contractor not affiliated with the site who conducts either Type A or Type B verification, depending on the project. Type A verification involves a review of remedial action plans, historical documents, site data, and final surveys in relation to cleanup criteria; Type B verification includes Type A activities supplemented by field measurements, split samples, and other confirmatory activities.

The process is a solid tool for DOE sites to reassure stakeholders that the work to release these sites from DOE control is proceeding according to plans, in a manner which emphasizes the long-term protection of human health and the environment.

INTRODUCTION

The ultimate goal of any environmental restoration cleanup action is to ensure that resulting radiological and (where appropriate) chemical conditions at the site or facility comply with established criteria, standards and/or guidelines and that the public and the environment are thereby protected. For Department of Energy projects

most elements of the completion of a project (i.e. collection of post cleanup data, preparation of documentation, and coordination with state and regional authorities) are implemented by the responsible field organization and the contractor performing the cleanup. As a quality assurance activity, the Department of Energy-Headquarters Office of Northwestern Area Programs conducts independent verification of their environmental restoration activities.

The purpose of independent verification (IV) is to validate the accuracy and completeness of the field measurements and attest to the credibility of the procedures followed during the cleanup operations. While the data collected during independent verification may supplement the data of the cleanup contractor, it is not considered a substitute for complete data from the cleanup contractor. The IV program managed by the Office of Northwestern Area Programs is intended to convey to the public and regulators that the cleanup of these sites and facilities have met established criteria.

NORTHWEST AREA PROGRAMS INDEPENDENT VERIFICATION TASK GROUP

On May 6, 1992, the DOE Office of Environmental Restoration established Independent Verification as a requirement for all Environmental Restoration cleanup projects. In July 1992, the Office of Northwestern Area Programs issued a policy statement containing implementation guidance for IV.

In order to implement IV consistently and efficiently throughout the Office of Northwestern Area Programs it became apparent that additional guidance with input from the Operations Offices was necessary. A task group was formed to develop this guidance and provide recommendations on improving the IV process. The task group is composed of representatives from the Office of Northwestern Area Programs and the Operations Offices: Chicago, Idaho, Oakland, and Richland. Initially the task group determined which types of projects needed IV. A key consideration in this determination was whether projects had sufficient regulatory oversight to render an independent evaluation of the cleanup unnecessary. In these cases the regulator's verification activities fulfilled the need to assure public acceptance of the project.

It was determined during the first meeting of the task group that to function properly, the Independent Verification Contractors (IVCs) who perform the verification activities needed to be included and consulted. It was also determined that maintaining the task group and meeting semi-annually to discuss issues related to the IV program would be beneficial for everyone involved.

The initial findings of the task group are summarized below;

The IV program should be managed from DOE Headquarters (HQ). The task group recommended this approach as the most efficient way to manage the program while allowing the Operations Offices to use it to its fullest extent. The Operations Office representatives on the task group believed that IV should be managed by HQ in order to convey to the public and regulators that DOE is providing independent confirmation that remediated sites and facilities meet the cleanup criteria. The IVCs believe that the management of IV from HQ would assure greater consistency between the different Operations Offices.

Determination of a project's candidacy for IV should be based on 1) the type of cleanup action; 2) future use of the property; and 3) the degree of verification being undertaken by regulators.

The draft procedures should include or address "all hazardous wastes as defined under RCRA and CERCLA that are included in the project plan."

Scoping visits by the IVC and HQ to the site and Operations Office should become a regular step in the IV process. A scoping visit at the beginning of the process provides an opportunity for the personnel involved in the project (the IVC, the Office of Northwestern Area Programs, the Operations Office, and the cleanup contractor) to discuss the project and one another's expectations in the IV process. From this develops a better understanding of the roles and responsibilities of the IVC, cleanup contractor, Operations Office and HQ, contributing to an efficient and complete IV process.

The draft procedures for performing IV should be reviewed, updated, and issued as formal procedures.

Depending on an IVC's level of involvement at a site, the IVC's participation in mid-year and year-end meetings should be considered.

DETERMINATION OF APPLICABILITY OF INDEPENDENT VERIFICATION

The determination of whether a cleanup action should be subject to IV by the Office

of Northwestern Area Programs is based on 1) the type of cleanup action; 2) future use of the property; and 3) the degree of verification being undertaken by the regulators. However, the Operations Office may request IV of any cleanup action, whether or not it is required, if necessary (e.g., to alleviate public concerns). Also, the Operations Office may recommend that IV is not necessary if, for example, there is already a high level of regulator verification activities in a particular project.

To assist both HQ and the Operations Offices the task group produced a matrix for determining which Environmental Restoration projects needed an IV performed (see Attachment 1). One of the more controversial areas of this matrix regards CERCLA and RCRA cleanup actions. The task group concluded that the Northwestern Area Programs should not conduct IV of these actions.

The task group gave the selection of activities that need an IV considerable thought. In order to provide input into the decision, a background paper was written on the verification activities that EPA region and State regulators in the Northwestern Area Programs perform, and sample Federal Register notices stating that no further action is needed were obtained and reviewed. Based on an evaluation of this information and the experience the task group members have had in implementing IV programs and managing environmental restoration projects, the task group concluded that to require the Northwestern Area Programs to perform an IV at either CERCLA sites on the National Priorities List (NPL) or RCRA sites would be a duplication of verification activities performed by regulators.

CERCLA Verification

Under CERCLA, DOE is appointed the lead agency for cleanup activities at DOE owned or operated facilities. DOE is therefore responsible for planning and implementation of removal or remedial actions. This responsibility is subject to EPA and State oversight. In the National Contingency Plan (NCP), authority to enter facilities to conduct investigations of CERCLA action is given to EPA, States, and OSHA (40CFR 300). Beyond this authority, there are other instances throughout the CERCLA process where EPA or the State are indirectly involved in oversight and decisions regarding cleanup actions. EPA and State involvement varies on a site-specific basis, depending on the type of action.

The Close-Out reports constitute one form of third party verification under CERCLA. Their purpose is to clearly demonstrate that activities at the site have been sufficient to meet the completion requirements specified by the Record of Decision (ROD). Close-out reports are prepared by the EPA Regional Administrator and must be approved by both the State and EPA Headquarters. Approval of a final close-out report indicates that the cleanup has achieved levels specified in the ROD, and that the site may be placed in either the "Construction Completion" category on the National Priority List (NPL) or the "no further action needed" category. The Close-Out report correlates to the IVC's final report, as both reports indicate that the cleanup has met the goals stated in the planning stages of the project.

RCRA Verification

For RCRA cleanup actions, it is in the course of a corrective action that environmental restoration activities are most likely to be subject to verification. Although neither EPA nor other party verification of corrective actions is required by law, EPA does administer oversight specific to corrective actions. Facilities are required to demonstrate the effectiveness of corrective actions by establishing a groundwater monitoring program in conjunction with the corrective action. This program must also be supplemented by semi-annual written reports submitted to EPA Regional Administrators. These reports are intended to inform EPA on the effectiveness of corrective actions in progress and can be interpreted as a type of verification activity. Further, RCRA Section 3007 (c) requires the EPA to undertake annual inspections at federal facilities for the purposes of enforcing compliance with RCRA Subtitle C.

The descriptions above illustrate the breadth of compliance and verification activities required under CERCLA and RCRA. For this reason, the task group concluded that the IV of CERCLA and RCRA cleanups would be a duplication of effort.

RESPONSIBILITIES OF PARTICIPANTS IN THE IV PROCESS

Their are four major participants involved in the IV process:

- The cleanup contractor
- The DOE Operations Office
- The DOE-HQ Office of Northwestern Area Programs

The Independent Verification Contractor.

The success and efficiency of the IV of any project is dependent upon the participants knowing what their responsibilities are, and performing their responsibilities in an efficient manner.

The cleanup contractor is responsible for documenting the technical aspects of the project and making them available to the IVC in a timely manner. Whether the cleanup contractor communicates directly with the IVC or provides the information to the Operations Office for transfer to the IVC is determined during the scoping visit. Initially, this information is typically in the form of reports, such as Annual Environmental Monitoring Reports, Characterization Plans and Reports, Decommissioning Plans, and Project Plans.

As the project proceeds through the Operations Phase the information is related more to status and scheduling. The cleanup contractor needs to inform the IVC when the site is ready for verification of interim areas (i.e. rooms, ditches, temporary excavations) and the final survey.

The DOE Operations Office is responsible for coordinating the flow of information on the project between the cleanup contractor and the IVC. Initially, the IVC will rely on the Operations Office to provide the status of the cleanup and background information on the project. The Operations Office is also responsible for evaluating proposed release criteria for soil and/or supplemental limits, and then forwarding these proposals to the DOE-HQ for approval. The Operations Office also must coordinate both the assembly of the Post Remedial Action Report and the closeout of the project.

As stated earlier, the Operations Office may request IV of any cleanup action, whether or not it is required, should the Operations Office believe such verification is necessary (e.g., to alleviate public or regulatory concerns). Also, the Operations Office may determine that IV is not necessary for a particular project. An example would be if the IV would duplicate the regulators' verification efforts.

The management of the IV process by the Office of Northwestern Area Programs provides an additional degree of independence between the Operations Office managing the cleanup project and the IVC, providing assurance to HQ, the Operations Office, the general public and regulators that the resulting radiological and chemical conditions meet the site cleanup goals.

The Office of Northwestern Area Programs is responsible for coordinating the IV Task Group. This group meets every 6 months to review the status and lessons learned from the implementation of IV, and to review and update the status of the projects that are undergoing IV. This group provides participant input into the IV process which is utilized by the Office of Northwestern Area Programs as a basis for decisions.

The HQ responsibilities for cleanup project activity begins with the selection of the IVC for the project. This selection is made with the intent to equally distribute the projects between the IVCs and to establish and maintain current relationships that the IVCs, Operations Offices, and cleanup contractors have established.

The next major responsibility for HQ is the determination of the type of verification needed for the project. HQ relies heavily on the recommendation of the IVC, but is responsible for making the determination (with input from the Operations Office as needed). During the operations phase of the cleanup project HQ monitors the progress of the cleanup and the IV, and assures that the cleanup contractor and the IVC maintain communication about the status of their portions of the project. This monitoring is done through monthly reports, as well as informal communications with the IVC and Operations Office.

When the IVC is preparing to perform a field survey, the Office of Northwestern Area Programs reviews and approves the IVC's survey plan. The review of this plan usually concentrates on whether the scope of the survey plan is sufficient to verify the cleanup, and whether the release criteria that the IVC will be verifying are the release criteria that was stated in the Project Plan. If there are discrepancies between the surveys of the IVC and the cleanup contractor, HQ coordinates their resolution. All discrepancies should be resolved by the time the IVC final report is issued. Discrepancies between the IVC and cleanup contractor, the scope of work for the IV project, and the release criteria are the major items HQ looks for in their review of the final report.

Fig. 1.

The IVC serves as an independent technical expert to the Office of Northwestern Area Programs who confirms the accuracy and completeness of cleanup actions. The responsibilities of the IVC can be segmented into the following phases: Document review and planning, Field work, and Statements and reports.

During the document review and planning phase the IVC is responsible for recommending the type of verification needed for the project and if the cleanup criteria is stated clearly enough to support verification. In the IVC's role as a technical expert, if inconsistencies are noted, or if more efficient means to accomplish the surveys or analysis exist, they should bring this to the attention of the Operations Office or cleanup contractor.

The field work portion of the project consists of performing the on-site surveys of the site. It is often carried out in conjunction with the remedial action contractor in order to limit the amount of time needed before restoration, especially if there is a large turn-around time for laboratory analysis. Background and baseline data must also be gathered at this time for purposes of comparison. Field work is often the most time-consuming part of the IV, as factors such as weather and changes in site conditions can greatly extend the schedule, and large areas require a great deal of radiation scanning and sampling.

An integral part of the IV process is the documentation. The IVC is responsible for producing a report documenting their verification of the project. As soon as the laboratory results are received on the IVC's field work, the IVC should send a letter report to HQ containing the results of the survey, identifying any discrepancies with the cleanup contractors data, and including a statement of whether the IVC's data will verify the cleanup contractor's findings. The IVC should then assemble a final report consisting of field and laboratory analysis results, any discrepancies, confirmation of site compliance with the projects guidelines, and corrective actions for discrepancies.

In addition to the final report the documentation phase includes the archival of representative samples from the remedial action and verification surveys. The samples will be maintained by the IVC over a five-year period, to be held as evidence of the adequacy of the cleanup project and to confirm its close-out. The majority of the samples are to be derived from the IVC field work; however, the samples of the cleanup and radiological contractor may be incorporated as appropriate. Upon obtaining these samples the IVC becomes responsible for their maintenance.

CONCLUSIONS

The Independent Verification process has been found to be a solid tool for DOE to reassure stakeholders that the work to release sites from the Office of Environmental Restoration's control has occurred according to plans, in a manner which emphasizes the long-term protection of human health and the environment. The formation of an IV Task Group has improved the IV process in the Office of Northwestern Area Programs by allowing IVCs, HQ, and Operations Offices opportunities to discuss projects and problems in a non-adversial setting.

Two important lessons learned from the task group have been:

The IV process is greatly improved when the roles and responsibilities of participants are identified and discussed early in the process at a scoping visit.

The earlier the IVC is involved in the project the better. Early involvement lessens the impact of the IVC's comments on the schedule, and allows DOE and cleanup contractors to take advantage of the IVC's expertise in a consulting role as well as a verification role.

Attachment 1

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THE NATIONAL TECHNOLOGY INFORMATION EXCHANGE WORKSHOPS: A TECHNOLOGY TRANSFER TOOL*

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ABSTRACT
The U.S. Department of Energy (DOE) National Technology Information Exchange (TIE) Workshops are sponsored by DOE Headquarter's Office of Environmental Restoration, Office of Program Integration. The goal of these workshops is to provide a quality forum that promotes the exchange of environmental restoration- and technology-related information across the DOE complex, including information needed to transfer technologies between sites and into DOE from outside sources. A significant percentage of technologies, necessary for conducting restoration activities, already exists. TIE Workshops offer an excellent opportunity and arena for sharing implementation strategies, experiences, and the results of these technologies with those individuals and programs who have a need for this information. Significant savings of time and money have already been realized by the transfer of information, knowledge, and technological solutions among the DOE field offices and sites.
INTRODUCTION
The DOE Office of Technology Development's 1991 midyear review identified a need for improved communications among DOE sites and laboratories regarding availability of both new and existing technologies capable of meeting the needs of environmental restoration programs. The DOE Office of Environmental Restoration initiated the TIE Workshops in response to this recognized need.
TIE Workshops promote teamwork and communication between environmental remediators and technology developers, with the focus on sharing actual field experiences, remediation or demonstration results, and lessons learned (1). The workshops also provide an excellent forum for sharing program needs, project implementation strategies, and information regarding existing and near-term technologies capable of providing solutions to problems. This sharing is accomplished by bringing those who actually do the work together, to interact with one another and to share ideas. This interaction provides an improved planning and communications network, that results in saving both money and time. TIE Workshops offer DOE site representatives an opportunity to learn and share information about solutions to common problems.
OE has also identified the need to recognize fully developed, usable technologies in the private sector and other Federal agencies and to place greater emphasis on transferring these technologies into DOE programs. Subsequent TIE Workshops devoted concurrent sessions on technology transfer barriers and potential solutions to mitigate those barriers.
TECHNOLOGY TRANSFER
TIE Workshops have demonstrated notable examples of technology transfer, with real savings in both cost and time. The following technology transfer examples demonstrate the value of these opportunities to share information between peers and sites.
During the Third National TIE Workshop (hosted by Lawrence Livermore National Laboratory in Pleasanton, California, in November of 1992), a Los Alamos National Laboratory (LANL) participant heard about the "rapid geophysical surveyor," used to locate waste burial pits and shafts. At the time, LANL project personnel were planning to use the "conventional" method, magnetometer surveys, that would have taken up to 2 months to complete the data gathering phase over approximately a 1-hectare area. An engineer from the Idaho National Engineering Laboratory demonstrated the rapid geophysical surveyor technology and capabilities at the LANL site. LANL personnel used the rapid geophysical surveyor to complete the 1-hectare survey to locate the waste burial pits and shafts (even in bad weather) in only 4 days. This survey produced superior data and resulted in enormous cost and time savings for the LANL project.
Rocky Flats Environmental Technology Site (RFETS), under an Interagency Agreement with the Colorado Department of Public Health and Environment and the U.S. Environmental Protection Agency, was required to close and remediate four solar evaporation ponds (Operable Unit 4) in accordance with an approved Interim Measures/Interim Remedial Action Program. An engineered cover capable of protecting human health and the environment for a 1,000-year period was required.

Participants at the Fifth National TIE Workshop, held in Denver, Colorado, and hosted by RFETS, heard about research and design efforts to develop an engineered surface-barrier cover that had been ongoing at the Hanford Site for a number of years.

As a result of this opportunity to exchange information and gain an understanding of each others needs and capabilities, RFETS scheduled a series of meetings with the Hanford Site personnel responsible for the design of the surface-barrier cover. These results enabled RFETS to use the research and design work, which had been completed at the Hanford Site, and apply it to their specific needs for a conceptual design of an engineered cover. This cooperation and information exchange resulted in saving both time and money.

At the Sixth National TIE Workshop (held in Kennewick, Washington, and hosted by the DOE Richland Operations Office in May of 1994), it was reported that the RFETS projected savings of $25 million by using the existing surface-barrier design being developed at the Hanford Site. In addition, significant intangible benefits in the form of time savings (estimated at 5-to-8 years), regulator satisfaction and response, and DOE credibility were gained from the experience.

Individuals from the Idaho National Engineering Laboratory and Sandia National Laboratories proclaimed that the DOE Idaho and Albuquerque Operations Offices may have a similar need for use of surface barriers in the future. If the need arises, TIE participants have an institutional memory of technology applications for various environmental restoration problems. There is always the potential to use information from past TIE Workshops and the TIE Quarterly publication to help sites that may need to address environmental restoration problems at a future date. The TIE workshops are an investment for future needs.

In addition to learning about existing technologies, TIE participants can investigate new technologies developed by industry and discuss strategies for working on common issues. One participant estimated savings of nearly $1.5 million by implementing new techniques (learned at the workshop) to reduce the number of wells at a Southwestern Area Program site.

The benefits of attending the TIE Workshops are many and difficult to quantify. However, numerous participants have gained valuable contacts and information from these workshops. Some additional examples of time and money savings identified from participants who have attended the TIE workshops are

Decontamination and decommissioning activity performed at one site can be performed at others by reusing equipment.

Expedited Exploratory Studies savings at one site will be approximately $250,000.

The Hanford Site saved $10,000 in hotel costs by combining efforts for Environmental Impact Statement preparation and attendance at the TIE Workshop.

Savannah River Site's specifications for decontamination and decommissioning efforts have been shared with others for an estimated savings of 320 work hours and $20,000.

The use of a database developed by HAZWRAP (Oak Ridge, Tennessee) resulted in a developmental cost savings of $150,000.

Long-term savings are anticipated when implementing better technologies.

The TIE Quarterly, a publication that reports on the TIE Workshops, allows for the continuation of information exchange. The TIE Quarterly also reports on current ongoing environmental restoration and technology development activities at various DOE field and site offices. The TIE Quarterly is a tool used to share success, failures, and the introduction of existing and new technologies to help improve communication and to transfer technological solutions throughout the environmental restoration communities of DOE and their contractors.

The World Wide Web (WWW) Internet link provides a means of sharing TIE program information and the use of environmental management technologies within the DOE complex. The WWW TIE web server contains several useful features, including current and past TIE workshop activities; technology contacts made within the DOE complex; and technology updates from DOE Headquarters, Operations Offices, laboratories, and the private sector. The TIE web server is accessible through Mosaic Software. The uniform resource locator is http://www.em.doe.gov.

The TIE Workshops and the TIE Quarterly have provided forums for people to communicate effectively concerning the use of technologies. The Technology Connection (TechCon) Program has used both tools as effective ways to bring problem holders and technology suppliers together in a meaningful way. The TechCon Program

is a DOE environmental restoration effort to identify and support the implementation of existing, usable technologies that have the potential to provide superior performance in solving environmental problems today. The TechCon Program has identified nearly 1,000 sources of technology from domestic and international firms and focuses principally on overcoming obstacles. The TIE Workshops and the TIE Quarterly have provided effective pathways for communications.
Face-to-face interactions, such as those provided through the TIE Workshops, have greatly improved the awareness of resources across the DOE system. Realizing the value of improved communications, an informal group calling itself the EM-40 Technology Network was formed in early 1993. They meet regularly at each TIE Workshop and share insights, challenges, and wisdom. The EM-40 Technology Network would prefer to meet more than once a year.

CONCLUSION

These examples demonstrate the usefulness and need to make available an opportunity to freely and openly exchange knowledge and information so DOE sites can maximize the use of research, technologies, and lessons learned from other DOE facilities. By attending TIE Workshops, participants have transferred technologies and technological information that have saved both time and money. TIE allows participants to build on the experience of others and to network with peers. The goal of the TIE Workshop is to identify and implement the most cost-effective and appropriate technology available to address DOE environmental restoration problems today. On the basis of participant feedback, the TIE Workshops achieve the goal of helping DOE environmental restoration personnel identify and implement the best available, cost-effective, and applicable technologies.

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GROUNDWATER MODELING TO SUPPORT HUMAN HEALTH RISK ASSESSMENT AND FEASIBILITY STUDY AT ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE, COLORADO

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ABSTRACT

Operable Unit No. 2 (OU2) of the Rocky Flats Environmental Technology Site (RFETS), formerly known as the Rocky Flats Nuclear Weapons Plant (RFP), located west of Denver, Colorado, has been the site of extensive environmental investigations related to contamination from past radioactive and chemical waste storage and disposal practices. As part of the OU2 Phase II RCRA Facility Investigation/Remedial Investigation (RFI/RI) process, numerical groundwater flow and contaminant fate and transport modeling using the MODFLOW, MT3D, and ONED3 codes was performed in a simplified manner to support the data needs of the OU2 Human Health Risk Assessment (HHRA). A more complicated three-dimensional flow model using MODFLOW is also being developed to support the OU2 Corrective Measures Study/Feasibility Study (CMS/FS). Characterization of OU2 physical conditions indicates that complex subsurface geologic conditions dominate the hydrogeology and contaminant distribution and transport within OU2. Due to the complex top-of-bedrock surface topography, and the influence of local precipitation as the dominant groundwater recharge mechanism, groundwater elevations and the areal extent of the saturated zone are highly variable during the year, influencing both the extent and migration of contamination. In addition, groundwater flow and contaminant distribution and movement are strongly influenced by interaction between the saturated alluvium and underlying bedrock.
Numerical modeling of the complex hydrogeologic and contaminant conditions in OU2 in a detailed manner to simultaneously meet both the HHRA and CMS/FS needs, although feasible, is costly and time consuming. In order to meet mandatory regulatory

schedules for completion of the HHRA and CMS/FS, separate modeling approaches were developed that addressed the specific data needs of each study. To support the HHRA, it is adequate to limit the degree of model complexity to that necessary to provide conservative estimates of contaminant loading to local creeks, thus resulting in conservative estimates of human health risk. To accomplish this, the complex, multi-layer, highly transient site conditions were simplified to an equivalent, single-layer, steady-state system that incorporated the large-scale behavior of the integrated OU2 system. To support the CMS/FS, it is necessary to preserve the geologic and temporal complexity of the flow system in order to simulate realistic flow system behavior in response to various remedial action alternatives. To accomplish this, a three-dimensional, multi-layer, transient model is being developed to simulate the detailed flow conditions within OU2.

PHYSICAL SETTING

RFETS is a government-owned and contractor-operated facility that was part of the nationwide nuclear weapons complex. The primary mission of the facility from its inception until the late 1980s was to produce metal components for nuclear weapons, including components fabricated from plutonium, uranium, beryllium, and steel. Both radioactive and chemical wastes were generated at the facility, and past waste handling procedures involved both on-site storage and disposal of wastes. In 1992, the primary mission of the facility changed from weapons production to environmental restoration and stabilization of the special nuclear material remaining on site. RFETS is located on a relatively flat, gently eastward-sloping alluvial fan deposit known as the Rocky Flats Alluvium, adjacent to and just east of the Front Range of the Rocky Mountains. RFETS is located in a sparsely populated area about 16 miles northwest of Denver, at an elevation of approximately 6,000 feet above mean sea level. The RFETS area has a semi-arid climate, with an annual precipitation of about 15 inches, most of which falls in the spring. Temperatures are moderate, but the area experiences very strong wind events.

OU2 is located in the eastern buffer zone portion of RFETS, outside the main industrial area. This buffer zone consists of open land in a relatively undeveloped state. Physiographically, OU2 consists of a relatively flat pediment or ridge between two creeks; South Walnut Creek to the north, and Woman Creek to the south (Fig. 1). OU2 includes three major contamination areas. These areas, known as Individual Hazardous Substance Sites Areas (IHSS Areas), include the 903 Pad, Mound, and East Trenches Areas. The 903 Pad IHSS Area includes the 903 Pad, an area previously used to store drums containing spent solvents laced with radioactive metal lathe shavings, and waste burial areas. Leakage of the drums and wastes resulted in soil and groundwater contamination with volatile organic compounds (VOCs) and radionuclides. The Mound Area was also previously used to store drums containing solvents and oils, some of which contained radioactive materials. Leakage of those drums also resulted in soil and groundwater contamination. The East Trenches Area actually consists of two separate areas, the Northeast Trenches and Southeast Trenches, where crushed drums and other waste were buried in trenches. Leakage of residual material in the drums and waste has resulted in subsurface soil and groundwater contamination, primarily in the Northeast Trenches Area.

The hydrostratigraphy of OU2 is complex. Figure 2 is a schematic cross-sectional representation of the subsurface conditions at OU2. The hydrogeologic system at OU2 is comprised of the Upper Hydrostratigraphic Unit (UHSU) and Lower Hydrostratigraphic Unit (LHSU). The UHSU, which is the focus of the modeling studies, is comprised of the Rocky Flats Alluvium and Arapahoe Formation No. 1 Sandstone (No. 1 Sandstone) beneath the OU2 pediment, and the colluvium/terrace deposits on the hillsides of the South Walnut and Woman Creek drainages. The LHSU is comprised of claystones, siltstones, and thin, laterally discontinuous clayey, silty sandstones, and does not appear to be substantially impacted by previous facility operations.

The UHSU flow system is characterized by flow in several directions, controlled by the geometry and physical relationships of the different geologic units, and several topographic features on the top-of-bedrock surface. The UHSU groundwater system is also highly influenced by areal recharge from local precipitation, which is the dominant recharge mechanism, resulting in substantial seasonal variations in groundwater elevations and the lateral extent of the saturated zones. The two most significant shallow permeable geologic units beneath the pediment of OU2 are the saturated Rocky Flats Alluvium and No. 1 Sandstone bedrock.

The principle feature that controls groundwater flow within the Rocky Flats Alluvium is a northeast-trending paleoscour in the underlying top-of-bedrock surface (Fig. 3). Alluvial groundwater collects within this scour and flows northeast along the scour to a point where the scour intersects the south hillside of the South Walnut Creek drainage. The alluvial groundwater then discharges to a major surface drainage gully seep on the hillside.

The No. 1 Sandstone is a paleostream channel laterally surrounded by claystone. Flow within the No. 1 Sandstone bedrock is controlled by the geometry of the sandstone channel and by the laterally intermittent connection with the overlying Rocky Flats Alluvium, which acts as a source of water to the sandstone. The hydrogeologic interaction between those two units is complex and varies both spatially and with time.

Because the Rocky Flats Alluvium and No. 1 Sandstone are entirely truncated by the South Walnut and Woman Creek drainage valleys, virtually all of the groundwater within those units discharges to seeps along the drainage hillsides. Once discharged at the seeps, the groundwater enters colluvium and terrace deposits that mantle the hillsides of OU2, and flows downslope to the creeks (Fig. 2).

Contamination within the Rocky Flats Alluvium and No. 1 Sandstone consists of VOCs and, to a lesser extent, radionuclides. Contamination plumes in the alluvium occur in the vicinity of the 903 Pad Area and extend northeastward along the paleoscour, and southward toward Woman Creek. Contamination plumes within the No. 1 Sandstone occur east of the 903 Pad and in the Northeast Trenches Areas and are related to downward migration from the overlying alluvium. The groundwater chemicals of concern (COCs) used for the OU2 HHRA consist of PCE, TCE, carbon tetrachloride, chloroform, methylene chloride, 1,1-DCE, plutonium, and americium.

As a result of the contamination, a number of environmental investigations have been conducted at OU2. The most recent work was performed as part of the OU2 Phase II RFI/RI investigation. As part of that investigation and ongoing feasibility study work, groundwater models were previously applied in support of the HHRA, and are currently being applied to support the CMS/FS. The HHRA evaluated potential risks associated with the exposure of hypothetical off-site residents to contaminants in surface water in South Walnut and Woman Creeks where they leave RFETS property on the eastern boundary of the facility. The CMS/FS will evaluate several potential remedial action alternatives designed to mitigate the groundwater contamination conditions within OU2.

MODELING

In selecting the appropriate modeling approaches, five general criteria were specified:

The selected models should be able to incorporate key processes and conditions known to occur at the site.

The selected models should be able to satisfy the objectives of the study.

The selected models should be verified using published equations and solutions.

The selected model codes should be complete and well documented and preferably available in the public domain.

The selected models should be practical and cost-effective in terms of actual application as well as resolution of uncertainty.

Initial Modeling Approach

Initially, two levels of model complexity were considered when scoping the groundwater modeling work: Option 1) a simple analytical model designed to assess contaminant concentrations at HHRA receptor locations; and Option 2) a complex three-dimensional, multi-layer, transient numerical flow and fate and transport model to accurately simulate detailed flow conditions, and contaminant transport to receptor locations. Based on assessment of each option relative to the aforementioned criteria, Option 2, the complex three-dimensional numerical model, was initially selected as the modeling approach because it could incorporate most of the complex site conditions and processes, and could simultaneously support both the HHRA and CMS/FS, originally a key objective of the modeling study.

Option 1, use of a simple analytical fate and transport model, was ruled out because it did not meet the criteria of incorporating most of the key processes or conditions known to occur at the OU2 site. These included multiple contaminant source areas, multiple groundwater and contaminant discharge points, complex saturated zone configurations, heterogeneous geologic and hydraulic properties, and variable groundwater flow directions, hydraulic gradients, and contaminant migration

pathways. Option 1 was also ruled out because, although it could support the HHRA needs, it could not support the CMS/FS.

Reevaluation of Project Needs

Approximately midway through the modeling study, the need to compress the project schedule to meet mandatory regulatory schedule deadlines for completion of the HHRA forced a reassessment of project objectives and modeling needs. During several project meetings, each project task was evaluated relative to its potential to affect the project schedule. Because modeling was a critical path item for meeting the project HHRA deadlines, and the application of the complex model involved some uncertainty with respect to the HHRA schedule (i.e., there was some concern that if the complex model encountered difficulties, it could possibly jeopardize the schedule), it was decided to explore simplification of the modeling approach in order to minimize the potential for schedule impacts.

In order to simplify the modeling approach, it was necessary to adjust the project objectives accordingly. A simplified model could not be expected to reliably simulate the details of a complex flow and contaminant transport system, particularly at the level of detail necessary for supporting the CMS/FS. Therefore, it was decided that the project modeling objectives would be limited to supporting the HHRA only, and that modeling in support of the CMS/FS would be conducted later for detailed evaluation of various potential remedial action alternatives.

Revised HHRA Modeling Approach

Given the new modeling objective of supporting the HHRA only, the modeling approach was reevaluated based on the data needs of the HHRA. The HHRA for OU2 addresses a number of potential exposure scenarios. However, the only exposure scenario requiring groundwater modeling involves the migration of groundwater contamination to South Walnut and Woman Creeks, and the subsequent migration of those contaminants in surface water in the creeks to hypothetical receptors at the eastern facility boundary. Direct ingestion of groundwater from an on-site water supply well did not require modeling because current observed contaminant concentrations were used as worst case conditions. Direct ingestion of groundwater at an off-site water supply well was not simulated because virtually all OU2 contaminated groundwater discharges to surface seeps prior to reaching the facility boundary.

The revised modeling approach was designed to simulate reasonable or conservative contaminant mass loading values to South Walnut and Woman Creeks. A separate surface water model then used the groundwater model results as an input parameter, added the estimated mass loading from surface sources via surface runoff processes, and estimated total contaminant concentrations at the receptor exposure points at the facility boundary. Because the HHRA groundwater modeling results were used only as input to the surface water model, the objectives of the HHRA groundwater modeling were selected to meet the data needs of the surface water model. Those data needs were: 1) an estimate of the total annual average groundwater flow into each creek, and 2) an estimate of the total annual average contaminant mass loading to each creek from groundwater. To meet those needs, the following HHRA groundwater modeling objectives were specified:

Simulate reasonable and/or conservative seepage flow rates at the alluvial and No. 1 Sandstone seeps along the south hillside of the South Walnut Creek drainage and the north hillside of the Woman Creek drainage to estimate total annual average groundwater flow to each creek.

Generate a composite groundwater flow field within the OU2 pediment to serve as input to a fate and transport model.

Simulate transport of UHSU groundwater COCs from source areas within OU2 to seeps along the hillsides of the South Walnut and Woman Creek drainages.

Simulate transport of COCs through colluvium/terrace deposits from seeps to discharge points along South Walnut and Woman Creeks to estimate total annual average contaminant loading from groundwater to each creek.

To meet these objectives, a revised modeling approach was selected that incorporated some aspects of the detailed, complex model while simplifying the hydrogeologic conditions, where appropriate, to avoid some of the potential problems associated with the complex model. During the simplification process, the following were considered:

The simplified model should be consistent with the modeling objectives.

The simplified model should be as representative as possible of the long-term overall behavior of the groundwater system at a level of detail consistent with the

needs of the HHRA. For example, because the simplification involved combination of multiple hydrogeologic units, the model should be representative of the equivalent effects at the creeks from flow and transport through the units separately.

The simplified model should be reasonable or conservative in terms of its input parameters and should simulate equivalent or conservative results relative to those that would have been simulated with more detailed and complex models.

Based on these considerations, simplified conceptual and mathematical models were developed. Wherever simplifications were applied, they were judged to be reasonable equivalent approximations of the large-scale behavior of the actual groundwater system, or were selected to result in conservative estimates of contaminant loading to the creeks, so as not to cause underestimates of human health risk at the exposure points.

Three site-specific groundwater mathematical models were used to simulate groundwater flow and contaminant transport; a single-layer, two-dimensional MODFLOW (1) numerical groundwater flow model to simulate the groundwater flow system in the Rocky Flats Alluvium and No. 1 Sandstone; a single-layer, two-dimensional MT3D (2) numerical fate and transport model to simulate the fate and transport of COCs in the Rocky Flats Alluvium and No. 1 Sandstone; and a one-dimensional ONED3 (3) analytical solute transport model to simulate COC migration in colluvium /terrace deposits on the OU2 hillsides from the seeps to the creeks.

The HHRA model domain (Fig. 4) incorporated those portions of the saturated Rocky Flats Alluvium and No. 1 Sandstone important with respect to contaminant transport by including most of the identified major IHSS Areas that contribute substantial contamination to the groundwater system. Boundary conditions were selected to represent seep conditions where they occur on the hillsides of OU2. Some of the major simplifications incorporated in the models included:

Combination of the Rocky Flats Alluvium and No. 1 Sandstone into a single equivalent composite hydrogeologic unit. This simplification was reasonable based on the direct hydraulic communication between the units (where they are in direct contact) and their similar hydraulic properties, and the fact that distinction between alluvial and sandstone seeps was not necessary for assessing total contaminant loading to the creeks.

Simulation of the system under steady-state conditions rather than the actual highly transient conditions. This simplification was reasonable for simulating the long-term average annual maximum concentrations needed for the risk assessment.

Simplification of contaminant source area behavior as constant steady sources or step sources. This simplification was conservative in that it applied worst case conditions over a longer period than is likely to occur in reality.

Simulation of radionuclide colloidal transport as dissolved phase contaminant transport. This simplification was applied using very conservative (i.e., low) estimates of adsorption coefficient so as not to overestimate retardation of the radionuclides.

Simulating one-dimensional transport of contaminants in the colluvium/terrace deposits. This is a conservative assumption relative to probable actual conditions.

Simplification of the geometry of the hydrogeologic boundaries into a rectangular model domain. In applying this simplification, care was taken to preserve the approximate locations of seep discharge areas so as to accurately distribute groundwater flow and contaminant discharge to each creek.

Model Simulations and Results

MODFLOW groundwater flow simulations were run under steady-state conditions and were calibrated against observed site conditions in terms of water levels and flow directions in the alluvium, and hydraulic gradients in the No. 1 Sandstone, and the relative spatial distribution of seeps and their flow rates. The results showed a high correlation with observed conditions in most cases.

MT3D fate and transport base case simulations were run for each COC for a simulation period of 20 years to simulate current conditions. The results from these simulations were then compared to observed contaminant conditions and it was concluded that the model results were similar or conservative relative to observed conditions, and thus, the model was appropriate for simulating reasonable or conservative future contaminant conditions. Predictive simulations were then run for each COC for simulation periods of 150 years for VOCs and 1,000 years for radionuclides. The results of the simulations were then used as input to the ONED3 analytical colluvium/terrace deposit model to simulate fate and transport of the

contaminants as they migrated to the creeks.
The results of the modeling were plotted as predicted curves of concentration versus time at each creek. The maximum concentration values were then converted to contaminant loading values by multiplying them with the estimated groundwater flow rate to each creek. The results indicated that maximum VOC mass loadings would occur to the creeks within the next 30 to 60 years, and maximum radionuclide mass loadings would occur within the next 300 years for americium, but would not occur within the next 1,000 years for plutonium because of its extremely high retardation factor (R=840).
Of note is that even though the groundwater model simulated substantial VOC concentrations (up to 3,100 mg/l, TCE) being discharged to the creeks within the 150 year simulation period, VOC concentrations in surface water at the exposure points at the facility boundary were low (a few mg/L). This is because the surface water model simulated VOC volatilization, reducing VOC concentrations in the creeks to near or below Maximum Contaminant Levels (MCLs) at the receptor locations at the eastern facility boundary.
Corrective Measures Study/Feasibility Study Modeling
Recently, work has begun on development of the groundwater flow model to support the data needs of the CMS/FS. Because the CMS/FS involves analysis of various potential remedial action alternatives within the contaminated portion of OU2, it requires data on the behavior of the groundwater flow system at a greater level of detail than was required for the HHRA. To address the data needs of the CMS/FS, a detailed three-dimensional, multi-layer, transient flow model is currently being developed. This model will simulate groundwater flow within the Rocky Flats Alluvium and No. 1 Sandstone, as separate hydrogeologic units in hydraulic communication. The model will be used to simulate hydraulic responses to various potential remedial action alternatives including no action, groundwater extraction by well array or interceptor trench, reduced groundwater recharge due to placement of a slurry wall or by limiting infiltration, and enhanced groundwater recharge by injection and spray field combinations. The results of the flow modeling will be used in conjunction with detailed contaminant fate and transport modeling to support selection of effective remedial actions for contaminated groundwater at OU2. The approach and results of the detailed flow modeling will be the subject of a future technical paper.
CONCLUSIONS
The effective and efficient application of groundwater modeling to the OU2 project required that the objectives and needs of the different aspects of the project be fully considered, even when those objectives and needs changed midway through the project. The application of separate groundwater modeling approaches has and will continue to allow mandatory regulatory schedule deadlines to be achieved, while meeting the differing technical needs of the OU2 HHRA and CMS/FS. Preliminary review comments received from the involved regulatory agencies indicate that the simplified modeling approach applied in this study for the purpose of supporting the HHRA will likely be accepted. We expect that the agencies will also accept the use of the more complex model to support the CMS/FS. This demonstrates a willingness to consider the project objectives when assessing whether the degree of model complexity is adequate.
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PROPOSED APPLICATION OF URANIUM SOLUTION MINING TECHNOLOGY TO GROUNDWATER RESTORATION: THE USDOE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT, OHIO
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ABSTRACT

Uranium plumes, associated with the former USDOE uranium-enrichment metal fabrication plant at Fernald Ohio, have contaminated areas of the sole-source Great Miami Aquifer. CERCLA removal actions on the major uranium plume offsite are utilizing a traditional Pump and Treat well network to arrest plume transport. Restoration of the impacted aquifer to the proposed drinking water standard for uranium (20 ug/l) could require decades unless in-situ recovery methods are implemented. Proposed hydraulic "push-pull" with or without injection of benign lixiviants can enhance recovery significantly. Preliminary laboratory testing and numerical modeling suggests that the major uranium plume in the GMA may be remediated in less than 10 years.

INTRODUCTION

The Fernald Environmental Management Project (FEMP) is located 29 km northwest of Cincinnati Ohio and is managed by the U.S. Department of Energy (USDOE). For almost 40 years, nuclear weapon grade materials and assemblies were produced at this 425 h USDOE complex. Total uranium concentrations in groundwater exceed several hundred ug/l off-site. A major uranium plume (Fig. 1; see (1)) was contained in 1993 with the installation of a 5-well network pumping at an average rate of 1.09E+7 l/d. Over 2.0E+8 liters of uranium-contaminated groundwater are contained offsite within the upper 35 m of a buried glaciofluvial valley (1). The unconfined Great Miami Aquifer (GMA) has weak vertical hydraulic gradients thereby promoting flooring of the uranium plume approximately 30 m below the water table. Horizontal hydraulic conductivity calculated from slug testing and pump testing range from approximately 60 to 200 m/d. Modeled advective groundwater velocities range from 0.33 to 0.95 m/d. Uranium partitioning onto the clay and silt fractions of the aquifer matrix appear to retard locally the transport of the uranium plumes up to 23% (2). Generally though the geochemcial conditions in the GMA are oxidizing and promote the mobilization and transport of U+6 (2;3). Streams and man-made conduits assist the aeration of the glaciofluvial aquifer which is largely dominated by carbonate sand and gravel.

The problem is that present Pump and Treat (P&T) induced zones of capture are not sufficient to recover areas of the plume which are associated with sluggish velocities and/or have weak influence on distal parts of their respective zones of influence. These large diameter wells (>4.7 cm) must pump at least 80 lpm to maintain capture of the plume. A solution is to utilize in-situ recovery (ISR) waterflooding and increase the hydraulic gradient; and hence increase velocities toward the well screens. Simulations by (2) suggest desorption rates of uranium that is adsorbed to "dirty" sands would increase (eg. 3.0 - 6.1 min-1 x 10-4). Additionally dissolved oxygen (DO) as a lixiviant could further enhance recovery rates of FEMP uranium akin to uranium solution mining.

Push-pull (PP) technology is the key ISR component and has been applied successfully for decades by enhanced petroleum recovery operations and is now the method of choice by uranium ore recovery operations. Variations of PP are being utilized to effect extraction of VOC's and LNAPL's in many environmental restoration programs. Assuming of course that the hydrogeologic setting is favorable, the use of injection wells for environmental recovery programs is advantageous.

WELLFIELD DESIGN

Basic wellfield patterns on recovery and injection wells for FEMP are patterned after the uranium solution mining industry. A schematic representation of symmetrical 5-spot well networks is illustrated in Fig. 2. Recovery wells are shown in the center of the grids with injection wells at the corners. In this example the reservoir is injection limited. The induced pressure sink collectively promotes flow inward. In this fashion the plume is concentrated and flow is increased toward the respective recovery wells. The 5-spot is one of the most efficient PP patterns with d/a = 0.5.

Factors influencing the choice of PP networks include:

- Distance between injection and recovery well
- Sweep efficiency
- Pore volumes injected

Recovery capacity
Formation anisotropy
Regional groundwater flow
Chemistry and contaminant recovery

Sweep area per well will vary depending on the well pattern and the distance between each injection and recovery well pair. The sweep efficiency (i.e. percentage of area contacted by injected fluid at a given time) will vary with the number of pore volumes injected. The pattern pore volume is calculated by multiplying the pattern area, thickness of the contaminated zone, and porosity. It is important to note that the areal sweep efficiency does not depend on the magnitude of the flow rate as it is calculated at steady-state conditions.

In addition to sweep efficiency and breakthrough time, the well recovery and injectivity will influence well patterns. If the aquifer is extraction limited as appears to be the case at FEMP, more recovery wells will be needed than injection wells. Anisotropy will influence well placement. Modeling streamline trajectories is useful to initially predict efficient well spacing. Pump tests and other estimates of local permeability changes should assist further ISR well placement. This is crucial if lixiviants are to be injected so as to determine their residence time. For a given formation anisotropy each well pattern (eg. 5-spot) will perform differently in response to directional permeability. For example Table I includes calculations of sweep efficiency (%) for 5-spot and line-drive well patterns per Kx/Ky.

These PP factors can be modeled to approximate an initial wellfield design. As operations proceed on recovering FEMP uranium, wells can be relocated and well hydraulics altered. Design of these wells is not outlined here. Suffice it to underscore that these ISR wells proposed for FEMP are PVC Schedule 40 and 15 cm dia. Compare this to P&T wells at FEMP which are 316 stainless steel and 30.5 cm dia. Standard low flow pumps are installed in ISR wells compared to large flow submersibles in traditional P&T wellfields.

HYDRAULIC PUSH-PULL MODELING

Preliminary two-dimensional steady-state flow governing the ISR modeling of FEMP is:

Eq. (1)

with transmissivity solved as (e.g.):

Eq. (2)

From Eq. 2 the characteristic 2-D equation for a pathline then is

Eq. (3)

and the complex velocity potential for the flow path model analysis is:

Eq. (4)

with:

Eq. (5)

Eq. (6)

The preliminary ISR modeling was conducted with GRANFLOW (4), a FORTRAN code based partly on PATH2D and FLOWPATH. The results (Fig. 3) indicate largely extraction-limited and irregular arrays of wellfield patterns which are superimposed on a 20 ug/l uranium isopleth of the major FEMP plume. The isopleth is contoured from a 1993 sampling event over a short time period. It therefore represents a crude "snapshot" of the uranium contamination in the upper GMA. The "snapshot" is part of a larger time-averaged representation (Fig. 1). Figure 1 illustrates the time-averaged plume(s) over 1988-1993 for the entire FEMP site.

Networks of recovery wells are placed near a major losing reach of Paddys Run (Fig. 3) and were simulated as a continuous array of injection points. The irregular nature of the wellfields is due to the factors of anisotropy and uranium distribution coefficients (Kd). The northern part of the plume outlined in Fig. 3 appears to have more local permeability variability but it is uncertain what the distribution of permeability is in the southern environs.

Although the flow was only modeled in two dimension, the vertical component is believed to be weak based on hydraulic gradient data (1). As a first approximation some cells required only 3 pore volumes to be replaced while others required 18 pore volumes in order to reduce the groundwater concentrations of uranium to 20 ug/l. Bleed rates of approximately 4% were permitted during mass balancing calculations in GRANFLOW between model injection and recovery wells. Operations at solution mining operations typically adjust bleed rates to accommodate local field conditions and economic factors. If lixiviants are not used then there are less variables to

control. The greatest uncertainty in the model runs were the variability of infiltration rates from the local streams and man-made conduits into the aquifer. Paddys Run (Fig. 1), often a losing stream across FEMP, was modeled as a continuous injection gallery.

The location of the P&T wells south of the 20 ug/l isopleth in Fig. 3 appears south of the plume but is an artifact of "snapshot" sampling. Actually the time-average location of this isopleth intersects and is captured by these wells. These wells maintain 80 lpm to effect containment of this major plume. Modeled maximum pumping rates from the ISR wells suggest pumping would be less than 39 lpm and more often near 21 lpm.

No significant excursions were predicted but additional monitoring wells are suggested to guard against errant streamlines. The FEMP site has many monitoring wells which are not shown in Fig. 3. In this model run, approximately 7.8 years would be required to restore the major uranium plume in the GMA to 20 ug/l. Of course it assumes that the uranium sources are removed. During operations, improved field parameter data would afford adjustments in order to further improve extraction efficiency in the wellfields.

LIXIVIANT TESTING

The use of lixiviants may not be necessary as the hydraulic component of PP results in significantly less model remediation times than existing P&T model remediation times at FEMP. However additional removals from higher Kd areas might require lixiviants. Alternatively one could simply exchange more pore volumes until the proposed MCL had been achieved.

Dissolved oxygen is benign and is suitable for the natural geochemical conditions in the GMA. Even entrained air which would occur in the delivery system is conducive for maintaining a mobilizing environment for U+6. An empirical relationship between DO and elevated uranium concentrations in the aquifer, particularly where losing streams across FEMP intersect the aquifer, is observed (2).

Basic reactions for uranium in the UO2 -O2 -CO2 -H2O system in the GMA beneath FEMP include:

Eq. (7)

Eq. (8)

Eq. (9)

Preliminary testing of core materials with dissolved oxygen along with aqueous speciation modeling by (2) using PHREEQE, including updated thermodynamic data for uranium species, suggests that UO2(CO3)3-4 in EQ9 is predominant. The basic DO reaction then is:

Eq. (10)

Laboratory simulation of DO as a lixiviant for injection wells was performed on aquifer solids with a Kd = 23.6. with an initial leachate concentration of 24.1 pCi/l. At 3.6 pore volumes, leaching of uranium was near equilibrium with the column of groundwater solution. At 8.9 pore volumes, two-phase flow began. Saturation with 2 phase flow at unsaturated pressure conditions present in the aquifer medium was 73 mg/l at 25oC.

Modeling a delivery system such as froth flow (5) with DO as the oxidant assumes:

- Radial geometry.
- Porosity is constant.
- Permeability varies in depth only.
- Fluid is incompressible
- Fluid occurs as a single phase.
- Mobility and density of injected fluid and formation fluid are equal and constant.
- Mass transfer rates are large enough such that reactions are kinetically controlled.

The governing equation for DO transport from the injection well into the aquifer is:

Eq. (11)

where the initial and boundary conditions for the injection cycle are:

Eq. (12)

Eq. (13)

Eq. (14)

Solving EQ11 through dimensionless analysis yields:

Eq. (15)

Solution of EQ12 yields a net 41% increase in uranium that is desorbed from the aquifer solids into solution. Desorption experiments are ongoing and it remains

uncertain if residual mass of uranium is significant compared to total uranium in solution. Presently, field investigations support the hypothesis that much of the uranium occurs as U+6 and probably is mobilized in the predominant UO2 -O2 -CO2 -H2O system. Injection of lixiviants while modeled to be significant in enhancing uranium recovery, when residual uranium masses are adsorbed, still may not be warranted if most total uranium is presently mobilized. DO and entrained air in delivery systems would have no adverse impact on the natural geochemistry of the GMA beneath FEMP. Losing reaches of streams aerate upper parts of this unconfined aquifer.

CONCLUSION

A ISR recovery wellfield patterned after uranium solution mining operations could be quite effective at FEMP. Preliminary modeling of hydraulic push-pull techniques suggest a very significant reduction in the time necessary to achieve lower uranium concentrations in the GMA. Compared to existing P&T methodology with or without pulsed pumping (Fig. 4) ISR should be more effective. Operationally, ISR wellfields are less costly to install and allow flexibility in manipulating the local hydraulic conditions to effect capture and recovery rates of the uranium contamination.
ISR could reduce the uranium mass in the major plume in less than 10 years and without the use of injected lixiviants. Favorable hydrogeology and geochemistry in the GMA make FEMP an attractive site for conducting ISR recovery. Ideally then ISR would permit a near zero tolerance discharge to the Great Miami River since the waste water treatment plant would recirculate treated waters to the delivery systems in the wellfields.

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NOMENCLATURE

ai stoichiometric coefficient of mass of oxidant I, M/M
Di Damkohler number of species, dimensionless
H plume intercept thickness, L
k thickness of weighted average permeability, L
kl longitudinal dispersion coefficient, L2/T
r radial distance from the test well, L
R hypothetical boundary distance, L
DWi leachable species initially present, M/M
a l longitudinal dispersivity, L
a t transverse dispersivity, L

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FASTER, CHEAPER RADIOANALYTICAL METHODS

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ABSTRACT

Methods for chemical characterization of the environment are being developed under a multitask project for the Analytical Services Division (EM-263) within the U.S. Department of Energy's (DOE) Office of Environmental Management. The project focuses on improvement of radioanalytical methods, with an emphasis on faster and cheaper routine methods. We have developed improved methods for the separation of environmental levels of technetium-99, radium, and actinides from soil and water; separation of actinides from soil and water matrix interferences; and isolation of strontium. Among the novel techniques for separation that are being used are element- and class-specific resins and membranes. We have also developed methods for the simultaneous detection of multiple isotopes by using inductively coupled plasma-mass spectrometry (ICP-MS). Because of its mass-selective mode of detection, ICP-MS requires less rigorous chemical separations than does radiochemical analysis. In addition, improvements in the limits of detection, the volume of the sample and the time of analysis were obtained by using other techniques of sample introduction, such as ultrasonic nebulization, automated batch separation and electrothermal vaporization. Investigation of integration and automation of the separation methods with the ICP-MS method using flow injection analysis is underway, with an objective of automating methods to achieve more reproducible results, to reduce the cost of labor, to cut the time of analysis, and to minimize the generation of secondary waste through miniaturization of the process. A solid scintillation device has been developed and tested for measurement and identification of low-energy beta-emitting radionuclides; application of this technique will eliminate the generation of mixed-waste counting cocktails. The final product of all activities will be methods that are available (published in the literature and in DOE's compendium of analytical methods) and are acceptable for use in regulatory situations. Concerted efforts are being made to facilitate implementation of the methods at DOE sites to realize the improvements offered.

INTRODUCTION

Our nation faces a daunting challenge to clean up and protect our environment; but before launching any cleanup, one must characterize the type, concentration, and extent of the contamination. During cleanup, one must monitor the progress; and after cleanup, one must often monitor the site to ensure that the cleanup was successful. Techniques and methods for chemical characterization and monitoring are essential in the execution of these efforts. Research- and-development efforts to improve capabilities can translate into major savings and improvements in environmental cleanups by reducing the unit cost of measurements (e.g., fewer steps in an analysis), reducing the time required to provide the information to the user (e.g., field analysis), or improving the quality of information (e.g., chemical speciation). Radiochemical analysis is of particular concern to the DOE, which expends over $300 million annually on this activity. Unlike organic and inorganic analytes, which are widely found as contaminants, comparatively little effort has been expended on improvements of radiochemical analyses. Desirable characteristics of any new method for characterization of the DOE's radiochemical contamination are the following:

- Faster, to reduce turnaround times
- Cheaper, to reduce the burden on taxpayers
- Better performance, to achieve the desired objectives for data quality
- Less generation of secondary mixed waste through reduction of scale and elimination of steps
- Closer to real-time analysis
- Quicker characterization of sites.

We have developed improved methods for the separation of environmental levels of technetium-99, radium, and actinides from soil and water; for separation of actinides from soil and water matrix interferences and for isolation of strontium. We are also developing methods for the simultaneous detection of multiple isotopes (including nonradionuclides) by using a new instrumental technique, inductively coupled plasma-mass spectrometry (ICP-MS). These accomplishments are discussed in the following sections. Additional research being conducted to improve other aspects of environmental radiochemical analysis is discussed in the section "Ongoing Activities."

RAPID DETERMINATION OF TECHNETIUM-99 IN SOIL AND WATER

Technetium-99 ($t_{1/2}$ = 2.1 x 105 yr) is a pure beta-particle emitter that is formed as a fission product from 235U and 239Pu. Technetium-99 is as abundantly produced as

137Cs and 90Sr. The main sources of 99Tc released to the environment are from the processing of nuclear fuel and the testing of nuclear weapons. Minor contributions come from radiopharmaceuticals and leachates from radioactive waste sites.
In an oxidizing environment, the 99Tc exists predominantly in the heptavalent state as the pertechnetate ion, TcO_4^-, which is soluble in water. Because this solubility tends to make the ion conservative in surface waters and mobile in aquifers, an increasing number of measurements are made each year to monitor technetium in the environment near radioactive waste sites and nuclear processing facilities. Current analytical methods are time consuming and generate a significant amount of mixed waste.
We have developed and validated a rapid, accurate procedure to collect and assay 99Tc in aqueous samples (1). All materials used in the procedure are commercially available. A standard volume (1 L) of sample water or prepared soil extract is passed through two sequential anion exchange membranes (3M Empore Anion Exchange - SR), which extract the pertechnetate anion selectively. Both membranes are then counted directly in a beta counter, and the total original concentration is calculated. Recovery of technetium via the membranes is usually on the order of 99%; but certain interferences, especially nitrate, may reduce recovery to as low as 30%. The preferred way to minimize this interference is to avoid the addition of nitrate to collected water samples. The primary advantages of the method are a short turnaround time (2.7 h) per sample and an average analysis time of 30 min (eight aqueous samples over a 240-min period).
Calculations
The efficiency of the tandem membrane setup was calculated as
$$E = 1 - B/A \qquad (1)$$
where E is the collection efficiency of each membrane, A is the number of beta counts per minute on the first membrane, and B is the number of beta counts per minute on the second membrane. The calculation assumes an equilibrium process and that the membranes are uniform in their functionality. The concentration (Ao) of analyte is calculated as
$$Ao = A/(E \times \text{volume} \times \text{counter efficiency}) \qquad (2)$$
where A and E are as defined previously and where counter efficiency is measured as counts per minute divided by disintegrations per minute (cpm/dpm).
Table I compares our method for analysis of 99Tc with other methods.
RAPID DETERMINATION OF RADIUM ISOTOPES BY ALPHA-SPECTROMETRY
Because radium and its daughters are radioactive and occur naturally in environmental matrices like drinking water, the determination of long-lived alpha emitters such as 226Ra ($t_{1/2}$ = 1600 yr) is very important from the standpoint of geochemical studies and human health. The most frequently used procedure for determining radium is the radon emanation procedure (HASL-300, Ra-03) (2). This technique involves the collection and measurement of 222Rn, a daughter of 226Ra. Large sample volumes, a multiweek waiting period for the ingrowth of the radon, and the absence of internal quality control detract from the utility of this method. Other methods in use involve coprecipitation with barium sulfate followed by counting with a gamma-ray spectrometer, application of thermal ionization mass spectrometry, or electrodeposition followed by alpha spectroscopy. The electrodeposition methods are direct, highly sensitive, and highly specific in comparison with other established procedures. In addition, the electrodeposition techniques decrease the sample size and reduce the turnaround time of the analysis to approximately 24 hr, including counting time.
We have developed an improved method for the determination of low levels of 226Ra and 224Ra in environmental samples by alpha spectrometry (4). A cation-exchange column (1 g. dry resin) is used to separate the analyte from other constituents in the sample (1-50 mL). Optimal separation of the radium isotopes from the matrix is achieved on a cation-exchange column when 100 mL of 1.5 M HCl is used for washing and 45 mL of 6 M HCl is used to elute the sample. After preconcentration and separation, the radium is electrodeposited onto a stainless steel disk from a solution of ammonium oxalate and hydrochloric acid. Electrodeposition conditions are optimal when 400 g of platinum are added, when ammonium oxalate and hydrochloric acid are used as the electrolyte solution, and when an electrodeposition current of 600 mA is used. Radium-224 is used as a yield tracer, through standard addition, to allow assessment of the quality of an individual analysis. Elapsed time for sample analysis is 24 hr, including both analysis and counting times.

Linear responses are greater than two orders of magnitude. The detection limits of the procedure, taken as three times the standard deviation of several reagent blank analyses, are (1.8 + 0.3) x 10-4 Bq and (2.9 + 0.3) x 10-4 Bq for 226Ra and 224Ra, respectively. Recoveries of 226Ra and 224Ra ranged from 90% to 100% when samples of drinking water, well water, and dissolved bones were analyzed. Precision was calculated to be less than 5% for the determination of 226Ra, with a 95% confidence. Matrix effects were studied for the salts of barium, magnesium, iron, and calcium. Future work will focus on preconcentration processes to analyze samples with very low levels of contaminants (e.g., surface water) and on the determination of 228Ra by using beta spectroscopy with the same electrodeposition disk that was used for the 226Ra determination.

DETERMINATION OF ACTINIDES IN SOIL

Techniques such as ion exchange, liquid-liquid extraction, and precipitation have been described for the separation and preconcentration of the actinides, however, these traditional methods are time consuming and generate large quantities of mixed wastes. Newer extraction chromatographic materials such as certain commercially available resins (TruSpec and TevaSpec) and a commercial ion-exchange material (Diphonix) are highly suitable for the concentration and separation of the actinide elements from soil matrices. These materials are highly selective and are the basis for efficient analytical procedures that generate significantly less waste than the more traditional procedures. The goal of our work (5,6) was to develop improved separation methodologies to reduce the manpower and waste costs associated with the laboratory analyses of actinides in soils. The analytical scheme was designed to satisfy the requirements of both radiometric and nonradiometric detection methods. The ICP-MS method requires only a group separation of the transuranics, whereas alpha spectrometry requires the sequential isolation of the actinides. In addition, the schemes for separation were designed to allow analysis of actinides in soils, whether acid leached or totally dissolved through fusion.

Acid Digestion Procedure

TruSpec SPS is used to perform group separations of the actinides, and TevaSpec is used to isolate the trivalent actinides from the lanthanide elements. Thorium and plutonium are individually isolated on a prepacked anion column (Bio-Rad AG 1-X8). Soil samples are dried, homogenized, ashed, repetitively leached with 6 M HCl followed by 8 M HNO3, and taken up in 2 M HNO3-0.5 M Al(NO3)3 after evaporative elimination of the HCl. Ascorbic acid is used to reduce trivalent iron, which interferes with the uptake of americium by TruSpec. The trivalent actinides, including americium, and the lanthanides are eluted from a TruSpec column with 4 M HCl. Plutonium and thorium are removed with 0.1 M tetrahydrofuran-2,3,4,5-tetracarboxylic acid (THFTCA). Finally, uranium is eluted with the addition of 0.1 M ammonium bioxalate. The trivalent actinides are separated from the lanthanides by using TevaSpec resin.

The THFTCA fraction containing the plutonium and thorium can be directly analyzed by ICP-MS; however, plutonium and thorium must be separated prior to alpha spectrometry on an equilibrated, prepacked anion-exchange column (Bio-Rad AG 1-X8).

The ammonium bioxalate fraction from the TruSpec column containing the uranium and residual thorium is dissolved in 8 M HNO3 and passed through an anion-exchange column that retains the thorium. The column effluent containing purified uranium can be directly analyzed by ICP-MS; however, for alpha spectrometry, the solution is treated with aqua regia and then with concentrated HNO3 prior to electrodeposition.

Fusion Dissolution Procedure

Soil samples are dried, homogenized, ashed, fused with flake NaOH, taken up in deionized water, and chemically reduced. The resulting Fe(OH)3, which coprecipitates the actinides, is separated by centrifugation. The Fe(OH)3 is dissolved in HCl, and silica is removed by centrifugation after formation of a colloid with polyethylene glycol. The actinides are separated on a Diphonix (60-100 mesh) column as a group by using 1-hydroxyethane-1,1-diphosphonic acid (HEDPA). The HEDPA solution is decomposed prior to elemental separations with a double-oxidation procedure. Elemental separations are performed as described under "Acid Digestion Procedure."

The extraction chromatographic resins used in the described procedures offer several distinct advantages over most ion-exchange, coprecipitation, and liquid-liquid extraction procedures. Extraction chromatography allows for specific solvent extraction with the convenience of column chromatography. Both the acid concentrations and volumes used in these systems are significantly lower than those

required for traditional methods, thereby significantly reducing waste volumes; however, these lower acid volumes and concentrations make it difficult to maintain large sample sizes, or soils digested with HF, in solution. As a result, Diphonix resin is used to preconcentrate the actinides from larger soil samples that have been totally dissolved by sodium hydroxide fusion.
Results (6) obtained through ICP-MS and alpha spectrometry agree quite well. For analytes present as contaminants (239/240Pu; 241Am), the laboratory results for standard reference materials were in good agreement with the accepted values (5); however, for the acid digestion procedure, the results for matrix constituents (238U, 234U, and 232Th) were quite low, because the silicate matrix was not decomposed. The NaOH fusion technique allows for accurate analysis of both matrix constituents and contaminants.
MEASUREMENT OF LONG-LIVED ACTINIDES BY INDUCTIVELY COUPLED PLASMA-MASS SPECTROMETRY
Inductively coupled plasma-mass spectrometry is relatively new in the radiochemical field and has been shown to be a very rapid technique for the determination of long-lived radionuclides. Quadrupole ICP-MS has been used to determine 99Tc, 129I, and actinide isotopes in environmental samples; however, many investigators have employed ICP-MS solely for the detection of single radioisotopes or radioactive elements. This approach fails to take advantage of the mass spectrometer's capability for rapid sequential analysis.
We have accomplished the determination of long-lived actinides in soil by ICP-MS (6,7). We selected soil samples for this study because of the difficulties inherent in their preparation for analysis. We employed ultrasonic nebulization to enhance the sensitivity of the instrument and used extraction chromatography to eliminate the sample matrix, concentrate the analyte, and separate uranium from the other actinides (to minimize spectral interferences) prior to instrumental analysis. Limits for instrumental detection of the actinides range from 50 mBq/L (239Pu) to 0.002 mBq/L (235U). Alpha-spectrometric determinations of 230Th and 239Pu and the 234U/238U activity ratio in soil leachates compare well with results from ICP-MS determinations.
We also studied hydrogen adducts of the major naturally occurring actinide isotopes, 232Th and 238U, by using ICP-MS and found that these adducts interfere with the determinations of 233U and 239Pu. When the instrumental sensitivity for uranium and thorium was optimized, the 232ThH+/232Th+ ratio was found to be (3.9 + 0.2) x 10-5 with pneumatic nebulization; with ultrasonic nebulization, the ratio was (2.10 + 0.07) x 10-5. Under the same conditions, 238UH+/238U+ ratio was found to be (3.2 + 0.2) x 10-5 and (1.8 + 0.1) x 10-5, respectively, for pneumatic and ultrasonic nebulization. Conditions that reduced hydrogen number density or increased plasma temperature (or both) decreased the hydride/atomic ratio.
ON-LINE LANTHANIDE AND ACTINIDE DETERMINATIONS USING FLOW INJECTION ANALYSIS/ICP-MS
Flow injection analysis (FIA) is a widely used nonchromatographic flow technique for automated quantitative analysis. Control of physical and chemical kinetics in FIA can be used to design powerful systems for environmental trace analysis. We have developed FIA systems that are based on liquid-solid separation and preconcentation for determining lanthanides and actinides in environmental samples. The sample is loaded onto a small column under well-defined conditions. After the sample matrix has been washed away, the solution conditions are altered so that the sample will elute abruptly into a small volume and pass to the detector. The FIA manifolds use off-the-shelf components to permit the precise manipulation of carrier, sample, eluent, and washing streams through column separation modules. The separation modules preconcentrate the sample by a factor of 10 or more and remove the potentially interfering environmental matrix. TRUSpec, a commercially available extraction chromatographic resin that is based on a bifunctional organophosphorus extractant, is packed into a separation column (3 x 50 mm) in the FIA manifold. Detection is performed by using ICP-MS. We have determined 151Eu and 238U in groundwater samples in a low-dispersion (D < 2) FIA system. The extraction column does not degrade when subjected to repetitive injections. A relative standard deviation of <2% was observed for consecutive (n=20) 100-ng/L 238U injections. Carryover between a
100-ng/L and a blank sample is <5% and the total time for analysis of each 5.0-mL aqueous sample is 6 min. We are developing an FIA manifold that allows for the separation of plutonium, thorium, and uranium thereby permitting the minimization of hydride interferences in ICP-MS determinations of actinides. We are also

exploring the use of the FIA approach for uranium preconcentration and matrix removal prior to kinetic phosphorimetric analysis.

ONGOING ACTIVITIES

To build on the accomplishments discussed previously, more research is being conducted to improve additional aspects of environmental radiochemical analysis. We are developing new separations for strontium and cesium, developing new scintillation detectors for alpha and beta emitters, and continuing investigations into applications of ICP-MS to radiochemical analysis. Method integration is key to improvement of efficiency. We are devoting significant effort to the application of FIA techniques to this problem, with an ultimate objective of a continuous-flow dissolution-separation-measurement system. We believe that this fully integrated system of radiochemical analysis is an achievable objective. Initial results indicate that we can achieve significant savings with comparable or even with improved figures of merit when we automate manual methods.

Development of techniques and methods must be followed by validation of the method. For example, a DOE Methods Compedium (8) procedure, we are validating based on extraction chromatography for the measurement of strontium-89 and strontium-90 in water. The validation analyses are being conducted by six commercial radioanalytical laboratories. Results will yield information on the precision and accuracy of the method, as well as an evaluation of the written draft procedure.

CONCLUSIONS

Innovative application of both classic and new techniques to radiochemical analytical methods can provide substantive improvements in the speed, cost, and other performance measures. Improved dissolution, novel separation chemistries and techniques, nonradiometric determination, automation, and waste minimization are some of the generic approaches that have been proven fruitful. A modest research investment has yielded several methods that can be directly applied to both environmental and waste samples and will result in substantial savings for the DOE and other customers.

Numerous additional improvements can be made to other inorganic, organic, and radiochemical methods using similar approaches. Given the projected costs, the increasing needs for real-time data, and the requirements for improved figures of merit, a pressing need exists for continued development of improved methods that can be applied to frequently requested analyte-matrix combinations.

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ABSTRACT

A technology demonstration that optimizes sampling strategies and real-time data collection was carried out at the Kirtland Air Force Base RB-11 Radioactive Burial Site, Albuquerque, New Mexico in August 1994. The project, which was funded by the Strategic Environmental Research and Development Program (SERDP), involved the application of a geostatistical-based "smart sampling" methodology and software with on-site field screening of soils for radiation, organic compounds and metals. The software, known as PlumeTM, was developed at Argonne National Laboratory as part of the DOE/OTD-funded Mixed Waste Landfill Integrated Demonstration (MWLID).

The objective of the investigation was to compare an innovative Adaptive Sampling approach that stressed real-time decision-making with a conventional RCRA-driven site characterization carried out by the Air Force. The latter investigation used a standard drilling and sampling plan as mandated by the EPA. To make the comparison realistic, the same contractors and sampling equipment (Geoprobe soil samplers) were used. In both investigations, soil samples were collected at several depths at numerous locations adjacent to burial trenches that contain low-level radioactive waste and animal carcasses; some trenches may also contain mixed waste. Neither study revealed the presence of contaminants appreciably above risk based action levels, indicating that minimal to no migration has occurred away from the trenches. The combination of Adaptive Sampling with field screening achieved a similar level of confidence compared to the RCRA investigation regarding the potential migration of contaminants at the site. By comparison, the Adaptive Sampling program drilled 28 locations (vs. 36 for the conventional investigation), collected 81 samples (vs. 163), and sent 15 samples (vs. 163) off-site for laboratory analysis. In addition, the field work took 3 1/2 days compared to 13 days for the RCRA investigation. These figures translate into large cost savings because 22% fewer boreholes were drilled, 50% fewer samples were collected, and 91% fewer samples were analyzed off-site. Of these costs, the most significant savings involved laboratory analyses which typically cost >$1K/sample. Additional costs associated with the increased level of field screening carried out and costs associated with the use of the Adaptive Sampling software are relatively minor compared to the savings achieved.

During the field demonstration, a SunSPARC workstation containing the geostatistical program was successfully linked via the Internet with an identical workstation at Argonne. In the near future, it will be possible to support real-time sampling decisions in the field from remote locations thousands of miles away.

INTRODUCTION

The Kirtland Air Force Base RB-11 radioactive waste site, located in Bernalillo County, southeast of Albuquerque (Fig. 1A), is a 0.02 sq. km. (4.5 acre) landfill containing nine or ten disposal trenches (the exact number is unknown). Incomplete records suggest that the four earliest trenches located at the southern end of the site (Fig. 1B) are 15 meters (~50 ft.) long by 3 meters (~9 ft.) deep by 0.6 meters (2 ft.) wide and have about 1.2 meters (4 ft.) of earth cover. Two of these trenches are covered with asphalt. The remaining trenches are described as being 30 meters long(~100 ft.), 6-7 meters (20-24 ft.) deep and 2 meters (6 ft.) wide with 1.2 meters (4 ft.) of earth cover.

The RB-11 landfill was used to dispose of laboratory wastes (gloves, wipes, etc.) and animal carcasses that had received varying exposure doses of radiation as a result of military research activities carried out in the 1960s and early 1970s. Most of the radioactivity was in the form of induced activity and short-lived radionuclides. However, based on interviews with former employees who worked at the site, it is likely that several millicuries of radionuclides with longer half lives are present, e.g., 137Cs (t1/2=30 yrs), 90Sr (t1/2=28 yrs). Only a small portion of the waste appears to have been buried in drums. In addition to the radioactive wastes, an undetermined amount of hazardous and toxic liquid wastes may also have been disposed of in the trenches. These included small amounts of acids, mercury, cyanides and silver.

The purpose of this paper is to describe a case study in which traditional site characterization methods currently approved by EPA, such as grid drilling and off-site laboratory analysis, are compared with an innovative approach that combines sample optimization with real-time field screening. The innovative approach achieves similar results but is considerably more cost-effective and time-efficient because fewer boreholes need to be drilled and fewer samples need be collected and analyzed off-site. In addition, the sample optimization strategy employed allows real-time decisions to be made in the field regarding additional sampling, thus obviating the need for more costly supplemental sampling programs during a revisit of the site. Our aim here is to present an alternative site characterization methodology that is equivalent to meeting the information needs of a regulatory-driven program, while being more efficient than traditional methods. We consider the present study to be a first step in demonstrating this new approach. Future, similar investigations will be necessary to demonstrate that data quality objectives (QA/QC), statistical validity, and regulatory satisfaction can be achieved at a broader spectrum of sites.

PREVIOUS STUDIES: KAFB RCRA INVESTIGATION

The U.S. Air Force is responsible for implementing a final remediation action plan for the RB-11 site as required by the Resource Conservation and Recovery Act (RCRA) and under the Air Force Installation Restoration Program. Previous investigative activities at the RB-11 site are summarized in an EPA-approved Stage 2B Work Plan for Kirtland Air Force Base, New Mexico (U.S. Geological Survey, 1). In addition, the RB-11 site was the focus of a minimally intrusive field demonstration of innovative site characterization technologies carried out by Sandia National Laboratories in 1993 (Floran, 2). None of these previous investigations conclusively identified any type of contamination at the site.

In July 1994, the Air Force Environmental Management Division and their contractor, Halliburton NUS, conducted a RCRA Facility Investigation (RFI) at RB-11 to fulfill requirements of their Part B Permit. The results of that investigation (Halliburton NUS, 3) are briefly summarized here.

Conventional geophysical surveys including ground penetrating radar and EM-31/61 electromagnetic surveys were used in the RCRA investigation to define nine irregular trench areas, called "disturbed areas". These data were also used to choose locations for subsurface soil sampling, which was subsequently carried out with a Geoprobe soil sampler. The objective of the sampling was to define the extent of contaminant migration, if any, away from the disturbed areas. As required by the Air Force, sampling locations were carefully chosen so that they were outside of the disturbed zones to prevent penetration of contaminant sources within the trenches. After initial field screening, each soil sample was analyzed in an off-site laboratory for gross alpha and beta radiation; 226Ra and 228Ra; volatile organic compounds (VOCs); semi-volatile organic compounds (SVOCs); cyanide; metals (including mercury); and petroleum hydrocarbons. The RCRA investigation concluded that there has been no significant migration of contaminants (organics, metals, radiation) away from the trenches. However, if a release not detected by field screening had been identified, the Air Force was prepared to revisit the site and conduct a detailed follow-up sampling program.

Over a span of 13 days, 36 boreholes were drilled and 163 soil samples were collected and sent to an off-site laboratory for analysis. These numbers do not include surface soil samples collected, additional drilling and sampling carried out for the EPA, and QA/QC samples that were required by RCRA (duplicates and blanks). If the latter activities were eliminated the total operation would have probably taken about 11 days. It should be noted that the Stage 2D-1 RFI report (Halliburton NUS, 3) revealed that radiation levels averaged slightly above background adjacent to one trench, although the data were insufficient to verify that radiological migration has taken place. In addition, trace amounts of mercury were detected in soil samples near three trenches, suggesting that limited migration of this metal may have occurred at the site.

ADAPTIVE SAMPLING INVESTIGATION

The Adaptive Sampling field demonstration took place during the first week in August 1994, approximately a week after completion of the RCRA investigation. The primary objective was to demonstrate that a "smart sampling" methodology that combines real-time field screening results with sample optimization could do an equivalent or better site characterization than could be achieved by using a conventional

approach. The traditional type of site characterization often involves grid sampling, a heavy reliance on costly off-site analyses, and multiple site visits and sampling programs. To accomplish this objective, the Adaptive Sampling plan was compared with the conventional work plan carried out by the Air Force. To make the comparison as realistic as possible, the same drilling contractors (Halliburton NUS) and the same sampling equipment (Geoprobe) were used in both investigations. Off-site laboratory analyses closely matched those specified in the KAFB Work Plan. The main objective of the analytical work performed during the investigation was to provide data that could be reliably compared with similar data obtained by the Air Force RCRA investigation.

However, there were significant differences in the way the two efforts were carried out. The Air Force followed the required conventional approach of collecting soil samples, conducting field screening (for organics and radiation only), and sending each sample to an off-site contract laboratory for confirmatory analysis as required by EPA in the RFI Work Plan. The RCRA investigation resulted in a four to six-week delay between collection of samples and obtaining analytical results. This time gap was potentially crucial because if any of the samples were found to be contaminated, an expensive follow-up investigation involving a new phase of sampling would have had to be conducted. Although both investigations employed field screening methods, the Adaptive Sampling strategy planned to use these results to obtain additional samples immediately if contamination were encountered, thus saving the added costs of revisiting the site.

A second major difference between the two investigations involved the number of samples collected. The modified RFI Work Plan, which addressed all EPA requirements, called for sampling every 5 feet to the bottom of each trench and then 10 feet below each trench. If contamination was encountered, sampling would continue every 3 meters (10 ft.) until no further contamination was detected. By using PlumeTM (discussed below), a site characterization can be performed quicker and with fewer samples compared to a conventional RCRA characterization, yet achieve a similar level of confidence regarding potential migration of contaminants.

APPLICATION OF THE SMART SAMPLING METHODOLOGY

A geostatistical-based computer program, PlumeTM, and "smart sampling" strategy was used to optimize drilling and sampling locations. PlumeTM was developed at Argonne National Laboratory and is a module of SitePlannerTM, a data management and display program marketed by Consolve, Inc., and currently being used by many government laboratories and private industry. Using PlumeTM, 28 borehole locations were sited adjacent to the disposal trenches. Sampling locations were numbered sequentially from roughly south to north beginning with S1 at the southwest and ending with S28 at the northeast (Fig. 2A). None of the bores drilled directly into the disturbed areas, which was also the case with the conventional RCRA investigation.

PlumeTM combines Bayesian analysis with geostatistics to assist in the location of sampling points. A more complete description of PlumeTM's methodology can be found in Johnson (4). Bayesian analysis allows a quantitative merging of "soft" information for a site with hard sampling data. Soft information can include historical records, aerial photographs, non-intrusive geophysical survey results, etc. This kind of information is used to form an initial conceptual image regarding the probable location and extent of contamination. PlumeTM uses indicator geostatistics to update and refine the conceptual image as hard sample data become available. Indicator geostatistics allow one to interpolate from areas where samples exist to areas where samples are absent. New sampling locations can then be selected so that the uncertainty associated with contamination extent is minimized.

At RB-11, the soft information available consisted of aerial photos; anecdotal information regarding the number, size, location, and content of each trench; and several non-intrusive geophysical survey results for the site. This information was used to construct a conceptual image of the contamination at the site. Figures 2A and 2B show a plan view and cross-section, respectively, of this conceptual image in which soils are gray-scale coded, ranging from white (highly unlikely that contamination is present) to black (contamination known to exist). Most of the site appears as variations of gray, since relatively little hard sampling data were available at the outset.

The conceptual image served as the basis for both the RCRA sampling program as well as the program designed with PlumeTM. The sampling strategy for the two investigations was the same: sample as close to trenches as possible without

actually penetrating them to determine the likelihood that lateral and/or vertical contaminant migration has taken place. In the case of the RCRA sampling program, soil bore locations were based on a modified grid pattern, with one set of soil bores located west of the trenches, four bores to the east of the trenches and the remainder between the trenches (Fig. 2A).

For the Adaptive Sampling program, soil bore locations (Fig. 2A) were selected incrementally with the aid of PlumeTM, so that information gain was maximized. Information gain was defined as maximizing the volume of soil in the vicinity of the RB-11 trenches that could be classified as clean at an 80% certainty level. This definition of information gain was equivalent to the stated objective of the RCRA investigation, which was to determine whether contaminant migration had occurred away from the trenches. RCRA investigations never completely remove uncertainty regarding the nature and extent of contamination. The level of uncertainty that can be tolerated during a traditional characterization activity has not been specified by the EPA. For the purposes of this comparison, an 80% certainty level was chosen. Using the 80% certainty level, the information expected from the RCRA sampling program was evaluated by assuming that the samples would have yielded "clean" results. These samples were used to update the initial conceptual image and measure the volume of soils that would be classified as clean. Locations for the Adaptive Sampling effort were then selected to provide the same information gain, while keeping the number of bores and sampling locations to a minimum. The assumption of clean samples reflects the best possible outcome from the conventional investigation --- confirmation that contaminant migration has not taken place.

In the first phase of the Adaptive Sampling program, enough soil bore locations were selected to provide the same base amount of information as expected from the conventional investigation. Because of the field analytical methods employed by the Adaptive Sampling program, analytical results for radiation and VOC analyses were available the same day that bores were drilled, while most of the metals analyses were available before the end of the first phase. Based on these results, the conceptual site model was updated using PlumeTM. If contamination had been encountered, a second phase of sampling would have immediately ensued, with PlumeTM providing the locations of new bores. Additional soil bores and sampling would have continued until the contamination extent had been fully characterized.

FIELD PROCEDURES

Background Sampling

Prior to the field demonstration, four subsurface soil samples were obtained for background determination of organic compounds, metals and radiation. The samples were collected using a Geoprobe at various depths between 5 and 8.5 meters (16-28 ft.) from areas known to be uncontaminated near the periphery of the site, approximately 90-120 meters (~300-400 ft.) from the ends of the closest trenches (Fig. 1B). These background samples were analyzed on-site by field screening methods and also in off-site laboratories. The data were used to help formulate a sampling strategy for the demonstration.

A procedure was designed to minimize the amount of material needed for both on-site field screening and off-site analysis. Each soil sample was divided into five splits; three of these were used initially for field screening, and two were sent off-site (Fig. 3). In addition, one of the splits used for radiological screening was also sent to an off-site laboratory for isotopic uranium and thorium determinations. This same procedure was followed during the main sampling phase of the investigation in August.

Drilling

During the August demonstration, a total of 81 soil samples were collected at depth using two Geoprobe soil samplers which operated simultaneously in the field (the Air Force also used the same two Geoprobes during their field sampling) . Twenty-two of the 28 borehole locations were sampled at approximately 3, 6 and 9 meters (10, 20 and 30 ft.) below the surface, while the remaining 6 holes were sampled at depths of 3 and 6 meters (10 and 20 ft). Three additional samples were obtained including two field replicates.

Field Screening Methods

On-site field screening of soil samples for radioactivity and volatile organics was obtained within minutes of bringing the samples to the surface. Similar screening for metals took longer but a substantial number of samples were analyzed prior to the end of the field work. Field screening instrumentation included a

photoionization detector (PID) for headspace analysis of organic vapors; a Geiger-Muller (GM) radiation instrument with an internal sodium iodide detector and "pancake" probe for gamma radiation; and x-ray fluorescence (XRF) analysis for metals. Soil samples were also scanned for alpha radiation using a scintillometer and beta radiation using a second GM tube. Field screening methods were employed sequentially, with headspace done first to minimize loss of volatile organics. In addition to headspace and gross alpha/beta and gamma detection, a full gamma-ray spectral scan of a split from each sample to be sent off-site was performed within several hours of sample collection (Fig. 3). Metals analysis by XRF was available within 24 hours. These analyses were done at Sandia in a nearby laboratory three miles away, but for the purposes of this paper, are considered to have been done "on-site". It should be stressed that the laboratory XRF unit is field transportable and could have been operated at the RB-11 site within a mobile lab. During background sampling, which preceded the field demonstration, XRF analyses were obtained in one hour. Quick turn-around time for metals was achieved by eliminating a time-consuming grinding step during sample preparation. Previous results using this innovative method have been shown to be reasonably accurate at identifying anomalous samples during field screening (Floran, 5). Such a rapid analysis strategy ensured quick determination of potential contamination at a particular drilling location.

RESULTS

Field Screening

No elevated values above background readings were obtained. All headspace values were 0 ppm (action levels were set at 10 ppm). XRF results were below RCRA action levels for all metals analyzed. Gamma radiation counts on each soil sample were below background plus two standard deviations, which was the action level used to identify radiological contamination.

Laboratory Analytical Results

Approximately 20% of the total number of soil samples collected were sent to off-site laboratories for confirmatory analyses. These included 15 samples plus two field replicate QA/QC samples. Separate splits of each sample (including background samples) were analyzed for a complete suite of organic compounds, metals and radioactivity. These included 34 volatile organic compounds, 67 semi-volatile species, 23 metals (TAL metals + mercury), three uranium isotopes, two thorium isotopes, cyanide, and pH. In addition, 73 radionuclide species were analyzed in an on-site laboratory at Sandia by gamma-ray spectroscopy. These data are tabulated elsewhere (Floran and Bujewski, 6); only the results are discussed below.

VOCs, SVOCs

Two volatile organic compounds, methylene chloride and toluene, were detected in a majority of the soil samples; acetone was found in just three samples. All of these occurrences were below human health risk based (HHRB) standards, as defined by the Air Force's RCRA investigation (Halliburton NUS, 3). The presence of methylene chloride and acetone are likely to be the result of laboratory contamination. Both compounds were noted in laboratory blanks. In addition, methylene chloride was also found in equipment and trip blanks. Toluene ranged from below detection to 22 mg/kg, well below the HHRB action level of 1.6 x 107 mg/kg.

Six semivolatile compounds were detected in the RB-11 soils, all below action levels. One compound, phenol, was found in every sample and another, bis(2-Ethylhexyl)-phthalate, was detected in most samples. The majority of these occurrences were near or below the reporting limit although in one sample, phenol had a concentration of 1700 mg/kg, and in two other samples, bis(2-Ethylhexyl)phthalate had values of 1500 and 1200 mg/kg. The presence of bis(2-Ethylhexyl)phthalate was noted in the equipment blank. Trace quantities of benzoic acid, chysene and benzo(b)fluoranthene were detected in one sample each, and di-n-ocylphthalate was found in two samples, all at levels below reporting limits.

Metals

Only one metal, beryllium, was detected above its action level. All of the samples exceeded this concentration (0.2 mg/kg), ranging from 0.28 to 0.64 mg/kg. These levels of beryllium are typical of the relatively high background values within the area being investigated (Halliburton NUS, 3).

Other (Cyanide; Soil pH)

No evidence of hydrogen cyanide or any other metallic salts of hydrocyanic acids were found. Two samples reported cyanide concentrations at or slightly above the

reporting limit (0.5, 0.7 mg/kg). The action level for cyanide is 2000 mg/kg. Soil pH ranged from 8.4 to 9.5.

Radiological Compounds

Three isotopes of uranium (233/234U, 235U,238U) and two isotopes of thorium (230Th, 232Th) were analyzed in an off-site laboratory. Ranges for the uranium isotopes were 0.82-4.8 pCi/g (233/234U), 0.021-0.11 pCi/g (235U), and 0.74-4.4 pCi/g (238U). Thorium isotopes ranged from 0.59-1.4 pCi/g (230Th) and 0.54-1.4 pCi/g (232Th). None of these values are appreciably above background values determined for the site (Adams, 7).

Radiological contaminants of concern at RB-11 include the source radioisotopes, 90Sr and 137Cs. 90Sr was not determined. 137Cs was measured during the field screening phase of the Adaptive Sampling investigation but no values above background were recorded. Analysis of these radioisotopes was not required by the EPA-approved Work Plan, although gross beta determinations provided an indirect indication that neither was present above background levels (Halliburton NUS, 3).

Data Transfer Using the Smart Sampling Methodology

A SunSPARC workstation at Sandia National Laboratories, New Mexico, where the geostatistical program resided (four miles from the RB-11 site), was successfully linked to an identical workstation at Argonne National Laboratory via the Internet. Communication between the workstations was instantaneous. Data from the RB-11 site were transmitted to both workstations in minutes using a cellular phone/modem hookup from the field. If contamination had been encountered during the RB-11 investigation, we had planned to use the PlumeTM software at Argonne to suggest additional sampling locations. In the near future, it is likely that these types of real-time sampling decisions in the field will be possible from remote locations thousands of miles away.

COST SAVINGS

Substantial cost savings can be obtained by optimizing the number of samples obtained during a site characterization investigation, as well as by judiciously choosing how many of these should be analyzed off-site. The Adaptive Sampling program resulted in drilling 22% fewer boreholes, collecting 50% fewer samples, and analyzing 91% fewer samples in an off-site laboratory. A synergistic cost savings was possible with the smart sampling approach because fewer samples were collected (compared to the conventional type of investigation) and only a small fraction of the reduced sample set was analyzed. Of these costs, the most significant savings involved laboratory analyses which typically cost >$1K/sample.

For the analyses performed in this investigation, undiscounted prices per soil sample were as follows: VOCs ($295), SVOCs ($585), TAL metals + mercury ($466), cyanide ($65), pH ($25), isotopic thorium ($178), and isotopic uranium ($166). However, discounts of up to 50% are typically available to long-term customers. Actual costs for the types of analyses done here might range between $1K (~50% discount) to $1.8K (no discount) per sample. Thus, there is a tremendous potential for savings in analytical costs if only 20% of the samples collected are sent to an off-site laboratory, as was done in the present study.

Cost savings achieved by reducing off-site analytical costs must be weighed against additional costs associated with the increased level of effort associated with field screening (labor, depreciation on analytical equipment, report writing, expendables, etc.), as well as the costs associated with the PlumeTM software (acquisition, training, personnel required to run the program). In the Adaptive Sampling program, the major additional field screening cost that was not borne by the Air Force investigation was the use of an XRF unit for metals analysis. However, these costs, estimated to be $30/hr or $1,600 for the total project (including labor), were minor compared to the savings in off-site analyses. The cost of the SitePlanner/PlumeTM software, which is now available on a PC, has been steadily dropping; present costs including training are ~$6K.

CONCLUSIONS

The combination of smart sampling with field screening enabled our project to achieve a similar level of confidence compared to the conventional investigation regarding potential migration of contaminants away from the trenches. By comparison, the Adaptive Sampling project drilled 28 locations (vs. 36 for the Air Force), collected 81 samples (vs. 163), and sent 15 samples (vs. 163) off-site for laboratory analysis. In addition, the field work took 3 1/2 days compared to 13 days for the Air Force. These figures translate into large cost savings: 22% fewer

boreholes drilled, 50% fewer samples collected, and 91% fewer samples analyzed off-site. Of these costs, the most significant savings involve laboratory analyses which typically cost >$1K/sample for the type of analyses done in the present study. Despite the large number of samples collectively screened and analyzed in both investigations, no significant contamination above background levels of any kind was found. These results suggest that no gross, systematic migration of contaminants away from the trenches has occurred at the site.

The type of site characterization effort described here, in which geostatistically-based iterative sampling is combined with real-time field screening, is best demonstrated when contamination is present. In such a situation, the value of extensive field screening and avoidance of subsequent sampling phases is more easily quantified. Future demonstrations of the smart sampling methodology at sites with known contamination are planned and will be more rigorously compared with conventional approaches to site characterization.

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INNOVATIVE GROUT RETRIEVAL DEMONSTRATION

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ABSTRACT

This paper presents the results of a field demonstration of an innovative transuranic buried waste retrieval technology involving grouting the buried soil/waste matrix followed by application of demolition grout and then retrieval. By grouting the waste in situ prior to retrieval, the highly mobile transuranic contaminants are locked into a soilcrete matrix, thus reducing the chance of aerosolization during retrieval. In this manner, the technology offers an enhanced contamination control. The demolition grout fractures the soilcrete/waste monolith to facilitate the retrieval process. The technology is applicable for both hot-spot retrieval or full-pit or trench retrieval. The grouting is accomplished using Portland cement and jet grouting directly into the soil/waste matrix at 6,000 psi. Demolition grout involved use of the commercially available product Bristar. Retrieval can be accomplished using standard excavation equipment or remotely controlled equipment. The results presented in the paper include: a) an assessment of the overall efficiency of the jet-grouting operation, demolition grout operation, and retrieval operation, b) an assessment of both the dust spread and rare earth tracer spread during both the grouting phases and retrieval phases, and c) a cost estimate to apply this technology at buried transuranic waste sites.

INTRODUCTION

This paper presents the results of full-scale field evaluation of an innovative retrieval technique for buried transuranic waste. The Idaho National Engineering Laboratory (INEL) has 2 million cubic feet of transuranic waste in shallow land burial (commingled with at least 6 million cubic feet of soil). Retrieval and disposal of this waste is one of the options being considered for this waste. Also being considered is the concept of removing selected hot spots and applying a capping action to the buried waste. Conventional retrieval using off-the-shelf remote excavators for either full-pit or hot-spot retrieval is expected to create considerable dust. Standard contamination control systems involve using misting sprays to reduce aerosolized soils, fixants to maintain the naturally occurring moisture in the soil/waste matrix, and ventilation. Using these conventional mining technologies in a prior retrieval demonstration (1), only a 70% reduction in dust spread was achieved over a baseline case of digging without any contamination control measures applied. During prior dust-control studies, it was identified that a 98% reduction in dust is required to allow for bubble-suited entry in a retrieval enclosure (2). Therefore, any technology that enhances the control over dust spread is mandatory to pursue.

DESCRIPTION OF THE TECHNOLOGY

The innovative retrieval technique involves grouting the buried soil waste matrix with a jet-grouting procedure, applying an expansive demolition grout to the matrix and retrieving the debris. The grouted matrix provides an agglomeration of fine soil particles and contaminants that results in an inherent contamination control during the otherwise dusty retrieval process. The jet grouting was accomplished using a Casa Grande drilling/jet-grouting rig (shown on Fig. 1). The grout was simple Portland cement mixed on a 1:1 ratio based on mass. The 4-9/16 in. drill stem was driven through the waste (consisting of a landfill-type mixture of drums and boxes containing simulated waste material including concrete, paper, cloth, sludge, metal, and wood).

Fig. 1. Casa Grande drilling/jet-grouting rig.

Once fully inserted, the jet grouting began using a bottom-to-top motion of the drill stem. The drill stem was programmed to withdraw in precise increments at a predetermined rate (5 cm every 6 seconds). There was a 2-revolutions-per-second rotation of the drill stem during the jet-grouting operation with two injection points near the bottom of the drill stem located 180 degrees opposite each other facing outward. Using a positive displacement pump, grout was pumped into the waste via the two nozzles at 6,000 psi. Once the jet-grouting operation was completed in any hole, a 2-in.-diameter, thin-walled, spiral-wrapped metal tube closed at the bottom end was inserted into the grouted hole and weighted in place until the grouted monolith was cured. These tubes were used to insert the demolition grout to facilitate the fracturing of the waste. The monolith was evaluated for temperature and once stabilized, the Bristar demolition grout was applied. This Bristar grout is designed to expand upon curing and during field trials near the simulated pit expansions on the order of 1 in. in diameter were observed. Retrieval of the grouted/fractured monolith can be accomplished by standard excavation techniques or using a gantry crane mounted clam-shell/grapple arrangement.

FIELD DEMONSTRATION RESULTS

A full-scale field demonstration of this retrieval technique was performed on a simulated waste pit at the INEL. Reference 3 discusses the important features of this proof-of-concept demonstration including the pit construction, jet-grouting activities, application of the demolition grout, retrieval of the grouted pit, and quantitative evaluation of aerosolized soils and rare earth tracer spread for all phases of the demonstration.

A simulated waste pit 10 ft x 10 ft x 10 ft was constructed using 55-gallon cardboard and metal drums and 4 ft x 4 ft x 4 ft cardboard boxes full of simulated waste. The simulated waste consisted of cloth, paper, metal, wood, sludge, and concrete identical to that which was shipped and buried at the INEL from the Rocky Flats Plant. Each container had a random dump of 200 grams of rare earth tracer to simulate the transuranic contaminant. The pit was built prototypical of pits found in the INEL Subsurface Disposal Area using backfilled lake-bed soil. A large weather shield was erected over the site and air samplers were used for all major portions of the demonstration.

The jet-grouting phase was accomplished with minimal dust spread and no rare earth

tracer spread above background. A total of 36 grout holes placed 24 cubic yards of Portland cement into the pit in a jet-grouting action (at nominally 6,000 psi). The jet grouting was accomplished by first driving the injection bit into the waste and withdrawing the bit in 5-cm increments while applying a slow rotation of the bit. The withdrawal rate was nominally 5 cm per 6 seconds.

The hole spacing was 24 in. on a triangular basis and the process took about 40 minutes to drill, jet grout each hole, and reposition the drill for the next hole. The hole spacing was based on a series of predemonstration field trials in compacted INEL soil in which up to 28-in.-diameter columns were created. After each jet-grouting operation, a bottom-sealed, 2-in.-diameter, spiral-wrapped tube was placed into the drill hole and allowed to cure in place. During the jet-grouting operation, soilcrete and some waste in the form of sludge came to the surface of the pit (in gallon quantities for some holes). Even though this material contained the rare earth tracer, no airborne spread of tracer was found above background on the air samplers spaced strategically around the pit. Figure 2 shows the pit following the grouting operation.

Fig. 2. Grouted pit with spiral wrapped tubes.

The application of Bristar as a demolition grout failed to fracture the waste as planned. The reason the grout did not expand upon curing was that there was an axial temperature gradient in the monolith with up to 60F higher temperatures in the middle regions of the pit than on the ends. This axial temperature gradient was caused by the heat of hydration as the Portland cement cured and the surrounding insulating properties of the soil. As a result, it was difficult to achieve fracturing. The grout appeared to expand at the near surface positions; however, these were the cooler positions. A separate application of the Bristar product in several holes drilled directly into the monolith showed a positive proof of fracturing. For this case, the temperature of the pit was more uniform and reduced from the postgrouting value. An alternative nondust-producing technique to the demolition grout would have been using a hydraulic rock spliter in the spiral-wrapped tubes to fracture the monolith.

Even though the monolith was not fractured by the demolition grout, the standard backhoe bucket with thumb attachment in the belowgrade orientation was successful in removing the monolith. The 10 ft x 10 ft x 10 ft monolith was removed in just under 5 hours of retrieval time. Grouted boxes containing metal pipe, wire, and plate steel caused the most difficulty in retrieval and a larger backhoe or front-end loader would have greatly facilitated the process. The general soilcrete mix was easy to flake off with the backhoe in 1 ft x 2 ft regions. Removal of computer paper resulted in tracer spread as the grout-encased paper easily disintegrated upon retrieval leaving the computer paper intact with visible tracer on the paper. Other waste forms such as pipes and wood also came apart from the grout during retrieval; however, for these materials there was a layer of grout on the surface and in the case of pipes the pipes were completely filled. If the waste pit contained all intact metal drums, there is an indication that this process would result in completely grouted drums including the space between the drum and the drum liner. One such specimen of a metal drum originally containing plastic pipe was completely full of grout and was sectioned for display purposes as shown on Fig. 3.

Fig. 3. Cross-sectional view of grouted drum (jet grouted 10 ft below surface).

The retrieval was accomplished with both overburden included and overburden removed. It was found that leaving the overburden on promotes both dust spread and tracer spread during retrieval. During retrieval with the overburden on resulted in a dust removal over a baseline retrieval case of between 30 and 40%, which compares unfavorably with conventional contamination control cases using misting and fixants where 70% removal was achieved. However, when performing retrieval with the overburden removed, a 90% dust removal was achieved, which compares favorably with the desired dust removal of 98%. During retrieval with overburden on, the airborne tracer concentration was 4,000 times background and with the overburden removed, the tracer concentration was only 1.35 times background. The overburden fell into the pit displacing air, which caused entrained particulate from the debris in the bottom of the pit to travel to the air monitors indicating a spread of contaminant.

In summary, a positive proof-of-concept for a enhanced contamination control was shown during the demonstration. The technology could be applied to buried heterogeneous waste for hot-spot retrieval or full-pit retrieval and could also be applied to contaminated soil zones. It is recommended that the retrieval technology

be further examined to obtain an efficient technique compatible with the grouting idea. For full-scale retrieval this would involve a large 235-CAT class excavator or larger and a large Balderson thumb end effector with companion shears. For this large excavator approach, the abovegrade digging action would pull the waste into the bucket and keep the waste from falling into the pit.

FULL-SCALE APPLICATION AND COST ESTIMATE

To apply this technology to full-scale operations would require very little change to the procedures used in the grouting phase; however, two outstanding questions remain relating to demolition grouting and retrieval of the monolith. The problem areas were stabilizing the axial temperature gradient in the monolith following grouting for application of the demolition grout and using the proper remotely controlled equipment to retrieve the waste. Additionally, it is mandatory to remove the relatively clean overburden prior to removing the monolith.

The cost estimate for this technology is incomplete; however, a first order engineering estimate is given. These costs only reflect the costs of grouting, fracturing, and retrieval. Costs associated with buildings and conveyance equipment, packaging, assay, and disposal would be separately estimated based on risk assessment. To perform grouting and retrieval on a 1-acre site would cost $16 million, excluding costs associated with containment and management of the project. The grouting phase is about half of this cost. To perform a hot-spot removal on a 40 ft x 60 ft site would cost just under $4 million, again neglecting the costs associated with management and containment. For either of these estimates, there are no costs given for treatment and final disposal.

CONCLUSIONS

1) On an overall basis, a positive proof-of-concept for the innovative grout/retrieval technology for retrieval of hot spots and full-scale retrieval has been shown for all major phases of the demonstration.

2) The grouting phase can be accomplished with a simple weather shield and still maintain control over contamination spread. This is based on the measured concentration of dust and rare earth oxide tracer spread during the grouting operation. The airborne tracer spread was measured at background values even though waste material was present in ejected grout/soil material around the drill stem. The simulated contaminants (rare earth tracers) in the "ejected" material was locked up in a slurry mix and did not spread to surrounding regions.

3) The jet-grouting technique created a monolithic block of waste/soil/Portland cement. Although there were striations of soil within the block, there were no continuous regions of soil that would be conducive to water migration through the block. There was no evidence of shadows using the nominal 24-in. hole spacing and all waste material had been encapsulated with the soilcrete mix. Additionally, there was no volume increase in the pit from the jet-grouting activity. There was a mass increase as 24 cubic yards of Portland cement was injected into the pit assessing about 60% voids.

4) It is concluded that the Bristar demolition grout causes extensive fracturing of the monolith if the temperature of the monolith is uniform and the correct product is applied.

5) For excavations involving an access pit [side (digface) retrieval], it is concluded that retrieval of the grouted monolith be accomplished only following overburden removal prior to the retrieval of the monolithic block. Overburden tends to fall into the access pit and cause entrained air to carry contaminants throughout the retrieval area. Comparison of tracer spread with and without overburden removal shows a factor of 4,000 times background of air samplers with overburden and a factor of 1.35 times background without overburden.

ACKNOWLEDGMENTS

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27-45
STATISTICALLY ROBUST MAPPING OF GAMMA-EMITTING CONTAMINANTS IN THE GROUND

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ABSTRACT
It is common practice at DOE sites to infer the three-dimensional distribution of contaminants in the ground from the analysis of samples from boreholes. In the case of gamma-emitting contaminants, spectral gamma-ray (SGR) borehole logging is sometimes used to augment or replace the sample data. In general, the same SGR analysis technology is used for both laboratory analysis and in situ analysis for gamma emitters. Laboratory spectrometers can give accurate results but the sample size is small. With SGR logging a single measurement represents an analysis of 1000 to 10,000 times as much material as a typical physical sample. Common practice is to make several SGR logging measurements per foot of depth, resulting in overlapping sample volumes and a well-analyzed borehole. Furthermore, while it is generally not feasible to obtain additional samples once the borehole is in place, SGR logging is a cost-effective method for monitoring changes in concentration over time along existing boreholes. Finally, SGR logging is very sensitive, with an estimated detection threshold on the order of 0.1 pCi/g for 137Cs. Typical background concentrations are an order of magnitude higher.
Because the results are not normally quantitative, Los Alamos is modifying and adapting existing data processing techniques developed for uranium exploration and other applications to the problem of assessing environmental contamination. The Quantitative Spectral Gamma-Ray Logging Project is also advancing the state of the art by eliminating limiting assumptions inherent in these existing data processing approaches, in particular the assumption that contaminant distribution varies with depth only. This new approach integrates nuclear geophysics, contaminant transport modeling, and geostatistics. We are developing data quality assurance tools that apply not only to borehole logging but to the collection and analysis of physical samples as well.

INTRODUCTION

Passive spectral gamma-ray (SGR) borehole logging can detect and identify the artificial gamma-emitting nuclides that are found in the ground at some DOE sites as well as natural gamma-emitting nuclides found in most rocks and soils. In addition to naturally occurring potassium, uranium-family and thorium-family gamma emitters, a number of contaminant nuclides can be detected readily with SGR logging (1,2), including 60Co, 137Cs, 152Eu, 154Eu, and 235U. Nuclides with lower energy or weaker gamma-ray lines can also be detected under favorable conditions.
SGR logging is very sensitive, giving an estimated detection threshold better than 0.1 pCi/g for 137Cs, an order of magnitude below typical background concentrations. SGR logging can give a continuous record of concentrations along the borehole and typically analyzes at least 1000 times more material than laboratory sample analyses. Supplementing a reduced sample analysis program with borehole logging can reduce overall costs while enhancing data confidence, as discussed in a General Accounting Office study of the subject (3). In the case of long-term monitoring, in situ measurements are superior to other technologies, offering continuous coverage, large sample volume, sensitivity, and accuracy, combined with the ability to repeat measurements in the same borehole year after year at reasonable cost.
SGR borehole logging services and instruments are available from a number of commercial vendors. The detectors used in SGR borehole instruments are usually low-energy-resolution scintillators (4-6), but a few instruments containing high-energy-resolution, solid-state cryogenic detectors have been built and are in use today. (1,7,8). In the case of low-energy-resolution detectors, full spectral processing is generally needed to identify specific nuclides because the broad photopeaks from different energy lines overlap (2). In the case of high-energy-resolution detectors, overlapping peaks pose much less of a problem, but automatic picking of specific characteristic energy lines from the complicated spectrum is desirable (9,10).
While SGR logging can be very accurate in identifying which contaminants are present, especially when a high-energy-resolution detector is used, the data are not

proportional to the concentrations of contaminants at a given depth except in thick, uniform zones (11). To address this limitation, data processing techniques have been developed for SGR logging, particularly for uranium exploration applications, which can convert the qualitative logs to quantitative concentration profiles for individual nuclides within limitations imposed by well-understood sources of error (12-14). The application of such data processing should make SGR logging acceptable to regulators as a replacement for most laboratory analyses for gamma-emitting contaminants along boreholes; this could yield savings of millions of dollars at DOE sites where gamma-emitting contaminants are a concern (3). In addition to site characterization, the advantages of continuous coverage, large sample volume, sensitivity, and accuracy, combined with the ability to run repeat measurements in the same borehole year after year at reasonable cost, yield a monitoring tool that is superior to all other technologies.

SYSTEM CALIBRATION AND CHARACTERIZATION

A vital step in applying existing borehole techniques to a new application such as environmental restoration (ER) is to calibrate the logging system for the conditions that will be encountered. To understand the scale of the calibration problem it is helpful to draw a comparison with the calibration of laboratory analytical instrumentation. To calibrate laboratory gamma-ray spectrometers, a number of standards must be prepared containing accurately known constituents in the same geometry as the unknown samples that are to be analyzed, and any deviation of the samples from the standards (for instance, different density) must be understood and corrections applied.

Precisely the same calibration requirements hold for borehole instrumentation except standards that simulate the borehole environment are needed rather than small cans or bottles of material. In the case of SGR logging, that typically means a model at least 1 m in diameter by 2 m high with a borehole of the appropriate diameter down the center. As a minimum requirement for quantitative results, the logging system must be calibrated for at least one set of known conditions. Many physical calibration models already exist, including a number in the DOE complex and at other government facilities. For instance, doped concrete models were used extensively in the NURE Program in the U.S. (15) and similar programs in other countries (16), as well as in the petroleum industry (17,18). Quarried rock models have been established at the U.S. Geological Survey in Denver (19), at the Nevada Test Site (20), in the petroleum industry, and elsewhere. In some cases, models are made of loose material encased in a shell of aluminum, plastic, or other material. Suitable experiments or computer modeling can be used to account for the effect of the shell. This approach has been used at the Nevada Test Site (21) and elsewhere.

In the ideal case, the calibration model(s) will mimic field conditions exactly; in practice, that may be difficult to achieve. Therefore, following the basic calibration, correction factors must be determined for field conditions that are different from the calibration conditions, such as a different borehole diameter. This can be accomplished using a number of additional physical calibration models or a combination of physical models and computer simulations. Due to cost and other constraints, the number of physical models available is generally not sufficient to cover all conditions encountered in the field. Computer simulations can be used to supply additional information (22-25). After benchmarking a computer simulation program against experimental data from physical models, other parameters that are difficult or impossible to model physically can be studied using the computer simulations. For instance, it is a relatively straightforward task to evaluate the response of a given logging system in a totally dry physical model and a second model totally saturated with water, but achieving known, intermediate values of saturation using physical models is difficult. However, computer simulation programs, once benchmarked at 0% and 100% saturation, can extend the calibration results to intermediate values. Simulations can also extend the calibration to include such factors as trace elements, different formation densities, and many other real-world conditions that may need to be considered.

QUANTITATIVE INTERPRETATION

Estimates of radioelement concentrations for a homogeneous layer of rock perpendicular to the borehole can be made based on a calibrated gamma-ray or SGR log. Such estimates are derived from the equation (12):

Eq. (1)

where G is the desired average radioelement concentration or weight fraction (the G

represents uranium ore grade in uranium mining), T is the thickness of the bed (meters), A is the area under the recorded gamma ray curve corresponding to the radioelement in question (counts per second times meters) and K is the calibration factor (K-factor) for that logging instrument or tool and that particular radioelement (weight fraction of a given radioelement per count per second). The assumptions upon which Eq. 1 is based limit its accuracy under certain conditions. This has been studied by various authors (12,26). For Eq. 1 to be valid for estimating uranium or thorium concentrations, those elements must be in secular equilibrium with their respective gamma-ray emitting daughters.

Note that the deflection of the recorded gamma-ray curve is generally not proportional to the radioelement concentration at a given depth (Fig. 1a). Only when the tool is in a thick, homogeneous zone (at least a meter or so thick as shown in Fig. 1b) does the concentration become proportional to the recorded gamma-ray intensity, or

Eq. (2)

where I is the recorded gamma radiation intensity for a given gamma emitter (counts per second) at a given depth.

Borehole Effects

Any deviation of the actual borehole environment from the calibration models must be understood and corrections must be applied (27-30). An example of a borehole effect that requires correction would be the effect of steel casing in the hole. If the system is calibrated in a model without casing and then run in a cased borehole, the shielding effect of the casing will cause the radioelement concentrations to be underestimated. A multiplicative correction factor based on casing thickness and composition can easily be applied to compensate.

Figure 1(a) shows the response of a calibrated SGR logging instrument (with detector D) in the vicinity of a thin radioactive bed. Concentration is underestimated and bed thickness is overestimated, but the product of the two is correct. Figure 1(b) shows that instrument response is proportional to concentration near the center of a thick, homogeneous radioactive bed.

Geologic Formation Effects

When an SGR logging tool is calibrated in a model that is different chemically from the lithology to be logged, the resulting calibration factors will not necessarily be correct (31). This could happen if the model is composed of a type of rock different from that encountered in the field, or if there is a different amount of water in the pore spaces of an otherwise identical rock type. Errors resulting from differences in the chemical composition of the matrix material between the calibration model and formations found in the field will generally be small at the higher energies most frequently used for radioelement estimates. Errors can be large at low photon energies when either the calibration model or the field formation contains substantial quantities of heavy elements not found in the other. This is known as the Z-effect, where Z is the symbol for atomic number. The Z-effect results from the fact that at low energies, highly nonlinear photoelectric absorption dominates the gamma-ray interactions, rather than the linear Compton scattering dominant at energies around 1 - 2 MeV.

Conventionally, radioelement concentration is expressed as a weight fraction of the bulk formation including water. This eliminates most of the error associated with varying water content. However, there remains a small, second-order error not accounted for by that convention due to changes in the gamma-ray transport properties of the medium. Results can be improved slightly by applying a multiplicative correction factor (31).

Thin Contaminated Layers

As stated above, data processing techniques have been developed for uranium exploration and other applications that can correct for the errors inherent in logging through thin radioactive layers within limitations imposed by well-understood sources of error. Such techniques are referred to as spatial deconvolution techniques. Spatial deconvolution is not a matter of simple scaling but involves the application of inverse theory to back out the "smearing" effect of the spatial response of the logging instrument. This smearing effect is a non-linear function of a number of borehole and formation parameters including borehole diameter, casing type and thickness, and borehole fluid as well as formation density, porosity, water saturation, and other factors that affect the passage of gamma radiation through matter (13,14). Errors caused by dipping beds can also be

corrected (32).
In an infinitesimally thin borehole using a point detector, the system response function j(z) can be approximated by the equation
Eq. (3)
as given by Davydov (34) based on work by Suppe and Khaykovich (35). In this equation, z is depth along the borehole (the thin zone is located at z = 0) and is a parameter referred to as the shape constant (not the same as the Th/U spectral stripping factor a). The shape constant a can be determined theoretically (36) or experimentally in model boreholes (37). Under favorable conditions a can be determined directly from field logs (37).
Equation 1 describes a double-sided exponential function, which has a corresponding 3-coefficient inverse filter (14):
Eq. (4)
where Dz is the sampling interval along the hole. This digital inverse filter is applied to the digital log using discrete convolution to recover the radioelement distribution information, thereby deconvolving the log. The advantage of the approximation given in Eq. 3 is that it leads to a simple inverse filter (Eq. 4) that may be adapted relatively easily to changing conditions by adjusting a.
In the real case of a finite borehole diameter and detector length, the system response function will no longer approximate Eq. 3 but will appear more rounded in the vicinity of the radioactive zone. Nonetheless, the system response function in these cases will generally approximate the double-sided exponential away from the vicinity of the radioactive zone. For the most part these various distorting effects can be removed from the data separately and in any order (at least to the extent that they can be removed at all). Thus, it is not necessary that the system response function correspond precisely with Eq. 3. The inverse filter may be used to remove the effect of the double-sided exponential part of the total spatial response function, leaving some residual distortion in the immediate vicinity of the thin radioactive zone depending on such factors as borehole diameter and detector length. These effects may also be considered individually using a similar empirical approach, although they are not readily tractable like the double-sided exponential.
The effects of borehole, formation, and instrumental parameters on the shape of the system response function have been considered by many authors (29,36,38,34,14). An increase in the value of the shape constant a indicates a more rapid decrease in radiation intensity with distance from the radioactive zone. This condition can be caused by many factors, including
a) An increase in formation bulk density, including an increase in pore fluid density
(e.g., replacing air with water);
b) A decrease in dip angle between the radioactive zone and the normal to the borehole, until perpendicularity or 0o dip;
c) A decrease in borehole diameter, allowing proportionately less radiation to travel
in the borehole;
d) An increase in borehole fluid density such as from air to water to heavy mud;
e) An increase in the equivalent atomic number Zeq of the formation;
f) An increase in steel casing thickness or density;
g) Restriction of the detected gamma rays to unscattered photons such as the 1.76-MeV 214Bi energy line of the uranium family; and
h) For unscattered gamma rays, a decrease in gamma-ray energy.
Spatial deconvolution techniques, which have proven superior to detector collimation and other approaches, were able to detect substantial errors in the published radionuclide concentrations of a thin-zone gamma-ray logging test and calibration facility at DOE's Grand Junction installation (33) during the National Uranium Resource Evaluation program, as shown in Fig. 2. Those errors, which had gone undetected throughout twenty years of logging instrument tests and calibrations in that model, were confirmed by subsequent physical sampling and laboratory analysis. The concentration values for that model have been corrected and re-published.
Figure 2. (a) Cross-sectional diagram of the mixed-zone model N-5 located at the DOE Grand Junction office. (b) A high-quality, calibrated SGR log from the borehole in N-5 using a small (2.5 cm x 2.5 cm) NaI(Tl) detector. (c) The same data shown in (b) after deconvolution using the inverse filter given in Equation 4. Disagreement with the published concentrations (marked by the arrows in Fig. 2c) caused DOE to

order additional core samples and other samples to be obtained and analyzed. The concentrations were revised based on the log and republished. (d) The deconvolved data compared with the revised published concentrations.

Nonhomogeneous Distributions of Contaminants

Existing approaches to the quantitative interpretation of SGR borehole logs assume that the gamma emitters are distributed in homogeneous layers. This can be a good approximation in some geologic environments such as the sedimentary lithology often associated with oil fields. It appears, however, that contaminants at DOE sites may more commonly be found in complex, three-dimensional distributions. For such cases, an approach is needed that extends the state of the art beyond linear deconvolution. We have proposed an approach which we are calling Optimized Geostatistical Inversion (OGI) that integrates nuclear geophysics, contaminant transport modeling, geostatistics, and other site data such as physical sample analyses. The OGI approach will help explain the relationship between SGR borehole logs and physical sample data, how well the physical sample data and the SGR data predict the actual distribution of contaminants in the ground, and what the trade-offs are among sample density, data quality, and cost. This approach will also provide statistical quality control tools to describe the confidence limits of both SGR logging data and physical sample data. Finally, OGI is a results-oriented approach tailored to each specific location and contamination problem based on a combination of simulations and analysis of field data. The OGI approach may be summarized as follows:

1. Using an available contaminant transport computer model, simulate the general geologic and contaminant transport characteristics of the target environment and use the simulations to gain an understanding of how contaminants of concern move through this type of environment and are deposited along their transport paths. Note that the goal is not to predict or duplicate the actual contaminant distributions, but to use realistic simulated distributions to gain a statistical understanding of the field data.
2. Emplace imaginary boreholes through the simulated contaminant distributions. Using a Monte Carlo radiation transport computer program or other suitable approach, develop synthetic SGR borehole logs based on the contaminant distributions developed in the contaminant transport modeling stage. The synthetic logs should consist of spectra generated at equal depth intervals along the borehole, similar to actual logs.
3. Estimate concentrations of gamma-emitting contaminants based on an analysis of the synthetic borehole gamma-ray spectra using the same procedures as will be followed for real data.
4. In the same imaginary boreholes, determine contaminant concentrations based on physical samples representative of typical sampling protocols. By repeating this for many sample depths and borehole locations, a geostatistical model of the sampling process can be developed.
5. Repeat steps 2-4 for other imaginary borehole locations to develop a geostatistical model of SGR logging and physical sampling for the target environment.
6. Compare the known simulated distribution of contaminants in the subsurface developed in step 1 with estimates that would be obtained by in situ SGR measurements and physical sampling (steps 2-5). In comparing the results of borehole sample analyses and SGR logs, it is important to remember that these samples and in situ measurements investigate two mutually exclusive but related volumes of material. The sample from a particular depth is a small volume of material, perhaps a few cubic centimeters, removed during the drilling process. The SGR measurement at that same depth represents a weighted analysis of perhaps 105 cm3 of material, including all of the material in the region around the borehole sensor except the physical sample itself and other material removed by the drilling process. Thus, in an inhomogeneous medium such as found at Hanford and other DOE sites, it may be expected that the physical sample data and SGR data will rarely agree in a point-by-point comparison. To assure an acceptable level of data quality it is essential to understand the statistical distributions involved in these analysis processes. This approach allows such an understanding to be developed cost-effectively.
7. Finally, using actual SGR data and physical samples from field boreholes, perform a geostatistical comparison of physical sampling and in situ measurements. Compare these results with the statistical distributions developed during the simulation

phase of the project. In this step, inconsistencies in the synthetic data can be identified and the model refined, if required.

Figure 3. The cutaway diagram of the borehole environment (a) shows two identical point gamma-ray sources located at different distances from the borehole. The resulting log (b) indicates that two very different spatial responses would result from the two sources. The gamma-ray energy spectra recorded at the center of the anomalies from sources 1 and 2 are shown in (c) and (d), respectively. The spectrum corresponding to source 1 shows a small peak-to-continuum ratio and lower overall intensity (notice the different intensity scales on the two plots). This information can be used to distinguish between 3-dimensional source distributions and layered distributions.

The simulation approach described above is important because a plausible distribution of contaminants is known completely and accurately, along with the other properties of the media that affect the accuracy of the contaminant estimates. Analysis of the detailed simulated data sets will lead to a detailed understanding of the problem that could otherwise be developed only by examining massive amounts of experimental data, a much more expensive procedure. Using simulation, the groundwork is carefully laid by developing a robust geostatistical framework in steps 1-6, into which the borehole data are folded in step 7.

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INNOVATIVE DESIGN AND CONSTRUCTION SOLUTIONS TO REMEDIATE RADIOLOGICALLY CONTAMINATED RESIDENTIAL COMMUNITIES IN ESSEX COUNTY NEW JERSEY

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ABSTRACT

The Montclair/West Orange and Glen Ridge Superfund Sites, located in Essex County, New Jersey, are contaminated to varying degrees with radioactive materials. Both sites are listed on the Superfund National Priorities List (NPL). The waste materials originated from radium processing facilities prevalent in the area during the early 1900's. The design for remediation of these sites is performed by Bechtel National, Inc. on the behalf of the United States Army Corps of Engineers (USACE) Kansas City District, which administers the project through an Interagency Agreement with the United States Environmental Protection Agency (EPA) Region II.

This paper will describe each part of the remedial design process: design field investigations, preparation of designs, final evaluation of designs, and actual construction using these designs. The paper will discuss the relationship between the interpretation of data and the remediation methods and technologies used for the remediation of structures by underpinning basements, on-grade structures, and chimneys; removal of contaminated material below supporting elements without resorting to underpinning; and demolition and reconstruction of basement foundation walls (where contaminated) without use of traditional "cribbing" methods.

SITE DESCRIPTION AND LOCATION

The two sites consist of 773 residential and commercial properties having a combined land area of approximately 210 acres. The Montclair/West Orange radium site covers approximately 120 acres and includes 297 properties in the Town of Montclair and 200 properties in the Town of West Orange. The Glen Ridge radium site covers approximately 90 acres and includes 276 properties in the Town of Glen Ridge and 32 properties in the City of East Orange.

The houses within the sites are generally wood-framed, multi-story with below ground basements, front and back porches, and detached garages. The architecture of the homes is typical of the 1930's era. The size of a property usually ranges from one-sixteenth to one-eighth of an acre.

SITE HISTORY

Radium research and processing facilities were prevalent in Northern New Jersey from the early 1900's to the late 1920's. The processing facilities produced luminous paint for watch dials, surveying equipment and airplane instruments; and later extracted the radium for research and medical applications. By the early 1930's, the effects of excessive exposure to radium were discovered, which subsequently caused an end to the processing facilities. However, the discarded processing waste materials from these facilities were left behind. While some of these materials remained at the facilities, various waste materials such as sand tailings, coal ash, bottles, etc. were disposed of in rural areas surrounding the facilities. Eventually, as these areas were developed with residences, commercial buildings, roadways, parks, etc. the contaminated material was moved to fill low-lying areas or mixed with uncontaminated soil and fill material. In many cases the contaminated material was mixed into concrete and mortar and used in the construction of buildings.

NATURE OF CONTAMINATION

The contamination at the Montclair/West Orange and Glen Ridge Radium Sites consists of elevated concentrations of radionuclides in the soil. The radionuclides present in the soil at these sites are primarily from the uranium-238 decay chain. Contaminated properties in the Montclair/West Orange and Glen Ridge sites have elevated levels of gamma radiation from radionuclides decaying within the soil and building materials, and alpha radiation emitted from radon and radon decay products.

REMEDIAL DESIGN FIELD INVESTIGATIONS

The primary methods used to perform the field investigations on each property included:

- Exterior/Interior gamma radiation surveys
- Subsurface Investigations

Exterior/Interior Gamma Radiation Survey

The exterior and interior gamma radiation surveys were conducted across an entire property (outside the home and on the lowest floor elevation of a home and garage) to locate areas of elevated gamma radiation and map out areas where potential soil contamination existed. Gamma readings were measured using a very sensitive gamma measurement instrument (gamma scintillometer) oscillated approximately six-in. from the ground surface across 6-ft. x 6-ft. grid blocks. This instrument was used to detect general areas where the radiation was above background.

The gamma survey and the readings recorded during the survey were used as indicators to proceed with one of two investigative strategies. The first included subsurface investigations by drilling; the second included a repertoire of investigative methods to classify a home as either "clean" or requiring additional field investigation. Only strategy one is relevant to the Remedial Design (RD) process.

Subsurface Investigations

A subsurface investigation was performed for all properties which exhibited elevated surface gamma radiation measurements. Drilling locations were selected after evaluating surface gamma radiation measurements, noting areas of elevated measurements and noticeable trends of contamination. Consideration was also given to the field investigations performed during the RI/FS; data results from contiguous properties; and any significant or noticeable features or trends observed from work in the immediate area.

Generally, the quantity of boreholes drilled was limited to minimize cost. Therefore, the drilling approach varied for each property and was usually dependent on the apparent extent of contamination as ascertained from the gamma survey and as the need to clarify some aspects related to the engineering design solution of the remediation.

Drilling

The drilling techniques used depended upon the depth of anticipated contamination; physical access onto a property; and the type of soil material that might be encountered.

The drilling was performed using the following types of drilling equipment:

- Skid Mounted Drilling Rig
- Tripod Mounted Sampling Rig
- Hand Augers

With drilling depths of up to 10 ft. several of the holes were drilled with a truck-mounted drill rig. Since soil samples were also taken, a 3-in. diameter split spoon sampler was used.

Due to the limited physical access to many properties by even the smallest drilling rigs, the great majority of boreholes were hand augered. As drilling proceeded for each borehole, a descriptive log of the material encountered in each 6-in. increment of drilling depth was documented. The soil was described by type and color. This information often helped to identify lenses of material that were frequently found to be contaminated.

Borehole Gamma Logging

When a borehole was drilled and all spoil material removed, the hole was gamma logged at every 6-in. increment to determine the rate of gamma radiation at varying depths. Gamma logging was accomplished by inserting a gamma radiation sensing probe into the borehole to determine radiation levels. The radiation measurements recorded during gamma logging were used as precursors for selecting soil samples to analyze and as indicators for depth of contamination.

Soil Sample Collection and Analysis

Soil samples were taken for each 6-in. increment of borehole drilled. Soil samples were selected to isolate the boundaries between contaminated and clean soil. Normally samples were taken after the downhole gamma logs were evaluated and soil samples were screened. Field screening of samples allowed for a quick count gamma measurement and comparison to the downhole gamma log to ensure that the sample is representative of the area and was not contaminated by material falling into the hole during the drilling. Soil samples were analyzed for Radium-226, Radium-228, Uranium-238, Thorium-230, Thorium 232 and Potassium-40.

REMEDIAL DESIGN

Designs were prepared to remove all material exceeding criteria. Preparation of remedial designs required the collective interpretation of gamma radiation surveys, downhole gamma logs and soil sample analysis.

A remedial design for a property included preparation of an excavation plan, restoration plan, sections, landscape schedule and subsurface borehole profiles. The excavation plan outlined the lateral limits of excavation and the depth of excavation; the restoration plan showed features which were to be restored by a contractor; sections of the excavation areas were taken to show actual depth of excavation in relation to existing ground lines; landscape inventories identified trees and shrubs which were to be restored; and borehole profiles showed the type of soil material encountered during drilling.

Excavation Plan

When defining the contamination boundaries, all data gathered during the RD field investigation were considered. The walkover gamma radiation surveys were used as guides in establishing the horizontal limits of contamination. In many cases, contaminated areas were delineated by elevated surface readings and were used to establish design excavation limits.

A detailed civil survey drawing was prepared for each affected property. Civil survey work consisted of preparing grid, contour, boundary and planimetric surveys. The excavation plan showed areas of known contamination exceeding cleanup standards. Excavation was designed to be conducted in two stages - stage 1, defined as primary excavation, and stage 2 as secondary excavation. The secondary excavation cut line represented the anticipated limit of material removal, to be verified by instrument readings during excavation. This cut line was controlled by the depth of contamination detected in boreholes.

Cross Sections

Cross sections were taken across areas of excavation to show depth of excavation in relation to the existing ground line. The sections consisted of the following elements: existing ground line, primary excavation line, and secondary excavation line.

Sections were always cut to intersect boreholes. This was necessary since the borehole data was the baseline for preparing the secondary excavation cut line. The secondary excavation cut line was created by drawing a line through the bottom of contamination (BOC) elevation in each borehole. The primary excavation line was then delineated within the secondary line to define the amount of contaminated material that could be removed before beginning the more time consuming excavation guided by instruments.

Restoration Plan

A restoration plan was developed for all properties disturbed by excavation. Restoration of a property required the replacement of features such as demolished structures (garages), driveways, sidewalks, curbs, lawns, trees, shrubs, and all other disturbed features.

Restoration plans were created to require a contractor to restore the property to preconstruction conditions. In virtually all cases, the appearance of the restored areas was superior to that of the preconstruction conditions because of the contractor's workmanship.

STRUCTURE REMEDIATION

The objectives of the design process were to develop innovative designs to eliminate the need and expense of demolition and cribbing methods while exploring the most cost effective and practical techniques of remediation.

Basement remediation, by common construction methods, would require support of a house by cribbing and demolishing the basement foundation walls, footings, and slab. Deep excavation to remove contamination beneath on-grade structures (e.g., porches, decks, or add-on rooms) typically requires demolition of the structure. However, the

use of innovative underpinning designs allowed remediation of structures and surrounding soils while limiting the required restoration.
Underpinning residential structures offered many challenges, both in RD and Remedial Action (RA). A pre-design reconnaissance consisting of inspecting the structural elements to determine their condition, size, and configuration was mandatory for a successful design effort. Since the construction, configuration, and condition of structures can vary greatly, the designer had to approach each structure as a new and separate problem. On the other hand, it is advantageous to develop solutions based on common elements that can be easily adapted for many structures in order to simplify construction.
Basement Remediation
The houses on these sites are typically wood-framed, multi-story construction resting on concrete masonry unit (CMU) basement walls. The method used to remove contamination from basements depended on the source and location of the contamination as well as the configuration and condition of the structure. Contaminated fill material was usually found either along basement walls or beneath slabs and footings. Remediation in these cases rarely required demolition of structural elements other than the basement floor slab. As previously noted, one of the primary contaminants at the sites was an ash fill. This ash was often used as backfill material and, on occasion, used as an admixture for concrete and mortar. These contaminated building materials were used in the construction of some residential structures.
Removal of the basement floor slab due to inherent contamination or to allow access to contaminated subgrade material proved to be the simplest of the required designs. The interior tubular supports often were founded on the slab itself, necessitating use of temporary supports while the existing supports were removed.
When the source of contamination was fill material, it was not unusual for that contamination to extend to a depth greater than the bottom of footing (BOF). Provided that the foundation or the basement wall was not contaminated, it was possible to remove the material under the footings without demolishing the walls. This was accomplished by excavating a series of alternate three-foot long trenches beneath the footing, while always leaving two trench lengths between open trenches. The trenches were excavated to BOC, then a concrete pier was poured beneath the existing footing to create a new foundation bearing at BOC. This method was used for trench excavations up to 3 ft. deep.
When it became necessary for a partial or entire wall to be demolished, there were many solutions that could have been used. Simplicity, cost, and safety were the driving forces in developing the most practical method for each house. When a major support wall (i. e., one carrying loads from the first floor joists) was demolished, the simplest underpinning method was to provide alternate temporary support for the joists.
For minor support walls (i.e., carrying only loads from upper story walls) and when framing configurations prohibited transfer of wall loads to joists, it was necessary to develop an alternate method. In these cases a concrete lintel was constructed over the walls to be demolished. Although this method can be labor intensive, the lintel allows demolition of large sections of wall.
On-Grade Structure Remediation
The variation in the construction of on-grade structures prohibited many generic solutions. Increased difficulty of structural characterization, magnitude of loads, and depth of anticipated remediation led to a comprehensive design approach to the support of add-on rooms.
Original methods developed required the excavation of shored pits. Once constructed, timber columns and wood needle beams were installed in place of the existing support system. The temporary supports and needle beams removed load from the piers. As design methods were continuously analyzed and improved, a new system was developed using augured piles with I-beams in place of the shored pits. Use of this method should be cheaper and safer than the shored pit method.
PROCESS EVALUATION
Throughout the remedial design process, the methods used to obtain data, interpret data, apply the data to design, and prepare the RD were evaluated to ensure the methods were appropriate.
The investigatory process used to obtain data (i.e. Exterior/Interior Gamma Radiation Surveys, Subsurface Drilling, Borehole Gamma Logging and Soil Sample

Collection and Analysis) has been successful. These data proved to be predictive when used to determine the presence of contaminated material and to define vertical and lateral boundaries of contamination. However, these methods of investigation were not successful in locating contaminated basement foundation wall footing material. When an interior gamma scan was performed across the basement slab, elevated gamma readings were attributed to and usually verified by soil sample analysis to be either a contaminated concrete slab, contaminated subgrade material beneath the slab, or a combination of both. RA work has shown that some foundation wall footings are also contaminated, presumably from using contaminated sand in the concrete mix. Subsequent interior investigations included use of other test parameters, such as interior radon working levels, as indicators to target potential sources of contamination. In addition, methods to investigate existing footings were employed. This was accomplished by core drilling a 12-in. diameter hole in the basement slab adjacent to a foundation wall. The subgrade material beneath the slab core was removed by hand until the footing was exposed. Once the footing was exposed, the footing was scanned for gamma radiation and a sample taken. This procedure has been successfully executed by field personnel.

Another area of investigation that evolved throughout this project was interpretation of data results from foundation wall gamma scans. In some cases, the elevated wall readings were attributed to shine from elevated slab readings, elevated exterior material backfilled against the wall, or a combination of both. To verify if the wall material itself was contaminated (wall material is typically concrete masonry block with mortar joints), a hammer drill was used to core a 2-in. diameter sample. Analysis of the sample confirmed if suspect building material was contaminated.

The methods for removal of material within basements, on-grade structures, etc. were used successfully during RA. The contractor was able to utilize the systems without incident. The support methods that used steel instead of timber members proved to be more cumbersome to construct than expected due to confined work areas and weight of materials.

The wall demolition methods (concrete lintel and joist support) have worked well. Entire and partial foundation walls were demolished with these systems in place. When full height wall demolition was required (i.e. from top of slab to bottom of joist) the exposed footings were often found to be structurally unsound for use. As a result, the contractor usually had to remove the footing section corresponding to the demolished wall section and reconstruct the footing and wall section. Subsequent design work anticipated existing structural conditions that were not suitable for restoration of structural elements.

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27-48

IDENTIFICATION OF MICROBIAL BYPRODUCTS INVOLVED WITH TRANSPORT OF URANIUM IN SATURATED SOIL

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ABSTRACT

Filtrates from Aspergillus niger and Penicillium simplicissimum mold cultures can remove 80-90 % of the uranium in Fernald soil when used in a soil washing process. Analyses of the mold filtrates were done to determine the basic components responsible for the uranium transport. High performance liquid chromatography analysis of the mold filtrates showed the primary organic acid components to be

citric, gluconic, oxalic and malic. Uranium removal was correlated with individual acid concentrations and gluconic acid showed the strongest correlation (.56) of any of the single acids. Slightly stronger correlations were shown for all acids combined (.62) and gluconic and citric acids combined (.61). Similar experiments using solutions made to contain 0.1% of individual acids in DI water showed in decreasing order malic, citric, gluconic, and oxalic to be effective in removing uranium. Solution filtered from soil washing experiments was separated by ultrafiltration (3000 dalton) to determine the contribution of small organic molecules to uranium transport. Preliminary results indicate 60-80% of the uranium is in the less than 3000 dalton fraction which contains the organic acids. The 20-40% of uranium which did not pass through the filter suggests that additional compounds other than organic acids are involved in uranium transport from soils. These data will be used to further develop and optimize the soil washing process and provide insight for transport phenomenon important to risk assessment strategies.

INTRODUCTION

The Biotechnology group at Idaho National Engineering Laboratory has been developing a bioprocess for removal of uranium contamination in soil at a former feed-metals plant near Fernald, Ohio (1). In light of the large volume of soil and low level of contamination at Fernald, a large-scale on site washing plant has been proposed. The process uses a solution of metabolic by-products filtered from broth cultures of common molds (referred to as mold filtrate in this paper) to wash the soil. Compounds in the mold filtrate compete with soil particles to form a mobile complex with uranium which moves with the liquid wash phase which is removed from the soil by filtration. This soil filtrate solution can then be treated to remove and concentrate the uranium. Advantages of the system are that it does not subject the soil to extremely acidic or basic conditions, so soil structure is preserved and it uses biodegradable mobilization agents.

Experiments conducted at INEL have shown filtrates from Aspergillus niger and Penicillium simplicissimum mold cultures grown in malt glucose solution are effective at mobilizing uranium from the Fernald soil (See Table I). Remaining to be resolved in the proposed system are issues of separation of complexed metal from solution, optimization of process, and development of waste treatment facilities to handle residual biological oxygen demand created in treatment. This paper addresses the question which is key to all of these issues: Which compounds in the mold filtrate are responsible for mobilizing the uranium?

Several different factors are involved with complexation of uranium in soil treatment. The fact that microbial activities may increase mobility of metal ions has been well documented (2,3,4). Microbes excrete a wide range of biochemicals that may complex with metals. Some of these are "chelating" agents which specifically increase metal concentration in solution and thus increase availability of trace metals necessary for growth (5). Other excreted chemicals may be simply by-products and only incidentally form complexes. Additionally, the presence of actively metabolizing microbes has an effect on the local electrochemistry and may change the soil properties such that metals attach to small particles to form micelles which persist in solution (6,7,8). This paper presents work done at INEL in FY 94 directed at separating the filtrate used in soil washing to determine the degree to which different size fractions and chemical components contribute to mobilization of metals.

This work has potential for application in many situations where similar forms of metal contamination are a problem. It also contributes to the growing body of knowledge related to microbial influence in geochemical cycling (9). Perception that these contributions are globally significant has increased very much in the last ten years (11,12). This knowledge has interested the mining industry and has potential as an extractive technology and as a remediation tool (10,11,13,14). Of particular interest for those involved in radiological waste management, the mechanisms discussed in this paper and the related electrochemical phenomena are of great importance in nuclide mobility (15, 16, 17). Understanding of these mechanisms is key to risk assessment, to containment of wastes, and to regulation of remediation.

Materials and Methods

Mold Filtrate and Soil: Aspergillus niger and Penicillium simplicissimum were cultured aerobically in broth initially containing 20 g/l malt extract, 20g/l dextrose, and 1 g/l peptone for 2 to 12 weeks depending on the experiment. The cultures were vacuum filtered to remove cellular material, and the mold filtrate was

collected. The pH of the mold filtrates range from 4 to 6. The initial concentration of uranium in the soil was 500 mg/kg. 20 mg of soil was placed in 250 ml Erlenmeyer flasks and 100 ml of extracting solution was added. The soil/solution mixtures were place on a model G24 Environmental Incubator Shaker (New Brunswick Scientific, Edison, NJ) set at 200 rpm. Experiments were done at room temperature, approximately 22oC; the pH of soil/solution mixtures ranged from 5.5 to 7.0.

Uranium Analysis: Uranium concentration in all soil filtrate samples was determined by inductively coupled plasma (ICP) with atomic emission spectroscopy analysis (Applied Research Laboratories, Sunland, CA). Samples were drawn from experiments and acidified (6.25% HNO3). It was found that filtrate from soil washing using some broth filtrates, particularly those from Aspergillus niger cultures, were extremely difficult to filter in preparation for analysis, and even after filtration these same mixtures could cause obstruction of the ICP burner nubulizer. Therefore, all samples were routinely refluxed with gentle heating for one hour before filtration. Although this process did not improve filterability much, it did eliminate difficulties with the ICP.

Size Separation of Soil Filtrate: Soil filtrate was separated into greater than and less than 3,000 dalton size fractions to determine the contribution of colloid and soil particles in solution to uranium transport. 20 g of soil was mixed with 100 ml of mold filtrate for 24 hrs (see soil and solution mixing section). Two 15 ml aliquots of soil filtrate were removed and spun separately at 1000 G to remove slow settling particulates. The supernatants were then placed in separate Centriprep model 3 (3,000 dalton molecular weight cut-off, cellulose filter) concentrator tubes (Amicon, Beverly MA). The tubes were spun at 3000 G in a Sorvall RC-5B centrifuge with an HS-4 rotor (Newtown, CT) and repeatedly drained until all liquid had passed through the 3000 dalton membrane. These experiments were conducted in duplicate using mold filtrate from four different mold cultures. Penicillium simplicissimum was grown in Difco malt extract broth (Ps 4/9) and glucose/malt sugar broth(Ps 9/27). Aspergillus niger was grown in Difco malt extract broth (An 4/9) and glucose/malt sugar broth (An 9/27). The cultures were significantly different in composition as determined by high performance liquid chromatography (HPLC) analysis, so conclusions could be made for a spectrum of fermentation types. Uranium concentration was determined for the non-fractionated soil filtrate and the filtrate (<3,000 daltons) fraction (see uranium analysis section).

Organic Acid Content of Mold Filtrate: Samples of broth filtrate from mold cultures were taken before application to soil and the small organic acid contents identified by HPLC separation (Alltech 325 pump and Linear UVIS 200 detector (Fremont, CA) set at 210 nm. The column used was an Aminex HPX-87H). Concentration data from repeated HPLC runs and uranium concentrations obtained from ICP analysis of extract from corresponding soil washings were compared by regression analysis (n=16 for oxalic, citric, and gluconic acids, n=2 for malic acid).

Synthetic Mold Filtrate and Soil: To verify correlation of uranium removal with specific acids, synthetic "mold filtrate" solutions consisting of deionized water and acids were made and applied to soil in the same manner as mold filtrate. Citric, gluconic, malic, and oxalic acid solutions at a concentration of 0.1 wt% were prepared. A solution of citric and malic acids (0.1 wt% each) was also prepared. The pH of the acid solutions ranged from 5.5 to 7.8. Acid concentrations were based on the range of those found in mold filtrate to aid in the comparison of results. All soil washing experiments were conducted in shake-flasks using 100 ml of solution and a 20% soil (dry weight) to solution ratio. Flasks were agitated for 24 hours, 200 rpm, at room temperature before sampling for uranium. Experiments were done using single and triple applications of synthetic mold filtrate as per the experiments with mold filtrate. The pH of the soil/solution mixtures ranged from 6.5 to 8.

Results and Discussion

Soil washing is an effective approach for removing uranium from soil, but the chemicals and conditions employed often subject soil to extremes in pH which can damage soil. Soil washing using mold filtrate from Aspergillus niger or Penicillium simplicissimum cultures can remove greater than 90% of the uranium in Fernald soils (initial uranium concentration is 500 mg/kg) while maintaining the pH of the soil in the 6-8 range. No uranium removal was seen when uninoculated broth was applied to the soil. Optimization of the soil washing step and development of the overall uranium removal system requires identification of the active components of the mold filtrate. This work identifies some of the specific active components and provides

evidence of some additional active components that are not yet fully identified. There was no significant difference between mold filtrate from Aspergillus niger and Penicillium simplicissimum in terms of effectiveness of uranium removal or pH of the soil/mold filtrate mixture (see Table I). Multiple applications of mold filtrate are more effective than single applications, although the majority of the uranium is removed by the first application of filtrate. The pH of the soil/solution mixture does not change with multiple applications of the mold filtrate, suggesting that the additional removal is not a pH effect. Aspergillus niger and Penicillium simplicissimum produce a variety of compounds including organic acids that can mobilize uranium from soils. Optimization of the uranium extraction process relies to a large degree on identifying the types of compounds involved in the uranium mobilization.

Separation by HPLC of filtrate from mold cultures before contact with soil showed citric, gluconic, and oxalic, acids to be the primary organic acids present (see Table II). As would be expected, a complex relationship between culture conditions and organic acid production was observed. In future work, culture conditions (pH, EH, dissolved oxygen, etc.) will be controlled to optimize production of desired components. When filtrate was used to wash soil and the results compared with organic acid composition, gluconic and citric acids were found to best correlate with uranium removal rates (see Table III).

Preliminary results from size separation of the soil filtrate indicate that 60 - 80% of uranium in solution is contained in the less than 3000 daltons size fraction(see Table IV). Four different mold filtrates were applied to separate soil samples and soil filtrate was collected from each and separated into two fractions by a 3,000 dalton filter. The organic acids identified in the mold filtrates most likely account the uranium contained in the less than 3,000 dalton fraction of the soil filtrate. The remaining 20 - 40% of uranium in the soil filtrate is in the greater than 3,000 dalton fraction. This strongly suggests that larger organic molecules play a significant role in transport of uranium through soil. The cellulose membrane used in the separation could also account for a percentage of the uranium in the upper size fraction (especially in the samples with a low concentration of uranium in the unfractionated soil filtrate) but would be unlikely to contain all of the uranium in that fraction (0.02 - 0.16 mg). Cellulose filters were used because they do not readily bind most compounds found in biological media. At this point it is not known whether the larger compounds remove uranium directly from the soil or serve as a sort of mobile sponge or capacitor.

Solutions consisting of organic acids (0.1% w/v) in DI water were used to treat soil in the same manner as with broth filtrate. When extract from washing was analyzed for uranium content, it was found that malic and citric acids were the most effective (see Table I). Overall removal rates were equivalent to those achieved with broth filtrate. A combination of malic and citric acids did not remove as much as would be expected from their individual performances but in fact did only slightly better as a pair. This was interpreted as an indication that complexation of different small organic acids takes place with the same mineral form of uranium in a competitive manner. This may explain the discrepancy between synthetic and fermentation solutions. Repeated treatments with combined citric and malic acids gave extractions equivalent to those achieved with filtrate form mold cultures. Oxalic acid was the least effective at removing uranium and presented the same kind of filtering problems encountered with Aspergillus niger extracts. Avoiding fermentation conditions or organisms which produce oxalic acid will yield more easily handled soil filtrates.

CONCLUSIONS

Significant amounts of uranium are transported from soil by solutions containing microbial by-products. Under the current experimental conditions, most mobility was attributed to small organic acids, however higher molecular weight molecules dissolved soil components may account for a significant fraction. Culture conditions effect solution components and can have important effects on mobilization. Mineral form of contamination likely has a bearing on rate and amount of mobilization. These conclusions support the hypothesis that microbial activities play important roles in distribution of metals in the environment. There is a potential to use this knowledge for control and removal of contamination in soil, and the significance for risk assessment is clear.

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27-49
A LASER INDUCED FLUORESCENCE TECHNIQUE FOR URANIUM TRANSPORT CHARACTERIZATION
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ABSTRACT
The feasibility of using laser-induced fluorescence (LIF) to detect small concentrations of uranium in wet sand was investigated. The application of this technique will be to observe the migration of uranium in saturated and unsaturated flow systems. This would compliment experiments looking at fingered (unstable) flow of water in certain types of unsaturated systems of importance to the Yucca Mountain Site Characterization Project (YMSCP).
Experiments were performed in which a UV laser pulse was used to excite aqueous uranium (as uranyl ion, UO22+) in a wet sand mixture. The characteristic green fluorescence (actually a long-lived phosphorescence) emitted by uranium in the decay transition, filtered with respect to frequency and decay time, was observed as a function of uranium concentration, solution chemistry, and solid-liquid contact time.

The specific goals of this study were to determine the detection limit of uranium in wet sand under appropriate (realistic) conditions, and to explore the feasibility of effective calibration of the LIF signal with uranium concentration at the ppm level. It was found that in addition to interference from autofluorescence of the sand and other materials, a key issue is the effect of the aqueous chemistry on uranium fluorescence strength, particularly with regard to pH. Therefore, quantitative measurements of uranium fluorescence signal strength in aqueous solutions without sand were also conducted, in order to map pH dependence.

Experiments revealed that in nitrate/carbonate solutions, uranium fluorescence goes through a maximum at a pH of 6.0-6.5, and decreases by orders of magnitude at 1.0-2.0 pH units away from this peak. The bright green emission or "superfluorescence" observed in this range is believed to be the result of one or more uranyl complexes. In experiments with wet sand, resolution of 1 ppm uranium from background was achieved.

MOTIVATION

In order to assess the health risk that would result from geologic burial of spent fuel, it is important to be able to predict the travel times for hazardous, long-lived radionuclides from a repository to the accessible environment. The mostly likely release scenario is thought to be groundwater infiltration into the repository resulting in dissolution of the waste form and migration of radionuclides with the water. It is expected that many of the dangerous, long-lived isotopes, particularly uranium and the transuranics, will migrate slowly due to sorption by clays, zeolites, and other sorptive material in the rock surrounding the repository. Connecting fractures of various sizes may be the dominant flow path through the unsaturated layer from the repository to the aquifer, and these fractures are also thought to contain sorptive substances that will retard the movement of the relatively dilute contaminants in the groundwater.

Predictions of radionuclide migration time often take the form of calculating the groundwater velocity and some form of retardation factor for a particular element in a particular ground material. Typically this factor is calculated from a partition coefficient which is found experimentally by looking at the equilibrium distribution between dissolved and sorbed amounts of the element in batch or column systems. Work at Sandia National Laboratory has demonstrated that downward flow under certain unsaturated conditions in a fracture, simulated by crushed rock between glass plates, can deviate strongly from uniform flow (1). When the ratio of total flux to saturated conductivity is less than one, flow instability causes the growth of "fingers" (preferred hydraulic pathways).

Fingered flow results in increased water velocities at the fingertips, and drains the bulk of the flow from the overlying layer. This has implications for the behavior of a contaminant in such a flow system. The question arises as to whether a contaminant would tend to move with the fingers or stay behind with the bulk water front, since "fingered migration" could significantly reduce radionuclide travel time from the repository to the aquifer located beneath.

An experimental technique for detecting minute uranium quantities in a wet sand mixture can be adapted to examine the behavior of aqueous uranium in a simulated groundwater flow system. In particular it could be used to investigate uranium movement in fingered flow; data in the form of uranium concentration profiles at several locations could be taken in conjunction with observations of the flow field, allowing characterization of fingered migration under various conditions of flow rate, pH, ground material, ionic strength, and other key parameters.

LASER INDUCED FLUORESCENCE (LIF) OF URANIUM

While there are many radionuclides of concern regarding long-term geologic burial, uranium was used in this experiment for the following reasons: uranium decay chains include daughters important for health risk assessment, particularly Ra-226 and Rn-222; uranium dissolution and movement in groundwater is thought to be representative of the actinides; under most conditions uranium forms uranyl ion (UO_2^{2+}) in aqueous solution, which can be detected by its characteristic green luminescence upon excitation by ultraviolet light.

The spectroscopic properties of uranyl ion are well known, and several researchers have shown the high sensitivity capable with a fluorescence technique involving intense UV excitation (2-5). One of the most important considerations for achieving this sensitivity is separation of the uranium signal from other induced fluorescence, such as the sample ground material (sand or crushed rock), organic

materials, and structural materials like the sample container. This is accomplished by time-resolution of the signal and frequency selection with a monochromator. Using time-gating, frequency selection, and controlled solution chemistry (acid enhancement), uranium levels on the order of parts per trillion have been detected (2).

Time-resolution is aided by the relatively long excited decay time for radiative transition in uranyl ion, and is the foundation on which the technique is based. Fluorescence lifetimes for most materials tend to be on the order of nanoseconds, while the uranyl phosphorescence decay time is on the order of microseconds. The uranium signal is therefore characterized by a relatively long tail remaining after other fluorescences have died out. The typical uranyl ion emission spectrum has peaks at approximately 494, 516, and 540 nm (3). (Phosphorescence correctly indicates a radiative transition with a much longer mean-life than fluorescence, however the word fluorescence will be used exclusively hereafter for simplicity).

EXPERIMENTAL SETUP

Figure 1 shows a schematic of the experimental setup. The excitation source was a tripled Nd:YAG laser (355 nm) operated at a range of 10-30 mJ per pulse. The sample chamber was black-anodized aluminum which has very low autofluorescence and stray reflection. Samples were held in square quartz cuvets (17 mm inner side length) which had low autofluorescence and minimized beam scatter. Emission light left the sample chamber, passed through a monochromator (5 nm resolution), and was collected by a photomultiplier tube (PMT). Colored glass filters were also used at the ports of the sample chamber for stray light protection. A monochromator setting of 520 nm, which corresponds to the central emission peak, gave the best signal-to-noise response.

The signal from the PMT was sent to a digitizing oscilloscope (20 MHz bandwidth); data acquisition and analysis was performed on LabviewTM for Macintosh. A shot consisted of a single pulse from the laser, which sent a synchronous trigger to the oscilloscope, programmed to wait a specific delay time before data collection.

For shots on sand, the beam illuminated the sample at a spot on one face, with the cuvet oriented at 45o (see Fig. 1). Beam penetration depth was judged to be on the order of a few grain diameters (1-2 mm). Acid-washed Wedron 510 sand of known mesh size was used. Even after HCl treatment the sand was observed to provide noticeable background fluorescence. Wet sand was prepared by adding (typically) 10.0 g of sand slowly to 20.0 ml solution to ensure a well mixed batch with no dry spots. For shots taken through water (no sand), the cuvet face was perpendicular to the beam, so the full path length was illuminated.

Titrations were performed by adjusting the pH of 200 ml solution with sodium hydroxide or sodium bicarbonate and nitric acid, and transferring 20 ml between bulk and cuvet for each fluorescence measurement. In all cases a blank was prepared and tested in a fashion identical to the corresponding uranyl solution, except that instead of uranyl nitrate it contained an amount of nitric acid equivalent to that in the uranyl solution.

RESULTS

We have found that solution chemistry, especially pH in the range 5-8, strongly affects the LIF signal of uranium. This is crucial since the pH range of interest for contaminant migration at Yucca Mountain is about 7-9. The effects of enhancers and quenchers have been extensively reported (e.g. (4,5)), but only for highly acidic solutions, so this phenomenon was unforeseen. (An enhancer is an acid, typically phosphoric, which forms uranyl complexes with ligands that block nonradiative decay. A 10% phosphoric acid solution can increase uranyl LIF intensity by three orders of magnitude. Quenchers, on the other hand, are ions which aid in nonradiative relaxation such as vibration.)

In order to use the LIF technique, either for laboratory or in-situ observations, it is therefore necessary to calibrate fluorescence signal with pH. Due to the buffering and sorptive properties of the sand, it further requires a means by which to measure local pH in a solid/aqueous system. A demonstration of this type of measurement was carried out with fiber optic pH sensors that are currently available but were not obtained for this study.

The calibration of fluorescence with pH, therefore, became a primary objective. Figure 2 shows three fluorescence vs. pH "maps" for 1 ppm uranium; trial 1 (squares) indicates a solution titrated with sodium hydroxide which had been exposed to air for six days, trial 2 (triangles) indicates the same solution after being exposed to

air for three weeks, and trial 3 (circles) indicates a fresh solution titrated with sodium bicarbonate. Each data point in the figure represents the time-integrated average luminescence of ten identical "shots" using the following time-gating: delay time (To) was 10 microseconds, and total scan time (Ts) was 50 microseconds. Background (as tested with blanks) for all these runs provided negligible contribution and was ignored.

Several maps such as these were generated to test reproducibility of the results. It was generally observed that in systems containing uranyl nitrate and sodium hydroxide and/or sodium bicarbonate, uranium exhibits a "superfluorescence" in the pH range 5-8 which peaks at pH 6.0-6.5. As seen in the figure, however, while generally consistent, the maps were not quantitatively reproducible.

It seems evident that CO2 has an effect since, as seen in Fig. 2 from the difference between trial 1 and trial 2, the fluorescence gains as much as an order of magnitude (at pH greater than 6.5) upon additional equilibration with air. Furthermore the well-equilibrated solution data agrees quite well with that containing added carbonate (trial 3). Further increases in carbonate concentration had no effect, however, and the role of carbonate is still unclear. It seems reasonable that CO2 level controls the formation of a superfluorescing uranyl complex containing carbonate ligands.

Calibration could likely be improved by more careful control of the solution chemistry. Careful measurement of key species concentrations in solution should resolve the role of CO2 or any other species which can effect the fluorescence. In many cases pH drift during titration was observed, probably due to CO2 exchange. Another possible source of error is precipitation of uranium compounds, since at pH 7-8 uranium solubility experiences a minimum at the ppm level (or below, depending on CO2 level and other factors) (6).

In order to verify that useful calibration is feasible, experiments were performed comparing fluorescence of solutions at pH 6.5 with 0.5 ppm U, 1 ppm U, and 2 ppm U. The results are shown in Figs. 3a-d, in which each curve is again an average of ten shots. The x-axis in these figures covers a total scan time indicated by Ts; delay time between pulse and the beginning of the scan is indicated by To. The background (i.e. blank) is not shown in these figures as it was very small (less than 0.04 V after 1 microsecond); thus the resolution in this pH neighborhood is considerably better than 0.5 ppm uranium.

As seen in Fig. 3a, 1 ppm U and 0.5 ppm U are easily distinguishable and the 0.5 ppm U signal is quite strong. The initial (first microsecond) shape of the curves is affected by background, after which a long tail persists for more than 50 microseconds, as seen in Fig. 3b. Figures 3c and 3d compare the same short-time and long-time behavior for 1 ppm vs. 2 ppm uranium. Sensitivity is clearly much better than the factor of two concentration differences used here. The integrated signal between 10 and 50 microseconds, as used above, appears to be a useful measure of fluorescence strength.

Preliminary tests with wet sand were made to verify that the uranium fluorescence can be resolved from the background (autofluorescence) imposed predominantly by the sand. Examples of the results are shown in Fig. 4a-d for shots of 1 and 10 ppm U vs. no uranium (blank) in sand. The 1 ppm U data were single shots taken with To=3 microseconds and Ts=10 microseconds; the solution was initially (prior to adding sand) pH 6.6. The 10 ppm U shots, for which the solution was initially pH 7.1, have Ts=10 and To=0 so the entire curve is shown.

Figure 4a shows that 1 ppm U is discernible above background from sand, but in these conditions 1 ppm is probably near the detection limit. Nevertheless it essentially demonstrates the feasibility of taking measurements with shots on sand. Figure 4b shows the difference between sample and blank and indicates an appropriate exponential decay constant of several microseconds. Figs. 4c and 4d show that the difference between sample and blank is considerably more exaggerated for 10 ppm U.

CONCLUSIONS

It has been shown that in systems containing uranyl nitrate and sodium hydroxide and/or sodium carbonate, uranium exhibits a superfluorescence that exists in the pH range 5-8 and peaks at pH 6.0-6.5. Signal calibration is required due to sensitivity to pH, CO2, and possibly other aspects of solution chemistry.

The results of this study indicate that the LIF technique is suitable for investigating fingered migration and for in-situ measurement provided that the local pH is known for calibration of the fluorescence signal. Thus capability for

measurement of pH in wet sand or rock, for which fiber optic pH probes are available, will allow dependence on this parameter to be accounted for.

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27-52

CHERNOBYL - A SOLUTION FOR THE CLEAN UP OF HIGHLY CONTAMINATED FORESTS AND WOODLANDS

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ABSTRACT
This study corresponds to a remediation action of contaminated woodlands and forests due to the fall out of radionuclides after the CHERNOBYL accident. This remediation action implies both incineration of contaminated wood products and foliage and the transformation of clean wood into paper pulp for the industry.
Based on existing industrial products, this remediation action can be justified by the sale of by-products such as electricity and card board which can pay for the necessary industrial investment.
The pay back return is estimated to be obtained in about five years.

INTRODUCTION
The purpose of this study is to present an industrial demonstration unit able to process wood contaminated by the fallout of radionuclides, particularly Cesium 137 (half life 30 years) from the accident of CHERNOBYL Nuclear Power plant Unit 4.

50,000 km of contaminated forests have to be cleaned up from cesium, strontium but also hot particles contamination PU241, which present a real danger for the environment,

Contaminated woodland and forests would be cleaned up by cutting up and chipping the contaminated wood or bark, converting it into small-size waste (chips measuring three to five centimeters), and incinerating the contaminated portion to generate steam and electricity in a waste heat recovery boiler associated with the fluidized bed incinerator,

The sound portions of the wood be converted into building materials or wood pulp by twins-crew extruders. These extruders are particularly economical in their use or energy and produce little secondary waste.
Several solutions could be used for treating wodd chips and for removing the residual contaminaton due to the Cesium 137.

PRINCIPLE OF WOOD TREATMENT
The principles chosen for wood treatment will follow the following main schemes:
A. Segregation of contaminated wood from/non contaminated wood,
B. Cutting down of contaminated trees
C. Removal of bark from clean wood

D. Incineration of contaminated bark
E. Transformation of clean wood into wood pulp
The contaminated portion of trees (50 to 100 bq/gr, forage, branches and bark could be processed in a circulating fluidized bed incinerator designed to generate steam in an atmospheric fluidized bed incinerator in order to produce electricity.
Secondary organic wastes resulting from top soil decontaminating processes such as the RESSAC process developed by the French IPSN could also be collected and incinerated in a fluidized bed incinerator in order to produce steam for electricity generation.
The central portion of trunks which generally present low contamination will be chipped into small pieces in a sawmill and will be used as feed for producing wood pulp for the cardboard industry.
FLUIDIZED BED INCINERATION AND STEAM PRODUCTION
The fluidized bed incinerator and the associated steam production unit include the following components see (Fig. 1).
A fluidized bed incinerator into which are injected the wood waste chips, the bark and the contaminated litter and also some refractory powder for the fluidized bed support.
A solids separating cyclone, installed at the furnace outlet in the high temperature gas (870C) permits reinjecting most of the solid particles leaving the reaction chamber into the system, with only a very small fraction of the ash produced released with the smoke. This limits erosion constraints in the down-steam lines.
The recovery boiler where the heat of the off gas stream coming from the cyclone is removed.
An external heat exchanger feed with fluidized solids removed from the bottom of the solids collected from the cyclone and cools it before their reinjection into the furnace. The distribution between hot and cooled recycled solids permits maintaining the desired temperature in the furnace.
The stability of the circulating bed implies maintaining a large quantity of solids recirculating. The quantity recirculated affects the differential pressure between the top and the bottom of the furnace. Controlling this recirculation is obtained either by regulating the rate of removal of the dusty ash or by regulating the wood injection.
For the incineration of contaminated wood and organic wastes, it is assumed that a large scale fluidized bed incinerator (6 tons/hour of contaminated wastes) is necessary.
This incinerator, as defined in Fig. 1, is coupled to a waste heat recovery boiler for superheated steam production which could be used in a steam turbine for electricity generation.
Protection against the dispersal of radionuclides is provided by a bag house filter located on the off gas stream at the outlet of the waste heat recovery boiler.
The bag house filter is equipped with high retention efficiency bags made of GORE-TEX material having an efficiency of 99.9% for particles of 0.2 to 0.3 m. The GORE-TEX felt is able to operate at 240C and the bags are periodically cleaned by pulsed air.
The incinerator operates a slightly negative pressure which is an additional safety feature.
WOOD PULP PROCESS FOR CLEAN WOOD
The twin-extruder contains two intermeshing and corotating screws are identical and consist of a series of modules each module having a conveying section (right-hand thread) and reverse thread (left-hand thread) section retaining the fibrous matter. Slots and windows are cut into these latter sections. The modules are added, one after the other, on two splined hafts which provide the necessary rigidity.
The thread profile, the right and left hand thread pitches, and the number and which of the reverse thread windows are determined in order to achieve gradual defiberization of the chips and to obtain the required pulp consistency.
The chips fed into the extruder are picked up by the first conveying section of the first module. At the first left-hand thread, a "plug" forms which is forced through the windows. The pressure and induced heating involved in this operation cause the defiberization to begin.
For deciduous and coniferous wood chips there are already two units in operation the most recent started up in February 1992 at the P. Jacquemin cardboard factory in the

Vosges, France. This twin-screw extruder has a capacity of 5 tons/hr of wood waste for cardboard pulp as shown on Fig. 2.
This wood pulp obtained from twin screw extruders is then transformed into high quality calibrated cardboard for the wrapping industry by the means of standardized equipment used in this industry, such as high speed refiners, steam dryers and rollers.
The significant advantage of using twin-screw extruders for paper pulp processing is a less energy consumption: 400 kwh saving per ton of paper pulp and a possible recycling of liquid effluent generated. Various low pollution of bleaching agent can be used such as ozone for obtaining high quality paper pulp.

CONCLUSIONS

The main conclusions of this study are that solutions for the clean up of contaminated woods and organic wastes exist:

Incineration of contaminated wood can generate heat recovery and steam production with environmental protection,

Off gas filtration can be achieved with a high degree of environmental protection. Even with high levels of contamination the emission releases are within the authorized limits.

Clean wood or low contaminated wood can be processed into wood pulp with appropriate decontamination and effluent treatment.

Sounds wood can be used also for construction timber.

Generated energy and wood pulp production can solve economic problems.

It seems that using fluidized bed incineration the wood pulp and timber industries can provide substantial benefit and revenues allowing the possibility of investing in equipment for the cleanup of Chernobyl contaminated areas (see Fig. 3).

ACKNOWLEDGMENTS

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27-54

INNOVATIVE INITIATIVES FOR ENVIRONMENTAL RESTORATION AT U.S. DEPARTMENT OF ENERGY FACILITIES

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ABSTRACT

Environmental restoration activities at U.S. Department of Energy (DOE) facilities are typically governed by a compliance agreement between DOE, the U.S. Environmental Protection Agency (EPA), and the applicable state regulatory agency. These agreements outline the structured process and timetable for remedial decisions and actions. Using the compliance agreement at the Rocky Flats Environmental Technology Site (RFETS) for typical examples, potential streamlined approaches for remedial decisions within the regulatory framework are discussed. Successful implementation of streamlined remediation approaches requires close coordination with the regulatory agencies. Application of these streamlined approaches will allow earlier

implementation of remedial actions at DOE facilities, earlier reductions in human health and environmental risks, and cost savings. Cost savings in environmental restoration continue to be a high priority with the increasing pressure on government budgets. These cost and schedule savings will allow continuing progress in clean-up at DOE facilities while meeting the commitments of the compliance agreements.

INTRODUCTION

Environmental restoration activities at U.S. Department of Energy (DOE) facilities are typically governed by a compliance agreement between DOE, the U.S. Environmental Protection Agency (EPA), and the appropriate state regulatory agency. The compliance agreement at the Rocky Flats Environmental Technology Site (RFETS) is commonly known as the Interagency Agreement (IAG). The IAG documents the commitments between DOE, Region VIII of the EPA, and the Colorado Department of Public Health and the Environment (CDPHE) with respect to the strategy and timetable for environmental restoration activities.

The RFETS IAG identifies a structured process for regulatory review and approval of planned remediation, and the schedule for the planned activities related to each Operable Unit (OU). The structured process is a sequential step-by-step approach based on methodical investigation and characterization, baseline risk assessment, and feasibility study to result in remedial decisions. Each step in the structured process includes regulatory review and public review as appropriate. The compliance agreements are also based on DOE development of a proposed remediation, followed by independent regulatory review. A typical timetable for an RFETS OU, as outlined in the IAG, is shown in Fig. 1.

There are opportunities to use streamlined approaches within the regulatory framework, adapted from successful applications at other facilities, to accelerate the selection, review, and approval process for planned remediations. Acceleration will allow earlier identification and implementation of the planned remediations and thereby will more quickly reduce risks to human health and the environment. Streamlined approaches will reduce the costs of investigations, studies, and analyses, and thereby allow a greater proportion of the limited government budgets to be directed toward implementing actions and reducing risks.

Streamlined approaches should include close coordination and interaction with the regulatory agencies for identification of regulatory requirements and interpretations, coordination of the approaches and alternatives to be considered, concurrence with the methodologies for screening, analysis, and evaluation, and development of the details of alternatives and recommendations. The benefit of close regulatory coordination, in early identification of regulatory requirements and issues, has been demonstrated at RFETS and at other facilities. Close coordination also provides a forum for early resolution of issues. Streamlined approaches that should be considered with close regulatory coordination include the "analogous site", "risk-based decisions", and "presumptive remedy" approaches.

ANALOGOUS SITE APPROACH

OUs have been typically identified in the compliance agreements based on functional areas or geographical locations. OUs such as those identified in the RFETS IAG have not been coordinated to consolidate the planned remediations for similar contaminants in common media (soils, groundwater, etc.). The "analogous site approach" makes beneficial and continuing use of the characterization and feasibility study information from earlier OUs, and applies the selected remediation alternative to later OUs determined to have similar contaminants in the same media. Several DOE facilities including RFETS have initiated efforts to develop sitewide Preliminary Remediation Goals (PRGs), to perform sitewide technology evaluations, and to develop a consistent framework for conducting feasibility studies for all of the OUs. A more progressive and streamlined approach would apply the results of early feasibility studies to all OUs with similar contamination.

As an example, an OU with plutonium contamination in surficial soils (within predetermined bounds) could proceed with a planned remediation based on the prior detailed feasibility study, alternative evaluation, and technology selection from an earlier OU. Initial investigation and characterization to select the proper remediation technology would be required, followed by implementation using an "observational approach". The "observational approach" simply applies a procedure of in-process monitoring to determine the extent of remediation. The potential substantial acceleration of the typical remedial investigation (RI)/feasibility

study (FS) process that would result from using the "analogous site approach" is shown in Fig. 2. The extent and efforts required for a feasibility study are significantly reduced. Note that proactive coordination between DOE facilities could ultimately lead to using the "analogous site approach" between DOE facilities, where a successful remedial approach at one facility might be directly applied at a different DOE facility for similar contamination in the same media.

RISK-BASED REMEDIATION DECISIONS

Typical DOE facility compliance agreements, including the RFETS IAG, rely on a Baseline Risk Assessment (BRA) to support remedial decisions. The BRA is generally performed after investigation and characterization of the contamination, and is used to determine the existing risks to human health and the environment, and to determine the remediation objectives. The risk assessment process is well defined and intended to support first the "No Action" alternative, and then to be used as a basis for establishing the PRGs and the Remedial Action Objectives (RAOs). Alternate streamlined risk-based approaches, coordinated with the regulatory agencies, have been successfully used at other facilities to accelerate this process and to provide early determination of the PRGs and RAOs. Once contaminants are identified during early investigation and characterization, or are known to exist based on historical record or process knowledge, risk analyses using conceptual fate and transport models may proceed directly. The Baseline Risk Assessment should not be required if it is known that a "No Action" alternative is not appropriate and that remediation is required. The risk analysis activities can therefore focus directly on determining the PRGs and RAOs based on the promulgated regulatory standards or the acceptable residual risk levels. The "risk-based decision" process makes the basic assumption that remediation will be required, and then determines the extent of remediation to reduce the risks to the acceptable levels.

This streamlining can still support the "No Action" alternative, in that if the risk-based PRGs are higher than the characterized contaminants, then no further action will be required. In addition, the requirements for remediation can be determined before detailed characterization is completed, and alternatives can be considered and evaluated somewhat in parallel to the completion of the characterization. The potential acceleration in the RI/FS process of "risk-based decisions" is also shown in Fig. 2. Acceleration is provided by parallel activities rather than by the methodical step-by-step process typically outlined in the compliance agreements. An iterative approach may also be used to evaluate composite risks from all media if the risk-based PRGs are considered to be too conservative. The risk-based process more directly supports an "observational approach", since determination of the PRGs and RAOs does not require detailed characterization. The development of the PRGs and RAOs relies on identification of the exposure scenarios and the potential receptors in conjunction with a target residual risk level. For example, the RFETS IAG requires a reduction in risk to less than a 1-in-one-million chance of contracting an excess cancer from exposure to the OUs. The exposure scenarios and potential receptors are determined specifically for each OU. Close coordination with the regulatory agencies is important for developing and implementing the "risk-based" approach to remediation decisions.

PRESUMPTIVE REMEDIES

Coupled with "analogous site" and/or "risk-based" approaches, presumptive remedies can also be employed to expedite the start of remediation. Risk-based determinations for contaminants in media generally apply across the facility. Appropriate remediation alternatives, as accepted by the regulatory agencies at an early OU, generally apply for similar contaminants. Historical records and process knowledge provide early evidence of the presence of contamination. Using these sets of information, identification of a contaminant based on historical record and process knowledge leads to an identified presumed remediation alternative with predetermined RAOs. A presumptive remedy can then be initiated while the nature and extent of contamination are being determined.

The potential acceleration of the RI/FS process using a "presumptive remedy" approach is also shown in Fig. 2. Remediation can begin much earlier in the overall timetable. Using "presumptive remedies" in an aggressive manner would more quickly reduce risks, and would reduce costs to allow further actions within the budgetary constraints.

STREAMLINING RISKS

There are inherent programmatic risks in these potential streamlined approaches.

Remediations may be implemented that are conservative, and therefore result in more substantial clean-up then required to meet regulatory, human health risk, or environmental risk requirements. This aspect of the programmatic risk is considered to be small, since remediation requirements at DOE facilities that have thus far been approved by the regulatory agencies have been quite stringent. The benefits of earlier remediation, earlier reduction of health and environmental risks, and reduced remediation costs out weigh the potential risk of overly conservative remediations.

It is possible that early remediations may be initiated that are later determined to be inadequate for ultimate remedies and closures. This represents a programmatic risk which is also considered to be small, since DOE facility remediations are being implemented with regulatory oversight and approval, the clean-up requirements are typically stringent, and most of the potential technologies have been successfully implemented with proven results. These programmatic risks must also be weighed and balanced against the gains of earlier remediations, earlier reductions of health and environmental risks, and cost savings. Close coordination with the regulatory agencies throughout the streamlined remediation process will reduce these programmatic risks.

CONCLUSION

Application of these streamlined approaches at DOE facilities with close regulatory coordination, review, and approval will save time and money. With increasing pressures on government budgets, innovation is warranted to continue to make progress in clean-up at DOE facilities, and to meet the compliance agreement commitments. Streamlining will reduce the cost of investigations, studies, and analyses, and will allow a greater proportion of the environmental restoration budgets to be used to implement clean-up actions and reduce risks. The potential acceleration of the typical IAG timetable for remediation of an RFETS OU using the streamlined "analogous site", "risk-based decisions", and "presumptive remedy" approaches is shown in Fig. 3. Broad application of these innovative streamlined approaches will substantially reduce the overall cost and timetable for remediation at DOE facilities.

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CHARACTERIZATION OF URANIUM- AND PLUTONIUM-CONTAMINATED SOILS BY ELECTRON MICROSCOPY

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ABSTRACT

Electron beam techniques have been used to characterize uranium-contaminated soils from the Fernald Site in Ohio, and also plutonium-bearing "hot particles" from Johnston Island in the Pacific Ocean. By examining Fernald samples that had undergone chemical leaching it was possible to observe the effect the treatment had on specific uranium-bearing phases. The technique of Heap leaching, using carbonate solution, was found to be the most successful in removing uranium from Fernald soils, the Heap process allows aeration, which facilitates the oxidation of uraninite. However, another refractory uranium(IV) phase, uranium metaphosphate, was not removed or affected by any soil-washing process. Examination of "hot particles" from Johnston Island revealed that plutonium and uranium were present in 50-200 nm particles, both amorphous and crystalline, within a partially amorphous aluminum oxide matrix. The aluminum oxide is believed to have undergone a crystalline-to-amorphous transition caused by alpha-particle bombardment during the decay of the plutonium.

INTRODUCTION

The clean-up of radionuclide-contaminated sites is a major problem facing the U.S. Department of Energy (DOE). Knowledge of the soil and contaminant characteristics can greatly facilitate the identification of suitable remediation techniques during

bench-scale tests, which otherwise must proceed by trial and error. More importantly, detailed characterization can allow the rejection of inappropriate technologies for a particular site on a sound scientific basis. At the DOE Fernald site in Ohio, several characterization techniques have been used to support the remediation efforts at both bench-scale and pilot-scale. These characterizations have described the exact nature of the uranium contamination and the effect that various chemical treatments had on the soil and contaminant (1). The application of electron microscopy in characterization studies at Fernald has been reported elsewhere (2-4). In this paper, further examples of soil characterization are presented, with emphasis on how they can benefit remediation efforts.

EXPERIMENTAL PROCEDURE

Soil characterization by analytical transmission electron microscopy (AEM) with electron diffraction provides the spatial resolution necessary to examine sub-micron phases. Using the combination of electron diffraction, X-ray energy dispersive spectroscopy (EDS), and electron energy loss spectroscopy (EELS), unknown phases can be identified. At Argonne National Laboratory (ANL) we have developed a small particle handling facility and analytical microscopy laboratory for the characterization of radioactive materials, soils, and residues. The facility uses micro-manipulation techniques to mount small particles and ultramicrotomy to prepare thin sections of the particles for AEM examination. The ultramicrotome is specially designed so that radioactive samples can be thin sectioned. A combination of optical microscopy and scanning electron microscopy (SEM) assures representative sampling.

Characterization Protocol

Soil characterization is divided into two tiers. In Tier I, the bulk physical soil characteristics are determined by particle sizing, separation, radiochemical and elemental analysis, optical microscopy, SEM, and X-ray diffraction (XRD). More detailed characterization falls under Tier II studies. Here, the oxidation state and speciation of contaminant radionuclides are determined by techniques such as AEM and X-ray absorption spectroscopy (XAS), and, if necessary, atomic and molecular spectroscopies. Often it is the combination of techniques which provide the most useful information on the soil contaminant.

Analysis of Soil Samples by Electron Microscopy

Radionuclide-bearing soil samples were infiltrated with a water-soluble melamine resin for examination in the SEM. The radionuclide-bearing particles were located by SEM backscattered electron imaging. Individual particles were isolated by trimming away excess material from the SEM mount. Thin sections suitable for AEM examination were prepared by ultramicrotomy (2). The ultramicrotome has been specially adapted to handle radioactive materials. The instrument itself is located inside a hood and the operator is required to wear a respirator during operation. Monitoring is carried out after sectioning to ensure that there has been no contamination. This method of sample preparation allows direct comparison of SEM and TEM images, which enables characterization of TEM samples to be representative of the bulk sample. The samples were analyzed in a JEOL 2000 FXII TEM operated at 200 kV and equipped with X-ray energy dispersive spectrometers (EDS) and a parallel electron energy loss (EELS) spectrometer. Phases were identified by a combination of EDS, EELS, and electron diffraction.

RESULTS AND DISCUSSION

Assisting Remediation at Fernald, Ohio

The uranium processing plant at Fernald, run by the National Lead Company of Ohio, was the starting point for weapons manufacture in the United States during the Cold War years. Uranium ore was brought from mines in North America and from the Belgian Congo for processing at Fernald (5). The soils around the Fernald processing plant have become contaminated with uranium after decades of uranium processing, and efforts are now being made to remediate the site. The plant is located 20 miles northwest of Cincinnati, by the Great Miami River. Radiological surveys of the site, conducted using a portable gamma spectrometer, located the plumes of high uranium contamination, typically around 500 ppm uranium. However, a number of highly contaminated regions were also found, where, in some instances, the uranium content was up to 5000 ppm (6). Efforts have been concentrated at two regions at Fernald; the incinerator site soils and storage pad site soils, as the soils from these regions were considered to encompass the range of contamination at the whole site. These soils all had an average uranium concentration of 500 ppm.

Lee and Marsh (6) identified by XRD the major non-uranium-bearing phases in the

soils, which were quartz, clays, calcite, and dolomite. Buck et al. (2-4) identified the major uranium-bearing phases by AEM as calcium uranyl (VI) phosphate (tetragonal meta-autunite), uranyl (VI) silicate (soddyite), calcium uranium (VI) oxide, uranium (IV) oxide (uraninite), and uranium metaphosphate [monoclinic-U(PO3)4]. The uranium metaphosphate was only found in the incinerator site soils. Bertsch et al. (7) and also Allen et al. (8) have used XAS to determine uranium oxidation state of bulk soil samples from Fernald. A shift to higher energies of the X-ray absorption uranium LIII edge suggested that 80% of the uranium was in the U(VI) oxidation state. Furthermore, micro-beam XAS by Bertsch et al. (7) suggested that the distribution of uranium-bearing particles in the incinerator site soils was inhomogeneous, as indicated by signal varying as the focused beam was moved across the sample. In contrast, in some storage pad soils, uranium appeared to be uniformly distributed through the soil. This observation agreed with radiochemical analysis and particle sizing performed by Lee and Marsh (6) and observations made by SEM and AEM (2-4).

Carbonate leaching has been selected as the most appropriate method for removing the uranium from Fernald soils. However, the U(IV) phases (uraninite and uranium metaphosphate) in bench-scale tests were not removed. Uraninite can be leached by carbonate according to the scheme (9);

$$UO_2 + 1/2O_2 + 3CO_3^{2-} + H_2O \rightarrow [UO_2(CO_3)_3]^{4-} + 2OH^- \quad (1)$$

In other words, as long as an oxidizing agent is present, uraninite will be attacked. Effective oxidation of tetravalent uranium can be achieved with molecular oxygen in carbonate solution, with the rate of oxidation being proportional to the oxygen partial pressure. Chemical methods for adding this oxygen such as hydrogen peroxide and potassium permanganate are either expensive and/or increase the amount of pollution. Permanganate, however, was used in the bench-scale tests, after TEM analysis had shown that uranium(IV) phases were still present in soils that had been treated with carbonate. Addition of the oxidizer was found to improve uranium extraction during carbonate leaching.

The technique used in the mining industry for the recovery of metals, termed "Heap leaching," may be an attractive alternative for introducing oxygen. The nature of the Heap, where soil is heaped (or piled) onto an impermeable pad, can allow some aeration. In the Heap-treated samples (see Fig. 1), there was evidence from TEM of uraninite dissolution (although some uraninite was still present in the treated samples); however, no evidence was found of dissolution of the uranium metaphosphate phase, and a number of uranyl phosphate phases were still present.

Characterization of "Hot Particles" from Johnston Island

Johnston Island, located in the Pacific Ocean, 1330 km southwest of Honolulu, became contaminated in 1962 when the Island was used for launching missiles to test the effects of high-altitude nuclear bursts (10). In a number of instances problems occurred with the Thor missiles and the nuclear devices were intentionally destroyed by chemical explosives. One intentional destruction 59 sec after launching deposited plutonium- and uranium-contaminated debris throughout the atoll, while a second destruct of a missile on the launch pad contaminated a smaller land area but to much higher levels. Plutonium was dispersed by the explosive high temperatures and pressures generated by the explosion. The growth of the plutonium daughter product 241Am has permitted isolation of "hot spots" by gamma detection.

Tier I size sieving studies and radiochemical analysis by Wolf et al. (11) demonstrated that 96.5% of the activity was located in the 2 to 0.063 mm range. Most of the activity was localized in small "hot particles" in the coral sand. Bramlitt has also indicated that some of these "hot particles" were magnetic. This suggests that they might be closely associated with iron (10). A mechanical soil sorting method has been developed by Moroney et al. (12), termed the segmented gate system, that screens out "hot particles" automatically by using NaI gamma detectors. Improvements in this system have helped to reduce the contaminated volume of soil by 98%. The clean soil has a total alpha radioactivity from the plutonium and americium of less than 500 Bq/Kg.

A Tier II study was undertaken to describe the nature of the plutonium and uranium contamination in the "hot particles" present in the contaminated soils, so that the movement of plutonium at Johnston Island can be explained and further dispersion into the environment predicted. In some sites plutonium has migrated to depths over 1.5 m, and the mechanism by which this has occurred is unknown. In addition, the fate of uranium is unclear.

The AEM examination showed that plutonium is not in direct contact with the coral but is present in "sols" of plutonium and uranium (50-200 nm in diameter) within a micro-crystalline (partially amorphous) aluminum oxide, as well as with other components such as iron. Plutonium-bearing iron particles may account for the magnetic effects observed by Bramlitt (10). Beryllium, a typical bomb component, which can be detected by EELS, was found in some samples. Particles enriched in gallium and plutonium and 50 nm in size were also found. Gallium is often alloyed with plutonium, but the levels found in some of the discrete 50 nm particles exceeded normal alloying levels. In Fig. 2, "sols" of plutonium and uranium can be seen contained within the partially amorphous aluminum oxide matrix. The nature of the aluminum oxide is in itself unusual; the amorphization of aluminum oxide may have occurred by the decay of the plutonium. Alpha decay of 239Pu would cause the most damage, though 239Pu with a half-life of 2.411 x 104 years, may have produced the amorphization of the aluminum oxide within 30 years. The plutonium content in the aluminum oxide varied between 0.5 and 3 wt%. This would produce only 1 x 1013 to 6 x 1013 alpha-events/mg. Assuming that each alpha-recoil event displaces 1500 atoms, the dose delivered would be 0.5 x 10-3 to 3 x 10-3 displacements per atom (dpa). Metamictization in some ancient mineral phases typically occurs after >1.0 x 1016 alpha-events/mg (~1 dpa) (13).

The crystalline-to-amorphous (c-a) phase transformation can be induced by variety of processes such as particle irradiation or inter-diffusion reactions. Elastic softening and volume changes of the lattice often results from the loss of atomic order. The softening effect meant the microtome was able to produce large continuous thin sections of the aluminum oxide. A completely crystalline or amorphous material would have exhibited deformation and/or fracturing. Furthermore, if expansion of the lattice occurred in the partially amorphous aluminum oxide, then this might have allowed the entry of carbonate-bearing water. However, no evidence of any carbonate precipitates was detected in the aluminum oxide thin sections. Alternatively, incorporation of plutonium in aluminum oxide may result in a c-a transition through stabilization of the amorphous structure.

We speculate that an explanation for the distribution of radionuclides in Johnston Island soils to be the following. The aluminum and other metals (iron, magnesium, etc.) used in the bomb casing and bomb materials (plutonium, uranium, gallium, etc.) underwent rapid oxidation when the missile firing was aborted. The plutonium and uranium was deposited in the form of fine droplets which became embedded in the larger amount of aluminum and iron components. Some of the uranium crystallized, which may have made it less susceptible to weathering. Over time plutonium was leached in the surrounding matrix, leaving uranium enriched globules surrounded by a matrix of amorphous aluminum oxide containing small amounts of plutonium. This plutonium was then able to leach into the calcium carbonate coral, where it moved rapidly over the coral surfaces.

The Tier II characterization of Johnston Island "hot particles" suggests that the segmented gate system will be effective at removing the plutonium, and that chemical soil washing will not be necessary. Most of the plutonium and uranium is mainly contained in oxide particles, although some plutonium is spread throughout the aluminum phase. In addition, the uranium also appears to be contained within the "hot particles," suggesting that removal of these particles will also remove any uranium in the soil.

CONCLUSIONS

These AEM investigations have shown a large variety of different contaminant phases in examined soils. These variations result from the broad range of sources of uranium and plutonium, and also the different chemical processes employed at sites such as Fernald. The range of uranium-bearing phases at Fernald has impeded the remediation efforts, as no single chemical process has been able to remove all the uranium without totally destroying the soil. However, carbonate Heap leaching has managed to remove a majority of the uranium phases, only leaving the uranium metaphosphate phase.

Removal of uranium from the Fernald soils and elsewhere can be made easier by detailed knowledge of the chemical and physical characteristics of the waste and its environment. A characterization technique must be able to determine the exact nature of the contaminant phase, as incorrect interpretations of data could lead to the selection of inappropriate remediation methods. A technique which provides both compositional and structural data, such as AEM, is advantageous because it can

determine the chemical form of unique phases. At Fernald, the identification of the ceramic-like uranium metaphosphate phase is a good example of this capability. At Johnston Island, AEM studies have confirmed that removal of plutonium using the segmented gate system will be effective and that it will also remove the uranium from the site, as the uranium is contained in the "hot particles." The characterization methods described above, in combination with other techniques such as XAS and radiochemical analysis, allow remediation technology groups to find more efficient ways of removing contamination. Characterization has been criticized as providing esoteric information. However, when used in the role of problem solving, it can provide information which improves the rate at which bench-scale testing can proceed to full-scale remediation.

The small particle handling facility and microscopy laboratory developed at ANL is equipped to handle radioactive materials for soil and residue characterization. The facility also includes radiochemical analytical instruments, inductively coupled plasma-mass spectroscopy for elemental analysis, and secondary ion mass spectroscopy for surface profiling of radioactive materials. We are currently engaged in establishing commercial sector projects for developing and testing soil and residue treatment technologies.

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27-56

INCREASING QUALITY ASSURANCE THROUGH AN INTEGRATED SURVEY SYSTEM

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ABSTRACT

The weakest link in a radiation survey is usually the operator. Repetition produces tedium which increases errors. On the other hand, the computer's strong suit is accurately performing repetitious tasks. The computer's role can be expanded from just data handling to one of retrieving the data from equipment. Through a U.S. Department of Energy Health Physics Faculty Research Award Program, the University of Florida, Department of Environmental Engineering and Science is developing an integrated, indoor survey platform. Two hardware aspects of this project are integration with survey equipment and determining positions. The heart of this survey platform is a Notebook computer with a PCMCIA data-acquisition board. Some survey equipment is available with RS-232 ports which can hook up directly to the Notebook computer. For equipment that is not RS-232 compatible, it is still possible, with the help of the manufacturer, to retrieve analog signals and feed them into a data acquisition card. The computer controls the survey equipment and results are displayed for the user and automatically stored in a database. There are limited alternatives for indoor positioning. Sonic rangefinders need a smooth surface to reflect against, GPS is unavailable indoors and tape measurements are tedious. Another alternative is currently being developed using a combination of triangulation and dead-reckoning. Both devices are controlled by the computer and results are displayed to the user and automatically stored in a database. With more computer control of a survey, the operator's responsibility shifts from just taking measurements to one of ensuring quality control.

INTRODUCTION

Quality assurance begins with the operator taking the survey and ends when the survey information is no longer needed. Quality assurance is the most potent at the beginning, during the survey. By ensuring the data is recorded properly, the operator is taking the first steps of quality assurance. But there are situations that lead to poor quality results such as inadequate training of the operator, fatigue as a result of environmental stresses and lack of concentration from performing repetitious tasks. To improve quality, refresher training can be provided, the operator could be relieved often during surveys and the amount of direct supervision could be increased. Another alternative is to provide "smart" instruments and "expert" systems to the operator.

"Smart" instrumentation can be defined as transferring equipment operations from human operators to computer control. This control only applies to the repetitious and routine procedures that are tedious to humans but for which computers are designed. There is a balancing point between how complex an operations will be to program into a computer and the simplicity for a human to perform the same operation. As computer processing speeds increase, they can take more responsibility for a survey. Computers will then be able to compensate for shortages of adequately trained personnel and increasingly stringent worker protection regulations. But, regardless of how "fast" a computer becomes, the higher order thinking processes required in surveys will still rely on human experience.

"Expert" systems can be defined as a "corporate" knowledge base of situations and responses. This knowledge comes from worker experience and will be accessible to all workers. To do this, the experts need to supply answers to questions that a novice may ask. Expert systems are increasingly needed to help cope with shortages of adequately trained personnel, increasingly stringent worker protection regulations, quality assurance and quality control requirements (1). Expert systems will be able to provide limited substitution for experience. A comprehensive system would allow a novice to operate equipment and perform surveys correctly and quickly. By providing a working knowledge of procedures, expert systems can also cost effectively facilitate the documentation of compliance with DOE, federal and state regulations (2). Development of these systems will put increasing pressure on the developers of software and survey systems to make them as user friendly as possible. These systems will require more from the computer and less from the operator. The first step in generating an expert system will result in a "cookbook" which can guide a novice through difficult situations. As knowledge is gained and integrated into the computer, the "smarter" the system will become.

COMPUTERIZATION OF INFORMATION

The first step towards computerization of surveys is to determine the Department of Energy's (DOE) viewpoint. The DOE allows for transferring records to computer storage as long as it is maintained in a retrievable form (3). The DOE recommends controls for indexing, quality assurance and security. A master index is required which lists the files on every disk and the type of information contained within each file. Additionally, the index should identify the program used to create the information and the format in which the information is saved. Most computer programs, such as WordPerfect, Quattro Pro, Lotus, etc., save information in their own format. Having survey results also saved as ASCII text would ensure that any other program can read the information and can also qualify as a backup to the original information. Quality control during data entry and analysis is ensured if the information is transferred automatically from the survey equipment directly into a database. Finally, safety of the information must be assured. Just as in the case for paper records of a survey, the computerized records should be accorded the same security. Theft and/or tampering with the information should be guarded against.

COMPUTERIZE SURVEY

The University of Florida, Department of Environmental Engineering and Sciences was awarded a contract by the DOE Health Physics Faculty Research Award Program entitled An Integrated Environmental Monitoring and Assessment System for the Evaluation of Indoor Remediation Projects." It outlined and contained five main objectives. These were 1) create databases for site information and test results, 2) automate survey equipment, 3) automate position equipment, 4) provide access to modeling programs and 5) provide access to laws and regulations. The intent of this project was to computerize the surveying process to relieve the operator of some of the responsibility during a survey and put this responsibility on the computer.

The computer for this project was a Toshiba T4600 Series Notebook computer with a 33 MHz IntelTM 486SL processor, 12 MByte RAM, 120 MByte hard disk and two Type II PCMCIA sockets. One PCMCIA socket was fitted with a National Instruments DAQCardTM-700. The data acquisition card allows for sixteen analog signals referenced to ground or eight pairs of analog signals referenced to each other, eight digital input and eight digital output lines and three counters.

An appropriate computer environment was needed which could implement these controls and still provide the flexibility for this project. Microsoft WindowsTM was chosen for three main reasons. First, Windows has become a popular and widely used environment that operates on computers that are decreasing in price and increasing in capabilities. A wide variety of Windows software is available for programming, word processing and graphics depending on the user's preference. Once the raw survey data is entered into the computer, the user can chose the programs they are most comfortable with to process the data. Secondly, most Windows programs have a similar texture. They use similar menus, pop-up panels and options which will help shorten the learning curve on new software. Finally, using the Windows environment will take the computer away from being dedicated to just surveys and allow its use in the office. This becomes important to small facilities that do not require frequent surveys and have limited budgets.

This survey program was developed using Microsoft Visual BasicTM. Visual Basic is a powerful programming tool that is designed to create Windows applications. It is hoped that running the surveys on Windows would require less initial training and be more comfortable to operators. The databases used for this program were created to run on the Microsoft AccessTM engine (4). The databases, tables and fields are created within the survey program and does not require the Access software. However, the user has the option of manipulating survey results and floorplan information, independent of this survey program, using the Access software.

Survey Equipment

For this project, the initial survey equipment used was a Ludlum Model 2350 Data Logger. The Data Logger is a portable digital ratemeter, timed counter or integrated dose counter and timer. This Data Logger can have attached a variety of Ludlum Alpha, Beta and Gamma Scintillation detectors. For this project the Ludlum Model 44-10 high energy 2" X 2" NaI(Tl) Gamma Scintillator and a Ludlum Model 43-1 Alpha Scintillation Detector were used. The Data Logger contains an RS-232 serial port which can be used to connect to a computer or a specialized handheld unit. The Data Logger contains sufficient memory to store the parameters for sixteen different detectors. Through a special handheld unit or a computer, the parameters can be read or changed. The Data Logger is also capable of storing up to 250 survey points with

position information or can supply real-time information to a computer or handheld unit. For this survey program, the computer controls the Data Logger and stores results into a database with information on positioning.

Positioning

The primary method of locating positions indoors has traditionally been by manually laying out grids. This alternative will always be available, but less laborious methods were explored. Eight alternative positioning methods were reviewed for this project. 1) The Global Positioning System (GPS) consists of twenty-seven satellites, twenty-four in geo-synchronous orbits with three satellites utilized as spares. At any time and place around the globe, four satellites are visible and can provide triangulation. However, signals from the GPS are attenuated by building materials making them unusable indoors. 2) The Ultrasonic Ranging and Data System (USRADS) uses three or more ultrasonic transceivers which receive a signal from a backpack worn by the operator. Indoors, this system would seem to be hampered by re-radiation of broadcasted signals causing multi-path errors, but currently the USRADS has had some success with indoor positioning. 3) Inertial guidance provides an alternative for indoor positioning by using "dead-reckoning." Starting at a known point within the room gyroscopes and accelerometers measure and compute changes in distances along all three vectors (X, Y and Z). However, this system has problems with gyro drift and accuracy is directly related to cost. 4) Sonic rangefinders provide the simplest alternative for positioning. The distance from walls can provide positioning and the rangefinder's signals can be directly entered into the computer. However, the signals have limited distance accuracy and are best for small, uncluttered rooms. 5) A sonic rangefinder repeater is available with some models. The signal from a sonic rangefinder is sent at one frequency, received by the repeater and retransmitted at a different frequency back to the sonic rangefinder. The biggest benefit with this system is an increased range. 6) A commercial laser rangefinder can provide accuracy to within a fraction of an inch over distances of several hundred yards. However, the cost for this accuracy makes it prohibitive. 7) One alternative developed for this project was the "floor mouse." It uses a known starting point and then dead-reckons on a roller ball similar to a computer mouse. Movement in the X and Y directions are calculated using optical readers and a spoked wheel. As the mouse moves, the spoked wheels rotate. The optical reader records the number of spokes that pass by and calculates the distance traveled. With calibration, the floor mouse can provide over 99% accuracy in determining position within ten feet of the starting point. This alternative is cost effective and the floor mouse can be discarded along with the other contaminated material. 8) The final alternative developed for this program uses an inexpensive laser pointer mounted on a potentiometer. As the laser is rotated from one landmark to another, the voltage across the potentiometer changes. The change in voltage equates to a change in angle. The angle differences between three known points within a room are entered into an algorithm which uses the known locations of the landmarks and calculates the laser's position within the room.

Both the floor mouse and the laser pointer system work individually and a combination of the two provide a viable option. Exact positioning can be determined by the laser pointer system and then the floor mouse can be used to dead-reckon to different points. A sensor attached to the floor mouse frame would indicate when the mouse has been lifted off the ground. This would require the operator to enter a new "fix" or exact position using the laser or from manual measurements. The combination of both systems will still remain cost competitive compared to other methods of indoor positioning.

Software Development

Modeling programs are still in their infancy. Programs that have been used for years were originally written on punched cards. Computer models were written for DOS-based or UNIX-based systems where the primary concern was with processing repetitive equations. The computers that were used were mainframes with several users all vying for computer time. Code had to be written with the emphasis on efficiency and not with the appearance of outputs. Today, desktop computers have the processing power that could rival the mainframes from the past with the benefit of being readily accessible. Programming languages have evolved from assembly language to Graphics User Interfaces (GUI). Code can be written and modified to appeal to a specific user. The learning curve for some programming languages is relatively short. For instance, anyone with experience with Basic would have no problem jumping to Visual

Basic and generating Windows applications. For hardware and software manufacturers, core processing modules can be generated and packaged into custom controls. Users could take these custom controls and apply them as needed for integrated with other applications. Such an example would be National Instrument's NI-DAQ software. Custom controls for Visual Basic are supplied when purchasing a National Instrument's data acquisition card. Instead of learning the intricacies of Standard Commands for Programmable Instruments (SCPI) and IEEE 488.2 standards, the user can hook up an analog signal to the computer and write a couple of lines of code that tells the computer to read the voltage on a certain input line or the state of a digital input line. The signal is immediately available for modeling applications. This level of flexibility allows users to apply their own modeling ideas to measured results.

Laws and Regulations

An informational source of standards & criteria is available from third party vendors. Installing a copy of standards and regulations onto the computer is easily accomplished by downloading from the appropriate Bulletin Board System (BBS). But, these are also available from third party vendors which also provide two additional benefits. First, monitoring the Federal Register for updates and revisions and then entering them into the appropriate regulations or standards is a full-time job. Second, not all the information contained in standards and regulations are applicable to each site. Third party vendors will take the needs of each site and ensure they receive only the pertinent information. These vendors can also provide ongoing support to ensure the user always has the correct and latest information.

CONCLUSION

The potential for human error increases during the process of surveying. There is no relief yet from having to perform surveys, but there are three immediate alternatives to speed up the process. First is to use the computer for data logging. This step alone will remove the redundancy of writing the information on paper and then transposing the results to a database. Computerizing results makes the data more portable. Information is readily available on-site without dragging along reams of paper. Secondly, the survey results can be directly entered into the computer by the survey equipment. The operator will still have the responsibility of entering the position and any comments. Finally, positioning equipment can be connected to the computer and automatically entered into the database. This removes the operator from the loop of entering positioning and survey information. This alone will improve the quality of the data and decrease the time of the survey.

At this time, we are at the beginnings of a paperless society. This process will come about over time, but methods should be explored that can wean ourselves from paper records. This project is a step in that direction by placing more of the responsibility of the survey with the computer and reducing the dependence of paper records.

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Session 28 -- Innovative Technology Acceptability: Technical & Institutional Issues/Barriers

Co-chairs: Paul Longsworth, USDOE;
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28-1

EXPEDITING ENVIRONMENTAL CLEANUP -- NATIONWIDE STAKEHOLDER INVOLVEMENT IN DOE'S PLUME FOCUS AREA

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ABSTRACT
The U.S. Department of Energy (DOE) has committed to expediting the cleanup of its contaminated facilities. This calls for using innovative technologies that will improve performance, cut schedules, and lower the costs of cleanup compared with using historically available technologies. DOE has organized its technology development and application activities nationwide into five focus areas. The mission of the Plume Focus Area is to enhance the deployment of innovative technologies for containing and cleaning up contaminant plumes in ground water and soil. New technologies must be driven by high-priority needs at the facilities nationwide, maximizing the benefits and widely deploying successful solutions.
The Plume Focus Area has committed to major involvement of stakeholders in its implementation, incorporating stakeholder values and input in identifying and prioritizing environmental management needs. Stakeholders will also participate in identifying new technologies to address those needs, and evaluating the acceptability of those technologies so that deployment decision makers can select remedies that address broad-based stakeholder concerns.
Stakeholders are defined to include technology users, public interest and environmental groups, civic organizations, Native American tribes, elected and appointed officials, and other interested parties. Early and substantive stakeholder involvement will identify information needs and potential obstacles to technology deployment, which can then be addressed through the focus area's technology development and evaluation efforts.
This paper describes the objectives, approach, and challenges of implementing a stakeholder involvement program that also addresses site-specific needs. It outlines the model that is being used, based in part on stakeholder involvement in DOE's Volatile Organic Compounds Arid Sites Integrated Demonstration. It defines near-term and longer-term activities, and indicates how stakeholder input will be integrated with technical, product line, and industry inputs to support a focused, needs-driven decision process.
INTRODUCTION
The U.S. Department of Energy (DOE) faces formidable challenges in remediating contaminated sites nationwide. A key concern is the presence of soil and ground-water plumes, which contain a range of radioactive and chemical contaminants. Today's technologies and approaches will not be enough to meet the technical, regulatory, and management challenges of the 30-year remediation process.
In Fiscal Year 1994, DOE put in place a new approach to develop and deploy the innovative approaches that are so badly needed. The new approach focuses technology development and deployment into five focus areas, including:

- Contaminant Plume Containment and Remediation
- Characterization, Treatment, and Disposal of Mixed Waste
- High-Level Waste Tanks
- Containment of Existing Landfills
- Decommissioning and Final Disposition

Innovative technologies are not worth developing if they are unacceptable to environmental restoration personnel, regulators, and stakeholders. The focus area is designed to cut across all the relevant DOE components, ensuring that technologies that are developed by DOE are actually used by the waste management and environmental restoration programs (EM-30 and EM-40). Furthermore, the focus area is looking for existing solutions to meet site needs that are available within other agencies and industry before deciding to support research and development.
To ensure that the actions of the Plume Focus Area reflect the perspectives of all the DOE sites and of the other groups and individuals in those site areas with a stake in their cleanup, the focus area team includes a sites' coordination function. The Sites' Coordination Team (SCT) develops and supports the implementation of methods for stakeholder involvement throughout all stages in plume technology development and application. This paper describes the objectives, approaches, and challenges of the SCT for the Plume Focus Area, and discusses how it may result in a model for direct stakeholder involvement in technical and highly complex technology

decision making.
PLUME FOCUS AREA ORGANIZATION AND MISSION
Each focus area includes representatives of all the involved elements of DOE. The Plume Focus Area consists of a DOE Headquarters management team, composed of representatives from EM-30, 40, and 50. A lead organization based at the Savannah River Site supports the management team in implementing the national objectives. The lead organization has formed an implementation team, bringing together DOE and contractor personnel from across the field offices who bring to the team specialized expertise in technology development, industrialization, stakeholder involvement, and other functions. Both teams work together to carry out the steps in the technology development/deployment process, incorporating site and stakeholder involvement at each step (See Fig.1). In summary, those steps include:
Identify environmental management needs
Identify potential technologies to meet those needs
Match potential technologies to needs
Verify needs and technology matches at site level
Issue solicitations and calls for proposal for needed technologies
Conduct research, development, and demonstrations of promising technologies
Evaluate research and demonstration results to assess technology performance and acceptability
Make technologies available broadly for deployment to meet environmental management needs
The SCT is involved in all of these steps in order to plan for, support, and coordinate the involvement of site personnel and other stakeholders, to drive technology development through high-priority site needs, and to improve broad acceptability of the technologies for deployment.
BACKGROUND FOR PLUME FOCUS AREA STAKEHOLDER INVOLVEMENT APPROACH
A foundation for the Plume Focus Area's approach to site and stakeholder involvement is a three-year DOE program, that was sponsored by the Office of Technology Development. The Volatile Organic Compounds Arid Sites Integrated Demonstration (VOC-Arid ID) pioneered a stakeholder involvement process that provided for early, meaningful participation in defining, demonstrating, and deploying acceptable technologies for cleanup of VOCs in arid sites' soils and ground water.
In the VOC-Arid ID, a range of "internal" and "external" stakeholders were involved at the site where the demonstrations are being conducted, the Hanford Site in Washington State. With a solid base of Hanford host-site stakeholder input, the ID broadened its efforts to include participation by stakeholders at other arid DOE sites where the technologies may ultimately be deployed. In this way, technologies' acceptability at other locations and in other regions were assessed, and stakeholders gained a better understanding of some of the technologies being developed to address VOC contamination problems at their sites.
What this earlier work confirmed is that technology planners need to consider both generic and site-specific issues when evaluating potential demonstration and deployment decisions. While the majority of stakeholder concerns are common across geographic areas and even among stakeholder categories, there is a significant level of specificity that, if ignored, would place future decisions at risk.
This is the model used as the basis for the Plume Focus Area SCT. While it has provided insights into design of an appropriate strategy, there are clear limitations when applied to a national program. The specific Plume Focus Area strategy has been tailored to the challenges of integrating ten DOE field offices, many with multiple sites, and each site with numerous environmental management needs and programs.
SITES' COORDINATION TEAM OBJECTIVES FOR PLUME FOCUS AREA
In order to achieve the Plume Focus Area mission, the SCT's objectives are integrated throughout. Specific SCT objectives include:
To define desired outcomes for sites and stakeholders to be involved in the focus area's technology evaluations, and provide a model(s) for achieving those outcomes while retaining site-specific flexibility.
To provide national-level strategic planning, guidance, and support to each DOE site to implement stakeholder involvement activities.
To provide the rest of the plume team with products of stakeholder input that are directly applicable to technology demonstration design, evaluation, and deployment decisions, in the formats and level of detail needed to support team decisions and

results.
To mesh these involvement efforts with those of the other four national focus areas and independent programs (e.g., robotics, characterization), with sensitivity to ongoing stakeholder involvement at the DOE sites.
These objectives and the resulting approach reflect a national vision for sites' coordination and stakeholder involvement. The SCT develops and aids in implementing stakeholder involvement throughout the country. Site-specific stakeholder involvement teams are being formed to carry out the strategy at each site, and the teams will coordinate closely with Site Technology Coordination Groups (STCGs) at each location. The SCT is responsible for identifying and working with those site-specific teams, supporting them to the degree needed (e.g., preparing needed guidance and strategies, helping the teams tailor strategies for their sites that meet the overall needs of the focus area), and facilitating inter-site coordination. Results will help in developing national technology acceptance reports on the innovative technologies being funded and evaluated under the Plume Focus Area.
The SCT's responsibility, then, is to serve as the catalyst to bring the site teams together in a national framework. The SCT will work with the site teams to identify required stakeholder involvement outcomes and assist them in achieving those results to support the national mission.
SITES' COORDINATION TEAM APPROACH
In this first year of operating the Plume Focus Area, the transition year, we have asked the sites to define their plume problems in soil and ground water, describe the technologies they are planning to use to address each problem, describe their environmental management needs, and prioritize these needs. Representatives from each of the DOE field offices have or are in the process of providing the Plume Focus Area with this information. The lead organization summarized this information for each site by grouping the problems into several categories: VOC contamination, heavy metal contamination, radionuclide (e.g., tritium) contamination, and dense non-aqueous phase liquid (DNAPL) problems. The matrices also included summary information on site characteristics and technologies currently planned to address site needs. In addition, descriptions of key operable units (OUs) with plume problems have been developed based on needs described by the field operable offices.
Information in the OU template includes:
- Site/OU designation
- Priority level
- OU Manager/address/telephone
- Regulatory and other drivers
- Schedule and milestones
- Overall objective for site/OU
- Performance requirements and goals for technology(s) to meet site objectives
- Current baseline approach/technologies
- Geologic setting
- Contaminant types, concentrations, volumes, and areal extent
- Issues and concerns

After these need sets were developed, the lead organization of the Plume Focus Area scheduled site visits to verify and prioritize the information. The first round of site visits is taking place during January - March, 1995. With the site needs compiled and verified, the goal will then be to match available and developing technologies with those site needs, and to identify problems for which there are no technologies either available or being developed by DOE that will address the problem. The site visits are critical to establishing the baseline program for the Plume Focus Area and building on it in future years. Environmental restoration program managers are being actively involved in order to obtain their perspective as well as their commitment to using the technologies being developed if they perform adequately in demonstration.
The SCT's role is also to ensure that stakeholders, other than DOE program managers, are involved to enhance the design of the technology demonstrations and ultimately the acceptance of the technologies. In support of this role, the SCT will work through the STCGs to help assemble site-specific stakeholder involvement teams. The SCT will also develop and disseminate strategies and guidance for conducting stakeholder involvement at the national and site-specific levels. To accomplish this, the SCT will coordinate closely with the site-specific stakeholder involvement teams, STCGs, and the Plume Focus Area lead organization and management teams, to

ensure that stakeholder involvement activities are conducted to produce the required results.
Again, the success of the SCT is largely dependent on the results of actions taken by the site-specific stakeholder involvement teams. In Fig. 2a,b,c, responsibilities and activities are shown as divided between the focus area team overall (top layer), the SCT (middle layer), and the stakeholder involvement teams and STCGs (lower third). The SCT, with critical support from these site-specific teams, will:

Develop a national strategy for the Plume Focus Area.

Establish tailored approaches to achieving site and stakeholder involvement in focus area activities and decisions.

Work with the sites to involve stakeholders in identifying environmental management needs at their sites, prioritizing those needs, and matching potentially applicable technologies with those needs.

Consult with stakeholders on the development of evaluation criteria and a prioritization process to rank environmental management needs across all sites; and support verification of the prioritization process by involving stakeholders.

Provide input to the focus area product line team on stakeholder concerns, issues, and data needs to help define the functional and technical requirements for the technology product lines.

Work with technology developers to develop information profiles on their technologies and use these profiles to consult with stakeholders on data requirements to include in the demonstration test plans.

Obtain stakeholder review of demonstration performance and results, receive their feedback, and prepare technology acceptance reports for each technology evaluated.

If there is considerable interest, or significant contrasting input on innovative technologies between sites, the SCT may convene regional focus groups with stakeholders from various sites to accomplish a shared resolution of issues. Such forums may be particularly useful to address interstate regulatory acceptance issues and approaches. Technology acceptance reports will support decisions on deployment of innovative technologies, and will be made available nationwide to communicate Plume Focus Area activities and results to a broad audience.

28-2

THE VALUE OF ADDING REGIONAL TO LOCAL STAKEHOLDER INVOLVEMENT IN EVALUATING THE ACCEPTABILITY OF INNOVATIVE TECHNOLOGIES

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ABSTRACT

Technology is urgently needed to clean up contamination by volatile organic compounds at United States Department of Energy (DOE) sites. In many cases, however, existing technology is too slow, inefficient, or expensive. The record of technology development is, in some cases, similarly disappointing. Remediation technologies developed at great expense and evaluated piecemeal over long periods have not been deployed because, in the end, the public judged them ineffective or unacceptable. The need for successful methods of remediation is too great and resources too limited to continue with ineffective technology evaluation. In order to make good decisions about which technologies to deploy, remedial project managers need to know stakeholders' requirements for the performance of proposed technologies. Expanding stakeholder involvement regionally identifies the concerns of a broad range of stakeholders at arid DOE sites throughout the West - issues that must be taken into account if technologies are to be accepted for wide deployment.

THE PURPOSE OF STAKEHOLDER INVOLVEMENT IN TECHNOLOGY EVALUATION

Volatile organic compounds (VOCs) are the most prevalent hazardous waste contaminants at Department of Energy sites throughout the United States. Effective, economical technologies are needed to clean them up. But innovative remediation technologies have proven difficult to deploy partly as the result of public

opposition, arising in some measure from stakeholders' sense that they have not played a meaningful, timely role in evaluating proposed approaches. Given this history, stakeholder involvement in technology evaluation has three purposes:
To define those stakeholder issues and concerns that must be addressed in order for a given technology to be accepted for deployment. These issues and requests for information ("data requirements") can then be incorporated into technology test plans.
To identify early in the process of technology development those technologies or aspects of technologies that require modification in order to be deployable. This avoids allocating resources to technologies that ultimately will be unacceptable.
To provide stakeholders with the results of technology demonstrations so they can make reasoned judgments about the acceptability of given approaches.
To achieve these purposes, to expedite the deployment of effective technologies, and ultimately to advance environmental cleanup, Battelle, with funding from the U.S. Department of Energy's Office of Technology Development, has carried out a three-year stakeholder involvement program.
EXPANDING STAKEHOLDER INVOLVEMENT
In its first phase this program, part of the VOC - Arid Sites Integrated Demonstration hosted at Hanford, Washington, involved Hanford stakeholders in evaluating six innovative technologies for the remediation of VOCs in soils and ground water. Because a guiding principle of this effort is "Test once, deploy broadly," it was necessary to determine data requirements at the arid sites other than Hanford that must be a taken into account if test plans are to be truly comprehensive and responsive to stakeholders' concerns. In addition, the team conducting the stakeholder involvement program wanted to subject the criteria for evaluating technology (Fig. 1) and the issues highlighted by Hanford stakeholders to a regional assessment. Were the priorities identified by Hanford stakeholders shared across the arid west? Were there site-specific issues that because of their surpassing local importance would determine the deployability of a technology at a particular site? The public involvement team wanted to identify those issues and concerns that must be taken into account in considering any technology for deployment at any site as well as those site-specific issues of overriding local concern.
Figure 1
To this end, the VOC-Arid public involvement team conducted 75 interviews with stakeholders at Sandia and Los Alamos, New Mexico, Rocky Flats, Colorado, and the Idaho National Engineering Laboratory. Those interviewed represented a cross section of the public concerned about the cleanup of particular DOE weapons complex sites in the West, and included local elected officials, regulators, Native Americans and Hispanic community members and representatives of environmental, public interest and civic groups.
Hanford-site Stakeholder Involvement in the Evaluation of Innovative Technologies for the Remediation of VOC Contamination
The foundation and context for these interviews was two years of consultation with Hanford stakeholders. Hanford stakeholders helped develop detailed criteria for evaluating innovative technologies to remediate VOC contamination. By participating in interviews, focus groups, and workshops, they defined data requirements that have been incorporated in the test plans now guiding technology demonstrations. (These data requirements are presented in a series of reports prepared by Battelle on ground water and soil remediation technologies.) Hanford stakeholders stipulated specific standards of performance for technologies. Overall, they have provided a detailed sense of what concerns the public about technologies and what technologies will have to do and be in order to be accepted and used.
Among the criteria for evaluating technologies, issues of technical effectiveness, cost, and time occurred most frequently in the comments of Hanford stakeholders. In addition to data requirements specific to each of the technologies under consideration, Hanford stakeholders identified the following data and performance requirements as pertinent to all the methods being assessed. (Please see Phase II Stakeholder Participation in Evaluating Innovative Technologies: VOC-Arid Integrated Demonstration, Ground Water Remediation System , Battelle Seattle Research Center, April 1994 for data requirements specific to particular technologies.)
1) Define remediation objectives to ensure that the technology truly contributes to the

objectives.
2) Conduct integrated comparisons of an innovative technology to the technology currently in use.
3) Design demonstrations to provide data on performance, cost, and time to reduce uncertainty and better define trade-offs.
4) Demonstrate the technology considering differing site conditions to measure its versatility.
5) Define the demonstration assumptions and expectations about secondary waste.
6) Define the risk management strategy and the elements of and process for assessing operational readiness.
7) Define how effectiveness of the technology, both in terms of its performance and its effect on the environment, will be monitored.
8) Plan for unintended consequences and define and test all potential failure control mechanisms.
9) Define the liability implications and insurance requirements for the deployment of the technology.
10) Demonstrate that future cleanup is not foreclosed by using the technology.
11) Have a credible third party evaluate demonstration data.

RESULTS OF REGIONAL STAKEHOLDER INVOLVEMENT

In expanding consultation to include stakeholders from four other DOE weapons complex sites in the arid west, the stakeholder involvement team intended to test the hypothesis that data requirements different from those identified by Hanford stakeholders will pertain at these other sites given geographic, cultural, regulatory, and institutional differences. The underlying assumption remains that if a technology is to be widely deployable, this broadened set of data requirements must be taken into account.

The regional stakeholder involvement program yielded a number of findings. Foremost is that a technology will be put into use beyond the location of its demonstration only if issues significant at other possible sites are identified and addressed. Failing to identify or address these site-specific concerns during technology demonstration will increase the time and cost of deployment, and could result in the rejection of the technology.

Regional stakeholders validated the criteria developed by the Hanford stakeholders. There was also agreement about which criteria are most important. Hanford and regional stakeholders placed the most emphasis on technical effectiveness. More comments from Hanford and regional stakeholders focused on performance than on any other criterion.

Similarly, regional stakeholders raised many of the same issues and concerns about the six VOC remediation technologies as Hanford stakeholders.

There are, however, certain local concerns that if not addressed in a technology's demonstration will prevent a technology's deployment at a site. These issues will dominate the local evaluation of a technology. Identifying them at the outset is essential to understanding a technology's potential for deployment.

Taken together, these requirements are basic conditions with which stakeholders will evaluate the acceptability of any remedial technology. They should be considered by those responsible for developing and selecting environmental remediation technology in the western states.

Requirements Identified at All Sites

Technologies should destroy contamination on site rather than concentrating it for transportation and destruction elsewhere. Those technologies that meet this requirement will be regarded more favorably than those that do not.

Technologies should not transfer contamination from one environmental medium to another. For example, the action of the technology should not result in moving contamination from ground water into air.

The number of steps in a technology's treatment system should be minimized as much as possible.

Technologies should be able to be operated and maintained by existing staff.

Technologies should be economical, understandable, and robust.

Technologies must be able to treat co-contaminants. In order to be accepted for deployment, a technology must be effective with more than just its target contaminant. Acceptable technologies will have the ability to deal with radioactive

co-contaminants. The inability to do so may be a "showstopper."

Acceptable technologies are those which solve all aspects of a contamination problem, not just part. It is not acceptable to solve one environmental problem only to create another in a different place or in a different environmental medium.

A technology's entire treatment system is important. Stakeholders will evaluate the entire system, not just the most visible or active component. The entire system must be maintainable and economical. Secondary waste from each component must be dealt with safely.

Across-the-Board Requirements with Site Specific Relevance

Certain site-specific concerns will determine the deployability of technologies:

Technologies will be evaluated within the regulatory framework prevailing at a particular site. To enhance a technology's acceptability, the technology and its demonstration must be presented in terms of that site's specific regulatory framework. Applicable regulations vary among states and tribal nations. For example, in New Mexico tribal governments may have more stringent cleanup standards than state government, particularly in terms of water quality. Also in New Mexico, RCRA alone will govern technology demonstration and deployment. At this time, DOE sites in New Mexico have no CERCLA liability. Conversely, environmental regulators in Idaho may use RCRA or CERCLA; for example, VOC vapor may be regulated under CERCLA and VOC liquid under RCRA.

Where a natural resource is considered special, impacts on that resource will determine a technology's evaluation and acceptance. For example, water is extremely important in the arid west. Communities, particularly in the Southwest, are discovering that the quantity of available ground water is significantly less than previously predicted. Therefore technologies that do not remove and reinject ground water are regarded favorably. In another example, the Snake River Plain Aquifer is of such economic, cultural, and political importance to southern Idaho that any proposed technology's impacts on it will take precedence in the evaluation of that technology. Similarly, Southwestern stakeholders prize the clarity of their air, and therefore will scrutinize any proposed technology's air emissions.

Remote sites place special requirements on technologies. These include the ability to withstand vandalism and operate reliably and automatically for long periods. Remoteness raises the question of power supply, an issue involving concerns about the visual impact of power lines and the air emissions from generators.

The ability to operate in locally prevailing weather will determine the acceptability of any technology. Extremes of temperature and humidity, high winds, particularly at Rocky Flats, and deep snows are among the conditions that must be taken into account.

Trade-offs among criteria will vary from site to site. Local conditions will determine how stakeholders weigh the benefits and drawbacks of some of a technology's characteristics in relation to its other capabilities.

Versatility applies to all criteria, not just to performance. To be acceptable, a technology must be versatile and adaptable in terms of all the criteria used to evaluate it. For example, regulatory compliance means something different in each of the states where stakeholders were interviewed.

In conclusion, not considering these issues in designing, demonstrating, and selecting a technology may result in decisions that are delayed or reversed, and technologies that are never deployed.

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MEANINGFUL STAKEHOLDER INVOLVEMENT: OVERCOMING EXISTING BARRIERS TO PUBLIC ACCEPTANCE OF INNOVATIVE TECHNOLOGIES

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ABSTRACT

We are currently at a crossroads regarding public funding for remediation of the chemical and radioactive contamination at United States Department of Energy (DOE) nuclear weapons complex sites across the country. Substantial criticism has been focused upon DOE's inability to clean up the widespread contamination at its facilities. And, the fiscal train wreck envisioned with the FY'96 budget has been dramatically elevated with last November's election. Regardless of the validity of the critique of DOE's fifty years of self-regulation, there is one ultimate truth -- the challenge to isolate these materials from the biosphere, regardless of the risk, is enormous.

Perhaps the biggest challenge faced by DOE today is the development of decision-making processes which welcome the views of interested stakeholders and actually use this input to improve both the quality and public acceptance of decisions. Future R & D decisions must convince the Congress and the public that the money invested in technology development has been spent well.

We submit that out experience in the Pacific NW laboratory's Volatile Organic Compound/Arid Lands Integrated Demonstration Project (VOC/Arid ID) has left us hope that R & D decisions can be made within a framework where money can be spent well, appropriate advanced technology developed, results produced and transferred to the private sector in a timely manner. We have concluded that a decision-making process can succeed if it is well conceived, engages and listens to a spectrum of stakeholder interests from its earliest stages, links principal investigators with both stakeholders and regulators, and does not create unrealistic expectations. Additionally, once established, the investigative and public participation processes employed in such a process can be applied to other sites across the complex.

In this paper, we will develop a comparative analysis of two different approaches to innovative technology acceptance, primarily related to Department of Energy Facilities, that have been initiated over the past two to three years. First, we will examine the Western Governors Association Committee on Development of Innovative Technologies (DOIT). The second approach to be examined, one in which both authors have extensively participated, is the stakeholder participation program supporting the Volatile Organic compounds (VOC) Arid Site Integrated Demonstration (VOC-ARID ID) conducted at Hanford, Washington.

Comparative analysis of these two approaches is intended to illustrate that successful development and deployment of technologies can only occur when those technologies, not only perform well during demonstration, but are found acceptable to the people and groups that have a stake in the cleanup process -- the regulators, the public, and the technical and business people. thus, stakeholder acceptance of innovative technologies is a significant existing "barrier" which can only be overcome through incorporation of a meaningful stakeholder involvement process or approach within the technology evaluation process.

INTRODUCTION

The question which is constantly asked in the realm of technology development is: How do we ensure or enhance the acceptability of innovative technologies? Substantial attention has been paid, as it should, to the technical issues associated with innovative technologies. However, little, if any, attention has been focused on the basis for public resistance to innovative technology deployment. It is well established that public stakeholders generally pay for technology development projects, either directly through financial contributions or indirectly via costs to the surrounding environment or loss of political momentum resulting from excess expenditure on "white elephant" projects. And, it is they who will have to live with the technologies once implemented or the slashed funding for remediation projects. Yet, too often effected stakeholders are not meaningfully involved in the process of selection, demonstration and evaluation of proposed

innovative technologies. Instead, they have been relegated to audience or spectator, invited to public meetings and told what technical experts believe are promising or necessary technological solutions to the problems they face and the reasons why the public should consider those proposed technologies acceptable. Not surprisingly, this approach has alienated the public and led to extreme public resistance to technologies which technical specialists conclude should be wholly acceptable.

We are currently at a crossroads regarding public funding for remediation of the chemical and radioactive contamination at United States Department of Energy (DOE) nuclear weapons complex sites across the country. Substantial criticism has been focused upon DOE's inability to clean up the widespread contamination at it facilities. And, the fiscal train wreck envisioned with the FY'96 budget has been dramatically elevated with last November's election. The critique of DOE's fifty years of self regulation has a number of valid points: lack of contractor accountability, minimal contractor oversight by DOE, enormous financial expenditure with little tangible results, an intractable and unresponsive
culture within both the department and its primary contractors oriented solely toward production of nuclear weapons materials, rather than the careful handling, packaging, storage and disposal of these dangerous wastes. Regardless of the validity of the critique, there is one ultimate truth -- the challenge to isolate these materials from the biosphere, regardless of the risk, is enormous.

Only in the past three years has DOE's effort to consider new and advanced technology for investigation and remediation borne any fruit at all. From the viewpoint of an outside stakeholder, the Department has used research and development (R&D) money primarily to enhance political agendas and provide the main contractors with more largesse. From the late 1980's until the end of 1994, much of the public financing has been poured into large white elephant projects like the Defense Waste Processing Facility at Savannah River -- which has yet to have a hot start and is billions of dollars over original estimates. It is obvious that such projects have not been conceived as components of an integrated, strategic, complex-wide effort to understand and then to design the most reasonable approaches to remediation, but as pork to elected officials. Clearly, these white elephants represent orientations and experiences of both the DOE and its contractors which may have made sense in the 1940's and 1950's but NOT today.

Perhaps the biggest challenge faced by DOE is to develop decision-making processes which welcome the views of interested stakeholders and actually use this input to improve both the quality and public acceptance of decisions. Obviously time and patience are required for members of the public to understand technical problems and options for solving them. Nevertheless, the time must be invested, for today the success of a decision is judged by its cost-effectiveness, its ability to address identified cleanup needs, and the investment of political support from a broad spectrum of stakeholders, (including the public). Thus, while public acceptance is inherently and ultimately a political issue, its value to federal agency decision-making cannot be understated. Future R & D decisions must convince the Congress and the public that the money invested in technology development has been spent well.

We submit that our experience in the Pacific NW Laboratory's Volatile Organic Compound/Arid Lands Integrated Demonstration Project (VOC/Arid ID) has left us hope that R & D decisions can be made within a framework where money can be spent well, appropriate advanced technology developed, results produced and transferred to the private sector in a timely manner. We have concluded that a decision-making process can succeed if it is well conceived, engages and listens to a spectrum of stakeholder interests from its earliest stages, links principal investigators with both stakeholders and regulators, and does not create unrealistic expectations. Additionally, once established, the investigative and public participation processes employed in such a process can be applied to other sites across the complex.

Of course, the front end investment of time for such a process is considerable -- the VOC/Arid ID took two and a half years between initiation and the demonstration phase. And, such projects are not a panacea for DOE's technology development or remediation needs. But, it is our belief that, within the VOC/Arid ID Pacific Northwest Laboratory (PNL), the stakeholders and the principal investigators have broken new ground. Consequently, it is critical that the funding for such projects continue either at current levels or increase. DOE must ensure that the early work at Hanford provides beneficial results throughout the complex. To indiscriminately

slash the Department's EM budget when it is about to make real headway is to "throw the baby out with the bath water".

To better illustrate our point, within the remainder of this paper we will develop a comparative analysis of the VOC/Arid Id with that employed by the Federal Advisory Committee to Develop On-Site Innovative Technologies for Environmental Restoration and Waste Management (DOIT) -- a program which cannot claim equal success. Comparative analysis of these two approaches is intended to illustrate that successful development and deployment of technologies can only occur when those technologies are developed in conjunction with a "meaningful" public participation processes which effectively and systematically involve a spectrum of stakeholders, including the regulators, the public, and the technical and business people.

THE FEDERAL ADVISORY COMMITTEE TO DEVELOP ON-SITE INNOVATIVE TECHNOLOGIES FOR ENVIRONMENTAL RESTORATION AND WASTE MANAGEMENT (DOIT)

In 1991 the Western Governors Association (WGA) entered into a Memorandum of Understanding (MOU) with the Department of Energy (DOE), the Department of Defense (DOD) and the Department of Interior (DOI) creating the Federal Advisory Committee to Develop On-Site Innovative Technologies for Environmental Restoration and Waste Management, commonly referred to as DOIT. Pursuant to its Charter, DOIT is to recommend a program to implement the tasks of the MOU, which include:

identifying technology needs at Federal facilities in western states;
identifying and assessing emerging technologies in Federal and private sectors;
assessing the effectiveness of technology selection processes;
identifying regulatory, institutional or other governmental barriers to technology development;
identifying work force planning/education requirements; and
identifying new approaches to more expeditiously overcome the barriers and get better technology into industry to expedite cleanup.

(Charter, Federal Advisory Committee to Develop On-Site Innovative Technologies for Environmental Restoration and Waste Management, Dec. 10, 1992, pp. 1-2).

As envisioned within the MOU, DOIT was the vehicle by which the signatories could establish a more cooperative approach to the development of technical solutions to environmental restoration and waste management problems shared by states, commercial entities and the federal government. Moreover, this regional approach would serve as a demonstration of principles and practices which could be adopted nationally. (Memorandum of Understanding, July 22 1991, at p. 1). The objectives of the MOU are laudable. However, the mechanism used to involve the public in this process has left much to be desired.

The DOIT charter contemplates the appointment of individuals to "technical workgroups (sic)". Work groups are to be used for any purpose consistent with the Charter and are intended as fact-finding bodies providing information to be used in preparation of an annual report addressing the enumerated tasks of the MOU.Individuals appointed to these Work Groups were to represent "an appropriate mix and balance" of representatives of state government, federal agencies, tribal governments, public interest groups, environmental groups, the waste cleanup industry, the waste cleanup support industry, federal laboratories, academia and other interested and affected parties. (Charter, Federal Advisory Committee to Develop On-Site Innovative Technologies for Environmental Restoration and Waste Management, Dec. 10, 1992, pp. 2-3).

Working groups were created by the DOIT Committee, without interaction or input from stakeholders, in 1991. They include the Mixed Waste, the Mine Waste, the Munitions Waste, and the Military Bases working groups. They were formed with little consideration of the objectives of the process, the proper mix of stakeholders required to meet the objectives, or the end product desired. Moreover, once formed, working group members failed to address vital process issues. Neither the decision-making processes nor the responsibilities of either DOIT Committee or Working Group members were ever defined. Not surprisingly, the working group quickly became different things to different stakeholders.

An appropriate understanding of the weaknesses in the approach to public participation employed by DOIT is best illustrated by examining the operation of one of the DOIT working groups. Since the authors' have extensive personal knowledge of the functioning of the Mixed Waste working group (MWWG), the following discussion will, in large part, be restricted to the functioning of that Group. However, the reader should be aware problems exist throughout the entire DOIT process.

Despite the stated objectives of the MWWG and/or the statement of tasks in the original MOU, the focus of the MWWG, and thus its effectiveness, has been extremely limited. For instance, MWWG fact-finding was initially confined to a review of eighty of the most "promising demonstration projects", largely sponsored by large Management and Operating contractors or DOE laboratories, already under consideration within DOE's "funding pipeline". Nine of those eighty technologies were endorsed by the working group for "demonstration" to allow "regulators and stakeholders to evaluate whether a technology is worth being considered for actual application, permitting or commercialization". Despite references to "selection" and/or "endorsement" for "demonstration", the activities of the MWWG were actually limited to enhancement of existing demonstrations through the addition of approaches to proactive stakeholder participation, existing regulatory barriers, and barriers to commercialization. (See, Id. at pps. 20-30).

The limitation of MWWG review to technologies within the existing funding pipeline caused much concern among public interest/environmental community representatives/members. They felt that "continued reliance on traditional DOE contractors [might] foster the promotion of traditional technologies.(Id. at 21). Concerns voiced by these members were met with assurances that the limitations were necessary if the MWWG was to "generate immediate and important technical and stakeholder participation results. (Id. at 16.) Additional concern was raised public interest/environmental community representatives when it became clear that a large percentage of the technologies being "endorsed for demonstration" by the MWWG were thermal technologies. These concerns were largely ignored. In fact, no mention of any opposition to thermal technologies by public interest\environmental representatives was noted in the 1993 Annual Report of the MWWG. As feared, the absence of opposition to thermal technologies within the 1993 Report was subsequently used by DOE as justification for its continued emphasis on thermal technologies in the face of continued public opposition.

Members representing the public interest/environmental community also questioned continued DOE requests for MWWG assistance in the development of a "proactive stakeholder process". Everyone agreed that effective stakeholder participation in DOE decision-making is important. However, DOE had already invested monies in the development of a credible and highly successful model in PNL's VOC/Arid ID. Many members felt that efforts in this regard were, therefore, duplicative and a waste of scarce resources. At the same time, participants in the VOC/Arid ID regularly questioned the relationship between PNL's effort and the DOIT process.

Complaints from MWWG members were not limited to the technology selection and enhancement process. Common complaints included the lack of a clear definition of the function of stakeholders within the process, and, for that matter, a total absence of any process by which stakeholders were assured that their input was communicated to the DOIT Committee, and/or response to that input received from that Committee. Serious questions also arose regarding the membership of the MWWG. Rather than seek to ensure a balanced membership, representatives of various interests had been invited to join in a hodgepodge fashion. Differences between public interest organizations and environmental groups were ignored. Additionally, the national, regional, and local nature of various environmental and public interest groups were never considered. Similarly, little or no effort was expended in the recruitment of representatives of Native American Tribes. Finally, DOIT Process Administrators had failed, even when questioned, to consider potential MWWG member conflicts of interest. Thus, members representing Management and Operating contractors or DOE laboratory proponents of technologies could not be excluded from Working Group decisions regarding those technologies.

So serious were stakeholders' concerns that they began to question the value of their continued participation in the process. Ultimately various public interest and environmental group representative/members of the MWWG, as well as other Working Groups, called for a roundtable to discuss stakeholder enhancements at demonstration sites. In response an "Implementation Guidance Roundtable" was convened in March 1994 to address "stakeholder" concerns. However, discussions at this roundtable revealed that the real concerns of Working Group members related to the structure and process DOIT was using to conduct its business rather than the implementation of stakeholder involvement processes at demonstration sites. Accordingly, DOIT agreed to hold a new roundtable to address structural and process concerns of the stakeholders.

In the interim, continual public interest/environmental concerns voiced within the Mixed Waste Working Group precipitated a crisis of sorts. In June, 1994, in response to rising levels of member frustration, the MWWG recommended the preparation of report assessing the concerns and recommendations of stakeholders. Paul Robinson, Research Director for the Southwest Research and Information Center was contracted to prepare the report. Interviews were conducted with eight members, representing tribal, public interest and environmental perspectives, and a report submitted on September 2, 1994. Upon completion, the report was presented to and discussed by the Working Group. In response, the MWWG developed a Management Plan. The Working Group also drafted work plans for its two phased analysis of technology demonstrations -- something the MWWG had never done before. It is noteworthy that unlike Phase I, the Phase II work plan more closely parallels the original objectives set forth within the MOU -- seeking to develop a model solicitation for mixed waste technology intended to identify and address barriers to technology development. (Work Plans for Phase One and Two are Appendices) (Contact the author for a copy of the Appendices)
The process followed by the MWWG was ultimately used a the framework for a Stakeholder Roundtable held September 28-30, 1994. Approximately eighty people attended the Roundtable and developed the following general recommendations, followed by specific recommendations and proposed action deadlines, for improvement of the DOIT process:

Improve the diversity and balance of participation
Clarify purpose, structure, and process of DOIT Committee Initiative;
Seek funding and improve access for a wider range of stakeholders to participate;
Link the initiative more directly to the decision-making process; . Develop a communications and outreach plan;
Develop a strategy to address education and training issues.

(DOIT Stakeholder Roundtable Report, Dec. 5, 1994 at pps. 4-9). (A copy of the Report can be obtained by contacting the author)
Needless to say, as originally implemented the DOIT initiative was not an effective public involvement effort. The DOIT process is not yet considered credible and remains seriously flawed. Much time and effort, which should have been spent addressing barriers to technology development, has instead been required to address problems with the process which could have been avoided.
This is not to say that the DOIT initiative is not a worthy effort and/or is not salvageable. The DOIT Committee apparently heard the message voiced initially within the Mixed Waste Working Group and then even more loudly at the Stakeholder Roundtable. As a result, the DOIT Process Administrators and the DOIT Coordinating Group are seeking to address the issues raised at that Roundtable. The remaining three Work Groups have been tasked with drafting management plans, patterned on that drafted by the MWWG. Moreover, the DOIT Committee itself is attempting to revise its Management Plan to address the concerns voiced at stakeholders. Thus, two years into the process DOIT and its Working Groups may are investing substantial time in an attempt to create a credible stakeholder process for achieving their objectives. As to the success of these efforts, only time will tell.*
What is evident to all concerned is that the key procedural issues which formed the basis of many stakeholder complaints should have been addressed, with the full participation of all stakeholder groups, at the beginning of the process. Moreover, to garner credibility and stakeholder support, working group members should have been: presented with a clear statement of the purpose and goals of the DOIT initiative and of each of its working groups; given ownership of and involved early and often within the process, and should have received frequent feedback from the DOIT Committee and/or the Coordinating Group about the input they provided.
PACIFIC NW LABORATORY'S VOLATILE ORGANIC COMPOUND/ARID LANDS INTEGRATED DEMONSTRATION PROJECT (VOC/ARID ID)
The VOC/Arid ID, in contrast to the DOIT Program, invested considerable time and energy at its inception to develop a coherent program that involved the spectrum of people and organizations who had a stake in the remediation of the Hanford site. The investigation had two phases. In the first phase, PNL identified a small cluster of potential new technologies; used Comprehensive Environmental Response Compensation and Liability Act (CERCLA) regulations to establish initial criteria for acceptability; developed an interactive computer program called ProTech to provide a visual representation of the site to be investigated, and identify the specific site location of carbon tetrachloride contamination and the initial technologies to be

evaluated.
Using this information, PNL and its subcontractors sent out approximately three hundred letters to people who had shown an interest in the Tri-party Agreement (the Agreement governing cleanup at Hanford) in particular and the cleanup at Hanford in general. Follow-up phone calls were then made to secure or verify participation. The staff then sought to interview forty individuals and organizational representatives with some stake in the outcome of remediation at Hanford. Stakeholders interviewed by the staff included public and environmental interest groups, regulatory agencies, Federal state and local governments, Native American tribes, Business, Labor, Education, retired Hanford workers, and DOE site contractors/technology users. ProTech was used by PNL during the interview process and considerably enhanced, for the stakeholders, the initial interviews. Interview results were then incorporated by PNL into a refined ProTech program.
Next, stakeholders were asked to assess PNL's draft criteria for the evaluation of technologies. Input obtained in individual interviews and small focus group meetings, was used by PNL to expand the criteria. In this way, the varied concerns of the many different stakeholders were used to create an expanded matrix for technology evaluation, referred to as Criteria Matrix. Although some significant additions were made to the initial list of criteria, the stakeholder input mainly consisted of refinement of existing criteria. It is important to note that the key program staff from PNL were involved in each of these efforts and were diligent in their efforts to incorporate the concerns and/or suggestions voiced by stakeholders. Once completed, the Criteria Matrix served as the basis for technology profiles used in subsequent meetings with stakeholders. The Matrix was also used to organize conceptual test plans being developed by the principal investigators, designing and testing innovative technologies. As stakeholders began to see their suggestions reflected in the succeeding documents, trust in the process began to develop.
Completion of the Criteria Matrix and the publication of a report entitled "Phase I Involvement for Potential Stakeholders of the VOC-Arid Integrated Demonstration" (December 1992) signaled the end of Phase One and the beginning of Phase Two of the program. Although many different stakeholders participated in Phase One and Phase Two of the program, a consistent but diverse set of stakeholders participated in all of the meetings. The continuity of participation and perspectives by this set of stakeholders enhanced the discussions and expedited the process. In other words, consistent participation by the same stakeholders prevented the need for a return to zero, on the learning curve, at every meeting. Phase Two of VOC/Arid ID began in early 1993 with the preparation of technology profiles and fact sheets for four innovative technologies to remediate VOCs in groundwater: sonic drilling, in-well vapor stripping, membrane separation and in-situ bio-remediation. These profiles, as well as profiles of baseline technologies, were sent to stakeholders. Stakeholders were then brought together in three focus groups, one for regulators, one for public and environmental interest groups and elected officials, and one for principal investigators and other technologists. The purpose of these meetings was application of the Criteria Matrix to the four groundwater remediation technologies slated for demonstration. Focus group summaries were subsequently prepared and provided to participants, as well as other stakeholders. Review of these summaries, and subsequent revisions to the technology profiles and fact sheets, incorporated stakeholder comments. Thus, stakeholders were pleasantly surprised by the responsiveness of the program. Efforts by PNL to incorporate stakeholder input made stakeholders feel that they were an integral part of the process, thereby increasing their trust in the process and support for the technology demonstration.
In early 1994, PNL convened an integrated workshop to bring together the regulators, principal investigators, and stakeholders. The principal investigators and the stakeholders engaged in a day long dialogue concerning the incorporation of the Criteria Matrix into test plans for the proposed technologies.* Stakeholders were then asked to review profiles and fact sheets for three additional technologies: tunable hybrid plasma (THP), passive soil vapor extraction using borehole flux (PSVE), and freeze wall barrier technology. During the summer of 1994, PNL also expanded its outreach efforts and conducted interviews with stakeholders from the other arid DOE sites, including: Sandia and Los Alamos National Laboratories, Rocky Flats, and Idaho National Engineering Laboratory concerning all six of the innovative technologies under investigation at Hanford. As a result, the Criteria Matrix representing data requirements was again expanded and the potential for

replication of the Arid Lands Program increased.
The process has now reached Phase Three, the demonstration phase. Stakeholders have been assured that data requirements set forth within the six test plans address concerns voiced by stakeholders within Phase One and Two. Upon completion, data from those demonstrations is to be provided to the stakeholders for review.
It is important to note that throughout the VOC Arid Program, stakeholders were kept informed, their advice sought and incorporated into the ongoing development of the Criteria Matrix, and public and environmental interest representative afforded full participation with the regulators and principal investigators in the evolution of the program. If PNL had a motto for this program, it must have been "Early and Often". The investment of time and resources from the inception of the program, while slowing down field demonstrations has enhanced the prospects for success of the VOC/Arid ID.

CONCLUSION

As we began to consider submission of this paper to Waste Management '95, the contrasts between the DOIT Program and the VOC Arid ID Program came into sharper focus. While the PNL effort had a clear, understandable design, a well-defined and focused stakeholder participation component, and a realistic scope of investigation, the DOIT initiative has been seriously lacking in all three. Additionally, despite the shared goals of the two processes, there has been no attempt to integrate them. PNL staff have been included as members of the MWWG and/or participants in DOIT activities. Yet, in most instances their input has met with systematic resistance, especially on the part of DOE staff, and the integration of the two processes intentionally thwarted. Instead, reliance for input on public participation has been placed on a process delineated in a report of the Colorado Center of Environmental Management -- a process which has received little, if any, support from the public interest/environmental communities.
The message which we wish to leave you with is this: acceptance and continued public funding of technology development efforts cannot and will not occur without "meaningful" stakeholder involvement process. "Meaningful" stakeholder involvement requires substantive involvement of interested/affected individuals in the very earliest decisions about technology suitability. Stakeholder input must be accurately incorporated into test plans and evaluation criteria. Additionally, stakeholders must be shown how their advice and guidance is actually used at each stage within the process. Although requiring a significant investment of time and resources, "meaningful" stakeholder involvement efforts, such as the VOC Arid ID, will ultimately facilitate rapid acceptance and implementation of appropriate technologies.

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INNOVATIVE ADMINISTRATIVE, TECHNICAL, AND PUBLIC INVOLVEMENT APPROACHES TO ENVIRONMENTAL RESTORATION AT AN INACTIVE LEAD-ZINC MINING AND MILLING COMPLEX NEAR PECOS, NEW MEXICO

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ABSTRACT

This paper will summarize innovative regulatory, technical, and public involvement activities associated with the investigation and remediation of lead and zinc ore mining and milling waste sites near Pecos, New Mexico. Resulting from unreclaimed mining operations conducted in the 1920s and 1930s, the site has been prominent during the past decade due to related fish kills in the Pecos River fish kills, state's role as a responsible party, potential for Superfund - CERCLA listing of the site, and state participation in funding of the multi-million dollar restoration effort.
The administrative framework and reclamation technology at the mill and tailings portion of the site, El Molino, is reviewed with an emphasis on potentially transferable innovative methods. The administrative framework responds to a local interest in a "Superfund level of cleanup without the perceived cost and delay problems of Superfund" and includes an administrative order on consent and statement of work which substitute for parallels within the Superfund process. This innovative approach provides a model of state enforcement of Superfund level remediation, if

the completed restoration efforts are fully effective and timely. The administrative process includes strong stakeholder involvement initiatives such as technical assistance and community relations contractors and informal technical review meetings to enhance and focus affected community participation. Technology innovations include site characterization techniques such as heavy metal analysis using x-ray fluorescence, and remedial design techniques such as subsurface flow cutoff trenches integrated into a surface flow diversion channel, construction of replacement wetlands, and a lined, multilayer tailings cover system.

INTRODUCTION

Successful environmental restoration of industrial waste disposal sites provides employment and post-restoration land use opportunities, as well as long-term human and ecological risk reduction. Attaining a goal of successful and sustainable environmental restoration requires effective and lasting solutions to the unique complex of administrative and technical problems presented by the individual sites. Environmental professionals seeking to effectively restore waste disposal sites can benefit from an understanding of successful approaches used for other wastes and at other sites in order to broaden the range of proven effective approaches available for consideration at sites of concern. Experience of this type, at restoration projects outside the radioactive waste management arena where innovative technologies and administrative solutions have been demonstrated, is particularly important for radioactive waste managers. Such sites provide excellent examples of effective solutions which are applicable to radioactive waste problem sites. They also offer a much wider set of cases and methods to learn from than the very limited number of completed radioactive waste sites restorations. Since the range of radioactive waste forms, sites and administrative settings is so wide due to the uniqueness of each site, the opportunity to apply experience from one radioactive waste problem to another may be very limited. Conversely, effective approaches from problem sites without a radioactive waste component but facing similar administrative and environmental restoration standards may be directly relevant. A prime source of such transferable restoration technology is the mining sector where large volume, heavy metal-laden wastes present complex, multi-pollutant problems where governmental and owner-operator roles are poorly defined and affected communities are actively involved in the restoration policy and design proceedings. Many mine and mill sites involve resolution of complex regulatory, technology and public involvement concerns in rural areas, often in the western US, but certainly found in other parts of the country and in every region around the globe, which are similar to the maze of decision-making and environmental technology design problems facing radioactive and chemical waste managers.

The environmental restoration process for lead-zinc mine and mill wastes near Pecos, New Mexico offers a wide array of specific examples of effective and innovative technologies and administrative and public involvement approaches potentially transferable to a variety of radioactive, and other large-volume, long-lived, chronic-hazard sites. Particularly relevant waste forms include those with dispersed heavy metal contamination, such as uranium extraction, processing and fabrication sites; low-level, buried and mixed radioactive waste sites; as well as other hard rock mining and milling and mixed metal-organic chemical sites.

BACKGROUND

The spectacularly beautiful Upper Pecos River Valley of northern New Mexico lies 20 miles west of Santa Fe. The valley is the setting for the Village of Pecos, an abandoned Pueblo Indian community now protected at Pecos National Monument, heavily used recreational attractions associated with enjoyment of the world-renowned Pecos Wilderness and other areas in the Sangre de Cristo Mountains within Santa Fe National Forest and the high-quality fishing and camping opportunities along the river. It is also the site of a long inactive mine and mill complex which produced some 2,200,000 tons of lead-zinc ore between 1926 and 1939. The mine site at Terrero (a Spanish term for mine dump) is well known by many of the 3,500 local residents and area visitors. It borders the highway which provides access to the high country of the Pecos Wilderness and is found at the mouth of Willow Creek, a local tributary to the Pecos River. Sixteen miles downstream of the mine site is the 50 acre mill and tailings complex - known locally as "El Molino" - which covers approximately one-half mile along, and with, the streambed of the Alamitos Creek. The southern end of the El Molino site is less than one-half mile from residences in the Village of Pecos and the Village's elementary and high school complex. Since state acquisition

of the surface rights to the mine and mill sites in 1950, portions of the highway, mine and mill wastes have been hauled off-site for use in construction at several nearby state and federally managed campgrounds and a state fish hatchery. This complex of mine and mill sites are titled the "Upper Pecos Site" and are being managed as five units: the Pecos Mine, El Molino, State Recreation Use Areas, State Highway 63, and the Lisboa Springs Fish Hatchery.1 This paper will focus on the El Molino unit, as the unit with the largest volume of waste, the unit closest to a residential community and the unit with the most advance environmental restoration program as of January 1995.

Though active mining ceased before World War II, substantial data is available on the nature of the materials handled at the mine and mill including the mill tailings. The ore body is reported to have been discovered in 1881, but not brought into production until 1927 by American Metals Company of New Mexico, The multi-mineral ore averaged 10.6% zinc, 3.3% lead, 0.5% copper, 2.8 oz./ton silver and 0.1 oz./ton gold, and was processed at a 600 ton per day selective flotation mill until "water problems, labor disputes and bad ground at depth contributed to the mine's shutdown in 1939". Ore was transported to the mill and tailings site, the area of the El Molino unit, via a 12 mile aerial tramway. (2) Following removal of values, mill tailings were discharged into two impoundments behind embankments across Alamitos Canyon. The Canyon is drained by Alamitos Creek, an intermittent stream which flows through the Village of Pecos before reaching the Pecos River and capable of generating substantial seasonal flows from snowmelt and peak precipitation runoff events. A third impoundment was constructed downstream of the other two dams, presumably to contain eroding tailings from the upstream dams, approximately one mile from the confluence of Alamitos Creek and the Pecos River in the Village of Pecos. See Fig. 1 Location Map for orientation.

In the tailings disposed of into the creek downstream of the mill site, concentrations reported for key metals and other contaminants include: 2800 to 10,000 ppm - lead; 150 to 5500 ppm - copper; 130 to 13,000 - ppm zinc; 15 to 20 ppm-silver; and 2.7 ppm - cyanide. (1)

Process chemicals used in the mill include: 1) in the mica circuit - Cresylic Acid - 0.175 lb/ton of ore; 2) in the lead circuit - Lime - 0.574 lb/ton, Potassium pentasol xanthate - 0.161 lb/ton, Zinc Sulfate - 1.395 lb/ton, Cyanide - 0.060 lb/ton; and 3) in the zinc circuit - Lime - 1.848 lb/ton, Copper Sulfate - 0.783 lb/ton, Sodium ethyl xanthate - 0.500 lb/ton, Pine oil- 0.151 lb/ton. (1)

In addition to the characterization of hazardous constituents in the tailings, extensive heavy metal contamination at the sites has been well documented by New Mexico state agency and operator investigations, though a range of specific data gathering activities are on-going. By the end of 1991, the following data had been developed.

Surface water samples from Alamitos Creek showed concentrations of: lead (ranging from <0.01 to 4.1 ppm), cadmium (<0.001 to 0.022 ppm), iron (<0.1 to 67 ppm) and manganese (<0.05 to 1.4 ppm) in excess of New Mexico Water Quality Control Commission (NMWQCC) standards (Pb - 0.05 ppm, Cd -0.01 ppm, Fe - 1.0 ppm, Mn - 0.2 ppm) and United States Environmental Protection Agency (EPA) Maximum Contaminant Levels (MCLs) - (40 CFR 141: Pb - 0.05 ppm, Cd - 0.01 ppm; 40 CFR 143: Fe - 0.3 ppm, Mn - 0.05 ppm).

Filtered ground water samples showed concentrations of sulfate and manganese as well as total dissolved solids (TDS) in excess of NMWQCC standards (SO4 - 600 ppm, Mn - 0.2 ppm and TDS - 1000 ppm) and EPA MCLs (40 CFR 143: SO4 - 250 ppm, Mn - 0.05 ppm, and TDS - 500 ppm).

Soil data showed significant contamination for lead (background 5 -30 ppm, tailings 2800 - 10000 ppm, downstream of tailings - 1400 ppm), zinc (background 30 - 60 ppm, tailings 190 - 13000 ppm, downstream - 350 ppm) copper (background 9 - 20 ppm, tailings 150 - 5500 ppm, downstream - 290 ppm) and iron (background 13000 - 17000, tailings 44000 - 51000 ppm, downstream - 31000 ppm). (1)

Though the mine and mill complex were privately operated, the State of New Mexico acquired the sites, except for mineral rights, in 1950. At that time, the New Mexico Game Commission purchased the mine and mill site from Pecos Estates, Inc., the corporation to which the mine operator American Metals of New Mexico had transferred the real property and minerals from the Pecos Mine and El Molino sites in 1939 after shutdown. American Metal Company Limited of New York was the majority shareholder in American Metal Company of New Mexico. The recently formed Cyprus-Amax Minerals

Company is the successor corporation to American Metals Company. (1) In addition to contamination at the state-owned mine and mill sites, other State Game and Fish Commission lands, and land administered the New Mexico State Highway Department and United States Forest Service lands were impacted by the use of mine waste in construction, off site uses which occurred after state surface rights acquisition. At various times since state acquisition, mine waste has been removed from the mine and mill areas for use as construction or maintenance materials within the Pecos River floodplain between the Village and the mine including state highways, building pads, government -owned campsites and the state's Lisboa Springs Fish Hatchery. (3) The Village of Pecos is governed by a mayor -council system typical of small incorporated rural communities in the West, with the Mayor serving as chair of the Village of Pecos Board of Trustees. Local community organizations have been active and visible in the consideration of environmental and natural resources issues in the Valley. La Gente del Rio Pecos, an organization addressing community development and natural resource protection concerns (and a successor to the earlier Upper Pecos Association), includes members active in raising the initial concern about contamination, assuring legislative support for state funding of cleanup, educating residents and reviewing site plans. Southwest Research and Information Center has served as a technical assistance provider for La Gente. A local People for the West chapter, including the store manager at Terrero, remedial contractor employees and other residents has also participated in many site decision-making activities. Individual local residents have verbally indicated that mine and mill waste has been transported to home sites for construction and yardfill use and selected residents report that children play, and they as children played, in the colorful tailings residue along Alamitos Creek. Active local resident use of the tailings area also include extensive shooting practice and off-road vehicle play, leading to potential lead exposure and on-site vandalism. A County Road which crosses the upper end of the tailings impoundment is the only road access to several private homesites upstream of the tailings.

While some hard rock exploration activity had occurred in the upper Pecos watershed in recent years, (4) American Metal and its successor AMAX had long since left the area except for mineral ownership and historical linkage to the Pecos Estates, Inc. When clean up concerns were raised in the 1980s, New Mexico had no "State Superfund" or an inactive hard rock mine reclamation program to define a regulatory framework for site management. Initial studies where funded by federal funds provided to the state through the Environmental Protection Agency to support site characterization and ranking activities pursuant to the federal Superfund legislation and associated program implementation.

Environmental investigations by the New Mexico Environmental Improvement Division, now the New Mexico Environment Department (NMED), at the site began with preliminary site assessments in 1985-6, (1) with NMED and Cyprus-AMAX staff and consultants conducting site characterization and design studies since that time. While Pecos Valley resident concerns about mining and water quality have periodically been raised during the 1980s, significant public involvement in the Pecos site contamination and remediation date from the Spring of 1991. In March of that year, spring snowmelt carried contaminants into the Pecos River, killing 90,000 fish. Two campsites where mine waste was used in construction were closed due to contamination and the drop in the number of Pecos area visitors dropped off significantly, hurting the local economy. Though state studies to develop contaminant distribution and hazardous ranking data were available to the public and AMAX unveiled a "remediation plan" in May 1991, no administrative framework had been adopted to allocate financial responsibility and define clean up standards at that time. (5)

INNOVATIONS IN THE REGULATORY AND ADMINISTRATIVE FRAMEWORK FOR SITE RESTORATION

Since that time, major hurdles in site administration, restoration standards setting and remedial design and construction have been crossed, and at a relatively fast pace. Less than five years have passed between significant public recognition of to the installation of an effective, at least in the short to medium range, pollution control and source containment system at El Molino. While different positions continue to be expressed as to the long-term adequacy of the remedy selected, and currently being implemented; many of the public policy steps, in terms of specific administrative and public participation measures, provide a use set of program elements for consideration at other environmental restoration site. Including sites like the Pecos complex, which at one time, appeared to have no clear administrative,

funding or design solution.
Following the fish kill in March 1991, a public meeting was convened in Pecos on May 22. In an innovative effort to work cooperatively with community interests, the meeting was convened not just by government officials but actively in conjunction with the most visible local non-governmental community-based organization, the Upper Pecos Association. Concerns raised at that time were well documented and continue to be reflected in site restoration activities. Citizen concerns included potential impacts on ground and surface water quality, active interest in prompt clean up, the need to reclaim sites for aesthetic and recreational uses, lead clean up standards and processes, downstream impacts, natural resource damage fishery impacts and use of local contractors. Parallel to the citizen concern for prompt clean up, both state officials and AMAX representatives expressed a perception that the area could be cleaned up more quickly, economically and correctly if EPA could be prevented from including the site on its Superfund (National Priority) List. (5) This "threat of Superfund" served as a common point of concern for the parties involved in the development of an "Administrative Order on Consent (AOC)" which defined the administrative and financial responsibility of the site owners and their successors, which include both the State of New Mexico and Cyprus-Amax (through its Amax Resource Conservation Company unit). The AOC was signed on December 2, 1992 by three representatives of the State of New Mexico - NMED, NM Game and Fish Department, and NM State Highway Department -each signed, as well as AMAX. (6)
The Pecos Administrative Order on Consent (7) has proven to be an effective and innovative approach to inactive waste site remediation decision-making in several critical ways. Firstly, it created an administrative framework for remediation and inter-agency communication where none had existed in state law; a framework within which governmental and private organizations agreed to share responsibility (without any party acknowledging their role in creation of the problem as stated in many "settlement" type agreements). Secondly, it contributed to the impressively strong and diverse support for appropriation of the $5,000,000 by the New Mexico Legislature, for the states on-fifth share of the estimated cleanup cost. Thirdly, the AOC has also been recognized outside New Mexico, where it has served as a working model in Idaho at the Triumph Mine site - also an inactive mine waste site with both state government and private participants and a common interest in avoiding Superfund Listing - where it has been a valuable example used in restoration decision-making by the responsible parties and local residents. (8)
The AOC, with its two attachments - a "Cost Allocation Agreement" and "Statement of Work" - establishes a site specific remediation program which is designed to provide a functional parallel to Superfund with respect to clean up criteria, remediation funding, timely performance and public involvement. The AOC and attachments restate directly and by reference the full range of Superfund criteria and criteria from other applicable state and federal requirements to be attained by the parties . Thus the AOC serves as a jointly-agreed upon demonstration to state and federal regulators that the remediation process will be at least as effective as a Superfund program, since all important Superfund milestones are incorporated in the AOC. This restatement of Superfund criteria allowed the responsible parties to provide EPA an enforceable foundation for successful site remediation without final Superfund listing, as long as substantial progress towards Superfund-level goals is maintained. An important benchmark for the responsible parties in their effort to accomplish reclamation outside the Superfund process was a Memorandum of Understanding between EPA and NMED which describes the cooperative roles of the agencies in the oversight and enforcement of response activities at the Terrero Mine Site. This cooperative role provides for implementation of the restoration program under the AOC, with EPA oversight to assure that site restoration is as effective as it would have been were Superfund, and other relevant federal laws, applied. (9)
Innovative elements in the Administrative Order on Consent include:

A Cost Allocation Agreement which establishes a commitment to fund the remediation with a cost sharing ratio of 80% Amax and 20% State funds, if the State of New Mexico appropriates it share of the funds through its legislative allocations. This condition was an effective incentive for New Mexico legislators to make a $5,000,000 special appropriation in 1993.

A Statement of Work (10) which provides for implementation of a schedule of itemized milestones including of health and environmental risk assessments, community relation plans, remedial investigations, feasibility studies, natural

resource damage assessments, long-term operation and maintenance plans and other performance objectives "in accordance with CERCLA and the National Contingency Plan (NCP) and shall meet the requirements and goals of a CERCLA RI/FS and RD/RA, ... Community Relations Plan consistent with the NCP", and technical assistance grants including reference to EPA guidance documents .

A set of numerical Applicable or Relevant and Appropriate Requirements (ARARs) to guide investigations and remedial design and demonstrate Superfund level remediation as well as compliance with other state and federal requirements;

Establishment of a document repository for all project materials in Santa Fe and Pecos;

Requirements for Site Health and Safety and Quality Assurance Plans, to meet federal standards for all site workers and data gathering activities;

A basis for selection of long-term remedial actions which demonstrate (among other criteria): long-term effectiveness and permanence; compliance with applicable public health and environmental standards; cost-effectiveness and an ability to be implemented; reduction of toxicity, mobility, and volume, and community acceptance.

A schedule, in the SOW, which outlines a timetable for development and implementation of Decision Documents for each of the five operable units including preparatory studies leading up to the selection of remedial action in the Decision Documents (DD).

A Dispute Resolution Process for the AOC parties to provide a mechanism for addressing disagreements without abandoning the overall framework.

Limitations in the AOC are also apparent, limitations which have partially undercut the achievement of timely and effective remediation. These include: 1) a lack of an explicit role in decision-making for affected community residents and organizations, interests who have continued to be active in the restoration process since the initial recognition of the problem and identification of issues to address during reclamation; 2) a lack of clarity in the roles and funding for the state agencies, which include both regulatory roles in the Environment Department, and "site owner" roles for both the in the Game and Fish and the Highway Departments; and 3) a lack of a clearly defined role for the state Office of the Natural Resource Trustee, an organization which had no operating experience prior to the AOC and which was lead by an individual with multiple roles in the AOC (the State Natural Resource Trustee was also the Chair of the state Game and Fish Commission).

INNOVATIVE PROCEDURAL AND DESIGN ELEMENTS IN THE FINAL DECISION DOCUMENT AND REMEDIAL DESIGN FOR THE EL MOLINO SITE

The transition from the need to reclaim a site to actual reclamation on the ground can be so problematic that simple attainment of successful progress toward overall reclamation goals is often the best measure of whether innovative actions taken are effective. Using this criteria, the progress to date at the El Molino site demonstrates broadly successful application of innovative environmental technology, though initially anticipated schedule dates have often proven to be unrealistic. The sequence of activities from the AOC stage through to the development and implementation of the site specific El Molino Decision Document provides extremely useful experience for interests attempting to identify successful cases of applied reclamation. These activities at El Molino are all the more notable in light of the limitations in the AOC that could have made El Molino reclamation particularly difficult. First off, El Molino became the testing ground for the AOC as the milestones set out in the SOW for the El Molino Site resulted in that site being the first site to be addressed by a remedial action Decision Document (DD). Second, El Molino initially was not recognized to be as major contamination and remedial design problem as its location and size merited, due to the press and responsible party focus on recreational and fishery impacts in the Pecos River near the Pecos Mine site.

These concerns led to public concerns about the timing and implementation of the DD, surfaced with the distribution of an original DD in April 1993. This interim Decision Document (11) identified a selected remedial design with conditions, presented updated contamination data a detailed set of Applicable or Relevant and Appropriate Requirements (ARARs), and a set of Statutory Determinations which concluded that "actual or threatened releases of contaminants and hazardous substances from this site, if not addressed by implementing the response action selected in this [DD], may present an imminent and substantial endangerment to public health, welfare or the environment" and that "community and state acceptance

is favorable to this remedy in comparison to other alternatives presented to the public."
The interim DD was responded to very strongly by La Gente del Rio Pecos, which asserted that the development of the document had been a very closed process limited to AOC parties with no effective opportunity for public comment, along the lines of the public meeting, workshops and newsletter which typify Superfund community involvement efforts when NCP guidance found in the AOC is followed. No public meetings had been scheduled prior the issuance of this "first final" DD. La Gente also raised a substantial array of procedural and design concerns with NMED including the failure of NMED to provide any basis for the conclusions that either the statutory requirements or community involvement requirements in the AOC and SOW had been meet. (12) The AOC included, directly and by reference, clear requirements for public meetings and a contractor-based community relations plan, including direct reference to NCP guidance, before issuance of a final DD. These were critical concerns for the residents who face the site-based risks daily and yet are provided no role in the AOC process other than through comment period opportunities and the (still anticipated) community relations plan.
NMED Secretary Judith Espinosa and staff responded positively to these concerns. They rapidly agree to meet with La Gente representatives and responded to the comments by agreeing to hold a public meeting in Pecos and extend a comment period after the meeting to allow additional written comments prior to the publication of a Final DD. This meeting of June 15, 1993 was advertized with bilingual flyers in local mailbox and hosted by the Village of Pecos attracted almost 100 attendees. The public meeting and extended comment period provided by NMED were important results of community-based stakeholder involvement in the reclamation process. The on-going benefits of these processes included heightened local community and local government involvement in the process, in addition to continued La Gente involvement, and significant improvement in the final DD issued in September 1993. (1) Village of Pecos involvement has been dramatically enhanced as a result of increased elected official involvement and the technical efforts of Village's technical assistance contractor on the Upper Pecos Site, New Mexico Engineering Research Institute. The Village was allocated $50,000 by the 1993 New Mexico Legislature for the technical assistance contractor, to provide a rough parallel to the Superfund Technical Assistance Grant (TAG) available for up to $50,000. In a twist from the Superfund framework though, the New Mexico Legislature provided the funds to a unit of local government, not a independent non-profit corporation, in sharp contrast to the Superfund framework which does not allow TAGs for municipalities and requires an independent non-profit as a grant recipient. Improvements in the September 5, 1993 Final DD were incorporated into the statutory determination and design condition portions and are reflected in the summary of the Final DD below.
The Final DD reviewed site contamination data and summarized six alternatives remedial designs, selected a preferred alternative and identified 19 additional items or conditions related to the NMED approval of the preferred alternative. The preferred alternative - called "Flood Conveyance Through a Channel"- was selected over the other five titled "No Action", Institutional Controls", Flood Conveyance Through a Pipe with Flood Attenuation", Flood Attenuation with Multiple Upstream Dams", and "Reprocessing of Tailings Either On- or Off-site". Design elements in the selected alternative, including the additional items include:
Consolidate all tailings and contaminated soil with lead concentrations above the health based risk level (set at 500 parts per million) into [the two largest] tailings ponds;
Convey surface water through tailings ponds 1 and 2 via a lined, stable channel designed for a 6-hour 100-year storm event;
Reinforce side drainages leading into the main channel to minimize erosion and design for a 10-year 24-hour storm and a safety factor of 1.5 for drainage bottoms and banks;
Cap tailings ponds to minimize erosion (to less than 1/16" per year) and ponding and revegetate with native plant species;
Stabilize dams to meet state embankment engineering standards;
Regrade and revegetate with native species all borrow areas;
Replacement of wetlands lost during remediation at an appropriate location within Alamitos Canyon;

Install piezometers in tailings ponds 1 and 2 and monitor along with existing and new ground water monitoring wells until compliance with ARARs for eight consecutive quarterly samples is approved by NMED;

Evaluate potential contamination of downstream water courses and agricultural fields near the site, including soil and tissue sampled, as part of Health and Ecological Risk Assessment, required before final approval of the remedial action;

Develop, install and maintain measures to protect newly reclaimed areas and prevent vandalism;

Conduct computer modelling of ground water flow and contaminant transport, develop contingency plans to protect community and private wells, and replace two existing private wells;

Develop and implement programs to assess effectiveness of the remedy including but not limited to assessment of moisture and metals movement from the tailings ponds, revegetation success, channel system and liner stability, tailings cap stability, wetlands replacement success, and surface and ground water quality;

Develop a long-term operation and maintenance plan to be approved by NMED upon attainment of ARARs and Remedial Action Criteria in the SOW, to be subject to review every five years.

INNOVATIONS IN SITE CHARACTERIZATION, REMEDIAL DESIGN AND CONSTRUCTION AT THE EL MOLINO SITE.

Site remedial design and construction activities have been conducted by Cyprus-Amax and their primary contractors on the project, Kenneth R. Paulsen Consulting, Woodward-Clyde Consultants and Daniel B. Stephens and Associates. All project costs are subject to the 80% Cyprus-Amax-20% New Mexico cost sharing arrangement in the CAA, resulting in Cyprus-Amax's consultants being subject, in part, to acceptance of their work products by the state as a condition of payment. This role of private contractors receiving payment from public funds appears to have enhanced the responsiveness of project contractors, as their staffs have continually recognized their responsibility to both their private contractor and the public, who through the state are also paying the consultant's bills. A full range of tailings, soil, water, and air quality sampling and analysis activities have been conducted at the site and documented in NMED-maintained document repositories in Pecos and Santa Fe.

Two innovative aspects of the site characterization efforts are notable, though highly accurate and reliable data gathering and analysis methods have been used throughout the project. A first key innovation in the process has been the early acceptance by the responsible parties of a Quality Assurance Project Plan (QAPP); a plan developed pursuant to the AOC and in conformity with EPA guidance documents identified in the SOW. The QAPP was prepared by Cyprus-Amax consultants and approved by NMED shortly after completion of the AOC. The early acceptance of this Quality Assurance Plan had several important results. The acceptance of the QAPP demonstrated that the parties could reach agreement on technical matters in addition to the administrative framework and allowed the cooperative approach to site characterization and reclamation to be further reinforced. Acceptance of the QAPP insured that all parties, including AOC parties and the public, would be able to rely on a common data base with results including reduced costs as redundant "competing data costs" were largely eliminated, allowing technical reviewers to focus on what-the-data-means, rather than was-the-data-accurate questions.

A second innovation has been the use of x-ray fluorescence (XRF) techniques for soil and tailings characterization for heavy metals, particularly lead. Public concerns for lead contamination and clean have lead to a need for the accurate mapping of the distribution of lead contamination and associated considerations of lead cleanup levels and clean up costs. The XRF method has been used to characterize lead levels at the Terrero Mine and El Molino Units in much greater detail and much faster than conventional techniques, such as off-site laboratory based inductively coupled plasma (ICP) and atomic absorption spectrometry (AAS) methods, due to the relatively low cost per sample, without sacrificing accuracy. NMED research shows a good correlation (correlation coefficients for lead and zinc comparative analyses were 0.70 and 0.74, respectively) between ICP and XRF data at the Terrero mine. (13) On-site chemical analysis using portable instrumentation, with essential real time data production allows up to 100 samples or more to be analyzed at the El Molino site. (14) Additional advantages to XRF include improved worker safety resulting from elimination of acid digestion steps in conventional analysis and the nondestructive nature of the method, which leaves samples and standards available

for future investigation. (15) XRF data has been used to develop 200 ppm, 500 ppm, and 1200 ppm contours for lead concentrations in soil at the El Molino site relying on several hundred individual data points.
The remedial design characterized in the Final DD has evolved in several important and innovative ways as a result of restoration designers effectively incorporating both new information gathered during investigation as well as construction and technology improvements identified though public comment and agency review. Key motivations for these design enhancements are technical comments provided by the Village of Pecos, La Gente del Rio Pecos and their technical assistance providers. Both parties have raised concerns about the long-term, the hundreds to thousands of year time, durability of the existing diversion channel, and ground water flow under and through the tailings as placed. (16) While this site is designed to pass surface water generated by a nominal 100-year event, substantial concern among residents has resulted in the Village of Pecos having passed a resolution recommending removal of the tailings to a "out of creek" site for permanent disposal, and consideration of the existing diversion mechanism as an interim, multi-year but not-multi-decade solution. (17) La Gente concerns have also included the need to insure that long-overdue health and ecological studies and community relations and long-term operation and maintenance plans are completed before, rather than after, the final remedial design is approved. These concerns for the reliability of the diversion channel concept was heightened when a leak of acid water from the tailings into the diversion channel shortly following liner installation was observed by Village and legislative committee representatives in the fall of 1994. (18)
These potentially volatile differences have been address substantively in an innovative approach to intergroup communication. The AOC and the public participants have maintained regularly, approximately monthly, working meeting with an informal yet technical detailed discussion focus using a consensus agenda and supporting technical working documents tone. These meetings have been held in response to La Gente and the Village Mayor's concerns expressed at Village of Pecos Board of Trustees meetings; initially at the invitation of Cyprus-Amax at the site, and most recently in Santa Fe hosted by NMED. Specific design elements of note include upgrade of the diversion channel liner and cutoff trench system, replacement wetlands plans and the long-term performance of the tailings cap.
A major environmental protection milestone achieved at El Molino is the control stabilization and control of off-site movement of tailings by embankment repair and construction of a 40-mil PVC and rip-rap lined diversion channel to carry surface water above the tailings. Engineered concrete cutoff trenches, footed into excavated bedrock according to the installers, complement the system by providing a barrier to subsurface flow in the Alamitos Creek alluvium and serve as a anchor for segment of the welded PVC liner. However the liner leaked within months of installation. (18) Cyprus-Amax and the state's response has been three- fold: 1) raise the channel base above the water table within the tailings (considered to the source of the leak); 2) maintain design flow volume by widening the channel; and 3) installing a horizontal piezometer and additional vertical piezometers at the cutoff trench site where the leak occurred. Benefits of this design modification are considered to be: reduced risk of leakage by locating the channel and liner above the local water level; availability of a drainage mechanism to prevent leakage even if the water table in the tailings again rises; and a more comprehensive groundwater monitoring network for the overall design.
Wetlands replacement is an integral part of the El Molino DD and is considered an important part of the plan by all parties. While debate continued on the acreage of wetlands replacement needed, initial construction of a wetlands has begun at the south end of the site, were previously deposited tailings have already been removed. Initial Cyprus-Amax plans includes a cattail dominated wetlands system utilizing seepage from the lower tailings dam site as a water source. Recent informal working group discussions has focused on the need for wetlands restoration to replace pre-existing subsurface flow-, rather than surface flow-, dominated wetlands as originally found at the site and the recognition of the uncertain volume and quality of the tailings dam seepage flow. The working group has also used pre-reclamation plant species list to establish revegetate planning and survival criteria. Design criteria and performance specifications for the replacement wetlands are currently being developed. (19)
The tailings cap has also been the subject of substantial working group discussion.

The original proposal of a 15 inch cap of borrow soil, with agricultural limestone and a bactericide below and revegetation by shallow-rooting plants was determined to be "inadequate, ... at the least, an additional layer must be added that will act as a barrier for water, plant roots systems and burrowing animals". (20) In response Cyprus-Amax proposed, in January 1995, a typical cap section which added a 30-mil PVC liner overlaid by a geoweb drainage mat with a fabric cover below the 15 inch soil cap. Public and regulatory agency review of this proposal were in process as this paper was prepared. While the design modification responds to agency and public concern for more effective barriers layers, documentation of the relative permanence and long-term operation and maintenance aspects of the design have yet to be distributed.

As a final element in the set of effective innovations at El Molino, construction activity at the site has relied heavily on local, meaning Pecos-area, construction contractors and heavy equipment operators. This reliance on a local workforce, can be considered innovative, as it responds to a strong community-based concern and has not been mandated by project decision documents.

CONCLUSION

This overview identifies a wide array of innovative approaches to environmental restoration currently being applied at a complex heavy-metal contaminated waste site. This "effectiveness-based-on-experience" measure of success provides a potentially more valuable assessment of specific innovations than, determinations of technical merit isolated from real world application. This transfer from proven model to full-scale application is so extremely difficult because site specific conditions, rather predetermined and controllable design conditions, usually dominate environmental restoration decision-making. As a result, the identification of effective innovations which are transferable among environmental restoration projects requires a clear understanding of the administrative, public policy and technical aspects for both the innovative demonstration setting and the potential transfer site.

Few if any sites are amenable to single step, "magic bullet" type of restoration decision-making and environmental restoration. Recognizing that full long-term restoration is accomplished in a step-by-step way allows site managers and technicians to focus on innovation designed to achieve the sequence of incremental milestones necessary to achieve site-wide goals. This summary of innovative aspects of the Upper Pecos Site, El Molino Unit demonstrates how well-prepared site specific solutions can lead to substantial environmental restoration progress on a broad front by carefully tailoring the application of design innovation to the full range of site concerns. The creative solutions at the Pecos site, offer a full range of innovative concepts and designs for resolution of site-specific waste management dilemmas, including radioactive waste sites, in many places.

The author wishes to acknowledge the many participants in the El Molino cleanup process. This includes Cyprus-Amax and their consultants, the State of New Mexico representatives, Village Trustees and their consultants, and in particular, La Gente del Rio Pecos and the residents of Pecos, who will depend on the environmental restoration of the El Molino site for the foreseeable future.

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TECHNOLOGY CERTIFICATION AND TECHNOLOGY ACCEPTANCE: PROMOTING INTERSTATE COOPERATION AND MARKET DEVELOPMENT FOR INNOVATIVE TECHNOLOGIES

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ABSTRACT

In the past two years, public and private efforts to promote development and deployment of innovative environmental technologies have shifted from the analysis of barriers to the implementation of a variety of initiatives aimed at surmounting those barriers. Particular attention has been directed at 1) streamlining fragmented technology acceptance processes within and among states, and 2) alleviating disincentives, created by inadequate or unverified technology cost and performance data, for users and regulators to choose innovative technologies. Market fragmentation currently imposes significant cost burdens on technology developers and inhibits the investment of private capital in environmental technology companies. Among the responses to these problems are state and federal technology certification/validation programs, efforts to standardize cost/performance data reporting, and initiatives aimed at promoting interstate cooperation in technology testing and evaluation. This paper reviews the current status of these initiatives, identifies critical challenges to their success, and recommends strategies for addressing those challenges.

Introduction

Public and private efforts to promote the development, deployment, and commercialization of innovative environmental technologies have accelerated in the past two years. During this recent period, the focus of federal, regional, and state

activities -- as well as the involvement of the private sector -- has shifted from the analysis of barriers to the selection of innovative technologies to the development and implementation of initiatives aimed at surmounting those barriers. Other published documents have catalogued and analyzed these barriers in some detail; that is not the focus of this paper*. Rather, this paper describes the status of several recent initiatives at the federal, regional, and state levels which attempt to address two of the most critical barriers identified by this earlier research. Specifically, these initiatives focus on:

streamlining fragmented technology acceptance processes within and among states; and

alleviating disincentives, created by inadequate or unverified technology cost and performance data, for technology users (e.g., engineering consulting firms, remedial project managers) and regulators to choose innovative technologies.

Fragmented technology acceptance processes within and among states, created by varying regulatory and procedural requirements as well as by differences in the values, interests, and influences of local community stakeholders, have the effect of fragmenting markets for environmental technologies. Markets are also difficult to assess and penetrate because acceptance of a technology for full-scale deployment depends upon the availability of credible technology cost and performance data, which is often insufficient in the eyes of users and regulators who are not generally rewarded for taking risks with new technologies.

Historically, technology vendors and developers seeking commercial acceptance of their technologies have been forced to adopt a strategy of conducting repeated treatability studies and technology demonstrations in multiple jurisdictions or markets. Eventually, sufficient experience with the technology is accumulated in enough separate locations that the technology becomes more readily accepted by users and regulators. This appears to have been the pattern with technologies such as soil vapor extraction and certain bioremediation techniques, which are now in such widespread use that the term "innovative" may no longer be entirely appropriate to describe them.

However, the time and cost burdens required by this approach have been identified by technology developers, especially small companies, as a major obstacle to innovation and a source of often unbearable financial strain on commercialization efforts*. Likewise, venture capitalists have indicated that the fragmentation of markets created by multiple state approval requirements and the uncertainties such repetitive demonstrations create for estimating return on investment significantly inhibits capital investment in environmental technology companies, further threatening small companies' financial viability*.

RESPONSES TO THE PROBLEM

Recent efforts at the federal, regional, and state levels to address these problems are of three types:

state and federal government certification or verification/validation programs, aimed at providing credible cost or performance information that users and regulators in multiple jurisdictions can use to make more certain judgments about the application of innovative technologies;

interstate regulatory cooperation initiatives ranging from multi-state involvement in individual technology demonstrations to more ambitious regional and national efforts to streamline permitting for innovative environmental technologies; and

federal and regional initiatives aimed at standardizing cost and performance data reporting protocols.

Programs and initiatives in these three areas are described below.

Certification and Verification/Validation Programs

While the U.S. EPA has conducted focused verification programs in a few select technology areas for several years (e.g., the SITE program), the agency has recently embarked on an ambitious new initiative -- EnTICE -- to extend validation of cost and performance to environmental technologies in more than a dozen technology categories. Beginning with a few pilot efforts, EnTICE will develop standardized testing protocols with the assistance of stakeholder advisory groups. Those protocols will then be used by a variety of selected verification entities (e.g., universities, federal labs, private testing firms) to verify the cost and performance of interested vendors' technologies.

In a related initiative, EPA's Environmental Monitoring Systems Laboratory has joined in a partnership with the Department of Defense, the Department of Energy,

the Department of Commerce and the Small Business Administration to form the Consortium for Site Characterization Technology. The agencies will collaborate on field demonstrations and evaluations of site characterization technologies, including a third party verification. One planned project involves DOE's Sandia National Laboratory, which will develop a field demonstration workplan for the U.S. Navy to test its Cone Penetrometer Laser-Induced Fluorescence technology (also known as SCAPS -- Site Characterization and Analysis Penetrometer System) in accordance with EPA guidelines for verifying effectiveness.

At the Department of Defense, the Environmental Security Technology Certification Program was initiated this fiscal year to help get technologies out of the laboratory and into the field with the help of third party validation of cost and performance data generated in demonstrations. The goal is to provide data that DOD users and federal and state regulators can accept so that repeated demonstrations of technologies can be avoided. Next year, the program plans to collaborate with the Department of Energy to test DOE technologies at military sites, with DOD providing the independent assessment and validation.

The State of California last year began a pilot program to certify the performance of hazardous waste technologies. Twelve technologies have been certified to date, based on reviews of existing test data submitted by applicants. The program, which operates on a fee-for-service basis, is partly intended to streamline intra-state regulatory acceptance by providing verified technical data for use by local and regional regulatory jurisdictions within the state that may not have the resources to validate vendor's claims. The local and regional entities could then focus their resources on assessing technologies against local or site-specific acceptance criteria.

Forthcoming regulations will also create a second type of certification -- regulatory certification -- that will streamline regulatory requirements for applicants by qualifying the technology for regulation under one of the less stringent tiers of the state's tiered hazardous waste permitting program.

California also is exploring the possibility of reciprocal arrangements with other states, whereby California's certification could be used by other states in their regulatory decisions and California, in turn, might accept cost and performance data generated under similar certification programs in other states. At this time, California is the only state with such a program. The State of Washington recently authorized the creation of a certification or validation program, in conjunction with Oregon and British Columbia, to promote streamlined permitting among those jurisdictions. In addition, the initiative will explore a linkage to DOE's Hanford site that would provide for technologies certified under the program to receive preferred status in procurements*.

Interstate Regulatory Cooperation Initiatives

The "DOIT" (Develop On-Site Innovative Technologies) initiative, a joint program of four federal agencies and the Western Governors' Association, began in December 1992 to expedite cleanups on western federal facilities and support the commercialization of new environmental technologies. Fourteen sites were chosen in 1994 to test the effects of specific demonstration process enhancements on federally funded technology demonstrations planned for 1994 and 1995. Several of the proposed DOIT demonstration enhancements involve regulators or other stakeholders from other sites in the region in demonstration planning or execution, with the goal of streamlining regulatory acceptance of the technologies in future applications:

Demonstration plans for a low temperature thermal desorption technology at DOE's Rocky Flats site in Colorado were discussed with regulators from five other interested states. Representatives from Kentucky, Ohio, New Mexico, Texas, and California were asked to supply information about regulatory requirements in those states that may be compared against performance measures to be assessed in the demonstration.

For DOIT demonstrations of off-gas treatment technologies at McClellan Air Force Base and the SCAPS technology at Port Hueneme Naval Base in California, CalEPA has invited other sites to participate in the design or review of demonstration plans and results.

In New Mexico, the demonstration of advanced landfill cover designs for mixed wastes at Sandia National Labs will incorporate review and comments from seven states on the demonstration design and acceptance criteria.

Information on bioventing demonstrations at DOD sites across the country will be

compiled by Hill Air Force Base in Utah for use by regulators in multiple states.
In December 1994, western governors charged the DOIT initiative with developing by June 1995 a framework for interstate cooperation on permitting innovative environmental technologies. Governors asked that the framework be operational by June 1996. According to Jim Souby, Executive Director of the Western Governors' Association (WGA), three possible levels of cooperation are being considered:

broader adoption by states of a Permit-By-Rule system, possibly patterned after California's, that would allow technologies meeting certain criteria to be permitted in states more expeditiously;

joint permitting activities, such as those currently being tested in some DOIT demonstrations, where multiple states would participate in a demonstration at a host site; and

cooperation based on common cost and performance protocols that would expedite subsequent state acceptance without active participation by those states in demonstrations themselves*.

Another outgrowth of the DOIT initiative is a proposal by the Western Governors' Association, which DOE has agreed to fund, to hire a regional "circuit rider" to assist western states with innovative technology issues and promote broader regional acceptance.

Elsewhere, the Southern States Energy Board, with DOE and EPA support, has initiated a pilot program with South Carolina and Georgia to streamline the permitting process between the two states on for technologies addressing contamination plumes. The pilot will also investigate the use of data management and integration technologies. If the pilot is successful, SSEB intends to expand the program regionally on a state-by-state basis*.

Finally, in an effort to share the experience of all of the above approaches and to identify additional strategies for states across the U.S., the states of California and Texas, WGA, SSEB, EPA and the Association of State and Territorial Solid Waste Managers conducted an interstate cooperation meeting February 7-8 in Denver. Participants, mostly state and federal agency representatives, discussed their activities and interests in promoting innovative environmental technologies and developed some initial action plans for facilitating interstate cooperation. Some of the specific strategies discussed by the group included:

EPA's Technology Innovation Office will establish a file in their CLU-IN electronic bulletin board for sharing information on technology demonstrations and other issues of interest to the group.

A task force formed by the group will develop summaries of interested states' regulations , standards, policies and other requirements governing the development, demonstration, and application of specific environmental technology types. A list of technical parameters for those technologies will also be developed and reviewed by participants in specific demonstration projects. The information will then be used to develop protocol templates for use in future applications of the technologies.

Another group will compile case studies, if available, on which innovative technologies have reached commercialization, which have not, any why, with recommendations for next steps.

Standardization of Cost and Performance Data Reporting

A number of groups have attempted, in various fora, to develop standardized cost and performance reporting formats for innovative environmental technologies. An effort led by the Western Governors Association last year generated a list of detailed questions to be addressed in technology demonstration reports.

In addition, an ongoing effort focused reporting data on full-scale technology applications by federal agencies is being pursued by the Federal Remediation Technologies Roundtable. The Roundtable issued a document last year describing the format, and representatives of the participating agencies will meet in Atlanta later this year to share examples of reports using the common format.

SOME CHALLENGES COMMON TO NEW INITIATIVES

While all of these related initiatives hold some promise in addressing barriers to innovative technology development, deployment, and commercialization, they also face several common underlying challenges. Research on regional stakeholder involvement in technology demonstrations conducted by Battelle, the Pacific Northwest Laboratory, and Environmental Issues Management, under the Department of Energy's VOCs in Arid Soils Integrated Demonstration Program, has shed new light on these challenges*.

First, no single, pre-defined set of technology acceptance criteria -- whether applied as part of a certification or validation program or utilized operationally as a testing protocol -- will address the full scope of data needs, institutional prerogatives, or implementation issues that decision makers, including communities, use to evaluate technologies, even within focused technology categories. Although some participants in regional discussions have openly wished for formal interstate reciprocity -- meaning acceptance by one state of another state's regulatory permitting decision -- most players in this debate recognize that the most realistic outcome is to identify a core set of data needs that are held in common by technology decision makers in multiple states. Then, perhaps, after a preliminary demonstration at one site successfully captures these core data, more streamlined and less costly demonstrations can be conducted at subsequent sites to address local or site-specific acceptance criteria.

Even so, a second challenge is that the proportion of the total scope of acceptance criteria can be captured in this core data set is uncertain. PNL's research indicates that considerable commonality exists among the general criteria most often raised by stakeholders from site to site; not surprisingly, technical effectiveness and cost factors are almost always mentioned. However, there is some evidence to indicate that greater specificity in the types of technologies being assessed (e.g., in situ bioremediation technologies) generates more specific, idiosyncratic data needs on the part of interested regional stakeholders. A greater focus on single technologies also creates another challenge to broader acceptability: for most stakeholders, acceptance decisions are not focused on single technologies, but on the entire technology system that is required. Thus, the acceptability of soil vapor extraction, for example, will probably be a function not only of the cost and performance of the extraction system, but the associated off-gas treatment technology as well.

A third challenge to technology acceptance is that the relative importance of different acceptance criteria will vary from locale to locale. In some cases, critical "deal-killers" may be criteria idiosyncratic to state- or site-specific concerns and not be captured in a core set of criteria. For example, while the creation of hazardous secondary wastes is an important evaluation criterion for treatment technologies everywhere, this criterion may receive primary consideration at sites, such as Rocky Flats, where storage capacity is a critical problem. This challenge creates a difficult dilemma for initiatives aimed primarily at promoting interstate regulatory cooperation, since many of these idiosyncratic, but "deal-killing" criteria are likely to be a function of public values and preferences not enshrined in regulations or policy.

Thus, the relationship between certification, or validated demonstration of a core set of cost and performance data, and the predictability of commercial deployment -- a key criterion for investors -- is not yet established.

A solution to these challenges may be found in combining a certification program like California's or EPA's with a planned outreach effort involving decision makers (and decision shapers) in targeted markets and designed to identify potential "deal-killers" in specific application settings as well as validate assumptions about the weighting and scope of acceptance criteria. Regulators and other stakeholders influencing technology acceptance may find greater assurance in both the validation provided by a regulatory body and the availability of credible data addressing their site-specific or local concerns.

Another method which could have value in promoting interstate cooperation and market definition is for more states to consider certification or validation programs similar to California's. The unpredictability of technology acceptance that is a major inhibitor to commercial investment could be significantly improved, at least in the short run, to the degree that such programs require states to more explicitly document the criteria that lead to market acceptance in their jurisdictions.

More generally, the growth in interest in this area is generating more information about acceptance criteria that, eventually, may be transformed into more robust assessments of technologies' maturity and commercial potential. That, in turn, will stimulate greater investment in these important technologies, allow users and regulators to tailor demonstrations and applications more appropriately to the capabilities of the technologies, and reduce the risks that make technology decisionmakers reluctant to employ them.

Session 29 -- Rules and Regulations
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29-1
WHATEVER THE STATE WANTS: RCRA PERMIT CONDITIONS
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ABSTRACT
DOE's mixed wastes have not been completely characterized, facilities for waste treatment were not built and operated, and few mixed wastes have been disposed. As a result, DOE has upwards of 600,000 cubic meters of mixed waste (HLW, TRU, and LLW) in inventory, with additional quantities of waste expected to be generated by D&D and waste cleanup activities.
The 1984 RCRA amendments require waste to be treated to EPA specifications before being disposed of. Waste treatment requires a permit (issued by a State or by EPA), and regulators can attach a variety of conditions to the permit when it is issued. Some of these conditions are nowhere to be found in regulations, nor are they specifically authorized by law. Rather, regulations, can rely on a general grant of authority to protect public health and the environment.
This broad grant of authority to the regulators, together with the 1992 Federal Facility Compliance Act, gives the States an unprecedented opportunity to plan for mixed waste management. Of course, any plans will have to be reviewed by the Congress, as DOE is not allowed to spend funds that Congress has not appropriated.
HISTORY
For several years after the 1980 RCRA hazardous waste regulations were promulgated, until the 1987 byproduct rule, DOE maintained that its mixed wastes were not subject to State or EPA regulation as hazardous waste. (Until 1984, DOE maintained that all of its wastes were not subject to hazardous waste laws.) A DOE order, since rescinded, required DOE units to obey the substantive standards set forth in the environmental laws, but not comply with administrative provisions (permits, inspections, etc.).
Since 1987, and until 1992, DOE claimed to be subject to both the administrative and substantive provisions of the environmental laws (in other words, DOE facilities apply for permits and allow inspectors on site), but not subject to the sanctions of fines and penalties. After the Supreme Court agreed with DOE on this point, the Federal Facility Compliance Act was written to require DOE (and other federal agencies) to be subject to fines and penalties as well.
One consequence of this history of seeming immunity from RCRA is that mixed wastes were not completely characterized, facilities for waste treatment were not built and operated, and few mixed wastes were disposed of. The emphasis at DOE during this time period was still one of nuclear materials and weapons production. The legacy for those of us in the waste management arena is upwards of 600,000 cubic meters of mixed waste (HLW, TRU, and LLW) in inventory, with additional quantities of waste expected to be generated by D & D and waste cleanup activities.
Now that DOE has this waste, it needs to plan for its management and eventual disposal. WIPP (the Waste Isolation Pilot Plant) was designed and built for safe disposal of transuranic mixed waste. DOE is investigating whether Yucca Mountain is an appropriate site for a repository for DOE and civilian high level waste and spent fuel. The remainder of DOE's mixed waste inventory, the low level mixed waste, also needs a disposal site or sites.
The 1984 RCRA amendments required waste to be treated to EPA specifications before being disposed of. They also required that storage be only for the purpose of accumulating quantities for proper treatment. No longer could hazardous (including mixed) wastes be lawfully stored indefinitely without proper treatment and disposal. Permits are required before anyone treats, stores, or disposes of RCRA hazardous waste, including mixed waste. Permits contain specific conditions for the operation of RCRA permitted units. Some DOE RCRA permits, for example, contain specific prohibitions against importing wastes from other sites.
RECENT COURT DECISIONS

In 1994, the Tenth U.S. Circuit Court of Appeals decided (U.S. v. New Mexico) that states are pretty much free to set whatever permit conditions they desire, so long as they are related to protection of human health and the environment. States are not required to first establish objective standards (by regulation) and then write detailed permit conditions to achieve the objective standards. Rather, a permit condition will be allowed so long as it appears calculated to achieve the goals of the RCRA statute. In this decision, the court followed the 1994 Supreme Court decision in the PUD No. 1 v. Washington Department of Ecology (114 S. Ct. 1900) case that stated criteria "are often expressed in broad, narrative terms, such as "there shall be no discharge of toxic pollutants in toxic amounts.'"

In this specific instance, the LANL incinerator applied for a hazardous waste permit. As permit conditions, the State required LANL to survey hazardous waste for radioactivity, to monitor radioactive emissions, and to shut down the incinerator if radioactive emissions exceed background by more than 10%. Although no New Mexico regulations require such actions, the permit writer felt that these conditions were necessary to protect human health and the environment.

Since neither DOE, nor NRC, nor any other agency has established a lower limit on what can be properly called "radioactive," state agencies (such as New Mexico's Health and Environment Department) can limit hazardous waste management facilities' receipt of mixed waste containing any detectable radioactivity. Rather than states simply regulating the management of hazardous constituents in hazardous waste, and leaving the management of any radioactivity to management under the provisions of the Atomic Energy Act, New Mexico has successfully asserted control of management of radionuclides under authority of its hazardous waste laws.

This decision injects even more uncertainty into RCRA permitting than existed before 1994. Courts are showing increased deference to EPA and state regulators, and allowing delegation to regulatory officials with lower and lower levels of responsibility. No longer will applicants be able to depend on the presence (or absence) of regulatory provisions that establish numerical (or even narrative) standards. Permit writers will be able to add requirements that are nowhere found in regulations, leaving applicants little written guidance as to what will be expected of them, and even less leverage for negotiation of permit conditions.

EPA and the States refer to the so-called "omnibus" provision of RCRA 3005(c)(3), that requires permit writers to include whatever provisions are protective of health and the environment:

Each permit issued under this section shall contain such terms and conditions as the Administrator (or the State) determines necessary to protect human health and the environment.

EPA has recently advised its permit writers and the states to use this authority to implement proposed regulations that have not yet been promulgated. Clearly, this provision affords permit writers much latitude: they are not constrained by promulgated regulations. Nor, according to the Tenth Circuit, must regulations be proposed. Rather, the permit writer need only find some relationship between the permit conditions and protection of human health and the environment.

FEDERAL FACILITY COMPLIANCE ACT

The Site Treatment Plan (STP) process required for DOE mixed waste facilities under the Federal Facility Compliance Act (FFCAct) provides even more State leverage with respect to permits. DOE submits STPs to the states, which can approve, disapprove, or modify the plans. DOE is required to develop treatment capacity according to binding orders to be issued by the States as part of the STP approval process. Since this treatment is regulated by RCRA, permits will be required before construction of the facility. Thus, while it is incumbent on DOE to construct the facility, it is incumbent on the States to issue the required permits.

Since the FFCAct requires facilities to be built and permits to be obtained, States will have a great latitude in determining which facilities to be built and the conditions of operation of the facilities. Both the recent United States v. New Mexico court decision and the FFCAct give great flexibility to the States to determine how wastes are treated. DOE has correspondingly lesser power to determine whether, where, when, and how its mixed wastes will be treated.

This STP process provides an unprecedented opportunity for DOE and the affected states to discuss the management of mixed waste. DOE has proposed treatment plans to the states; the states' responsibility is to now approve, disapprove, or modify those treatment plans. While it is clear that DOE will have to abide by approved or

modified plans, it is less clear what DOE's responsibility will be if a plan is disapproved.
Compliance orders issued by the states must be obeyed by DOE. Clearly, these orders must be limited to what is feasible. DOE should not be expected to comply with an order that, for example, directs it to do something that is impossible. While the authority of the states is great, it is not absolute. Congress will decide whether to appropriate all funds for compliance with state orders. DOE is subject to the same fiscal constraints that other federal agencies are; DOE managers can't obligate DOE to spend money that Congress has not appropriated. It is up to DOE and the states to ensure that the STP process results in a pragmatic approach to managing mixed waste that will result in Congressional appropriation of the funds necessary to implement the plans.

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AN OVERVIEW OF THE CORRECTIVE ACTION MANAGEMENT UNIT AND TEMPORARY UNIT REGULATIONS
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ABSTRACT
In February 1993, the U.S. Environmental Protection Agency (EPA) promulgated the corrective action management unit (CAMU) and temporary unit (TU) regulations as 40 CFR Part 264, Subpart S. These regulations are intended to foster the selection of protective and cost-effective remedies for the restoration of sites contaminated by hazardous substances, hazardous wastes, and hazardous waste constituents by removing certain regulatory impediments to implementing those remedies, most notably those of the land disposal restrictions (LDRs). This paper provides a brief overview the CAMU and TU regulations.
REGULATORY OVERVIEW
The Resource Conservation and Recovery Act (RCRA), as amended by the Hazardous and Solid Waste Amendments (HSWA), established a broad new mandate for EPA and the States to implement corrective action at hazardous waste treatment, storage, and disposal facilities (TSDFs). RCRA 3004(u) requires that permits issued to such facilities address corrective action for all releases from solid waste management units (SWMUs) at the facility and RCRA 3004(v) established the authority to compel remediation of releases that have migrated beyond a permitted facility's boundary. Under RCRA 3008(h), EPA may issue administrative orders to compel corrective action at interim status facilities. RCRA 7003 provided EPA authority to require corrective action whenever a release of solid or hazardous waste poses an imminent or substantial endangerment to human health or the environment .
On July 27, 1990, EPA published a proposed rule under these authorities to establish 40 CFR Part 264, Subpart S, a comprehensive regulatory framework for implementing RCRA corrective action. The proposal establishes detailed technical requirements and administrative procedures for investigating and responding to releases of hazardous wastes and hazardous waste constituents at RCRA facilities. The proposed Subpart S regulations also contained two provisions, the requirements for establishing CAMUs and TUs, intended to address some of the unusual problems associated with the management of wastes generated during environmental restoration activities. Based on public comments, EPA recognized the need to revise the proposed CAMU and TU regulations and to expedite their promulgation. As a result, EPA revised the regulations and published them as a final rule on February 16, 1993 (see 58 FR 8658).
The proposed Subpart S rule is a significant step toward a comprehensive set of regulations addressing corrective action; however, because EPA has not finalized the bulk of the Subpart S proposed rule (even though it is used as guidance by EPA), 40 CFR 264.100-101 (which essentially codified the statutory provisions) remains the governing regulation. An amendment to 40 CFR 264.101 promulgated in the CAMU and TU rule did, however, create a link between the general requirements of 40 CFR 264.101 and the specific requirements of the CAMU and TU regulations. Thus, it is necessary to make clear that until the balance of the Subpart S proposed rule is promulgated, 40 CFR 264.100-101 and the CAMU and TU regulations together are the regulations

governing corrective action under RCRA 3004(u) and (v), and RCRA 3008(h).
WHAT IS A CORRECTIVE ACTION MANAGEMENT UNIT?
The definition of a CAMU, found at 40 CFR 260.10, is "... an area within a facility that is designated by the Regional Administrator [RA] under Part 264 Subpart S, for the purpose of implementing corrective action requirements under 40 CFR 264.101 and RCRA 3008(h). A CAMU shall only be used for the management of remediation wastes pursuant to implementing such corrective action requirements at the facility." Note that the definition does not explicitly state that a CAMU must involve land-based units (e.g., landfills, land treatment units, surface impoundments, waste piles); this is however implied, in that it is most likely that a CAMU used to manage remediation wastes would be some form of land-based unit.
In some cases, land-based waste management activities within a CAMU that may otherwise be subject to unit-specific standards under 40 CFR Parts 264 or 265 may be incorporated into a CAMU rather than remaining a distinct and separate unit. For example, wastes are often excavated and staged in piles before being transported to a treatment unit. Under the approach outlined by EPA, the area where the wastes are piled would not be considered a separate "waste pile" unit for RCRA purposes. In this case the RA would specify technical standards for that area of the CAMU (e.g., liners, wind dispersion controls, closure requirements) according to the decision criteria in 40 CFR 264.552(c). Similarly, areas of a CAMU could also be used for land-based treatment processes, such as bioremediation systems that involve structures or equipment to maintain optimal treatment conditions.
One significant change between the proposed and final CAMU regulations merits special mention. Under the proposed regulations a non-land-based unit could not be physically located within the boundaries of a CAMU. This requirement was changed in the final regulations. Now a non-land-based unit (e.g., a tank) can be located within the boundaries of a CAMU, but that unit will not actually be a part of the CAMU; it would maintain its separate regulatory identity, and all applicable Subtitle C requirements (e.g., design and operation requirements under 40 CFR 264) continue to apply to that unit and to the wastes managed in that unit. This scenario, where a hazardous waste management unit lies within the boundary of a CAMU but is not a part of the CAMU, poses a significant compliance challenge to facilities operating under such conditions. Clearly, additional waste tracking and segregation practices would be required to ensure compliance with the applicable regulations.
WHAT IS A TEMPORARY UNIT?
Under the final regulations, a TU can only be a tank or container storage unit (so long as the unit does not require permitting under 40 CFR 264 - Subpart X) located at the facility conducting corrective action and which are used solely for the short-term treatment or storage of remediation wastes. A TU has an operational life of up to one year, with a one-year extension available at the RA's discretion. Tus can be located either inside or outside the physical boundaries of a CAMU and such location will not affect the requirements that apply to the TU; however, a TU must be located within the boundary of the facility. This requirement ensures that the regulatory agencies maintain a direct oversight of the unit and that the alternate standards specified for the unit by the RA are appropriate given the context of the site-specific assessment. Because of the narrow scope of the TU definition, Tus are used when there is a need for temporary relief from the compliance requirements of 40 CFR 264 applicable to these types of units. Tus cannot be used for the management of "as-generated wastes," cannot be any other type of unit (e.g., a waste pile, incinerator), are limited to a one-year operational life (with a single one-year extension available at the RA's discretion), and must provide the same degree of protectiveness under the alternative design, operational, or closure requirements as would be achieved by full compliance with all applicable Subtitle C requirements.
WHAT WASTES MAY BE MANAGED IN A CORRECTIVE ACTION MANAGEMENT UNIT OR A TEMPORARY UNIT?
Only remediation wastes can be managed in a CAMU or TU. Remediation wastes are defined at 40 CFR 260.10 as "... all solid and hazardous wastes, and all media (including ground water, surface water, soils and sediments) and debris that contain listed hazardous wastes, or which themselves exhibit a hazardous waste characteristic, that are managed at a facility for the purpose of implementing corrective action requirements under 40 CFR 264.101 and RCRA section 3008(h). For a given facility, remediation wastes may originate only from within the facility

boundary, but may include waste managed in implementing RCRA 3004(v) or RCRA 3008(h) for releases beyond the facility boundary." Wastes generated as part of the site investigations (e.g., drilling muds) are also considered to be remediation wastes. In addition, remediation wastes must have originated from corrective action at that facility, but does include those wastes generated as a result of RCRA 3004(v) or RCRA 3008(h) corrective action activities to address a release that has migrated offsite when the waste is returned directly to the facility for subsequent management. The definition of remediation waste excludes "new" or as-generated wastes (either hazardous or non-hazardous) that are generated from ongoing operations at a facility. In limiting remediation wastes to those that have "originated" from the facility, it should be made clear that this term refers to wastes that originate from remedial activities at the facility, rather than applying to the "as-generated" wastes that created the problem. For example, some facilities may have accepted wastes from offsite, and these wastes have subsequently contributed to contamination problems at the facility. Such waste would be considered remediation wastes for that facility when they are managed in the course of conducting corrective action under 40 CFR 264.101 or RCRA 3008(h).

Notwithstanding the example cited above, if wastes are transported to the facility from an outside source, they would not be considered remediation waste for that facility, regardless of whether those wastes were the result of some type of remedial action conducted at another facility. Similarly, wastes that are excavated, transported to an offsite treatment facility, and returned to the facility are not remediation wastes under these regulations.

The requirement that CAMUs and TUs be used only for the management of remediation wastes is of particular importance. Only wastes that are generated as a result of implementing environmental restoration activities at a facility can be managed within a CAMU or TU. These units cannot be used to manage "as-generated" hazardous wastes; that is, those wastes generated from ongoing production processes or other industrial activities. Further, CAMUs should be limited to the management of wastes that are, or are contaminated by, listed hazardous wastes (i.e., wastes that have "F", "K", "P", or "U" hazardous waste codes). This is because it is these types of wastes that present the greatest challenge when compliance with the full range of the RCRA Subtitle C regulations, most notably the LDR, is required. EPA has suggested that whenever possible, remediation wastes that exhibit only a characteristic of a hazardous waste (which usually do not pose as significant an LDR compliance challenge as do remediation wastes contaminated by listed wastes) and all non-hazardous solid wastes should not be managed in a CAMU, since there are other waste management alternatives that would conserve the capacity in the CAMU.

THE ADVANTAGES OF USING CORRECTIVE ACTION MANAGEMENT UNITS AND TEMPORARY UNITS IN ENVIRONMENTAL RESTORATION ACTIVITIES

There are two advantages to the designation of a CAMU. The first is that remediation wastes generated at a facility and managed in a CAMU (subject to certain restrictions regarding offsite transportation) are not subject to the LDR. This applies to remediation wastes generated during corrective action at any location within the boundary of the facility; that is, remediation wastes may be consolidated from several locations at the facility into a single CAMU. Further, lateral expansion of an existing unit designated as a CAMU, or other activities related to the construction and operation of a new CAMU, are exempted from the minimum technology requirements (MTRs) (e.g., liners, leachate collection systems); however, as a practical matter, these design features may be required by a regulator. The use of TUs during environmental restoration activities allows the RA set less stringent design and operating standards than otherwise required under 40 CFR Part 264 for the temporary operation of tanks or container storage areas used to manage remediation wastes, so long as the alternative standards provide adequate protection of human health and the environment.

In addition, the use of CAMUs may reduce the cost and/or enhance the environmental effectiveness of closure of hazardous waste management units under the provisions of 40 CFR Part 264/265, Subpart G. For example, a unit that would otherwise be capped with the untreated waste left in place could be incorporated into a CAMU, the waste excavated, treated to reduce the toxicity, mobility, or volume of the contaminants, the treatment residues returned to the unit, and the unit capped, all without having to meet the burdensome requirements of the LDR. Another advantage to the use of CAMUs is that remedies selected under the CAMU alternatives would likely be more

acceptable to the communities in the area surrounding the facility relative to those selected that do not include a CAMU, due to reduced reliance on incineration and/or off-site transportation and disposal. This potential for greater acceptance by the surrounding community would be of benefit not only in the context of RCRA corrective action, but extends to the evaluation of remedial actions under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA).

IMPLEMENTATION AUTHORITY

The final CAMU and TU regulations apply to interim status facilities undergoing corrective action under RCRA 3008(h) authority, as well as to permitted facilities conducting corrective action under RCRA 3004(u) and (v). Although the original proposed regulations for CAMUs did not explicitly state that CAMUs could be implemented under RCRA 3008(h) orders, EPA intends that the Subpart S regulations, when promulgated, would be implemented at interim status facilities as well as at permitted facilities. In order to clarify this point the final regulatory definition of a CAMU contains explicit reference to their use under RCRA 3008(h) orders. Further, at the RA's discretion, a CAMU or TU may be used in a response conducted under RCRA 7003 authority (i.e., imminent hazard authority) even if the response is not at a facility subject to 40 CFR Parts 264 or 265. In addition, the CAMU regulations are an applicable or relevant and appropriate requirement (ARAR) at sites being addressed under CERCLA.

The CAMU and TU regulations provide the RA with the authority to designate and approve such units for the purpose of managing remediation waste. While a facility owner or operator may request designating an area as a CAMU, or may request permission to use a TU, the decision rests with the RA but within the context of the RA's authority, there are limits which merit special mention. For example, the RA can, under 40 CFR 264.552(b), designate a "regulated unit" as a CAMU, or include a regulated unit as part of a larger CAMU. This authority does, however, have two important limitations: (1) only closed or closing units could be so designated, and (2) such a designation may only occur if doing so will enhance implementation of an effective, protective, and reliable remedy for the facility. The first limitation excludes operating regulated units, including regulated units continuing to operate under delay of closure provisions (in 40 CFR 264.113 or 40 CFR 265.113), from being eligible for designation as CAMUs because such units will continue to receive and manage "as-generated" wastes.

CONSIDERATIONS FOR DESIGNATING AN AREA AS A CORRECTIVE ACTION MANAGEMENT UNIT

With the exception of the limitations discussed above, 40 CFR 264.552(c) specifies seven decision criteria applicable to CAMUs which form the basis for designating an area as a CAMU. The RA will review the documentation supplied by the owner or operator and consider each of these decision criteria below in designating a CAMU.

Decision Criterion One: Facilitation of Reliable, Effective, Protective, and Cost-Effective Remedies

The first decision criterion [see 40 CFR 264.552(c)(1)] is that the CAMU will facilitate the implementation of a reliable, effective, protective, and cost-effective remedy. Under this criterion, a CAMU is not intended as a mechanism that will undercut the protectiveness of remedies; rather, CAMUs will facilitate the implementation of more reliable, effective, protective, and cost-effective remedies. If an owner/operator cannot provide information to support their claim that designating an area as a CAMU will result in remediation activities with these qualities, then that area will not be designated as a CAMU. In the preamble to the final CAMU regulations, EPA states that evaluation of this CAMU decision criterion will not necessarily require a detailed cost/benefit or other quantitative analyses to support this claim; however, this type of information, as well as other protectiveness, effectiveness, reliability, and cost information, would probably be necessary for the evaluation and should be provided with the submission to the RA.

Decision Criterion Two: Risks During Remediation

The second decision criterion [40 CFR 264.552(c)(2)] specifies that remediation waste management associated with CAMUs cannot create unacceptable risks to human health or the environment from exposure to hazardous wastes or hazardous waste constituents. This provision is intended to ensure that remediation waste management activities are conducted so as to control the short-term risks arising from environmental restoration activities. For example, corrective measures often involve management of large volumes of wastes that could potentially lead to exposure from windblown particulates, air emissions during excavation and transportation, or other

short-term risks. Other considerations include situations where waste characteristics are such that risks to workers are high and special protective measures are required to minimize this risk. Evaluation of this criterion requires that potential short-term risks from remediation activities be carefully examined prior to, and carefully controlled during, implementation of the corrective measure. As with the first criterion, although the regulations do not require a quantitative risk assessment, one probably will be necessary for the evaluation.

Decision Criterion Three: Uncontaminated Areas

The third decision criterion [40 CFR 264.552(c)(3)] requires the RA ensure that uncontaminated land will be included within a CAMU only if remediation waste management at that area will be more protective than at already contaminated areas of the facility. Although it would generally be inadvisable to establish a CAMU in an area that was not contaminated, it will not always be possible to avoid incorporating uncontaminated areas into the CAMU. For example, small uncontaminated areas often exist within an area broadly contaminated by releases of hazardous wastes or hazardous waste constituents, and to simplify the delineation of the areal extent of the CAMU, these uncontaminated areas would be included in the CAMU. Another case is that it may be appropriate to include small portions of uncontaminated land within a CAMU when the remediation activity cannot be conducted on or within the contaminated area itself. For example, it might be necessary to include in a CAMU for a surface impoundment where wastes are being excavated, a small parcel of uncontaminated land adjacent to the impoundment where excavation equipment can be located. In those cases where uncontaminated land is included in a CAMU, the RA will generally include in the permit or order conditions requiring the owner or operator prevent contaminating the uncontaminated parcel during the remediation activities. As with the first two criteria, while the regulations do not require formal risk assessments or other quantitative analyses be performed, such analyses are likely to be necessary in order to demonstrate compliance with this standard.

Decision Criterion Four: Minimizing Future Releases

The fourth decision criterion [40 CFR 264.552(c)(4)] specifies that areas within a CAMU where wastes will remain in place after closure of the CAMU are to be managed and contained so as to minimize future releases, to the extent practicable. This decision criterion closely parallels the closure provision for CAMUs, and is intended to make clear that the RA must consider at the time of CAMU designation whether long-term reliability and effectiveness of the remedy will be ensured through the implementation of a CAMU, particularly when it is necessary to leave wastes in place. Any CAMU recommendation by the owner or operator or designation by the RA must consider, as a primary objective, the long-term (i.e., post-closure) reliability and effectiveness of CAMU-related remediation actions.

Decision Criterion Five: Timing

The fifth decision criterion [40 CFR 264.552(c)(5)] specifies that the CAMU will expedite the timing of remedy implementation, when appropriate and practicable. The use of CAMUs is encouraged when doing so will eliminate unnecessary delays and will encourage more rapid implementation of corrective measures. However, it should be understood that CAMUs may not always result in a more rapid implementation of the corrective measure, or in complete remediation of a unit. By allowing for onsite waste management and encouraging the use of innovative technologies, the resulting corrective measures selected by the regulatory agency may take longer to complete than other options. For example, excavating all wastes and transporting them to an offsite commercial treatment or disposal facility can be accomplished quickly, but optimization of an onsite bioremediation system can take considerably longer, particularly if the system has not yet been used for a full-scale operation. Thus, as provided in the final regulations, this decision criterion only requires that a CAMU expedite remediation of the unit only when it is appropriate and practicable, in consideration of the technological limitations of the selected treatment option and other remedial objectives for the facility.

Decision Criterion Six: Enhancing Long-term Effectiveness

The sixth decision criterion [40 CFR 264.552(c)(6)] requires the use of treatment technologies to enhance the long-term effectiveness of the corrective measure by reducing the toxicity, mobility, or volume of wastes that will remain in place after closure of the CAMU. Long-term reliability and protectiveness is directly tied to the effective treatment of wastes that pose a potential for a future release. This

criterion does not preclude consideration of alternatives that do not employ treatment, as long as they are capable of ensuring long term effectiveness. As a general rule, however, treatment provides greater long term effectiveness than containment, but in certain circumstances containment may be sufficiently effective. In making this evaluation, there is no preference between toxicity reduction, mobility reduction, or volume reduction.

Decision Criterion Seven: Minimizing Land Areas Where Wastes Will Remain in Place

The seventh decision criterion [40 CFR 264.552(c)(7)] requires that the designation of a CAMU minimize the land area where wastes will remain in place after closure, to the extent practicable. CAMUs, by their very nature, promote consolidation of remediation wastes into smaller, discrete areas of the facility suitable as long-term repositories for the wastes. Such units can be effectively managed and monitored over the long term. This criterion of minimizing the land area where remediation wastes will remain in place after closure of the CAMU ties in with the overall goal of achieving effective, protective remedies with long-term reliability. In addition, as a practical matter, development of the facility property (for future beneficial uses or by the owner/operator) may be less constrained if a relatively small area of the facility were dedicated to continued long-term containment of remediation wastes than if remediation wastes were managed at or near their point of generation. This is particularly true for Federal facilities which often cover extremely large tracts of land that could be used for other purposes once the operating agency no longer needs use of the land.

DOCUMENTING A CORRECTIVE ACTION MANAGEMENT UNIT DESIGNATION

Documentation of CAMU decisions is analogous to the documentation the EPA must currently make to support the selection of a RCRA corrective measure or a CERCLA remedial action. Therefore, if a CAMU is selected as part of a final remedy under RCRA corrective action authorities, such an explanation will need to be incorporated into the Statement of Basis [a document analogous to a Record of Decision (ROD)] for the RCRA permit, corrective action order, or RCRA permit modification or order to require implementation of the selected alternative. If a CAMU is to be designated as part of a remedial action under CERCLA, an explanation of this designation will need to be documented in the Proposed Plan and included in the ROD.

In designating an area as a CAMU, the RA will review the documentation supplied by the owner/operator and consider each of the seven decision criteria discussed above. Based upon this review, in accordance with 40 CFR 264.552(f), the RA will document the rationale for designating the CAMU and will explain the basis for such designation. Such rationale will be incorporated as part of the RCRA permit or corrective action order, or in the remedy selection documentation (e.g., the ROD) for that facility, and will be made available to the public. The rationale given for a CAMU decision in the supporting documentation will generally address only those criteria that are considered important to a given CAMU designation. For example, when a CAMU includes uncontaminated land on which remediation waste management will occur, the rationale supporting this inclusion will be specified. However, if remediation wastes will only be managed on contaminated land, this criterion need not be specifically addressed.

CONTENTS OF INFORMATION PACKAGES REQUIRED FOR DESIGNATING A CORRECTIVE ACTION MANAGEMENT UNIT

Under 40 CFR 264.552(d), the owner/operator typically will be required to submit all the necessary information and documentation, such as the results of a RCRA Facility Investigation, Corrective Measures Study (CMS), or other site-specific analyses, for the RA to use in assessing the decision criteria discussed above. Therefore, it is in the interest of the owner/operator to ensure that the documents supplied to the RA address each of the specific decision factors in sufficient detail to allow an informed decision. Further, the facility should also consider how best to integrate the analysis of the specific criteria for selecting a corrective measure under RCRA and the criteria for CERCLA remedial actions with the documentation required for a CAMU. These analyses, while not identical, do share common elements, and careful integration of the evaluation processes may provide a significant savings in the level of effort required.

CAMU designations made through the permit process will generally be approved (or denied) according to the EPA-initiated permit modification procedures under 40 CFR 270.41, or the Class III permit modification procedures under 40 CFR 270.42. Class III permit modifications are similar to EPA-initiated modifications in terms of the

amount and type of public review and comment that are provided. Typically, an EPA-initiated permit modification requires compliance with 40 CFR Part 124 provisions for public notices, comment periods, and a public meeting. Class III modifications require similar actions such as publication of a public notice, a comment period, and public meetings, if requested. In the case of a CAMU implemented through the use of a RCRA 3008(h) order, the order would generally require that the same information be provided as for permitted facilities under 40 CFR 264.552(e). Therefore, the need to approve a CAMU early in the process (e.g., to support an interim measure or "stabilization" action) will pertain to facilities subject to RCRA 3008(h) orders, as well as RCRA-permitted facilities. Thus, to implement a CAMU under an existing RCRA 3008(h) order, the order may need to be amended to reflect the addition of the CAMU.

In 40 CFR 264.552(e)(1), EPA states that the RA will specify in the permit or order the actual areal extent or configuration of the CAMU. Because permits and orders will generally identify the physical boundaries of CAMUs on a facility map, together with a specific description of the physical boundaries or dimensions of the CAMU, the owner/operator should supply this information in the documentation supplied to EPA. 40 CFR 264.552(e)(2) states that the permit or order must specify how remediation wastes will actually be managed in, or as part of, a designated CAMU, including specification of design, operating and closure requirements. For example, if wastes were to be excavated and bioremediated in an enclosure located within the CAMU, the permit or order would specify the requirements for the bioremediation technology, the design and operation of any structures used for the bioremediation process, the disposition of the treatment residuals, and other associated requirements for those wastes and the areas of the CAMU to be used in managing them. Again, it is incumbent on the owner/operator to supply this information to the RA. Under 264.552(e)(3), the permit or order must also establish the groundwater monitoring requirements for the CAMU. Because CAMUs will typically be implemented following studies to determine the extent and nature of surface and subsurface contamination, in most cases groundwater monitoring systems will already have been installed to characterize releases to groundwater. 40 CFR 264.552(e)(3) is intended to guarantee that the owner/operators continue to monitor groundwater quality in the vicinity of the CAMU to ensure that any releases of contaminants from the CAMU are detected and addressed. It is important to note that the groundwater monitoring requirements specified in the final regulations are not detailed, specific requirements and in no way addressed the numerous technical elements of installing and operating an effective groundwater monitoring system. Instead this requirement provides only a general performance standard, leaving the detailed specifications and performance standards for a groundwater monitoring program to be developed based on site-specific information and conditions, and then be specified in the permit or order. Also, at 40 CFR 264.552(e)(4), the final regulations for CAMUs promulgate those provisions addressing closure and post-closure requirements for CAMUs that must be incorporated in permits or orders. These requirements address the inclusion of such closure activities as excavation, removal, treatment, capping or containment of wastes, capping of areas where wastes will remain in place, and removal and decontamination of equipment, devices, and structures used for remediation waste management in the permit or order.

DECISION FACTORS FOR DESIGNATING A TEMPORARY UNIT

40 CFR 264.553(c) specifies seven decision criteria that will be used by EPA to evaluate TU designations. The RA will review the documentation supplied by the owner/operator and consider each of these decision criteria in designating a TU. According to 40 CFR 264.553(c), the RA shall consider the following factors in establishing standards to be applied to TUs: 1) length of time the unit will be in operation; 2) type of unit; 3) volumes of waste to be managed; 4) physical and chemical characteristics of the wastes to be managed; 5) potential for releases from the unit; 6) hydrogeological and other relevant environmental conditions at the facility which may influence the migration of any potential releases; and 7) potential for exposure of humans and environmental receptors if releases were to occur from the unit. It is incumbent on the owner or operator to supply all the necessary information to support such a designation, and to recommend the alternative design, operational, and closure standards for the TU.

DOCUMENTING TEMPORARY UNITS DESIGNATIONS

Documentation of TU decisions is analogous to the documentation required to support

the selection of a RCRA corrective measure or the designation of a CAMU. Therefore, if a TU is incorporated as part of a corrective measure, an explanation would be incorporated into the Statement of Basis for a permit modification or a new or revised RCRA 3008(h) order. Under 40 CFR 264.553(g) the RA is required to document the rationale for designating a TU or granting a time extension for a TU, and to explain the basis for such a designation.

As required under 40 CFR 264.553(d), the RA will specify requirements for TUs in the facility's operating permit or in a RCRA 3008(h) order. The requirements specified will include the design, operating, and closure requirements for such units, and will reflect the decision factors described above. This section of the permit or order will also specify a one-year time limit for operation of the TU. At the end of the specified time limit for a TU, or at the end of an extension granted by the RA, the owner/operator will be required to cease management of remediation wastes in that unit and initiate the closure requirements prescribed in the permit or order. Incorporation of a TU designation into an existing permit will be conducted in accordance with the procedures for EPA-initiated permit modifications under 40 CFR 270.41, or the owner/operator of a permitted facility may request approval for a TU through a Class II permit modification. Class II owner/operator initiated permit modifications follow the procedures set forth in 40 CFR 270.42. In both cases, there are requirements for public notices, comment periods, and public meetings, if requested.

EXTENSIONS TO SPECIFIED OPERATING TIME LIMITS FOR TEMPORARY UNITS

In some cases, due to unexpected circumstances, a TU may have to remain in service beyond the one-year time limit. 40 CFR 264.553(e) specifies the criteria the RA will consider prior to approving an extension to the time limit originally specified. If such an extension is requested, the RA will have to determine that continued operation of the unit will not pose a threat to human health and the environment and that continued use of the unit is necessary to ensure the timely and efficient implementation of corrective measures at the facility. Upon approval of an extension, the RA will identify the specific time limit for the extension in a modification to the permit or order. As mentioned above, 40 CFR 264.553(g) requires the RA to document the rationale for granting a time extension for a TU and to explain the basis for such designation. The rationale for such decisions will be incorporated as part of the Statement of Basis in a permit or order modification. Approval for extensions for TUs that are not addressed under a Class III permit modification or that are not part of an EPA-initiated permit modification will be processed as Class II permit modifications, subject to the somewhat less stringent requirements of 40 CFR 270.42. In cases where it is necessary or desirable to continue the waste management activity that was conducted in the TU, the owner/operator will be required to retrofit the unit to meet the applicable standards specified in 40 CFR Part 264 or Part 265 for that type of unit, arrange for an alternative unit in which to continue conducting the activity, or otherwise modify the waste management practices so that the unit is no longer used as a TU. If the owner/operator chooses to retrofit the unit, but such changes to the unit cannot be made before the end of the extension period, the owner/operator will be required to cease management of the waste until the retrofitting has been completed. Changes to TUs (e.g., retrofitting) or to other remediation waste operations at the end of the operating time limit for a TU will be subject to approval through modifications to the permit or order.

INTEGRATION WITH STATE LAWS AND REGULATIONS

Under RCRA 3006, EPA may authorize States to administer and enforce the RCRA program within the State. Following authorization, EPA retains enforcement authority under RCRA 3008, 7003, and 3013, even though authorized States have primary enforcement authority. Under RCRA 3006(g)(1), as amended by HSWA, new requirements and prohibitions imposed under HSWA authority take effect in authorized States at the same time that they take effect in unauthorized States. EPA is directed to carry out these requirements and prohibitions in authorized States, including the issuance of permits, until the State is granted authorization to do so. Since the CAMU and TU rule was promulgated pursuant to RCRA 3004(u) and (v), and RCRA 3005(c) (i.e., interim status), all of which are HSWA provisions, EPA intends to implement the CAMU and TU provisions immediately in all States and territories in which the EPA now administers the RCRA 3004(u) and (v) corrective action authorities. Thus, the rule takes effect immediately in: (1) States that are not authorized for the RCRA base

program; and (2) States that are authorized for the RCRA base program, but are not yet authorized for the HSWA corrective action program. These regulations do not, however, automatically apply in States that are authorized for the HSWA corrective action requirements.
A complicating factor is that under RCRA 3009, States may impose more stringent or broader regulations than are included in the Federal program. Because the CAMU and TU regulations reduce regulatory requirements for certain types of waste management conducted during corrective action, EPA considers them to be less stringent than, and to reduce the scope of, other existing Federal corrective action requirements. Therefore, the CAMU and TU regulations will not apply in those States authorized for corrective action, until those States have adopted comparable provisions under their own State law. Furthermore, because the rule is less stringent than other existing corrective action requirements, authorized States are under no obligation to adopt these regulations, and States not yet authorized for corrective action are not required to include these provisions in their programs when they seek authorization.
Another complicating factor is that many States have laws and programs to address environmental contamination problems that are not addressed under RCRA or CERCLA authorities. As a general rule, since CAMUs are defined as units to be used in connection with 40 CFR 264.101 or RCRA 3008(h) actions, they can be employed only at a facility regulated under Subtitle C of RCRA, or at CERCLA sites where the CAMU or TU provisions are determined to be ARARs. Therefore, sites being addressed under State laws may not be able to have an area designated as a CAMU or TU.
In sum, although the CAMU and TU provisions have been adopted by EPA, these provisions may not necessarily take effect due to the specific requirements of State implementation of the Federal RCRA program, and of other State environmental laws. Thus, a facility may wish to designate an area as a CAMU under the Federal program in order to gain relief from some regulatory requirement (e.g., the RCRA LDR), but be prohibited from doing so under State law. It will be incumbent upon the owner/operator to determine the applicable requirements under both the Federal and State programs, assess how the use of CAMUs or TUs integrates with those requirements, and successfully demonstrate to the agencies concerned that their proposed CAMU or TU will meet those requirements. It may be possible that the use of a CAMU or TU to gain relief from a more stringent regulatory requirement is not possible due to a conflict between Federal and State regulations.
SUMMARY
The CAMU and TU rule finalizes provisions of the proposed Subpart S regulations for CAMUs and TUs. Both of these units function solely to manage remediation wastes that are generated at a RCRA facility during corrective action or by a CERCLA response action (if determined to be an ARAR). These units cannot be used to manage "as-generated" hazardous wastes from ongoing production processes or other industrial activities. In adopting these regulations, EPA has provided added regulatory flexibility in order to expedite and improve actions to address releases of hazardous wastes or hazardous waste constituents. For example, the final CAMU and TU regulations provides a means of gaining relief from the LDR and MTR requirements for wastes managed in a CAMU. Although these regulations provide additional flexibility when selecting a corrective measure, it is extremely important to recognize that other requirements, policies, and guidelines for establishing site-specific cleanup goals and for selecting remedies under both the EPA- and State-administered programs remain in effect, and may significantly influence the usefulness of the CAMU and TU provisions.

29-7
DEVELOPMENT OF THE EPA RADIATION SITE CLEANUP REGULATION
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ABSTRACT
The United States Environmental Protection Agency is developing a proposed rule that will establish a uniform standard for cleaning up radioactively contaminated sites. The rule will apply to federal facility sites and to sites licensed by the Nuclear Regulatory Commission or an Agreement State. The cleanup standard must be met if a site is to be released for public use. The regulation will not mandate the cleanup of any sites or provide remedy selection criteria on when cleanup is appropriate.

EPA has coordinated the development of the rule with a wide range of governmental organizations and affected groups. A regulatory impact analysis of the rule's economic effects is underway, and the Agency plans to develop implementation guidance to help agencies responsible for radiation site cleanups meet the cleanup standard.

This paper discusses the latest technical and policy developments for the radiation site cleanup rule. The information presented is preliminary in nature and is subject to change as the regulatory development process continues. It is intended to be used primarily to maximize discussion and comment with interested parties.

INTRODUCTION

The United States Environmental Protection Agency (EPA) estimates that there are approximately 5,000 sites known or presumed to be contaminated with radioactive material in the United States. Currently, radiation site cleanups are regulated or managed by several federal agencies, including EPA, the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), and the Department of Defense (DoD).

The lack of specific cleanup standards for radioactive materials has been a major impediment to progress in cleaning up many contaminated sites. Under current procedures, sites are evaluated on a site by site basis to determine the appropriate cleanup level. This process has caused delays and uncertainty in the cleanup of sites, as well as large costs due to legal disputes over determining an acceptable cleanup level. To address these concerns, the Agency is developing a radiation site cleanup regulation which will establish a health-based cleanup standard that will reduce the costs and delays in determining the level of cleanup to be achieved at contaminated sites. EPA anticipates publishing a proposed radiation site cleanup rule in the Federal Register during the summer of 1995.

Coordination Of The Rulemaking

The radiation site cleanup rule has generated extensive interest among a variety of constituencies, so EPA is coordinating its development with several organizations and groups. These include federal agencies such as DOE, DoD, and NRC; state and local governments; Native American tribes; and organizations representing environmentalists, industries, and other affected groups. The Agency also strongly encourages members of the public to participate throughout the process to ensure that their concerns are understood and addressed.

As part of its coordination effort, EPA has established a radiation site cleanup subcommittee under the National Advisory Council for Environmental Policy and Technology (NACEPT).a The subcommittee provides advice on issues related to the formation of the radiation site cleanup standards and helps foster scientific and technical objectivity and public openness. To ensure balanced representation and a wide range of viewpoints, the NACEPT subcommittee includes representatives of state and local government agencies, industry, and Native American and public interest groups. The meetings of the subcommittee are open to the public and provide opportunity for public comment.

The Agency is coordinating the development of the rule with state officials through the Conference of Radiation Control Program Directors (CRCPD), in particular with the CRCPD Committee on Decontamination and Decommissioning, which is examining similar technical issues. In addition, the Agency is exchanging information with other state organizations such as the Association of State and Territorial Solid Waste Management Officials.

EPA is coordinating with DOE, DoD, NRC, and EPA program offices through an Interagency Steering Committee and interagency workgroups. The Steering Committee was established by EPA to ensure that the radiation site cleanup rulemaking effort receives appropriate resources and priority. The interagency workgroup examines specific technical issues related to developing and implementing the radiation site cleanup standards.

The Agency also is working closely with the NRC on the Commission's development of separate regulations governing the decommissioning of NRC-licensed facilities. EPA and the NRC regularly share information that pertains to their respective rulemaking efforts, and EPA participated in the NRC's Enhanced Participatory Rulemaking Workshops. EPA and NRC believe that this parallel approach to rule development will ensure that EPA's cleanup regulation and NRC's decommissioning standards will be consistent, fully protective of public health, and issued in a timely manner.

Responsibility of EPA

In developing the radiation site cleanup regulation, EPA plans to exercise its authority under the Atomic Energy Act (AEA), which sets forth the Agency's responsibility and authority to promulgate regulations to protect people and the environment from the harmful effects of ionizing radiation. The Agency also has the authority to prepare radiation guidance documents. Under Executive Order 10831, the EPA Administrator is charged to "advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with States."
Reorganization Plan No. 3 of 1970 transferred from the Atomic Energy Commission (AEC) to EPA the authority to establish standards for the protection of people and the environment from the effects of all radioactive material. It should be noted that EPA is not the only federal agency to derive its authority to regulate radioactive material from the AEA; the NRC and DOE do as well. However, these agencies must implement EPA's generally applicable radiation standards.
OVERVIEW OF REGULATION
Purpose of the Regulation
EPA's goal in the development of the Radiation Site Cleanup Regulation is to establish clear, consistent, and protective health-based cleanup standards that are implementable. By doing so, the Agency will facilitate the cleanup of radioactively contaminated sites and promote the beneficial reuse of land. The Radiation Site Cleanup Regulation will not, however, mandate the cleanup of radioactively contaminated sites.
Applicability Of The Regulation
The regulation will apply to sites contaminated with radioactive material that are under the control of a federal agency, such as DOE or DoD, and to sites licensed by the NRC or an "Agreement State."b The regulation may also apply to Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)/Superfund sites as a potential applicable or relevant and appropriate requirement (ARAR). The cleanup regulation will set a standard that must be met if a site is to be released for public use. The standard will limit radiation exposures from all sources, including soils, ground water, surface water, air, and structures.
The proposed rule will specifically exempt the following facilities/sites from coverage:
Any uranium mill tailings pile after it has been disposed of under 40 CFR Part 192, Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings.
CERCLA sites that have been cleaned up under 40 CFR Part 300, the National Contingency Plan, with signed Records of Decision as of the effective date of the rule.
EPA may exempt NRC and Agreement State licensees from the cleanup regulation if the Agency finds that the decommissioning standards being developed by NRC are sufficiently protective of human health and the environment. This is in accord with a Memorandum of Understanding (MOU) between EPA and NRC (57 FR 54127, November 16, 1992), which discusses how the agencies will avoid overlapping regulations.
Health-Based Dose Limit
The current staff working draft of the radiation site cleanup rule limits total exposures from residual radiation at a site to 0.15 mSv/yr (15 mrem/yr) above background radiation levels if the site is to be released for public use. To ensure protection of underground sources of drinking water, EPA also plans to set a separate standard for ground water based on the Safe Drinking Water Act (SDWA) standards. In order for a site to be released, compliance with these standards would have to be demonstrated for a period of 1,000 years.
The annual committed effective dose limit of 0.15 mSv/yr (15 mrem/yr) above background corresponds to a lifetime excess cancer risk level of approximately 3 x 10-4 over 30 years of exposure. A 3 x 10-4 risk is generally consistent with other environmental cleanup programs, as well as other radiation protection standards. Because many radionuclides are long-lived, EPA believes it is important that compliance with the standard be demonstrated for a 1,000 year period.
Use of Active Control Measures
Achieving the health-based cleanup level established by the proposed rule would permit the release of a site without the use active control measures and would ensure that individuals located at the site would not be exposed to radioactive

material at levels above 0.15 mSv/yr (15 mrem/yr) in excess of background radiation. The term "active control measures" means measures that control radiation dose or prevent exposure to members of the public by methods other than physical removal of radioactive material. This definition includes, but is not limited to, deed restrictions, deed notices, and the plugging of existing ground water wells.
EPA recognizes that some sites cannot be cleaned up sufficiently to allow such release due to technological limitations, worker safety, ecological concerns, or other factors. In these cases, active control measures can be implemented to ensure that members of the public do not receive doses over 0.15 mSv/yr (15 mrem/yr) in excess of background radiation levels. The active control measures would also have to ensure that exposures to the public from contaminated ground water do not exceed the ground water standard. The active control measures must be selected through a public notice and comment process which involves individuals potentially affected by site cleanup activities. In addition, sites that employ active control measures will be re-evaluated every five years to ensure that the active control measures are still working properly.
EPA also is proposing an annual committed effective dose limit of 0.75 mSv/yr (75 mrem/yr), in excess of background radiation levels, in the event that all active control measures fail at a site. While EPA fully expects implemented controls to be effective, this additional requirement provides assurance for cases where the effectiveness of controls must be projected well into the future.
Public Participation
The draft radiation site cleanup rule enumerates five situations in which implementing agencies will be required to provide public notice and opportunities for public comment. The public notice and comment requirements would apply whenever any of the following occurs:
The implementing agency determines that a site, or any portion thereof, will be evaluated for remediation for release from under the control of the implementing agency or its licensee.
The implementing agency proposes its plan for achieving compliance with the radiation site cleanup regulation, including:
-Whether remediation will occur, the nature of the proposed remediation including the selection of any active control measures, and the anticipated future land use for the site, or
-That the land will not be released for public use.
The implementing agency remediates a site and intends to release it within 60 days.
For sites that use active control measures, the implementing agency verifies whether the controls are continuing to function.
The public notice and comment procedures are similar to those required by Superfund when proposing to list a site on the National Priorities List, proposing a remedial action prior to the signing of a record of decision (ROD), and when a fundamental change occurs to the remedy selected in a ROD. These procedures include notification to EPA; State, local and Tribal governments; and other communities that may be affected by site remediation (such as communities along a corridor for waste transport), as well as publication in the Federal Register of a notice with request for comment from affected parties. The proposed rule also requires the implementing agency to establish and maintain an information repository containing relevant site remediation documents for accessibility to the interested public.
The draft rule's public participation requirements are intended to be flexible. In all cases, there is a requirement that the public be informed of the relevant action, decision, or proposal, and have an opportunity to provide a response. Agencies may provide additional opportunities for public information and participation, such as public meetings, announcements, fact sheets, and other measures as appropriate under the circumstances. The rule does not require a response by the agency, but EPA anticipates that agencies will choose to provide responses in many cases, either because they are required to do so by other laws, or because community support is a practical necessity before property can be released. The purpose of the public participation requirements is to make the public aware of cleanup decisions, and give the public an opportunity to provide information that may assist the agency in conducting an evaluation or in carrying out the release of the property.
EPA has been working with other federal agencies and the public to develop

innovative methods for involving communities in the cleanup process. To this end, EPA established the Federal Facilities Environmental Restoration Dialogue Committee (FFERDC), whose members represent 40 federal agencies, tribal and state governments and associations, and local, national, environmental, community, and labor organizations. The FFERDC's mission is to develop consensus policy recommendations aimed at improving the FFER decision-making process to ensure that cleanup decisions reflect the priorities and concerns of all stakeholders.

One recommendation of the FFERDC is the establishment of site-specific advisory boards (SSABs) at federal facilities. SSABs would assist in providing:

Consistent opportunities for affected stakeholder involvement in federal facility cleanups,

Regular, early, and effective public participation in federal cleanup programs, and

Consolidation of the many public involvement initiatives addressing cleanup.

Although EPA is not requiring the establishment of SSABs, the Agency strongly encourages the use of SSABs (or similar mechanisms) and believes that they would promote early, direct, and meaningful participation through the cleanup process. The Agency believes that such participation is consistent with the goals of the AEA and other environmental statutes.

OVERVIEW OF ANALYSES PERFORMED IN SUPPORT OF THE RULE

In order to evaluate options for the cleanup standard, EPA has been conducting technical and economic analyses in support of this rulemaking.

Risk Analyses

Like metals, radioactive wastes cannot be destroyed by treatment techniques such as incineration or chemical reaction. Many radioactive wastes remain hazardous for long periods of time. Consequent ly, radioactive site cleanup means reducing the long-term risk to people who live on or near a contaminated site, often by excavating the waste and moving contaminated material to a disposal facility. Cleaning up and transporting radioactive waste, however, may lead to short-term risks, including increased traffic fatalities, that are nonradiological in nature. Therefore, EPA is assessing the net health impacts resulting from remedial activity by subtracting the short-term risks due to remedial activity from the fatalities averted due to the reduction in long-term risks as a result of remedial activity. The risk analyses evaluate the net health impacts at various cleanup levels and under various cleanup scenarios.

EPA used computerized models to assess the potential radiation doses and associated risks to the public from all significant exposure scenarios, media, and exposure pathways. The risk analysis focused on estimating: (1) the change in risk to an individual under reasonable maximum exposure (RME) conditions, and (2) the number of cancers per year in the exposed population. The methodology is consistent with that described in the EPA Risk Assessment Guidance for Superfund (RAGS) for baseline risk assessments.c

Mathematical models also have been used to estimate, based on RME assumptions, the cleanup levels of radionuclides in soil to be achieved under rural residential and commercial/industrial exposure scenarios. The scenarios address land-use and exposure situations expected after the cleanup of sites covered by the proposed rule.

The rural residential exposure scenario addresses long-term risks to individuals expected to have unrestricted use of a site after cleanup. It assumes that individuals live on the site and are constantly exposed, both indoors and outdoors, to residual concentrations of radionuclides in soil through the maximum number of exposure pathways, including:

External radiation exposure from photon-emitting radionuclides in the soil,

Inhalation of resuspended soil and dust,

Inhalation of radon and radon decay products from soil containing radium,

Incidental ingestion of soil containing radionuclides,

Ingestion of drinking water containing radionuclides transported from soil to potable ground water sources,

Ingestion of home-grown fruits and vegetables contaminated with radionuclides taken up from the soil,

Ingestion of meat or milk containing radionuclides taken up by cows grazing on contaminated plants and fodder, and

Ingestion of locally caught fish containing radionuclides.

The commercial/industrial exposure scenario addresses long-term exposures and risks

to commercial or industrial workers assuming a site was released with a restriction allowing only commercial or industrial development. Under this scenario, the model assumes workers would be exposed to residual levels of radionuclides in soil during an average eight-hour workday, both indoors and outdoors, at a site. This scenario does not consider exposures to site cleanup workers or to construction workers, nor does it address risks to workers from contaminated structures or building materials.
Five pathways were evaluated under the commercial/industrial exposure scenario:
External radiation exposure from photon-emitting radionuclides in soil,
Inhalation of resuspended soil and dust containing radionuclides,
Inhalation of radon and radon decay products from soil containing radium,
Incidental ingestion of soil containing radionuclides, and
Ingestion of drinking water containing radionuclides transported from soil to potable ground water sources.

Regulatory Impact Analysis

As required under Executive Order 12866, the Agency is developing a Regulatory Impact Analysis (RIA) to estimate the anticipated economic effects of the proposed radiation site cleanup rule. The RIA estimates the incremental cost of the proposed regulation by assessing two of the regulation's cost impacts. First, the establishment of a uniform cleanup standard is expected to provide cost savings through:
Streamlined site assessments,
Reduced legal costs associated with selecting a cleanup risk level, and
Reduction of market barriers that inhibit the use of existing waste minimization technologies.

Second, to the extent that the rule establishes a level of cleanup that is more stringent than what would be achieved without the regulation, the proposed regulation will cause an increase in costs.

To assess these cost impacts, EPA has developed a computer-based cost model that estimates the change in costs arising from the new regulation compared to a baseline case, which assumes no new rule for radiation site cleanups. To do this, the model computes the cost of cleaning up each of 16 reference site categories that are representative of the universe of federal facility sites to be remediated. EPA's computer model calculates cleanup costs under baseline conditions (i.e., no new EPA cleanup rule) and under alternative dose limits considered for the proposed rule. Summing the cleanup cost estimates for all 16 site categories provides estimates of national-scale impacts.

There are five major components of total cost associated with the cleanup of radioactively contaminated sites that are estimated in the RIA:
The cost of precleanup activities, which includes initial site characterization and legal costs,
The cost of cleanup,
The cost of transporting, by rail or truck, the contaminated soil excavated from a site to a disposal facility,
The cost of disposing of contaminated materials, and
The cost of verifying that a site cleanup has been successful and the target risk level has been achieved.

CONCLUSION

The Agency plans to publish a proposed rule in the summer of 1995, and a final rule is expected a year later. The publication of the proposed radiation site cleanup rule is expected to generate a great deal of public interest, and EPA looks forward to receiving additional comments at that time. The Agency believes that by involving all interested parties and by examining comments and information from a wide range of viewpoints, the rule will be consistent, implementable, and protective of human health and the environment.

29-8

ACHIEVING INDEPENDENT EXTERNAL REGULATION OF DEPARTMENT OF ENERGY ENVIRONMENTAL MANAGEMENT: TWO ALTERNATIVE MODELS

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ABSTRACT
In its strategic plan, Fueling a Competitive Economy,(1) the Department of Energy commits to developing external regulation of its environmental management activities. The Department, in fulfilling this commitment, developed two models that reveal the merits of alternative regulatory structures. This paper describes the results of applying the models to develop external regulation at the Rocky Flats Environmental Technology Site and other environmental management sites of the Department. The linear model relates each activity at a site to the multiple regulators prescribed by law. Results show that the Department can work most efficiently with regulators to eliminate overlapping requirements with the regulatory structure of the linear model. Also, the Department can enhance efficiency by customizing communications for efficient information transfer to the regulators.
The integration model, the second model, groups responsibilities under one regulator. To implement the structure of the integration model for external regulation, the Department must work with the regulator that monitors compliance prescribed by law. The Department can eliminate conflicting requirements to gain credible regulation and uses a dispute resolution process to resolve jurisdictional issues efficiently. With attention to the structure developed for regulation, the Department can operate effectively in the regulated environment and achieve its strategic goal of credible, external regulation.

Background

In the strategic plan the Department of Energy recognized the importance of increasing external regulation of its environmental management operations. The fruition of this process will be increased stakeholder knowledge of the ensured safety and health of workers and the public, and restoration of the environment as the Department transfers facilities from production to reuse or ultimate disposal. The Department has incorporated this goal throughout its operations and has identified critical success factors to measure the process' rate of achievement. The Strategic Plan commits the Environmental Quality business area to the following:

> Achieve independent and credible regulation of activities and facilities, and eliminate conflicting requirements.

In meeting the goal of the Strategic Plan, the Department must operate within several parameters such as federal laws, regulations, and court orders to attain external regulation. However, the Department maintains flexibility in establishing the regulatory structure in several ways. For example: the Department negotiates authority and scope of activities with various regulatory agencies; the Department regularly proposes and agrees to specific arrangements for regulation; the Department establishes stakeholder groups, initiates public hearings, and conducts the National Environmental Policy Act (2) process for public reviews. Finally, the Department proposes to the Administration legislation for efficient external regulation. Through these activities, the Department helps form the regulatory structure.

The Department recognizes the merits of a thorough analysis of the shift to external regulation of environmental management activities. The Department has a strong interest in external regulation that ensures responsible management and operation, and instills public confidence. In an evaluation of regulatory responsibility, the Department works to coordinate regulations to improve its cleanup activities.(3) Generally, the Department has retained internal orders and procedures. The Department has developed a basis for external regulation in the Standards it maintains for its self regulation function.(4) However, the Department is evolving from a position of self regulation to external regulation through interaction with regulators, such as state and federal agencies, stakeholders, judicial orders and decrees, and through Congressional mandates. Unfortunately, this process has produced procedures for environmental management overlap and often conflict. This paper examines two models which have been used to help implement the goal of external regulation. The models are presented from the perspective of a user that can be the regulator, the regulated entity, or other stakeholders. The models are tools for the user to conceptualize a particular regulatory environment, for example; once the goals of the regulator, regulated, and stakeholders are identified. Alternatively, the models can be tools to examine a regulatory process and to understand the consequence of establishing specific regulatory relationships. The application of these models is particularly useful during negotiations, in

developing regulatory relationships, and in fostering public understanding of the regulatory structure.
The first model, called the Linear Model, seeks to identify a diverse set of regulators for all environmental management activities. The user of the model groups expertise and activities to minimize the overlap among regulators. The user concludes with details that identify the existing and required external regulators.
The second model, called the Integrated Model, seeks to direct responsibility to principal regulators. The user of the model attempts to gain efficiency by focusing all regulation to a few parties. The user concludes with a structure that minimizes the number of regulators.
The Department has applied the mechanisms of both models implicitly and to expand the roles of regulators in such circumstances as interagency agreements and plans under the Federal Facility Compliance Act.(e) The models can be used to identify beneficial approaches in the negotiation of these documents. This paper will demonstrate the beneficial application of these models to the development of external regulation at the Rocky Flats Environmental Technology Site (RFETS).
Descriptions of the Models
Linear Model: Activities Correlated to Individual Regulators
The Linear Model links each activity at a site with specific regulations. The first model can be implemented as follows:
To achieve independent and credible regulation, the user identifies the relationships of activities to current regulations, and proposes additional relationships. The user then examines the current and proposed regulation in terms of comprehensiveness and reduced overlap.
1. Assessment of the status of the existing regulatory environment
The first step in application of the Linear Model is to assess the status of existing regulations. This assessment requires identification of existing external regulators for activities at the site, identification of appropriate regulations, assessment of the credibility of existing regulations, and identification of overlapping and conflicting requirements.
2. Development of a systematic process and development of initiatives
The second step in this model is to associate the appropriate regulatory agencies with activities. Individual regulatory responsibilities for specific activities must be identified to assure that each activity has a recognized regulator. Both employees and the public must recognize the independent regulator for specific actions.
Once the relationship between activity and the appropriate external regulator is identified, then an analysis of appropriate initiatives can be started. This will require identification of the existing parameters within which the model must operate such as legislation, court orders, and existing agreements with regulators and stakeholders. Initiatives can then be proposed to modify the regulatory environment towards that identified in the model.
Integration Model: Activities Integrated Under One Outside Regulator
The second model integrates regulatory responsibilities under one regulation and a single external regulator.
The mechanics of this approach can be illustrated by an example. The user of the model identifies one statute as the integrating vehicle for the site. The agency authorized to enforce the statute acts as a gatekeeper regulator and, through negotiations, determines the appropriate level of external regulation. If the gatekeeper regulator determines the regulated entities existing regulations and procedures are sufficient, then the regulated entity continues its management of activities. If, however, the gatekeeper determines that the regulated entities management is insufficient then the gatekeeper retains jurisdiction or, if other regulations are appropriate, the gatekeeper assigns the activity to another regulatory entity.
1. Identification of the Appropriate Vehicle and Regulator
The first step requires the user to identify the appropriate statute to act as the vehicle for regulation of the activities at a site. This selection is weighted toward regulations that are established within a regulatory agency with the resources and a technically-experienced staff. Of course, the regulatory agency must have appropriate jurisdiction. Therefore, the selection favors an external regulatory agency with access to a vehicle which gives it jurisdiction over the site or a group of activities at the site. The gatekeeper agency may also be limited to

certain activities. Therefore, this step of the model requires recognition of existing parameters within which the regulated agency must conduct its activities as in the previous model.

2. Determination of Appropriate Level for Operation

In this step, the user establishes a level of regulation that will meet the regulator and stakeholder expectations. The user can conduct the minimum level of regulation from existing controls such as Standard Operating Procedures (SOPs) for the site, standards and safety rules developed by the regulated agency. Often these standards and rules are published in the Code of Federal Regulations published in the Federal Register. Existing controls which have had the most extensive public review and have been implemented successfully should be given the most deference.

3. Assignment of Activities to Appropriate Regulatory Agencies

In the third step of the model, the user assigns activities to the gatekeeper regulator. With this assignment, the gatekeeper would review the activity to determine if they are being implemented by the regulated entity.

If the gatekeeper's review determines that the activities are being conducted within the acceptable regulatory limits, then the activity continues under the regulated agency. If, however, the gatekeeper determines that there is a unacceptable level of risk involved, then it is incumbent on the gatekeeper to review other potential regulatory vehicles to determine whether the activity could be regulated by another agency. In a situation where the gatekeeper agency determines that another agency exists and has jurisdiction, the gatekeeper agency would notify the other agency of the need to regulate the activity.

If the gatekeeper agency determines the site is not operating within acceptable regulatory levels and is not within the jurisdiction of another regulating agency, the gatekeeper regulator imposes requirements to address the deficiency. As part of this step, the gatekeeper agency must review the rules and regulations of other agencies to determine if jurisdiction exists. However, the point of demarkation for a gatekeeper's jurisdiction and another agency's jurisdiction may be uncertain and the gatekeeper's determination of an inadequate regulatory posture by the site should be resolved through an established dispute resolution process. Once the gatekeeper establishes the regulatory relationship, he must establish mechanisms that will prevent conflicts with each regulator's jurisdiction. Therefore, critical to the success of regulation, success of regulation structured by the model is a dispute resolution process that can efficiently resolve jurisdictional issues.

Application of the Models

The two models described above have been applied to environmental management activities of the Department of Energy. Applications described in this section are analyses of possible regulatory structures at the RFETS. The work was conducted principally to develop a headquarters perspective in support of the negotiations of the Rocky Flats Cleanup Agreement. The applications were developed for a specific site but are appropriate to most Department sites with environmental management activities.

Application of the Linear Model

Step 1. Assessment of the Status of the Existing Regulatory Environment

In most cases, the existing regulations of environmental activities are comprehensive and potential regulators are independent of the managing office. For example, at a site with environmental management activities, projects for downsizing, dismantlement, deactivation, restoration, managed for environmental remediation construction, operations, maintenance, production, maintenance, operation and construction projects proceed simultaneously.(6) However, there are several areas where conflicting requirements may exist. Conflicting requirements originate from requirements and responsibilities as defined in DOE Orders and Procedures, multiple external regulators and stakeholders, judicial orders and decrees, and specifically legislated Congressional mandates.(7)

Step 2. Development of a Systematic Process and Development of Initiatives

Figure 1 suggests appropriate regulators for several activities within the different projects. The Fig. exhibits major regulators within projects to illustrate the one-to-one relationship of regulator to activity. The results exhibit gaps and overlaps in the external regulation of site activities. One systematic process that can be used by the Department to gain comprehensive external regulation is to map every activity in the site budget to an independent regulator.

Fig. 1.

Initiatives can be undertaken to address specific issues identified in the analysis. For example, in the experience of the authors the credibility of regulations is degraded by: regulations and Orders that have evolved over time without attention to redundancies; reduced budgets that preclude compliance; uncertain regulators for specific activities; and poorly defined responsibilities. For example, cleanup activities and decontamination, decommissioning, deactivation, and dismantlement activities at Department facilities are covered under overlapping requirements of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA),(8) the Atomic Energy Act of 1954 (AEA),(9) and the Price-Anderson Act Amendments (PAAA).(10) Analyses using the models can be used to demonstrate to the public or regulators that existing or even reduced regulations can lead to desired remedied conditions.

Application of the linear model requires a review of laws from which the Department derives its authority. The analysis identifies some of the limitations on external regulation that must be addressed through legislation. For example, under the current legislation, certain regulations can only be developed by the Department. This is the case for activities prescribed by the AEA or the PAAA. The AEA defines the responsibilities of the Department in managing certain types of nuclear materials. The AEA, therefore, may limit the ability of the Department to achieve external regulation without legislative action. In the law, there is evidence that Congress intended to make the Department's authority exclusive in activities dealing with Special Nuclear Material (SNM). Congress limited the Department's ability to enter into agreements with a state that would discontinue the Department's regulatory authority over SNM materials in at least two instances. The first instance is section 274 (b) of the Act which explicitly prohibits the Department from entering into agreements with states to discontinue its regulatory role in cases where quantities of SNM are sufficient to form a critical mass. Another instance is found in Section 274 (c) which requires the Department to retain authority and responsibility with respect to regulation of operation of any production facility.

Another limitation on external regulation of the Department can be found in the Price-Anderson Act Amendments (PAAA), passed in 1988, which requires that the Department enter into agreements for indemnification with those conducting nuclear activities that present a liability to the public. The mandate from Congress required the Department to develop an enforcement program. Subsequently, the Department will promulgate many Department Nuclear Safety Requirements in the Code of Federal Regulations (CFR). The (proposed) Department Final Rule makes it clear that Department will assess civil penalties for any violation of: 1) any Department Nuclear Safety Requirement set forth in the CFR, 2) a violation of a Compliance Order, or 3) any program or plan explicitly required by a provision of the CFR or by a Compliance Order.(11) The PAAA also gives the Department the authority to issue fines and penalties for failure to operate facilities safely as well as to forward evidence of potential criminal violations to the Attorney General of the United States.

Initiatives can be identified that resolve identified issues. In this regard, conflicting regulations can be eliminated by encouraging regulators to focus on specialized areas consistent with their resources. Tools to accomplish this step are risk measures to identify priority actions, models comparing alternatives to reveal equivalencies, and other analyses to demonstrate resolution of conflicting requirements. For example, in order to expedite cleanup, the Department should review the benefits of legislation modifying the restrictions on SNM so that this material can be regulated by another federal agency, a federal agency in coordination with state agencies, or a state.

Another initiative that the Department could undertake to increase credibility of regulation is to assure that information is transferred consistent with the resources of the regulators and stakeholders. This is especially true in instances where the Department is mandated by Congress as the regulator. For example, the Department can work with regulators such as state agencies to develop resources and expertise in the nuclear field. Another example is the Department lessening the burden on citizen groups in the regulatory process. As with all entities, the public has limited time and resources to support regulatory activities. Upon implementing the linear model of regulation, the Department must use efficient mechanisms to transfer information to the public and the regulators. In addition, a systematic

periodic review of the external regulation should be conducted by the Department to meet the regulated goal of the Strategic Plan. This will serve as an important measure of the regulatory structure for both the public and regulators as well as the Department.

Application of the Integration Model

1. Identification of Appropriate Vehicle and Regulator

Many sites of the Department of Energy with environmental management activities are in the Environmental Protection Agency (EPA) National Priorities List (NPL). One consideration is that CERCLA is an appropriate vehicle to manage the Department's environmental management activities. For this consideration CERCLA would give the EPA overall jurisdiction at the site and, therefore, CERCLA is the appropriate integrating vehicle at these sites. This argument has validity in the sense that under CERCLA Section 104, the Administrator has very broad reaching jurisdictional authority and is directed to promulgate regulations establishing reportable quantities:

> Whenever (A) any hazardous substance is released or there is a substantial threat of such a release into the environment, or (B) there is a release or substantial release into the environment of any pollutant or contaminant which may present an imminent and substantial danger to the public health or welfare.

In conflict with this consideration are the restrictions found in parameters mentioned in the linear model. Any scheme established under an integrating gatekeeper agency must accommodate the Congressional mandate of the AEA and Department regulation through the authority of the PAAA.

2. Determination of Appropriate Level of Operation

During the negotiations of the Cleanup Agreement, the parties attempted to establish appropriate standards for operation for meeting state and Federal requirements. This was to be based on a compromise between the state's authority to enforce the Resource Conservation and Recovery Act (RCRA)(9) and the EPA authority under CERCLA. The state's standard was to be based on the RCRA operating permit for the site while an alternative standard for other activities was the the EPA determination of a substantial threat of release to the environment. Federally permitted releases which under the integrating vehicle of CERCLA provide exclusions to CERCLA requirements would be recognized as operating within adequate regulatory levels. For example, emissions that are within the permitted limits of a National Emissions Standards for Hazardous Air Pollutants(m) (NESHAPS) would not be categorized as a substantial threat of release to the environment.

3. Assignment of Activities to Appropriate Regulatory Agencies

In specific instances at sites with environmental management activities such as RFETS, limitations on the jurisdiction of the regulator depend on whether the facility is surplus or operational. Two types of facilities must be considered at environmental management sites: operating facilities which consist of facilities involved in maintenance of special nuclear material, waste storage, waste treatment, infrastructure support, and maintenance of resources; and surplus facilities which have been designate as surplus and are ready for decommissioning, deactivation, and dismantlement. Regulatory authority under CERCLA for environmental restoration may not apply to a facility that is operational. In such a case, regulatory jurisdiction may exist in a state RCRA operating permit, DOE Orders, or other outside regulations. Figure 2 illustrates the application of the integration model.

Fig. 2.

RESULTS

Use of the models reveals relationships of activities and external regulation that can be achieved at an operating site. The models highlight the impediments and changes required to achieve external regulation. The application of these models to regulation at sites with environmental management activities revealed three topics of major concern. First, current regulatory constraints limit external regulation and challenge the full implementation of external regulation. The review of the current regulations as required by both models reveals constraints to external regulation. Consequently, many regulations can only be developed by the Department. This is the case for activities prescribed by the AEA or PAAA. Several changes were recommended in the previous section which could reduce these constraints.

To achieve its strategic goal, the Department can either support legislation to transfer this regulatory authority to other agencies or oversee these activities in a manner credible to stakeholders. The Department should participate with other

agencies to identify conflicting requirements and to develop the specific delegation of responsibility. Where appropriate, recommendations and draft language should be submitted jointly to Congress so that legislative actions can be initiated. The legislative action must clearly define participation of the external regulator.
The second topic of concern revealed by the model is conflicting regulations. Conflicting regulations are easily recognized by application of the models and can be eliminated. Left unaddressed, conflicting regulations reduce the efficiency of operations and add to the cost of external regulation. The models recognize conflicting or overlapping regulations by the relationships of each activity to the external regulators. Conflicting regulations can be eliminated by encouraging regulators to focus on specialized areas consistent with their resources. Tools to accomplish this step are risk measures to identify priority actions and equivalencies as revealed by the models. In this regard, the Department can review the benefits of modifying the restrictions on SNM so that this material can be regulated by another federal agency, a federal agency in coordination with state agencies, or a state agency.
A final concern the models address is the development of functional relationships between the regulators and the regulated entity. An initiative identified by the models is transfer of information consistent with the resources of the regulators and stakeholders. A regulatory structure based on the linear model requires attention to information transfer to serve the many regulators of the Department's diverse activities. The efficient transfer of information is one key to successful working relationships in regulation. Broadcasting information is inefficient since each regulator must discern the relevance of all information to its responsibilities. The Department must be prepared at all times to respond to information requests. The regulators can participate effectively using published notices, draft findings, hearings, formal question response procedures, and other planned media to communicate.
The DOE has success with efficient and focused information transfer of benefit to the regulators. For example, in administering the National Environmental Policy Act, the Department lessens the burden on citizen groups by communicating information at planned intervals and concluding public interactions with summaries of concerns and responses. The summaries form the basis for advancing in the regulatory process. To implement external regulation, the Department must ensure that the regulators understand the methods of communication.

SUMMARY

The models presented in this paper represent a conceptual archetype. In actuality, the relationship between the regulator and the regulated identified by the user may be a hybrid of both models.
After the models have been applied, the user can evaluate the success of proposed initiatives by examining specific figures of merit. Suggested figures of merit include the independence of regulation, credibility of the regulation experienced by the public, the reduced conflict and overlap of the regulation, and the cost to implement the regulation. The evaluation of the models should also consider a cost benefit analysis to evaluate options. Effective parameters for a cost benefit analysis include the marginal benefit obtained in terms of additional safety to the public and workers compared to the marginal cost of implementation of the new regulatory structure. Figure 3 assesses the value of both models.
Fig. 3.

CONCLUSION

By developing external regulation that is credible and efficient, the Department can proceed in a coherent fashion to achieve the regulatory goal of the Strategic Plan. The Department can judge the potential of actions to support the goal of external regulation using two models. The linear model uses the complexity of existing regulatory structure to develop credible external regulation. Using the linear model, the Department must achieve efficiency through tailored communication to each regulator. The integration model minimizes the complexity of regulatory structure, but requires that the Department nurture an effective dispute resolution process to treat jurisdictional issues efficiently. Certain actions support the goal as revealed through application of the models and can be implemented. Other actions could detract from the goal and can be avoided. With attention to the structure developed for external regulation, the Department can operate effectively in the regulated environment and achieve its objectives within environmental remediation.

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THE FAST FLUX TEST FACILITY SHUTDOWN PROGRAM PLAN

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ABSTRACT
The Fast Flux Test Facility (FFTF) is a 400 MWt sodium-cooled research reactor owned by the U.S. Department of Energy (DOE) and operated by the Westinghouse Hanford Company (WHC) on the Hanford Site in southeastern Washington State. The decision was made by the DOE in December, 1993, to initiate shutdown of the FFTF. This paper describes the FFTF Transition Project Plan (1) (formerly the FFTF Shutdown Program Plan) which provides the strategy, major elements, and project baseline for transitioning the FFTF to an industrially and radiologically safe shutdown condition. The Plan, and its resource loaded schedule, indicate this transition can be achieved in a period of six to seven years at a cost of approximately $359 million. The transition activities include reactor defueling, fuel offload to dry cask storage, sodium drain and reaction, management of sodium residuals, shutdown of auxiliary systems, and preparation of appropriate environmental and regulatory documentation. Completion of these activities will involve resolution of many challenging and unique issues associated with shutdown of a large sodium reactor facility. At the conclusion of these activities, the FFTF will be in a safe condition for turnover to the Hanford Site Environmental Restoration Contractor for a long term surveillance and maintenance phase and decommissioning.

INTRODUCTION

The FFTF, located on the DOE Hanford Site near Richland, Washington, is the largest, most modern sodium-cooled test reactor in the world. Originally constructed to support the U.S. Liquid Metal Fast Breeder Reactor Program, the 400 thermal megawatt reactor began operation in 1980, and was ordered to shut down in December, 1993, when the DOE concluded there was no possibility of financial viability and the reactor was no longer needed to support its missions. The DOE requested WHC to develop a Plan to transition the FFTF to an industrially and radiologically safe

shutdown condition in approximately five years.
The decommissioning process for the FFTF, and other major transition facilities on the Hanford Site, will be accomplished in three phases: Phase I (Facility Transition), Phase II (Surveillance and Maintenance), and Phase III (Disposition). The FFTF transition phase started with termination of operations and includes those activities required to place the plant in a safe, stable and environmentally secure end-point condition with reduced risk to plant workers, the public, and the environment. Present planning is that FFTF will be unoccupied and locked, with the exception of maintaining a minimal amount of lighting, fire protection equipment, inert gas supply to the drained sodium systems, and ventilation required to support routine surveillance. The goal is to achieve this low cost surveillance and maintenance state as quickly as possible so funds can be redirected for higher priority environmental restoration activities at the Hanford Site.
At the completion of the Phase I activities, the FFTF will be turned over to the Hanford Site Environmental Restoration Contractor under the guidance of the DOE Office of Environmental Restoration (EM-40) for an extended surveillance and maintenance phase (Phase II) prior to the disposition phase (Phase III). During the surveillance and maintenance phase, the plant will be routinely monitored until decommissioning is completed. The DOE is currently developing a long-term facility decommissioning plan covering all key Hanford transition facilities. This decommissioning plan will facilitate integration and prioritization of the decommissioning activities with other Hanford cleanup efforts.
SCOPE OF THE FFTF TRANSITION PROJECT PLAN
The Plan provides the major elements and project baseline for completing Phase I transition activities and was the basis for development of a resource loaded schedule. The major activities include: reactor defueling, fuel offload to dry cask storage, sodium drain and storage, management of sodium residuals, shutdown of the auxiliary systems, and preparation of the appropriate environmental and regulatory documentation. When the transition end-state is achieved, the irradiated fuel will be washed and located in storage casks at the Interim Storage Area in the northeast corner of the FFTF complex, the approximate 260,000 gallons of sodium coolant will be drained and stored in the new Sodium Storage Facility adjacent to the reactor complex, the frozen sodium residuals remaining in the plant will be maintained under an inert gas blanket, and the auxiliary systems will be drained and deactivated.♀
Preliminary transition phase end-point criteria was developed for the FFTF early in the transition process. A detailed end-point criteria document is being developed to define the physical state of each plant system at the conclusion of the transition phase. This document will require extensive interaction between the FFTF staff, DOE, Environmental Restoration Contractor, and the regulators (i.e., State of Washington Department of Ecology and the U.S. Environmental Protection Agency). Once the transition phase is complete, the plant will remain in the low cost surveillance and maintenance phase for possibly decades before the start of disposition activities. Therefore, the plant will be configured with only a minimal number of operational systems.
The transition activities will be managed and directed utilizing a projectized approach for control and reporting to achieve the best schedule and cost effectiveness. The Resource Loaded Schedule (2) indicates that completion of the transition phase will require approximately six to seven years, at a cost of approximately $359 million. The summary schedule for the FFTF Transition Project is graphically shown in Fig. 1. The DOE goal is to accomplish the transition in approximately five years. The transition schedule extends beyond this goal due to many unique challenges associated with completing the sodium drain, fuel offload, and integrating these activities with shutdown of the plant's auxiliary systems. These challenges will require innovative engineering solutions and new and complex plant evolutions. During the first 4 years of the transition, approximately 90 percent of the plant systems are required to support hot sodium circulation. The minimum level of funding necessary to maintain the FFTF in a safe and stable condition prior to the sodium drain is estimated to be $32 million per year. Clearly, an expeditious completion of the transition will result in significant cost savings. Therefore, efforts are focused on accelerating the sodium drain and other critical path transition activities to the maximum extent possible, within budgetary and regulatory constraints in order to achieve schedule compression and the associated major cost reductions.

The key transition activities and associated issues are discussed in the following sections in the order depicted in the Fig. 1 summary schedule.

Reactor Defueling

The reactor vessel is currently being defueled to Interim Decay Storage and the Fuel Storage Facility using standard FFTF refueling equipment and operating procedures. The Interim Decay Storage vessel is located in-containment adjacent to the reactor vessel and is designed to provide temporary storage in a liquid sodium environment for new components en route to the reactor, and for irradiated components removed from the reactor. The Fuel Storage Facility is a separate building adjacent to the Reactor Service Building, and is designed to store spent fuel in a liquid sodium environment.

As each fuel component is removed from the reactor vessel, it must be replaced with a non-fueled component in order to maintain the upright geometry of the core necessary for the in-vessel handling machines to remotely grapple remaining core components. Initial shutdown planning called for inserting simulated core components that were used during the pre-critical checkout of the reactor system during this defueling evolution. An innovative approach was implemented to use the irradiated non-fuel core components (e.g., reflectors, control rods, etc.) that would have been excessed in order to maintain the necessary geometry. When defueling is complete, most of the non-fuel core components will be consolidated in the reactor vessel. This use of non-fuel irradiated hardware will result in waste minimization and circumvent the associated washing (removal of sodium from) and disposal costs for this hardware, culminating in a cost avoidance of $38 million. At the completion of reactor defueling, there will be 243 non-fueled components in the reactor vessel, 112 fueled components in the Interim Decay Storage and 258 fueled components in the Fuel Storage Facility. The milestone for completion of reactor defueling is September 6, 1995; however, the activity is ahead of schedule and is expected to be complete several months early.

Environmental Compliance

Significant activities are required to achieve compliance with the environmental regulations which impact the Phase I transition activities. The major regulations include the National Environmental Policy Act of 1969 (NEPA), Resource Conservation and Recovery Act of 1976 (RCRA), Clean Air Act and Amendments of 1990, Toxic Substances and Control Act of 1976, and National Historic Preservation Act of 1966. In addition, environmental restoration activities at the Hanford Site are governed by negotiated milestones established in the Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement).

Fig. 1.

Summary Schedule

The January 1994 amendment to the Tri-Party Agreement, required DOE to enter into negotiations with the State of Washington Department of Ecology and the U.S. Environmental Protection Agency, Region 10, to establish milestones for major Hanford Site facilities undergoing the decommissioning process. These milestones are legally binding and subject to judicial enforcement. A tentative agreement was signed by the three parties on January 17, 1995. The agreement includes milestones for transition of the FFTF corresponding to the critical path transition activities. The tentative agreement will undergo a 150 day public review and comment period prior to finalization. The goal of the FFTF agreement is to ensure that the facility completes the transition phase as expeditiously as possible so that funds can be redirected to higher priority environmental restoration activities at the Hanford Site.

The appropriate level of NEPA documentation required for the transition activities was determined to be an Environmental Assessment (EA). An EA was prepared and, following appropriate WHC and DOE review, was submitted to the affected states and Indian Nations on January 11, 1995 for a 30 day public review and comment period. The transition activities specifically delineated in the EA cannot commence prior to completion of the NEPA process. The associated constraints are depicted in Fig. 1, the most immediate being construction of the Sodium Storage Facility and initiation of fuel offload. The DOE-Richland Field Office recently received delegated authority from the DOE-Headquarters to approve NEPA documentation. This has expedited the EA approval process and removed it from the critical path. Following resolution of any comments, the Finding of No Significant Impact (FONSI) is anticipated by the end of February, 1995.

RCRA establishes requirements for the management of hazardous waste. The State of Washington Department of Ecology administers the requirements of this Act through the "Dangerous Waste Regulations" contained in the Washington Administration Code 173-303 (WAC). If a future decision is made to dispose of the approximately 260,000 gallons of FFTF sodium coolant as waste, management of the material would be regulated pursuant to these requirements. Currently, the sodium provides cooling for the irradiated fuel components. Although this inventory of sodium will not be needed for cooling after the fuel is removed, a new purpose for the sodium has been identified by the Tank Waste Remediation System (TWRS) Program at the Hanford Site. The TWRS Program plans to use the sodium, converted to sodium hydroxide, for caustic washing as part of the high level waste tank sludge pretreatment process. The TWRS process requires 21,000 metric tons of sodium hydroxide. Sodium hydroxide produced from the sodium metal at FFTF represents approximately 8 percent of this requirement. The sodium will be managed as product material for use at TWRS until an evaluation is completed in June, 1998, to confirm the final sodium disposition and form (i.e., product or waste).

In the interim, activities have been initiated to provide RCRA and WAC compliant storage for the sodium in the Sodium Storage Facility and treatment in the Sodium Reaction Facility, in the event the 1998 evaluation determines the sodium use at TWRS is not viable and the material must be managed as dangerous waste. A Notice of Intent for the new sodium facilities was submitted to the public for a 150 day public review/comment period. Following this period, a Part A Permit Application will be submitted to the State of Washington Department of Ecology.

A Notice of Construction is being prepared according to requirements of the WAC 246-247, "Radiation Protection - Air Emissions" and 40 CFR 61 Subpart H for the Sodium Storage Facility. The Notice of Construction is scheduled for submittal to the Washington State Department of Health by February 1, 1995 and approval is required to initiate construction of the Sodium Storage Facility.

IEM Cell/Fuel Offload Washing

The highly-radioactive fueled components will be moved to interim storage at the Interim Storage Area. The fueled components that are currently stored in a sodium environment will be transferred one at a time to the Interim Examination and Maintenance (IEM) Cell for washing, utilizing existing fuel handling equipment. The fuel is subjected to a moist argon atmosphere (within a closed sodium reaction system) to slowly react residual sodium in a controlled manner. The initial reaction is followed by several water rinses and, subsequently, the fuel is dried. The cleaned fuel is loaded into a Core Component Container in the IEM Cell, inerted, sealed and then transferred to the Reactor Service Building Cask Loading Station for placement into an Interim Storage Cask for transfer to above-ground dry cask interim storage at the Interim Storage Area. The cask is designed for storage up to 50 years pending a decision on final disposition (e.g., repository emplacement).

The rate at which fuel can be transferred to the IEM Cell, washed, dried and transferred into a Core Component Container and an Interim Storage Cask is limited by the time it takes to wash the assemblies. Fuel washing operations will be continuous (round-the-clock) seven days per week during the offload period. Since these IEM Cell activities, i.e., transfers, washing, and drying, have been performed routinely as part of the experiment processing cycle, rate estimates are presented with a high degree of confidence. Assuming a conservative availability factor of 50% for the entire fuel offload process, the current inventory of fuel assemblies and pin containers (371), which also includes 32 unirradiated assemblies and some special assemblies that will be stored at other locations, can be washed in approximately 2.8 years.

Several enhancements to the IEM Cell and cask loading station are in progress to support the fuel offload process and are scheduled for completion by September, 1995. An ion exchange system is being installed in the radioactive liquid waste system to essentially eliminate the waste that is currently generated during use of the sodium removal system (approximately 500 gallons/wash). This represents a significant waste minimization effort that will reduce the radioactive liquid waste that would have been generated from approximately 185,000 gallons to less than 1,000 gallons. The savings in radioactive liquid waste disposal costs are expected to be greater than $2 million.

There are two intact assemblies that produced a delayed neutron-monitoring signal while in the reactor, which indicates that a breach occurred in the fuel cladding.

These assemblies will be disassembled in the IEM Cell. The failed pin(s) will be identified and encapsulated, placed in pin containers with the remaining pins and dispositioned as the highly-radioactive fuel discussed above. Additionally, several fuel assemblies are known to have pin hole cladding failures which resulted in release of fission gases only while in the reactor; these assemblies will be processed last to minimize the consequences of potential contamination release and resultant deposition in the sodium removal equipment, which would make equipment maintenance more difficult.

While the modifications are being completed in the IEM Cell, selected fuel pins and ducts will be washed and transferred to off-site facilities (i.e., Idaho [Hot Fuel Examination Facility] or Pacific Northwest Laboratory [300 Area, Hanford Site]) for further examination. These materials were irradiated in the FFTF under the terms of a Specific Memorandum of Agreement between the DOE and the Power Reactor and Nuclear Fuel Development Corporation of Japan (PNC). Planning shows that this activity can be completed by the end of Fiscal Year 1995 without affecting the transition project critical path. Funding for this work will be provided by PNC.

Interim Storage Casks

The Core Component Container accommodates six or seven fuel assemblies or pin containers and will ultimately be stored in an Interim Storage Cask. In addition to providing the primary confinement boundary for the 50-year design life of the fuel dry storage system, the Core Component Container provides the necessary handling capability for transfer operations. The fabrication contract for fifty Core Component Containers was placed on January 13, 1995.

The Interim Storage Cask design and fabrication contract schedule is a critical path element for successful completion of the FFTF transition. Initiation of fuel washing activities is dependent on having storage available. Above ground, dry cask storage is currently used both nationally (e.g., by Virginia Power at the Surrey and North Anna plants), and internationally (e.g., nuclear power plants in Canada and Europe). The concept has been thoroughly studied and documented [additional details pertaining to this mode of operation for interim storage may be found in the Final Version Dry Cask Storage Study (3)].

The Interim Storage Cask design is a passively ventilated concrete and steel shielded cask with a stainless steel secondary confinement boundary. The maximum weight of the cask, with the 5,000 pound Core Component Container payload, will be 114,200 pounds. The Interim Storage Cask design is modeled after an approved U.S. Nuclear Regulatory Commission (NRC) design and meets the general licensing provisions of 10 CFR 72 for storage of commercial fuel. Provisions have been incorporated into the ISC design to accommodate the possibility of future on-site fuel transfers; however, this unit is a storage cask, not a licensed transportation cask. Fuel transportation casks are typically large, involve complex loading and transfer operations, and are very expensive. Design, procurement, and licensing of the 50 to 60 Interim Storage Casks required for storage of the FFTF fuel to meet requirements for off-site shipping would be cost prohibitive and inconsistent with industry practice.

The purchase order for the design and procurement of the first ten Interim Storage Casks was placed with General Atomics on September 16, 1993 at a cost of $4.7 million. After the first cask is received and tested, the purchase order for the remaining forty casks will be placed by September 1995. Original planning was to place six fueled components into each Core Component Container. However, an alternate loading configuration is being evaluated in which seven assemblies would be placed in each Core Component Container. This would reduce the number of Core Component Containers and Interim Storage Casks needed from 57 to 50. This approach would result in a substantial cost savings to the Project on the order of $2.6 million. There would also be additional cost savings achieved as a result of fewer cask loading/handling operations.

The Interim Storage Area, located in the northeast corner of the FFTF complex, will be surrounded by a fence with locked access to permit controlled loading, unloading and inspection of the Interim Storage Casks. There will be overhead lighting and crash barriers installed, as required. The design and construction of the Interim Storage Area is estimated at $300,000 with a construction schedule as depicted in the Fig. 1 summary schedule.

Unirradiated/Low-radioactive Special Nuclear Material (SNM)

Eleven slightly irradiated fueled components will require additional safeguards and

security measures because there will be insufficient radioactivity levels for the assemblies to be self-protecting. By the year 2030, seven assemblies will be classified as slightly-radioactive, and the remaining four will be moderately-radioactive. The current approach is to place the seven future low-radioactivity fuel in a single Interim Storage Cask and ship it to the Hanford Plutonium Finishing Plant (PFP) for storage in their existing protected area. The four future moderately-radioactive fuel would be intermixed (one each) with five or six highly-radioactive components and stored at the Interim Storage Area.
The fuel offload also involves sending 32 unirradiated fueled assemblies to the PFP for storage. As with irradiated fuel, the unirradiated assemblies will be washed in the IEM Cell. An engineering study is evaluating the preferred storage option at PFP. At present, the lowest cost option appears to entail loading the fuel into Interim Storage Casks for transfer and storage of the fuel on a storage pad at the PFP protected area. Before the Interim Storage Casks can be shipped to the PFP, a Safety Analysis Report for Packaging (SARP) must be prepared to allow the Interim Storage Casks to be transported across the Hanford Site. An estimated $5 to $7 million savings over earlier storage options could be achieved by this approach.
Sodium Storage Facility
The metallic sodium coolant will be maintained in a molten state until the fuel assemblies are removed from their respective storage location (i.e., the reactor vessel, Interim Decay Storage, or the Fuel Storage Facility) and the sodium transferred to the new Sodium Storage Facility. The Sodium Storage Facility will be designed and constructed adjacent to the FFTF complex to support the sodium drain operations and will house the FFTF primary (140,000 gallons), secondary (66,000 gallons), Interim Decay Storage (23,000 gallons) and Fuel Storage Facility (31,000 gallons) sodium inventories.
A detailed review of the design and construction schedule for the Sodium Storage Facility indicated that a six month acceleration could be achieved, contingent upon availability of appropriate funding, and would culminate in a substantial savings to the Project of approximately $3 million for every month the project is completed early. Detailed planning and supporting activities are proceeding accordingly on an accelerated basis. The Fig. 1 summary schedule reflects this acceleration. The conceptual design was approved on December 16, 1994 with a total project cost of $10.4 million (including required transfer piping). Project authorization was received from RL on December 30, 1994 for $5.2 million in Fiscal Year 1995 to complete the definitive design, site preparation work, and contract award for starting construction on/or before the end of the fiscal year.
Conceptually, the Sodium Storage Facility will be a concrete building approximately 110 feet long by 170 feet wide. Because of the uncertainty in the final sodium regulatory designation, and the fact this is a new facility, the Sodium Storage Facility will be designed and constructed to meet Resource Conservation and Recovery Act of 1976 (RCRA) requirements, as implemented by the WAC 173-303, Dangerous Waste Regulations. This will eliminate modifying the facility to meet WAC 173-303 storage requirements following the 1998 TWRS evaluation should the sodium designation change to waste. Four sodium tanks (three 80,000 gallon and one 52,000 gallon capacity), originally procured for the Clinch River Breeder ♀
Reactor Project, will be installed in the facility. The tanks are currently stored at the Hanford Area's 300 Area. Each tank is expected to require a separate move, using a crawler type transporter. A large, portable crane at each end of the move will be required to handle the tanks. An access route to the Sodium Storage Facility site will need to be cleared (power lines moved, fences removed/installed, underground utilities protected from weight, etc.). The facility will provide shielding to reduce radiation levels to acceptable limits outside the building walls. Radiation levels at the top of the tanks will be approximately 85 millirem per hour.
Piping will be provided from the FFTF sodium systems to the new Sodium Storage Facility. Insulation and trace heat will be added to maintain a 300F to 400F temperature during the sodium transfer. The tanks will be inerted prior to transfer of the sodium and an inert gas blanket will be maintained over the sodium during storage. The sodium will be pressure transferred to the facility, and subsequently allowed to cool and solidify. The sodium will be stored in a frozen state pending the decision on final sodium disposition and form (product or waste). The sodium would not be converted to sodium hydroxide until closer to the TWRS projected need

date in the year 2008.
Sodium Drain
Approximately 90 percent of the plant systems are required to support hot sodium circulation at an estimated minimum surveillance and maintenance cost of $32 million per year. Therefore, efforts have been ongoing to attempt to accelerate availability of the Sodium Storage Facility, as discussed above. An innovative approach will be implemented to transfer the secondary sodium loops to the three in-plant storage/drain vessels on an expedited basis prior to the availability of the Sodium Storage Facility (April, 1997). This secondary sodium transfer will be initiated after the reactor core has been defueled, and two immersion heater assemblies have been installed in the reactor vessel fuel transfer ports. This transfer sequence is estimated to save approximately $2.6 million resulting from reduced electrical power costs and elimination of related equipment maintenance and surveillance. The primary system will be drained when the Sodium Storage Facility is available followed by the Interim Decay Storage and Fuel Storage Facility drain when all the fuel has been removed.
Every reasonable effort will be taken to remove as much of the sodium as practicable from the plant. However, sodium residuals (approximately 3,000 to 4,000 gallons) will remain within the systems at the conclusion of the transition phase. It is planned to maintain an inert gas blanket over the frozen residuals to minimize any chemical reactions during the long-term surveillance and maintenance period. This will provide the greatest flexibility for later decommissioning activities. A small quantity (600 gallons) of sodium-potassium (NaK) alloy exists in specialized cooling systems within the plant. Chemically, NaK is more reactive than sodium, especially when exposed to air, and can become shock sensitive. As a result, we believe it is expedient not to leave Nak residuals in the plant. An innovative approach is being evaluated that includes cross connecting appropriate sodium and NaK piping systems and using the sodium to flush the NaK into the bulk sodium. The total NaK inventory is a small fraction of 1 percent of the sodium volume, and sodium properties (e.g., freezing point) would not be measurably affected by the presence of this small quantity of NaK. This option will require confirmation by the TWRS Program to ensure the potassium impurity would not impact the proposed use of the FFTF sodium in the tank waste pretreatment process.
Due to the relatively simple secondary loop layout and component design, most of the sodium in the secondary loops will drain by gravity. However, the Intermediate Heat Exchangers will not drain completely. Each Intermediate Heat Exchanger will have approximately 1500 gallons of secondary sodium remaining in the tube sides of the units following the gravity drain. This sodium must be removed by cutting into the boundary at the top of the inlet pipe of each unit and installing a special dip tube through which the sodium can be removed. This activity must be performed within the primary Heat Transport System cells inside containment where the Intermediate Heat Exchangers are located.
Drain of the primary system provides a unique challenge. With the exception of the reactor vessel, all of the plant sodium systems are trace-heated and system temperature can be maintained while draining. The reactor vessel is currently maintained at temperature by heat transfer into the primary system from the secondary loops (which are heated by pump work and/or oil fired preheaters). As soon as primary flow is stopped, the reactor vessel will start cooling. The reactor has no permanently installed heaters and there is virtually no temperature instrumentation on the vessel itself. During original preheat during the sodium fill, two temporary heating units were utilized to preheat the reactor vessel. Only one of these units is still available, the Nitrogen Blower Heater Unit. This unit circulates heated nitrogen around the bottom of the reactor vessel through a permanently installed piping manifold within the guard vessel. In addition, electrical immersion heaters are being purchased and will be installed through the fuel transfer ports to provide a redundant heating mechanism.
The three primary loops, the upper portion of the reactor vessel, and the reactor overflow tank can be drained. At the point where the reactor vessel outlet nozzles are uncovered, circulation will stop and the sodium will begin a slow cooldown if auxiliary heating is not provided. At this point approximately 51,000 gallons of sodium will remain in the reactor vessel and all vessel heat will come from the Nitrogen Blower Heater Unit and immersion heaters. Sodium draining can continue through the cold leg piping until the sodium level is reduced to the vicinity of the

reactor inlet nozzles. However, drain of the reactor vessel below the level of the inlet nozzles is precluded by design. Approximately 16,000 gallons of sodium will remain in the vessel at this point. An access hole will be drilled through the core basket and the wall of the low pressure inlet plenum to permit installation of a dip tube directly from the top through the reactor vessel head. This will require design and fabrication of remotely installed drilling equipment over forty feet long for drilling under sodium.

The Interim Decay Storage vessel has an installed drain line to the bottom and can be drained. The drain will be complicated by the time consuming task of removing sodium from the Core Component Pots within the vessel, either by siphoning or displacement. The Fuel Storage Facility vessel drain will also have to utilize a dip tube. In addition, the Fuel Storage Facility is not cross-connected with the rest of the FFTF sodium systems, and approximately 400 feet of sodium line will have be installed to connect to the Sodium Storage Facility.

General Plant Support/Auxiliary Systems Shutdown

The level of general plant support for the FFTF during the first four years of transition will be comparable to that required for hot standby because approximately 90% of the plant systems are required to support hot sodium circulation prior to sodium draining. Some reduction will result from the planned early drain of the secondary sodium systems. Future efforts will focus on compressing the logic for shutdown of the auxiliary systems. Disposition of the polychlorinated biphenyl transformers following the completion of the sodium drain was a major contributor to the two year extension to the transition schedule. Emphasis is being placed on integrating this activity earlier in the transition activities by shifting plant power loads as possible to free up transformers for removal. This effort is expected to result in compression of the overall schedule.

As the plant systems become available for shutdown, there will be a corresponding reduction in the need for general maintenance and plant support. As the level of required support decreases, the displaced resources will then be available to assist in configuring the plant for turnover to the Environmental Restoration Contractor. The required level of support will be minimal after fuel offload and sodium drain and cooldown to ambient temperature. Essentially all of the plant systems will be deactivated. Actual facility support will be limited to minimal maintenance, inert gas system positive pressure checks, and facility walkdowns. The cost for the facility in this low cost surveillance and maintenance state at the completion of the transition phase is expected to be approximately $1 million per year.

Minimal monitoring will also be required for the Sodium Storage Facility up to the time it is drained and the sodium is processed through the Sodium Reaction Facility. Fire detection will be required in both facilities where storage of sodium and processing will be centralized.

Sodium Reaction Facility

The Sodium Reaction Facility will be constructed as a separate building near the Sodium Storage Facility. Since the FFTF sodium may not be required by the TWRS Program until 2008, for planning purposes the construction of the Sodium Reaction Facility is being deferred, since the long term storage of solidified sodium is preferred to the long term storage of sodium hydroxide. The Sodium Reaction Facility schedule depicted in Fig. 1 presents the schedule that would be pursued if the sodium has to be disposed of as waste. It is retained on the schedule to maintain visibility for the project. The technical baseline for both the TWRS and FFTF Programs is to utilize the sodium, converted to sodium hydroxide, for the high level waste tank pretreatment process.

The sodium reaction process utilized by Argonne National Laboratory-West currently forms the technical baseline for the Sodium Reaction Facility. Pending operational verification of the Idaho unit, a duplicate processing facility will be constructed at FFTF, thus reducing development and design costs. The process consists of injecting molten sodium and water into a reaction vessel partially filled with 30% to 50% sodium hydroxide at about 240F. The vigorous reaction produces more sodium hydroxide and hydrogen gas. The gas is swept out of the vessel by a nitrogen cover gas purge and ♀
maintained at sufficiently low dilution so as not to be flammable when mixed with air. The FFTF sodium metal will be processed in two years, assuming a plant efficiency of 70% at a throughput of about 30 gallons/hour. If disposal of the sodium as waste is required, the sodium hydroxide solution would be reacted with

sulfuric acid to produce sodium sulfate. The sodium sulfate would be dried and collected into containers and transported to a Hanford Site disposal facility.

Documentation

The FFTF Transition Project Plan will be maintained as a living document that will be updated as the FFTF transition progresses in order to provide the current logic and project baseline. Implementation of a Technical Specification Reduction Project eliminated 79% of the current Technical Specifications and provided a new, cohesive safety basis for the FFTF deactivation (estimated savings of $1.1 million). Revisions will be required to reflect new plant evolutions (i.e., fuel offload and sodium drain) and evolving plant conditions as the transition phase proceeds. The End-Point Criteria Document will define the actions required prior to turnover of the FFTF to the Environmental Restoration Contractor and the Surveillance and Maintenance Plan will address the requirements for the Surveillance and Maintenance Phase.

CONCLUSION

To date, the Transition Project has been extremely successful and continues to be ahead of schedule and under cost. Challenging and innovative work is in progress to resolve technical issues and to support efforts to compress the transition schedule. The erosion of experienced FFTF staff, from natural attrition and the recent early retirement incentive, has begun to be felt and dwindling resources will make achievement of the aggressive transition schedule a challenge.

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PUREX/UO3 PLANT DEACTIVATION PROJECT STATUS: THE TRANSITION TO DECONTAMINATION AND DECOMMISSIONING

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ABSTRACT

The U.S. Department of Energy's Plutonium-Uranium Extraction (PUREX) Plant came on line in mid-1955. The PUREX plant extracted plutonium and uranium from spent nuclear fuel for use in the nation's defense program. The plutonium was sent to the Plutonium Finishing Plant, and the uranium was sent in liquid form to the Uranium Trioxide (UO3) Plant. At the UO3 plant, the liquid was converted to a solid uranium oxide powder that was then shipped to other Department of Energy defense sites. Now, Hanford's mission has changed from making plutonium to environmental cleanup. The Department of Energy shut down the PUREX plant in 1990 and the UO3 Plant in 1993. The two plants are to be deactivated--brought to a state where they can be maintained with minimal surveillance while awaiting final deposition.
Deactivation begins the transition of a nuclear facility to decontamination and decommissioning (D&D), the final phase of a facility's life cycle. The goal of deactivation is to place the facilities in a safe and environmentally secure configuration. The five-year PUREX/UO3, Deactivation Project Management Plan (Project Management Plan) outlines the proposed methods by which that goal is to be achieved. The Project Management Plan is being reviewed by the Washington State Department of Ecology, the Department of Health, the U.S. Environmental Protection Agency, citizens and interest groups effected by Hanford cleanup decisions. Their input helps decide how cleanup will proceed.

DEACTIVATION

Deactivation activities include eliminating or reducing the plants' major chemical

and radioactive hazards. The bulk of the equipment and the structural components will remain, and some contamination will remain on surfaces within the plants. Such contamination is fixed in place or isolated, to keep it away from people and the environment.

When deactivation is complete, the buildings will be unoccupied and locked, and will be monitored and inspected periodically. The PUREX and UO3 plants can remain in this safe condition for many years, until final D&D takes place.

WHY DEACTIVATE?

The Department of Energy wants to deactivate the PUREX and UO3 plants in order to progress with Hanford cleanup and to save money. Surveillance and upkeep alone for these two plants costs about $34 million per year. Deactivation activities are expected to cost between $10 and $16 million each year of the five-year deactivation project. Once the plants are deactivated, expenses will decease to less than $2 million annually.

THE DEACTIVATION PROJECT MANAGEMENT PLAN

In December 1992, planning was initiated for deactivation of the PUREX and UO3 Plants. The objective was to identify the activities needed to make both plants safe and environmentally secure until their final disposition.

The PUREX/UO3 Deactivation Project Management Plan describes the proposed deactivation approach and the estimated supporting technical, cost, and schedule baselines. For planning purposes, an option was chosen for each deactivation activity so that cost and schedule estimates could be developed for the project. However, other public interests and values may be associated with deactivation. Interested public parties, the Department of Energy, and Westinghouse Hanford Company and its regulatory agencies will all define together how the PUREX/UO3 deactivation project will proceed.

The objects of the Project Management Plan support the Department of Energy's goal to develop swift, uniform methods for deactivating facilities similar to PUREX/UO3. The objectives are:

Establish a safe and environmentally secure configuration of the plants that can be maintained for at least 10 years

bring yearly surveillance and maintenance costs to less than $2 million

Use a cost-effective, innovative approach to achieve and maintain deactivation

Comply with environmental, safety, and health codes and standards

Involve interested groups, citizens and affected parties in major policy decisions during development and execution of the Project Management Plan

Upon completion of deactivation, help the PUREX/UO3 workforces find new jobs

Apply lessons learned from commercial deactivation experience

Establish the PUREX/UO3 Deactivation Project as a model for other Department of Energy facilities.

ISSUES CONFRONTING PUREX/UO3 DEACTIVATION

Several issues challenge the PUREX/UO3 deactivation project. The project management plan explains how Westinghouse Hanford Company proposes to deal with these issues; however, public values will play an important role in making the final decisions. Some of the issues are:

Spent Fuel Storage

When PUREX shut down, it had not processed all the spent fuel in the plant. The remaining fuel must be moved and stored safely. The department of energy's current plan is to move the PUREX spent fuel to the 100 Area K Basins, where it will be encapsulated and stored with other Hanford spent fuel.

Nitric Acid

Contaminated nitric acid is a byproduct of the PUREX process. Approximately 200,000 gallons are stored at PUREX. One option is to reduce the quantity and concentration of nitric acid by a process called sugar denitration. This process would discharge nitrogen oxide gases to the air through the PUREX main stack. The preferred option is to re-use the nitric acid product at an off-site facility. Current plans are to send the acid to a facility in England.

Reducing Waste Volume

Large volumes of liquid waste will be generated from process flushing during deactivation. This waste will be transferred to the Hanford Site waste storage tanks. The amount of waste transferred can be significantly reduced by operating a PUREX concentrator that would create non-contaminated steam condensate, which would be disposed of in the 216-B-3 Pond System.

Plutonium/Uranium Solutions

There are approximately 6,000 gallons of plutonium/uranium solution in tanks in the PUREX canyon. The preferred alternative is to neutralize the solution and send it to the Hanford Site Tank Farms' double-shell waste tanks, where it would be mixed with similar waste.

Effluent Path

Currently, the PUREX ventilation system operates 11 stacks. The current planning basis option is to consolidate all the ventilation systems in the 202-A Building into a single effluent path. This action would reduce the volume of air discharged and the amount of stack surveillance required. Another option is to completely shut down the ventilation system.

Development of End States for Turnover to D&D

For each of the facilities' systems and spaces, a logical process is being followed to determine a satisfactory end state. The end state is the condition the system or space will be left in when deactivation is complete. The process looks at the objectives, tasks and expected later uses for that system or space. The end states are developed in successively more detailed levels, until each task necessary to achieve a satisfactory end state has been defined.

CONCLUSION

The PUREX/UO3 plants have had a long and productive history. Deactivation begins the transition of a nuclear facility to decontamination and decommissioning and is an important step in achieving a safe, stable condition that minimizes surveillance and maintenance costs. Deactivation also brings to an end a historic phase in the lives of the PUREX/UO3 plants.

30-3

DEACTIVATION AND CLEANOUT OF THE 308 FUELS LABORATORY AND THE 232-Z INCINERATOR AT THE HANFORD SITE

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ABSTRACT

This paper describes the deactivation and source term reduction activities conducted over the recent past in two plutonium-contaminated Hanford Site buildings: the 308 Fuels Development Laboratory and the 232-Z Incinerator. Both of these facilities belong to the U.S. Department of Energy, and the projects are unique success stories carried out in direct support of EM-60 functions and requirements. In both cases the buildings, for different reasons, contained unacceptable amounts of plutonium, and were stabilized and placed in a safe, pre-D&D (decontamination and decommissioning) mode.

The concept of deactivation as the last step in the operating life of a facility will be discussed. The need for and requirements of EM-60 transition between operations and D&D, the costs savings, techniques, regulations and lessons learned also will be discussed. This paper describes the strategies that led to successful source term reduction: accurate characterization, cooperation among different divisions within DOE and the Hanford Site, attention to regulations (especially unique in this case since the 232-Z Incinerator has been nominated as a Historic Structure to the National Register of Historic Places), and stakeholder concerns involving the proximity of the 308 Building to the Columbia River. The paper also weaves in the history, missions, and plutonium accumulation of the two buildings. The lessons learned are cogent to many other present and future deactivation activities across the DOE complex and indeed across the world.

DEACTIVATION ACTIVITIES AT THE HANFORD SITE

The Hanford Site, located in southeastern Washington state, was one of America's primary arsenals of nuclear defense production for nearly 50 years beginning in World War II. Approximately 53 metric tons of weapons grade plutonium, over half of the national supply and about one quarter of the world's supply, were produced at Hanford between 1944 and 1989. Today, many Site buildings are undergoing deactivation, an EM-60 activity that is a precursor to decontamination and decommissioning (D&D). The primary difference between the two activities is that equipment and structural items are not removed or torn down in deactivation. However, utilities are disconnected, and special nuclear materials (SNM) as well as hazardous and pyrophoric substances are removed from structures undergoing this

process.

308 FUELS DEVELOPMENT LABORATORY

Deactivation was completed as of March 31, 1994 at the Hanford Site's 308 Fuels Development Laboratory (FDL), a 94,000-square foot structure that opened in 1960 to support fuels diversification and modeling in the pre-breeder reactor era. Throughout its active lifetime, the 308 Building was the scene of an almost dazzling array of pioneering developments in fuels technology and fabrication. Primarily a non-defense facility, the 308 Building was the Atomic Energy Commission's (AEC - predecessor agency to the DOE) largest research effort to demonstrate the effectiveness of various PuO2 and MOX fuels in the "Atoms for Peace" program.(1) The first fuel mixtures produced in the PFPP (Plutonium Fabrication Pilot Plant - original name for the 308 Building) were metallic, but ceramic fuel blends were being worked in the facility within five years. For a brief time in the late 1960s, neptunium-aluminum alloy fuel target elements and lithium aluminate (LiAlO2) fuel targets were produced in the 308 Building for defense production testing in Hanford's N-Reactor.

However, the building's largest mission came when the Hanford Site was chosen in 1969 as the location for the DOE's prototype advanced sodium cooled reactor (the FFTF) that would develop and test fuels for breeder reactors.

The earliest fuels made in the 308 Building for potential use in the FFTF were vibration packed ("vi-pack") powders, but testing demonstrated a low density. This concept soon was discarded in favor of oxide pellet fuel made in the shape of cylinders approximately 3/10" high and 2/10" in diameter. The 308 Building's last mission was repackaging work involving MOX powders for purposes of consolidation This work took place from late 1991 through early 1992.(2)

The decision to deactivate the structure was driven by a 1980s Department of Energy (DOE) decision that plutonium fuels should not be fabricated in areas near the Site's boundaries, as well as by changing facility structural requirements. The 300 Area is located just four miles north of Richland, along the Columbia River, an area of intense interest and concern to regional stakeholders.

Removal of the building's working inventory of plutonium, used in making mixed oxide (MOX) fuel pellets for the Fast Flux Test Facility (FFTF) and other test reactors, took place over the course of a year beginning in mid-1991. Completed in May 1992, this work decreased the yearly security costs for the facility by approximately $3-million. During the peak of its years as a fabrication facility, the 308 Building held as much as 3 metric tons of encapsulated plutonium and approximately 200 kilograms (kg) of MOX powder.

Inventory transfer has been followed by the cleanout and stabilization of plutonium oxide (PuO2) and enriched uranium oxide (UO2) residues and powders in the facility's equipment and duct work. This additional effort, along with the transfer of all resident personnel from the building, lowered the annual surveillance budget by another $1 million.

ALPHA CONTAMINATION STABILIZED IN GLOVE BOXES AND HOODS

Over the past three years, a small crew, the remnant of a once-large FFTF fuel supply staff, worked in the 308 Building to wipe, spray and seal the 50 glove boxes and six open-faced hoods that are being left inside the facility until complete D&D occurs at a future date. A majority of the glove boxes are standard size, and contain front and back windows, numerous glove entry ports, equipment doorways and ports, and entryways for electrical and fluid/gaseous services. In these glove boxes, MOX powders and pellets were pressed and then sintered into reactor fuel, However, six of the glove boxes are approximately 30' long by 3' wide and 3' high, with 40 entry ports and multiple windows, larger equipment ports, and other penetrations. These glove boxes held the furnaces and other large, pilot equipment pieces crucial in the many fuels fabrication research and development activities that were pioneered in the building. The open-faced hoods are approximately 4' long by 3' wide and 3' high, and stand on legs about 4' high. These hoods, although they accumulated alpha contamination, did not provide the level of confinement of a glove box, and so were used primarily for work with uranium materials and processes.

The deactivation crew, after removing much of the instrumentation and other small equipment from the building, donned special anti-contamination clothing and extracted small equipment from inside the glove boxes and hoods. They then performed multiple wipe-downs of the inner surfaces, using damp rags that later were

dried and disposed as solid waste. Next, they sprayed the insides of the glove boxes and hoods with a modified acrylic latex, contamination fixant that appears cloudy at first but dries to a nearly clear state. Lastly, they covered all of the glove ports with specially fitted metal plates, and placed over them a polyolefin "shrink-wrap" material that contained an adhesive on the inside. The material is the same as that used to protect welds in industrial pipelines. Using a hot air treatment, they activated the tar-like adhesive so that it melted and flowed into all the crevices between the plates and the ports, thus creating a very rugged seal. The plutonium inventory currently remaining ("held-up") in this equipment totals less than 400 grams.

ATTENTION ALSO GIVEN TO DUCT WORK, TEST REACTOR AND OTHER BUILDING COMPONENTS

In the meanwhile, non-destructive assay was performed on 2,000 feet of duct work in the 308 Building. Although only small amounts of contamination were found, the flanges were caulked with silicone sealants. Lastly, the fasteners on the gaskets were painted with a high-grade interior household sealant. Uncontaminated equipment in the building, such as wire-wrap machines used to spiral wrap the outside of each FFTF fuel pin, pulse magnetic welding (PMW) equipment, and profilometers used to make precise measurements of the outside diameter of finished fuel pins, were either excessed or sent to offsite storage.(4)

The deactivation of a 250-KW (kilowatt) TRIGA (Training Research Isotopes, General Atomics)* reactor that was emplaced in Room 162 of the 308 Building's Annex in the late 1970s will occur on a slower schedule. The reactor operated for 13 years to perform neutron radiography testing on fuel pellets and pins, to irradiate materials, and to provide reactor operator training. Currently, the TRIGA's 68 fuel elements have been removed from its core and placed in racks in the water-filled pit or reactor pool. An Environmental Assessment (EA) for the disposition of this spent fuel, which has been irradiated to only a low burnup level (less than 1 percent), currently is under review.

Specially designed irradiated fuel shipping/storage casks are being designed by the staff of the Hanford Site's operating contractor, Westinghouse Hanford Company (WHC). Until the fuel leaves the 308 Building Annex however, regular surveillance of this area will be necessary. Following fuel removal, the control rods, other neutron sources, and some instrumentation will be taken out of the TRIGA and the water pool will be drained. Most of the large, fixed equipment will be left for future D&D.(5)

232-Z INCINERATOR

During 1994 in Hanford's 200-West Area, cleanout of plutonium-bearing equipment, parts and residues took place at the 232-Z Incinerator. Located in the PFP (Plutonium Finishing Plant) complex, the incinerator operated from 1962 through 1973. It was conceived and built as a way to salvage additional plutonium (Pu) that was being buried as residual material on contaminated solid wastes in the late 1950s, a time when the per unit value of plutonium was extremely high.

Candidate wastes for the incinerator included cartons containing contaminated filters, rags, paper, special work clothing, gloves that were not rubber or plastic, and other combustible items used in the PFP and in Hanford's REDOX and PUREX operations. Still more Pu was recovered from non-burnable scrap in one (later two) leach pot(s) in the 232-Z Building. The leachate then was run through solvent extraction processes in the 234-5Z facility to complete the Pu recovery operation.(6)

Preliminary isolation and terminal cleanout activities began in 1982, and continued on a sporadic basis until inspections undertaken in 1989-90 demonstrated the need for more intensive action to stabilize the old facility. At that time, when seismic evaluations demonstrated the 232-Z Building's inability to meet modern criteria, the structure was elevated to high priority status for cleanout. Further, inspections of the duct work and service piping estimated the presence of 848 grams of "held-up" plutonium, levels considerably higher than the "one-third of critical mass" amounts necessary for the facility to be considered "isolated."

The 1994 cleanout was conducted as a joint effort between the WHC Transition Projects and Decontamination and Decommissioning (D&D) organizations. Between January and September, Nuclear Process Operators (NPOs) from the PFP removed moveable equipment, instruments, portions of contaminated piping, duct work and other parts.

At the completion of this effort, the 232-Z Building was left standing, with its

five glove boxes, the scrubber (exhaust filtration) cell, the ventilation system, the in-floor filter and the filter box still in place. Approximately 600 grams of Pu were removed, and the effort cost about $1-million.

USQ DECLARED AND RESOLVED

However, just at that point, NDA (non-destructive analysis) undertaken to ascertain the Pu inventory of the incinerator glove box itself still showed the presence of between 1-10 kilograms (kg). Since the amount at the high end of that scale falls outside the confirmed safety boundaries, a USQ (Unreviewed Safety Question) was declared in late August, and administrative controls (including no unnecessary entries) were placed on the 232-Z Building.

Immediately, two separate criticality analyses were launched, to model the situation in the glove box under worst case scenarios. No conditions that could lead to a criticality event were found. Within just a few days, an extensive, independent re-analysis to better characterize gamma and neutron emissions from the Pu was started by a team of national experts in NDA. The team also looked at attenuation (shielding) factors in connection with the approximately 2,900 pounds of fire bricks in the glove box itself.(7)

Using state-of-the-art neutron detection equipment, as well as the long and careful germanium method of gamma detection, the team placed a highly radioactive cesium-137 source at various points outside the glove box to determine and compare their readings. No intrusive entry into the glove box was allowed. Their findings, released in October, demonstrated the presence of only 1-5 kg of Pu.

At the same time, Hanford Site personnel moved to re-study the structural characteristics of the 1962 concrete block building itself. While re-verification took place, consideration was given to placing a heavy steel shroud around the incinerator glove box itself. However, the re-study showed that such a shroud would not be necessary.

In a parallel path, safety re-analysis was taking place. Accident scenarios ranging up to total collapse of the building (with toppling of the glove box) were studied. The outcome of these analyses demonstrated only a 10 percent "release factor" for the glove box itself, but a 100 percent release factor for the building. In other words, should an earthquake occur, only 10 percent of the material in the glove box would be expected to escape the glove box (through two top glove ports), but 100 percent of that material would be expected to escape the building.

Given this result, no further protective actions were required, but WHC decided to install a special new anchoring system for the glovebox as an added safety measure. This anchor assures that the glove box will not topple over under any foreseeable accident conditions. Design, procurement and installation concluded in December. Meanwhile, the USQ was officially resolved.

232-Z FACILITY HISTORIC

Meanwhile, in mid-1994, the 232-Z facility had been named to the National Register of Historic Places due to its unusual equipment design and first-of-a-kind method of automating and executing the incineration of plutonium-bearing combustibles. Opened in 1963, the 232-Z operation surpassed previous Pu scrap burning efforts at other AEC sites in terms of complexity and automation.

Because of its unique past, the 232-Z facility required thorough historic documentation and photographing under the National Historic Preservation Act. A Historic American Engineering Record (HAER) preservation package was rushed to completion during the hectic days right after the USQ was declared and the criticality analysis was completed. The building is now being readied for final cleanout and eventual D&D.

CONCLUSION

In conclusion, the deactivation work carried out at Hanford's 308 Fuels Development Laboratory and at the 232-Z Incinerator demonstrate lessons that are cogent to many other present and future deactivation activities across the DOE complex and indeed across the world. The lessons in techniques of contamination fixation, cooperation among various organizations, attention to regulations and stakeholder views and priorities can be applied elsewhere to stabilize buildings that may have to wait years for D&D but that must be rendered safe for the interim period.

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30-4

METHODS TO ESTIMATE EQUIPMENT AND MATERIALS THAT ARE CANDIDATES FOR REMOVAL DURING THE DECONTAMINATION AND DECOMMISSIONING OF FUEL PROCESSING FACILITIES

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ABSTRACT

The methodology presented in this report provides a model for estimating the volume and types of waste expected from the removal of equipment and other materials during Decontamination and Decommissioning (D&D) of canyon-type fuel reprocessing facilities. This methodology offers a rough estimation technique based on a comparative analysis for a similar, previously studied, reprocessing facility. This approach is especially useful as a planning tool to save time and money while preparing for final D&D. The basic methodology described here can be extended for use at other types of facilities, such as glovebox or reactor facilities.

INTRODUCTION

Located in the southeastern region of the state of Washington, the Hanford Nuclear Site began chemical processing operations in 1944. Now over five decades later, a number of facilities on the Hanford Site are preparing for final decontamination and decommissioning (D&D). To assure that the necessary resources are available to handle the waste generated as a result of D&D, estimates of the volume and characteristics of the expected solid waste are required.

The Plutonium-Uranium Extraction (PUREX) Plant (Canyon 202-A) is a fuel reprocessing canyon facility that is similar in construction to other onsite and offsite fuel reprocessing canyon facilities. It is also one of the first major facilities that will be scheduled for D&D at the Hanford Site (1). This report presents the methodology used to estimate the characteristics and volumes of equipment and other materials that might be removed and processed as solid waste during the D&D of the PUREX Plant. Comparison of the PUREX D&D solid waste volumes to other canyon facilities yields solid waste estimates that may benefit future waste estimation and decontamination activities at Hanford as well as other U. S. Department of Energy (DOE) facilities.

PUREX PLANT

The PUREX facility consists of a canyon, four galleries, and a service annex. The canyon is a thick-walled, heavily shielded concrete structure that houses the equipment used for radioactive processing. The canyon area is 306.3 m (1,005 ft) long, 9.3 m (30.5 ft) wide, and 31.7 (104 ft) high. A single row of 12 process cells is contained within the canyon (1). The functions of these 12 cells, as well as the equipment they contain, are described in Table I.

A 1.8-m- (6-ft-) thick concrete wall separates the cells from the galleries (1). The storage, sample, pipe and operations, and crane cab galleries parallel the north wall of the canyon and are located at different levels, one above the other. The pipe and operations gallery, located below the crane cab gallery, contains instrument transmitter racks, electrical motor controls, steam and cooling water

supply lines, and the piping and associated valves used for transferring nonradioactive solutions. The next level down is the sample gallery, which contains the remote samplers used for obtaining process solution sampled from the cell equipment. The storage gallery area, the level below the sample gallery, was used for storage of dry chemicals and spare equipment.

The service annex is adjacent to the north of the gallery section and consists of two separate areas (1). The larger, main area houses the maintenance shops, offices, lunchroom, locker room, radiation zone entry, ventilation air and supply room, a switch gear room, compressor room, central control room, and the aqueous makeup facility. The smaller laboratory are contains the analytical laboratory, the headend control room, and a switch gear room.

In addition, PUREX contains two tunnels used for interim storage of failed or obsolete process equipment that was too radioactive or bulky for removal from the PUREX Plant. Since the estimation methodology presented in this report is based on the PUREX canyon cells, more detailed information on the PUREX tunnels is not presented.

In 1956, the PUREX Plant began recovering uranium and weapons grade plutonium from irradiated aluminum-clad uranium metal fuel from the Hanford Site reactors. The plant was modified in 1967, to reprocess zirconium alloy clad fuel from the N Reactor in order to recover plutonium, uranium, and neptunium.

Three semi-distinct operating areas reside within the PUREX Plant: the head end, where fuel elements were chemically declad and the irradiated fissile materials dissolved; solvent extraction, where the dissolved fuel was separated and purified; and the N Cell area, where purified plutonium was either loaded out as nitrate solution or transformed into an oxide.

In October 1990, the DOE - Richland Operations Office (RL) directed Westinghouse Hanford Company to initiate transition-to-standby activities for PUREX. The standby condition was achieved in September 1992. In December 1992, the Assistant Secretary for Environmental Restoration and Waste Management authorized plans to terminate the PUREX Plant and directed DOE-RL to proceed with shutdown planning and terminal cleanout activities (1).

At the completion of the Stabilization Campaign in 1990, the feed stock left in PUREX from the 1988 shutdown had been processed and removed from the plant. Bulk chemicals, solutions used to test the processing equipment, the PUREX process solvent, recovered nitric acid, and a small quantity of pre-1972 reactor fuel were left in the plant. During the subsequent transition-to-standby phase, these materials were left untouched.

The deactivation project will remove, reduce, and/or stabilize the major remaining radioactive sources and hazardous chemicals within the PUREX Plant. Completing these activities will reduce the risk to workers and the public and will allow for a reduced level of the PUREX Plant will be transferred to the Hanford Surplus Facilities Program pending eventual D&D (2).

Final D&D will include closure of secondary containment; the end-state of the equipment, systems and material left in place, including material in the "containment building;" final disposition of the vessels and equipment in the tunnels; and closure of the tunnels (2).

PUREX ESTIMATE

The PUREX Plant contains many process vessels, chemical storage tanks and other types of equipment that are candidates for removal and decontamination. To develop an estimate of the type of waste and volume of removal candidates, it was necessary to obtain information on the size, weight, material of construction, internal equipment and contents/contamination of the canyon process cells.

The large number of items involved and the allowable time precluded a survey of dimensions and weights for each piece of PUREX equipment based on inspection of individual as-built drawings. The as-built drawings represent the most up-to-date information, but would involve a fairly slow process of review and reproduction. For the PUREX estimate, approximately 5% of dimension and weight data were obtained from drawings, and 37% from certified vendor information files (CVI) (1). The only disadvantage is that without the as-built drawings there is no positive verification that the items described in the files are the items actually installed in the plant. Many of the CVI files specifically call out vessel or assembly weights, therefore they were very useful.

The items not identified using prints or CVI files contain estimates based, when

available, on information located in the PUREX Technical Manual (3), or on known values for similar pieces of equipment. Dimensions obtained from the technical manual do not include flanges or connectors or support structures. The technical manual also does not provide equipment weights. An item by item inspection and consultation of prints would be necessary to support actual D&D work.

The following waste types, based on the definition documented in the Hanford Solid Waste Acceptance Criteria current at the time of publication, were recognized in the development of this estimate (4).

Transuranic (TRU) Waste - Without regard to source or form, TRU waste is contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g of waste matrix. Transuranium radionuclides are radionuclides, radium sources and 233U in concentrations greater that 100 nCi/g of the waste matrix are managed as TRU waste.

TRU Mixed Waste - TRU mixed waste meets the definition above for TRU waste and contains dangerous waste in addition to the radioactive components. Dangerous wastes are defined in the Washington Administrative Code (WAC 173-303-040).

Low Level Waste (LLW) - LLW, as defined in DOE Orders 5820.2A and 5400.3, is radioactive waste not defined as high-level waste, TRU waste, spent nuclear fuel, or by-product material. All LLW is also classified according to Category 1, 3, and greater than Category 3 concentration limits. These limits are based on the waste classification system developed by the Nuclear Regulatory Commission (NRC) in 10 CFR 61; however, it should be noted that these categories are not synonymous with Class A, C, and >C definitions.

Low Level Mixed Waste (LLMW) - LLMW is low level waste that contains dangerous waste, as defined in WAC 173-303-040.

Nonradioactive Dangerous Waste - Any nonradioactive solid waste that has been contaminated by hazardous chemicals, as defined in WAC 173-303, is regulated as dangerous waste.

The goal of the current deactivation effort is to minimize dangerous wastes; therefore, equipment is assumed to be non-hazardous, unless a known hazardous component is present (1). Equipment classified as hazardous or mixed wastes include the dissolvers, which contain zirconium hulls and mercury thermowells, and the silver reactors, which contain silver salts. Nonradioactive, nonhazardous wastes are identified only as solid waste.

The waste volume estimates reflect those expected to be generated by all D&D activities prior to actual building structural component disposal. This includes volumes expected to result from removal of all the process equipment, piping, and fixtures that can be removed with relative ease, such as unbolting or cutting, leaving bare cell and canyon walls.

METHOD APPLICATION

Once a detailed study of a model facility is complete, other similar facilities may be estimated by scale. This methodology can save time and money while providing critical information to support planning activities for current and future solid waste treatment, storage, and disposal (TSD) facilities and operations. In this case, facilities similar in construction and process to the PUREX facility were selected.

The estimates obtained in the PUREX Plant study were applied to several other major canyon facilities located at the Hanford Site to characterize and estimate the volume of solid waste that will be generated during D&D. These facilities include the following: B Plant (221-B), T Plant (221-T), U Plant (221-U), the Uranium Trioxide (UO3) Plant (224-U & 224-UA), the Reduction Oxidation (REDOX) or S Plant (202-S), the Plutonium Concentration Facility for B Plant (224-B), and the Concentration Facility for the Plutonium Finishing Plant (PFP) and REDOX (233-S) (5).

In order to estimate the waste volume, the cells (or process areas) in each facility were matched with the most similar cell(s) in PUREX. Cells were correlated based on knowledge of the equipment present, physical layout, and process history.

To develop the volumetric estimate of waste, similar cell processes were identified, and it was assumed that the density of equipment in the unknown cell was the same as the matching PUREX cell. The volumetric ratios for similar cells were calculated then multiplied by the amounts of solid waste (volume and weight) reported for each of the PUREX cells (1) to yield the estimates for each unknown cell. These cell estimates were summed over the entire facility using the following equations(5):

Eq. (1)

Eq. (2)

where X represents the individual facilities. The same calculation was made for the pipe galleries:

Eq. (3)

Once the volumetric estimation was completed for each of the cells (or process areas) in a given facility, a waste class was assigned to the cell (5). The designation of waste classes was based solely on the best judgement of the authors and was grounded on knowledge of the processes and equipment present in a given cell. The piping in a facility was assumed to be designated in the same waste classes and proportions as the equipment in that facility.

EXAMPLE

The B Plant (221-B) is located in the center of the 200 East Area on the Hanford Site. The B Plant was originally constructed to chemically separate irradiated uranium fuel using the bismuth-phosphate process. The plant commenced operation in April 1945 and continued until 1952, when the bismuth-phosphate process became obsolete with the development of the PUREX method. The plant shut down until the early 1960's, when a series of renovations were made to support the extraction of various fission products from high-level liquid waste (5).

The final renovation, completed in 1968, facilitated a high-capacity solvent extraction and ion-exchange process. The campaign lasted from approximately 1968 until 1985 and resulted in the extraction of over 100 MCi of cesium and strontium (5).

A subsequent and final campaign involved 38,000 L of neutralized current acid waste (CAW) from double-shelled tank 101-AZ. Waste from this campaign still remains in the tanks in B Plant, but will be removed prior to decontamination and decommissioning (5).

Currently, B Plant stores chemicals used to treat LLW generated at the B Plant and the Waste Encapsulation and Storage Facility. It is used for the generation of demineralized water and for the conditioning of water used in heating, ventilation, and air conditioning units (6).

The B Plant is a lightly-reinforced concrete building, approximately 259-m (long), by 21-m (wide), and 23-m tall, covering an area of about 5,370-m2 and occupying a volume of approximately 121,100-m3 (5). The canyon portion is about 247-m long and contains 40 canyon cells in a single row running the length of the canyon. The building is divided into three main areas: galleries (listed from the bottom upelectrical, piping, operating, and crane), canyon deck, and the cells. All interior and exterior walls, roof and floor slabs, and cell dividing walls are constructed of thick concrete to provide shielding from ionizing radiation.

To estimate the volume, weight, and type of solid waste present in the B Plant, the following conditions were assumed (5):

The original equipment used for bismuth-phosphate separations accounted for the bulk of the equipment described in the B Plant Safety Analysis Report (7). This allowed the use of the Hanford Engineering Works Technical Manual (8) to corroborate information and supply missing information.

All process chemicals from separations activities were removed from the plant, either by flushing or by use of the plant for the most recent cesium-strontium extraction process (the neutralized current acid waste campaign).

The B Plant cells, which either performed the same general function or contained the same equipment as the PUREX facility, would produce the same D&D waste volumes. To estimate these volumes, B Plant cells were scaled to PUREX according to the volume of each processing cell and volume of equipment that cell contained.

For example, cell 22 in B Plant was used for vessel ventilation and contains condensers, an ammonia scrubber, filters, heaters and a tank. PUREX cell F is most similar based primarily on process knowledge and supported by equipment similarity. It was used for waste treatment and process ventilation and contains a nitric acid absorber, condensers, pumps, tanks, samplers, concentrators, jets and agitators (see Table I). Cell 28, in B Plant, was used for solvent extraction and contains an ion exchanger column and tanks. PUREX cell G is most similar based on process equipment. It was used for solvent treatment and contains a pulse column and generator, tanks, a decanter, a turbo-mixer, pumps, agitators, and samplers (see Table I). Table II summarizes cells 22 and 28: the use of each cell, the approximate sizes of the equipment within the cells, and the cell volumes (5).

Using the calculated ratios between PUREX cells and similar B Plant cells, an estimate of the volume and weight of equipment considered candidates for D&D was developed for each cell in the B Plant. Results for cells 22 and 28 are given as an example in Table III. The total equipment volume, considered candidates for removal during D&D, of B Plant is estimated at 1,585 m3. Its weight is estimated at 316,816 kg.

Once the volume of equipment was estimated for each cell, the type of waste expected also was estimated. This resulted in a total estimate of the volume and waste type expected from B Plant.

CONCLUSIONS

This study was designed to achieve an initial rough estimate of solid waste that may be generated during D&D from the facilities included in this study. The estimates were not only based on historical data, available documents, and process knowledge, but also a structural or process area comparison of the facilities with the results presented in the PUREX study (1). Although a more detailed study would be required to maximize characterization accuracy, the information presented in this report provides a rough estimate that will help facilitate future planning and activities associated with solid waste treatment, storage, and disposal.

The estimation methodology provides a model for estimating the volume and types of waste expected from the D&D of other similar DOE facilities. The facilities with the greatest similarities to the PUREX Plant include the H and F Canyon Facilities at the Savannah River Site, which also use a PUREX process, and the Idaho Chemical Processing Plant, which uses another solvent extraction process. Although these four facilities differ in their size, capacity, and specific chemistry (due to plant specific feed fuel constituents, cladding, and the final product produced), the methods applied to the PUREX Plant provide a substantial knowledge base for the subsequent deactivation and D&D of these facilities.

The method of volume ratio estimation can be applied to other types of facilities such as nuclear reactors or glovebox line facilities. In fact, a detailed study has been completed for the Plutonium Finishing Plant (PFP or Z Plant) at the Hanford Site (9), which may serve as a scaling standard for other glovebox line facilities.

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30-5

CONSIDERATIONS FOR TRANSITION PLANNING AND DEACTIVATION OF FACILITIES AT DEPARTMENT OF ENERGY NUCLEAR FACILITIES

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ABSTRACT

The U.S. Department of Energy (DOE) has focused internal efforts for the transition of surplus facilities, including nuclear facilities, within the DOE Office of Transition Management (EM-60). This Office has the challenge of transitioning facilities from former operations and production within Defense Programs to an inactive, safely deactivated mode awaiting receipt by the DOE Office of Environmental Restoration and Waste Management (EM-40). The responsibility for delivering nuclear facilities to EM-40 in a `D&D ready' mode (or in a condition that would allow preparation for reuse) has been assigned as a function of EM-60.
As a primary engineering, remediation, and waste management contractor to a number of DOE sites, Parsons Engineering Science, Inc. (Parsons) has provided support for the development and implementation of several site-specific transition, deactivation, and D&D programs. The information provided in this presentation is based on this experience and interactive participation.
The initial phase of facility remediation activities involves the transitioning of surplus DOE facilities from operations (within the mission of DOE Defense Programs) into the DOE EM-60 [Transition Management] organization. This organization is responsible for transition planning and management of initial activities prior to acceptance by the DOE EM-40 Environmental Restoration organization, which is responsible for oversight of decontamination and decommissioning (D&D) operations. Included in this presentation are considerations for transition planning and deactivation of nuclear facilities at the DOE site operations level. A discussion is provided on the establishment of a typical transition management program and `lessons learned' in program development.
The transitioning of D&D facilities to non-operational cleanup programs is a recent development within the DOE, and programs such as transition management and deactivation are not well defined in many cases. This programmatic overview provides some insight into concerns that many DOE sites may face.
This presentation has been prepare for the perspective of anyone involved in DOE waste management, environmental restoration, decontamination and decommissioning, transition management, and/or deactivation programs. Conclusions and suggestions are provided for future programs development. The analysis involved in this presentation indicates that a great deal of preplanning and evaluation is necessary in order to implement a cost-effective, efficient, and functional transition management program.

INTRODUCTION

The Department of Energy (DOE) has a very complex and convoluted mission as it undertakes to disassemble the unnecessary elements of the national nuclear defense program and dismantle surplus nuclear defense production facilities. The DOE nuclear weapons complex involves manufacturing, production, assembly, testing, and research and development installations that may contain a variety of hazardous, toxic and/or radiological materials. Proper planning, staging, transitioning, and management of these surplus facilities is a key element in this mission of remediation and restoration.
The facility transition and deactivation mission has been assigned to the Office of Facility Transition and Management (EM-60). EM-60 functions in an intermediate capacity, receiving facilities from the responsible production program operations office and preparing the facility for future transfer to another organization. In most instances the management responsibility for a facility is transferred to DOE EM-40, the Environmental Restoration and Waste Management organization. However, in some cases, ownership of the facility may be transferred to another organization for future use or may be retained by EM-60 until ultimate disposition of the facility is decided.
Many DOE sites have facilities that currently are in a cold shut-down (transition) mode, and are in the process of developing and implementing program procedures and protocol. The management structure to accomplish transition, deactivation, and implementation of D&D programs and policies, in many cases, is developmental or in a state of reorganization seeking optimal performance. These management programs are being developed to organize, staff, direct, and control activities necessary for managing and implementing transition, deactivation, and D&D program requirements and to assist in site-wide transition efforts. Subsequently, each site will be transformed from its previous operational mode of nuclear weapon components production to a condition of deactivation, decontamination, and environmental restoration.

Since it is not possible for all of the facilities to be transitioned simultaneously, DOE has adopted an overall strategy of deactivating facilities on a priority basis. This program involves selecting facilities for transitioning on a risk-based priority schedule. Once selected, the transition process requires that the selected facility be placed in a condition which provides safety, stability, and minimal monitoring, surveillance and maintenance requirements for an extended period while awaiting final decommissioning. This method was selected in order for DOE to utilize limited resources to accomplish the greatest net gains in facility safety and stability in the shortest time.

THE FACILITY TRANSITION PROCESS

Transition begins with the termination of operations. This phase includes surveillance and maintenance, and terminates with deactivation, the achievement of safe shutdown, and transfer to the future custodial organization. As the program has been conceptualized, the responsible environmental management program for a site is to receive `D&D-ready' facilities from the transition management program. Transition activities include deactivation planning for surplus facilities, placing the subject facility into a `D&D-ready' condition, and application to the environmental restoration program (under EM-40) for acceptance and transition of a facility for reuse or D&D operations. Specific elements of the transition program include shutdown of operations, deactivation of processes, removal of radioactive substances and Special Nuclear Materials (SNM), and placing the facility in a condition where D&D operations may be conducted.

Some of the concerns with transitioning and transfer of DOE facilities to EM-40 are:

The need for accurate preliminary characterization and hazards analysis;
Requirements for establishing an effective surveillance and maintenance (S&M) program;
Initiation of the EM-40 budget cycle (which could require up to three years); and
Achieving compliance with EM-40 acceptance criteria.

This phase ends with the execution of a memorandum of agreement (MOU) and acceptance of ownership by EM-40.

DOCUMENTATION FOR TRANSITION AND DEACTIVATION

The transition and deactivation process requires a great deal of certification, verification, and validation in order for custodial ownership to be transferred. These efforts require the preparation of documentation and reports that protect both the previous and future owners of a facility, and allow the involved parties to estimate the level of funding, risks, and schedules associated with the required deactivation project. This documentation interacts within the responsible DOE organizations at various levels.

Programmatic Documentation

The primary program elements and decisions that interact with the transition planning project phase include:

1) Generating a master list of surplus facilities;
2) Preparing a D&D Project Management Plan;
3) Developing and maintaining a Five-Year Plan; and
4) Providing project authorization (related to fiscal year budgets and the site Activity Data Sheets).

Specific documentation from these activities are:

Master List of Surplus Facilities

DOE Order 5820.2A requires that each DOE field organization prepare and maintain a complete list of contaminated facilities (both operational and excess). This list of facilities is maintained by responsible site DOE field office in order to identify and assign decommissioning responsibility to the appropriate responsible manager.

The surplus facility inventory will be used for developing a master plan for all transition and D&D subprojects (i.e., the phasing of overall D&D activities). The ultimate land use, utilization of facilities to facilitate D&D operations or other restoration programs, and the need to maintain utility services for on-going plant operations must be considered when developing the master schedule.

One component of the surplus facility list is the formal transfer of landlord responsibilities from the specific plant operations manager to transition management. As specified in DOE Order 5820.2A, Chapter V, 3.a.(5), contaminated facilities may be transferred from one program organization to other by mutual

agreement of the programs involved. This agreement typically will involve a Memorandum of Understand (MOU) between the appropriate managers. The transition management organization to which the contaminated facility is transferred is required to accept full responsibility for surveillance, standby operations, maintenance, and deactivation of the facility until a decision is made on the final facility disposition.

DOE-RFO Five-Year Plan

The Five-Year Plan represents the long-range planning action for ensuring the proper allocation of funds. This budget cycle includes establishing and maintaining a master D&D budget, and preparing Activity Data Sheets.

The inventory of surplus buildings and equipment forms the basis for assembling budgets to obtain required funding to execute the transition and D&D programs. The budget cycle is initiated with the development of the Five-Year Plan and is carried forward with the submission of annual budget requests for specific subprojects. The quantity of the information used to develop the budget estimates is a critical factor in ensuring that sufficient funds are available when required and to provide the justification for the funding. Activity Data Sheets will be used as the information resource for developing the budgets. The format and content of the Activity Data Sheets should be supplemented, if required, to ensure that program needs and goals are properly funded.

Project Authorization

Documentation may be required for submittal in support of project/subproject authorization. The DOE Project Management System (DOE Order 4700.1) provides guidance for project authorization documentation. Most deactivation and D&D projects will be managed under the specific requirements established for Major System Acquisitions (in which funding for a project is handled in a manner similar to cost/expense funding). However, the basic information required by the project management system for general project management may be applied. The D&D Manager must determine the level of detail and format appropriate on a task-by-task basis. Factors which determine which organizations or individuals must approve or authorize a transition project include, but are not limited to cost, funding, safety issues, required input from external organizations, schedule limitations, high project risks, and impending regulatory issues.

Project-Specific Documentation

Following the establishment of the program components, the development of plans for implementation of individual transition and deactivation projects can commence. This portion of the process typically includes preparation of plans and completion of activities, including: the Facility Operation History; Baseline Characterization Plan; and Facility Characterization.

The preparation of this information has been handled in a number of configurations by various DOE sites. In some instances, much of the information required for project-specific documentation is obtained by the operating organization prior to transfer. In other cases, the transition management function assumes the responsibility to characterize the facility and to determine the end state of deactivation prior to transfer. In still other instances, it has been the responsibility of the EM-40 organization to obtain the information required to prepare project documentation and to proceed with D&D operations. While the specific terminology may vary from site to site, the following documentation typically is prepared:

Facility Operation History

Information regarding the operational history of the facility will be reviewed to assess the nature and extent required of activities required for deactivation, including worker protection requirements. The goal of this effort has two components. The first goal of the review is to obtain as much information about the facility as possible early in the planning stages to allow the project scope, budget and schedule to be defined. If it is determined that additional information is required, the second goal of the review to identify the data requirements for the development of a Baseline Characterization Plan.

Baseline Characterization Plan

Once the data requirements have been identified, a Baseline Characterization Plan can be developed to obtain the missing information to support implementation of the deactivation project. Depending on the requirements for deactivation, the characterization plan may include the establishment of Data Quality Objectives to

ensure that the information obtained will be of a quality to meet future D&D project requirements. Field Sampling and Quality Assurance Project Plans may be utilized to define the sample locations and the sample collection/analytical procedures. To ensure proper protection of the field characterization team, existing Health and Safety procedures and plans will be reviewed and amended as needed to address any specific hazards associated with implementing the Baseline Characterization Plan.

Facility Characterization

Following approval of the Baseline Characterization Plan, the facility characterization activities will be initiated. The characterization activities include sample collection, laboratory analyses, data validation and data management. The characterization results will be used to select the appropriate decontamination methods, to classify the waste materials into the correct management categories, and to determine the extent of decontamination required to achieve subproject clean-up goals.

Facility characterization may involve two distinct types of documentation: the Facility Characterization Report and the Preliminary Hazards Analysis.

The Preliminary Characterization Report focuses on identifying the nature (nuclides, chemical constituents) and the general locations of the contaminants in the facility. General mapping of dose rates and airborne contamination (rad and non-rad), if present, should be included. Information on quantities of materials (if available) should be included; however, quantitative information is not a primary interest in this preliminary scope.

The Preliminary Hazards Analysis addresses, in general, the hazards present in the facility and the risks presented by those hazards. The principal use of this preliminary analysis is to help frame the S&M program so that risks to persons and the environment are at a low and acceptable level. There should be an evaluation as to whether a formal safety analysis report is required in accordance with DOE Order 5480.23, or whether the concept of a safety analysis as defined in DOE Order 5481.1B would be appropriate.

From a scheduling perspective, it is preferable to initiate the facility characterization as early as possible in the project. Delays in characterization could create detrimental effects on the project critical path planning and scheduling. The evaluation of the analytical results will be factored into the development of the deactivation management and implementation plans.

Safe Shutdown and Deactivation Report

This report addresses the manner in which the facility was taken out of active service and placed in a stable shutdown configuration in compliance with the appropriate DOE Orders. The status of the facility at the time of turnover to EM is to be described, and the measures taken to maintain the safe configuration provided for in an S&M plan. If additional safety-related actions are to be taken in the future, a schedule for such actions should be included. A safety assessment (in accordance with the residual hazards) should consider the presence of an unreviewed safety issues associated with the shutdown configuration. As appropriate to the facility, it may be necessary to modify the Technical Specifications or prepare a Limiting Conditions Document. In some cases it may be appropriate to include in this report the information required by the Preliminary Characterization Report and the Preliminary Hazards Analysis.

DEACTIVATION ACTIVITIES AND PROGRAMS

The most important physical activity of the transition process is the deactivation of the facility subject to transition. Consistent with DOE Order 5820.2A, the deactivation goals for a facility should complement subsequent D&D activities, and should include:

Elimination or reduction to acceptable levels the hazards and risks associated with facility/equipment contamination to ensure worker protection during the dismantling of surplus facilities;

Removal of materials necessary to minimize the potential for spreading contamination and to ensure proper management of wastes; and

Elimination of imminent contaminant hazards to allow the facility and/or equipment to be subject to less stringent controls for S&M or to be reused for other purposes.

The deactivation process should establish the conditions for surveillance and maintenance, and provide the basis for subsequent facility actions.

Deactivation Concepts

The deactivation strategy promulgated by the EM-60 organization has several

underlying concepts that formulate the basis of the proposed program. These include:
1. In many instances, there is little or no difference in the activities conducted at a facility during the deactivation phase and activities that might be conducted as a continuation of typical operations;
2. Facility organization and management typically will require alteration to accommodate the nature of the project-specific deactivation project;
3. The final condition, or end state, of a facility must well be defined in order to determine whether a deactivation project has met the EM-60 objectives for the project; and
4. The process of turnover of a facility to the future custodial organization is important in defining the scope and end states of deactivation activities.
The overall purpose of the deactivation period is to conduct shutdown activities that will assure public, environment, and worker safety. As the deactivation process proceeds, the facility becomes more inoperable, the hazards are reduced, and the requirements for surveillance and maintenance burden are minimized. During this period, work is conducted to put the facility in a stable, low risk condition which is economically and technically practical to maintain for an undefined and/or extended period. Activities that may be conducted during this period include removal of radiological source materials, disposal of hazardous chemicals, isolation of systems and equipment, removal of valuable surplus equipment, and others related services.
Goals for Deactivation
The goals of the deactivation process must be commensurate with the anticipated future of the facility. The options currently considered for most DOE facilities are:
Turnover to the responsible Environmental Restoration field organization for managing final decontamination and dismantlement;
Turnover to another DOE department or government agency for refurbishment and reuse; or
Granting, leasing or selling the facility to a public or private party for future use.
In some cases, it is simple for DOE to determine the future of a facility early in the transition phase. In other cases, additional planning and consideration will be needed to select among options for the future of the facility and therefore the final conditions of deactivation.
Turnover After Deactivation
After making preliminary arrangements with the organization that is to take custody of the facility in transition, it is the responsibility of the EM-60 organization to negotiate the conditions to be achieved prior to turnover. This agreement should be negotiated as early in the process as possible in order for the receiving organization to plan post-transition activities. Topics that should be addressed in this negotiation and agreement should include:
Discussion on the proposed end state of the facility, and whether this end state can be reasonably achieved;
Review of the anticipated condition of the facility at the end of transition, and the requirements to meet the proposed end state; and
Evaluation of the likely near-term and long-term surveillance an maintenance requirements for the facility after transition but prior to ultimate disposition.
This process may lead to the development of a plan for the long-term surveillance and maintenance period, based on the anticipated final transition state. The details of this planning should assist the receiving organization in mitigating potential threat to workers, the public, and the environment.
Turnover is completed when deactivation and end states are achieved, conditions for surveillance and maintenance are set, and the receiving organization takes responsibility.
INNOVATIVE TRANSITION/DEACTIVATION MANAGEMENT TECHNIQUES
The variation in which DOE sites apply the transition and deactivation processes has let to some ingenuity among the managers responsible for these processes. In order to meet schedules, comply with site regulatory requirements and milestones, and to comply with the future mission of the Department of Energy, each responsible manager has been required to identify innovative methods of program implementation. Some of these innovative management techniques include:

1. Avoiding Redundant Oversight -- Minimizing the perceived dual regulation of facilities to be subjected to D&D by remediating waste management facilities as elements of a RCRA or FFCA cleanup process rather than a D&D project;
2. Minimizing Future Work Requirements -- Working with the responsible facilities operations organizations to complete as much mitigative waste management and decontamination actions as possible during the final operating phases, reducing the amount of field activities (and therefore the amount of oversight and paperwork) required for deactivation, safe shutdown, and transition;
3. Using a Systems Engineering Approach -- Utilizing a Systems Engineering/Integration approach to surplus facility management and deactivation programs, which would implement the systems engineering process (as applicable) as promulgated by DOE Order 4700.1. Such an approach to D&D planning could integrate the elements of D&D operations specified in DOE Orders and directives (e.g., EM-40 guidance, DOE/EM-0142P, DOE Order 5820.2A, Sec. V, etc.) and guidance extracted from Nuclear Regulatory Commission guidance (e.g., 10 CFR and various NUREGS) with the system of project management established by DOE Order 4700.1. Since the program requirements for D&D at DOE facilities have not been completely developed and implemented for all field office requirements, the Systems Engineering/Integration approach inherently would involve a graded approach to the development of documentation and the application of directives.
4. Graded Approach to Safety Documentation -- Since the deactivation phase largely consists of ramping down operation of the facility subsystems and equipment, and making it increasingly inoperable, the intent is to stay within the existing bases for operating safety and environmental protection. There will be few, if any, activities during deactivation that fall outside existing analyses or safety requirements. Since most of the hazards that existed during operations will be removed, the need for detailed safety documentation after deactivation should be minimal, and can be handled under the `graded approach' promoted by DOE Order 5480.23. In many cases, it is anticipated that no safety documentation will be required other than an assessment of the remaining hazards in the facility D&D plan.
5. Revised Management Structures -- The requirements for deactivation management are not the same as for production operations; therefore, a revision to a project-based, schedule-intensive organization is more conducive to deactivation and transition efforts. The revised management organization should reflect an effort to condense the decision-making process and to allow the project to be handled with a minimum of administrative participants.
6. Stakeholder Involvement -- In order to promote a smooth transition, it is unquestionably required that the involved stakeholders (both in the transition process and in the post-transition condition) be recognized and involved in the transition as soon as possible. Early involvement of external and internal stakeholders is critical so that any concerns (primarily in the specification of the end states to be achieved) may be addressed. Early coordination and stakeholder involvement is required for efficient and effective termination of operations, transition of ownership, and deactivation of facilities to avoid costly delays and overruns in schedule and budget.
7. Utilization of Experienced Staff -- Many of the activities required to shut down operations, remove materials, isolate systems, and prepare a facility for a post-transition S&M period require the same knowledge and skills as facility/systems operation. These activities may require a special knowledge of the facility, including its design bases, procedures, equipment, and controls. Usually the existing operating staff is the best source of this knowledge; they are a valuable and unique resource that should be fully utilized in the deactivation work.
8. Contribution to the Transition Data Base -- Each transition/deactivation activity presents a unique opportunity to develop information, data, and lessons learned on the processes related to the specific activities. This information is valuable to future efforts, and should be compiled in a concise, retrievable form for future use by other DOE operations.

FUTURE CONSIDERATIONS AND LESSONS LEARNED

The analysis of transition management and deactivation programs supported by Parsons at DOE sites resulted in a number of concerns and considerations that may be applied to all such programs. In developing programs for transition from operational status and deactivation in preparation for D&D, several lessons were learned that may be of value to other sites that are anticipating the transition and deactivation of DOE

facilities. These concerns, considerations, and lessons may be summarized by the following items:

Evaluation of Future Regulatory Considerations

When planning for the transition of a facility, it is prudent to consider the type of regulatory agreements that exist at the site. The future regulatory climate may dictate how subsequent D&D activities are to be managed. It should be evaluated whether the deactivation and shutdown processes can be utilized to mitigate the amount of regulatory concern that will be applied to a facility transitioned to EM-40 for environmental restoration and waste management activities.

Future Use Considerations

There are very few private ventures (with the exception of programs such as the National Conversion Pilot Project) that are awaiting the availability of DOE facilities for commercial/industrial purposes. In order to entice reuse, government will have to [at least partially] subsidize reuse of federal facilities. Also, there is a question as to who would be responsible for maintaining infrastructure systems. Since this is a continuing investment in a terminal program, the wisdom of these investments should be questioned.

Indefinite Facility Disposition

Since future users may not be common, there may be a tendency to start transition processes with the hope of identifying an ultimate user for a facility that would seem to have some utility after use by DOE. However, without a clear mission for the facility, these indecisive plans can be very costly in terms of end state cleanup and S&M costs awaiting a final disposition. Decision up front in the transition process on the final disposition or status of the facility in question is necessary for orderly planning and conduct of activities.

Single Program Management

In some instances the progress of site-specific programs has been hampered by unclear directives and internal `turf battles' over the delineation between operations, transition and deactivation, and D&D activities. While the termination of operations should be manageable, it may be prudent to consider the coordination of all of the activities of transition, deactivation, and D&D under a single program direction. Although funding could be allocated from different DOE-HQ sources, the coordination of deactivation, transition, and D&D under one organization would avoid competing goals and objectives among site technical groups.

Defined Funding Sources

Programs have failed or have been limited in progress due to lack of committed funding for transition and deactivation committees. In order to avoid unnecessary delays and false starts, sources of funding must be identified and committed for all of the activities required for transition and deactivation. If committed funding cannot be defined, initiation of activities (regardless of apparent priority) may not be justified.

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30-6

CLEANUP OF A DOE NONREACTOR NUCLEAR FACILITY: EXPERIENCE AT THE LOS ALAMOS NATIONAL LABORATORY HIGH PRESSURE TRITIUM LABORATORY

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ABSTRACT
On October 25, 1990, Los Alamos National Laboratory (LANL) ceased programmatic operations at the High Pressure Tritium Laboratory (HPTL). Since that time, LANL has been preparing the facility for transfer into the Department of Energy's (Doe's) Decontamination and Decommissioning Program. LANL staff now has considerable operational experience with the cleanup of a 40-year-old facility used exclusively to conduct experiments in the use of tritium, the radioactive isotope of hydrogen. Tritium and its compounds have permeated the HPTL structure and equipment, have affected operations and procedures, and now dominate efforts at cleanup and disposal. At the time of shutdown, the HPTL still had a tritium inventory of over 100 grams in a variety of forms and containers.

INTRODUCTION
On October 25, 1990, Los Alamos National Laboratory (LANL) ceased programmatic operations at the High Pressure Tritium Laboratory (HPTL), an outdated experimental facility whose design does not meet the requirements of the modern Department of Energy (DOE) complex. The tritium operations formerly conducted at the HPTL will be conducted at the Weapons Engineering Tritium Facility (WETF) at TA-16-205. WETF enables LANL researchers to handle tritium using modern equipment and up-to-date procedures. The Engineering Sciences and Applications Division (ESA) has no further use for the HPTL and plans to remove the accountable tritium and make the facility available for decontamination and decommissioning (D&D), thereby eliminating the risk and cost of operating this now-surplus facility. Early, close communication with DOE led to an approach different from the one originally planned. The revised approach resulted in a reasonable cleanup of the facility and satisfied more stakeholders.

APPROACH
Statement of the Problem
The cleanup process must ensure that the HPTL
  poses a minimal risk to the environment and the safety and health of workers and the public,
  requires a minimum of support and maintenance, and
  is acceptable to DOE Environmental Restoration and Waste Management (EM) for D&D.
Safe shutdown required close attention to a number of critical factors.
  The HPTL is an old, tritium-contaminated facility housing outdated tritium process systems.
  The HPTL contained a large (>100 g) tritium inventory stored in unusual and uncertified containers
  Tritium contaminated equipment was in place for operation or stored in drums.
 A number of other hazardous materials were present at the facility.
  The HPTL had several potentially contaminated outfalls.
  Management consisted of informal oversight and inadequate staffing.
  The HPTL faced serious issues in regulatory and order compliance.
  The HPTL had inadequate documentation and operational procedures.
  The maintenance and surveillance were inadequate
Description
The 7500-square-foot HPTL had potential tritium contamination everywhere. In particular, the tritium-contaminated process systems that had evolved during its 35 years of operation were still in place and in use at shutdown. These systems were a high-pressure fill system, a low-pressure gas transfer system, and an experimental system. The systems consisted of pumps, valves, tubing, cryotraps, cylinders, instrumentation, and so forth.
The HPTL's tritium inventory of over 100 grams resulted in its designation as a Category II Nonreactor Nuclear Facility - the highest possible hazard category for tritium facilities. The inventory was approximately half tritium gas and half tritiated water on molecular sieve. This tritium inventory was in custom-made molecular sieve towers, experimental apparatus (such as a fish float), and uncertified shipping containers consisting of LP-50 and LP-12 gas containers and an AL-M1 tritiated water shipping container. Other tritium-contaminated equipment also remained at the site, including 36 drums of highly contaminated equipment ready for shipment to the waste area pending resolution of a mixed-waste issue. Other hazardous materials at the site included lead, mercury, caustics, flammables, and Class C explosives. The site had four potentially contaminated outfalls: a septic system, a National Pollutant Discharge Elimination System (NPDES) outfall, an acid

sump, and a roof drain. The facility had performed a tremendous amount of work over the years, but as the WETF approached startup, the HPTL was no longer viewed as an important experimental resource and received little attention from management and minimal staffing. At shutdown, the HPTL staff consisted of a full-time staff member operator, a part-time operator technician, a full-time radiation control technician (RCT), and a part-time custodian. This staff was inadequate to operate the facility in accordance with DOE-required formality of operations. The facility design antedated the new DOE for the weapons complex and was used to meet temporary demands until WETF became operational, at which time plans for final D&D of HPTL could be implemented.

Proposed Solution

LANL's prioritized approach for cleanup of the HPTL consisted of removing the tritium inventory, the highly contaminated process systems, the hazardous and mixed wastes, and other contaminated items. We also proposed to clean up the facility using the existing systems and documentation. Our plan included operation of the old process system and reprocessing and recovery of as much of the tritiated water as possible. Stressing technological issues, the original proposal comprised the following major elements.

Consolidation and transfer of tritium gas to the WETF for repackaging and measurement

Recovery of tritiated water and shipment of the tritiated water to Mound for recovery

Removal of contaminated equipment

Disassembly and removal of the contaminated process system

Removal of waste

Compliance with the National Environmental Policy Act (NEPA)

Preparation of updated Technical Safety Requirements (TSRs)

Improvement in facility management

At the time the HPTL was shutdown, EM-60 had not been organized, so HPTL cleanup began with DOE Defense Programs (DP) Research, Development, and Testing (RD&T) funding. Defense Programs through DP652 has been the funding source to date.

RESULTS

To address regulatory issues,

An environmental assessment (EA) was written and then included in the site-wide environmental impact statement (EIS).

Two categorical exclusions from NEPA were obtained.

Stack monitoring was brought into compliance.

National Emission Standards for Hazardous Air Pollutants (NESHAP) were Addressed.

Hazardous materials were stored correctly.

Most of the drains were plugged; only two outfalls remain.

To address operations issues,

A Management Control Plan, embodying a graded approach, was developed and implemented.

A project leader and full-time building manager were assigned to the facility.

Maintenance control increased and maintenance improved.

A general housecleaning was accomplished and maintained, and custodial work was minimized.

A modern liquid scintillation analyzer was installed on site.

Radiological postings and surveys were improved.

A bubble suit system was made available.

To address documentation issues,

A draft graded Safety Analysis Report was written but deferred.

Safe Shutdown Procedures were written.

A Building Emergency Plan was prepared.

Entry procedures were established.

A hazard communication (HAZCOM) program was established.

A Waste Management Plan was written.

Routine Monitoring Instructions were updated.

A draft Maintenance Implementation Plan was written but deferred.

Work was done under Standard Operating Procedures (SOPs), Operational Instructions (OIs), Special Work Permits (SWPs), and Radiation Work Permits (RWPs).

Five process system schematics were developed.

To address technical issues,

More than 70 grams of tritium were removed from the HPTL.
Two thirds of the containers of accountable tritium have been removed.
Thirty-six drums of previously packaged tritium contaminated hardware were removed.
Approximately 50 drums of other contaminated items were removed.

DISCUSSION

Early inspections and other interactions with DOE indicated that LANL would be expected to operate the facility in as close to the new (and evolving) DOE philosophy as possible. Deviations were at the sole discretion of the DOE. To accommodate the DOE needs, we inverted our original approach of addressing technical problems first and procedural problems last. Another critical factor in the HPTL cleanup was the end of the Cold War which changed LANL's operating environment in a number of ways: funding was reduced, the need for tritium was diminished, and tritium-supporting elements of the DOE complex were becoming unavailable. We, therefore, to changed and constantly revise our plan to follow evolving DOE guidance.

Regulatory Issues

NEPA was addressed with two categorical exclusions and an EA. The two categorical exclusions were for interim storage of the molecular sieve towers and for removal and interim storage of the process systems. The draft EA included both transition and D&D and was approved by the Albuquerque Operations Office. DOE Headquarters was reviewing the EA when the decision was reached to prepare a site-wide EIS for Los Alamos. In the course of review and as a result of the categorical exclusions, the EA for the HPTL had been reduced to only the D&D work. Continuing ongoing operations and the two categorical exclusions covered the transition work. The D&D work will be covered in the LANL site-wide EIS.

Stack monitoring was brought into compliance by upgrading the stack tritium monitoring system almost completely. This activity was part of a Laboratorywide compliance agreement with the EPA. The HPTL now has a state-of-the-art stack monitoring system with a bubbler and two real-time monitors. A NESHAP interpretation of stack emissions was obtained. We did not have to perform a preconstruction review because we planned to keep our tritium emissions well below historical levels. In fact, during cleanup such emissions have been about 5% of historical levels. There are no nonradioactive hazardous emissions.

Hazardous materials were dealt with correctly. We identified the hazardous materials at the facility and categorized them as either in-use or waste. The in-use materials were stored according to the type of hazard they posed(such as flammable). The waste was considered mixed because of the potential tritium contamination. It was stored according to its hazard potential in satellite accumulation areas and when possible, shipped to the Laboratory's mixed waste storage area.

As a result of a survey of the HPTL outfalls, we plugged most of the drains and obtained an agreement with the New Mexico Environmental Division to operate those we still needed. We retained a septic system, an NPDES outfall, and a roof drain. The NPDES outfall will soon be plugged because we plan to discontinue its use.

Operations Issues

The Laboratory operates the HPTL under a Management Control Plan that was negotiated with DOE. This plan addresses the various sections of DOE Order 5480.19. In developing this plan, graded approach was used whenever possible. A project leader, a building manager, and an RCT carry out the increased formality of operations. The project leader guided the project, developed much of the documentation, and served as liaison with DOE and other outside organizations. The building manager oversaw the day-to-day operation of the facility. Maintenance and its control improved as a result of ESA-TSE and Labwide initiatives. A general housecleaning was accomplished and has been maintained. Waste minimization procedures were established. The custodial work was minimized by evaluating the effectiveness of mopping on reducing floor contamination. The RCT upgraded the on-site liquid scintillation analyzer to improve posting and surveys. The existing contaminated supplied air (bubble) suit system was replaced with a portable system. Specific operations were staffed on a case-by-case basis using the previous system operators (now retired) and personnel from other Laboratory tritium facilities, such as the WETF.

Documentation

A graded approach was followed to documentation. Operation basis, routine operations, and specific tasks were addressed in that order.

A draft graded Safety Analysis Report (SAR) was written. The DOE Albuquerque Operations Office evaluated the status of the facility and deferred the complete approval of the SAR and associated Technical Safety Requirements. We are operating the facility according to the Operational Safety Requirements written before the shutdown.
Our Safe-Shutdown Procedures were approved by DOE/Defense Programs. These conformed to the procedure format that was uniform across Los Alamos tritium facilities.
The Building Emergency Plan was upgraded to meet modern standards and to reflect improvements and other changes in the facility. Entry procedures were written for normal and non-normal conditions. Site-specific HAZCOM was established. A Waste Management Plan and Routine Monitoring Instructions were written to supplement Los Alamos policies and administrative requirements. The Maintenance Implementation Plan was deferred by DOE based on the cleanup status of the facility.
Non routine tasks were performed under SOPs, OIs, SWPs, and RWPs. SOPs were used for operation of the process system, maintenance of the process system, supplied air (bubble) suit operations, and so forth. OIs were used for decontamination and packaging of nonstandard tritium containers and similar activities. SWPs were used for decontamination and preparation for shipment of containers and equipment. RWPs were used for the remainder of the operations involving tritium. Process system schematics using industrywide notation were drawn to document the system and guide its disassembly.
Technical Problems
The major technical emphasis in the HPTL cleanup was the removal of the tritium inventory. We started with over 100 grams in 50 containers. We have removed over 70 grams of accountable tritium in two thirds of the containers from the HPTL. Approximately half of the tritium was in the form of gas and half in the form of tritiated water adsorbed on molecular sieve.
The gas was contained in uncertified shipping containers (LP-12s and LP-50s) and experimental vessels. The shipping containers were packaged using an SOP as if for normal shipment but were then shipped over a closed road to WETF. The experimental vessels were secondarily contained and also sent over a closed road to WETF. This gas will be consolidated and sent to Savannah River for reuse.
The tritiated water on molecular sieve was in one uncertified shipping container (an AL-M1), in traps and cryopumps, and in several custom containers called towers. The molecular sieve towers were previously used to trap the effluent from the process system. The amounts of tritium in these molecular sieve towers were determined by plotting the pressure rise over time caused by helium-3 generated by tritium decay. The amount was confirmed by a mass spectrometer analysis of the gas in each tower at a given time. The other items will be put in a calorimeter.
Contaminated hardware was sent to the waste disposal facility at Los Alamos. Thirty-six drums of previously packaged tritium-contaminated hardware were removed. These drums were radiographed using Los Alamos Accident Response Group (ARG)/Nuclear Emergency Search Team (NEST) video radiography equipment to determine that no lead was in the drums. Other contaminated hardware was packaged as required by Los Alamos policies and administrative requirements and shipped to the disposal area.
RECOMMENDATION
In the process of transition, proceed with formality first and address technical issues second. In particular, communicate with DOE early, before safe shutdown, to establish the level of formality required under a graded approach. Establish safe shutdown procedures early. Plan ahead and document. Doing so establishes credibility and gives stakeholders confidence in the cleanup effort. Such a process leads to a safe, expeditious transition.

Session 31 -- Business Opportunities and Barriers
Co-chairs: Chuck Little, Fluor Daniel;
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31-1
NUCLEAR RESEARCH AND TECHNOLOGY TRANSFER TROUGH SPIN-OFF COMPANIES
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ABSTRACT
The paper reviews briefly the history of nuclear research centers from technology transfer point of view with particular emphasis to spin-off companies as agents of

transfer. The role and nature of firms spun off from nuclear research centers are examined through a case example from Finland involving three small high -tech companies. In referring to certain traditions and characteristics of nuclear technology and to the referred cases the author concludes that still a remarkable potential for entrepreneurial spin-offs relies in nuclear research. The paper also recognizes that the development of nuclear technology has reached the stage of maturity and therefore the domain of new firms to spin-off would typically be in the field of sophisticated services; such as consultancy, expert-, and educational services.

INTRODUCTION

Society while supporting the research institutions, such as universities and other research establishments, wants to have not only return to the investments but to utilize the accumulated knowledge on a wide basis and more efficiently today than before. The importance of technology as a central factor in creating competitive advantage both at the level of national economy as well as of an individual firm has commonly been recognized. Along the increasing importance of technology its development has become more and more resource intensive. Consequently the firms must rely also in increasing extent on external sources to build up the needed technology competence. This altogether has made transfer of technology a very up to date issue. Spinn- off companies constitute a demonstrative and direct input from research institutions to the economic environment; and typically in a sector considered as particular desirable i.e. small high-tech firms. As a technology transfer agent a spin-off company is potentially efficient, although the main channels in transferring the technology rely elsewhere.

Nuclear research is an interesting example due to several reasons. It is an approach making exercise as a large scale interdisciplinary concerted action to develop science into technology. Further nuclear research has been a forerunner in internationalization of scientific research and later on technology development. Nuclear research and technology have been objects of national investments and inputs of exceptional scale. The countries pioneering in the nuclear research and technology had military applications as primary objectives. In the countries following the early ones, the research objects consisted of peaceful civilian applications. In those countries the large investments were supported much by the argumentation that nuclear research and technology would promote the overall economic and technological development in the country. Besides the potential prosperity brought by the nuclear power the nuclear research establishments were expected to serve as a kind of central source for technology to be transferred to many fields of science, industry and business. Spin-off companies provide a relevant view to the above argumentation.

This paper reviews briefly the history of nuclear research centers from technology transfer point of view with specific emphasis to spin-off companies. In spite of its specific character the case of the Finnish nuclear research and the referred institute it reveals certain aspects of general interest related to the efficiency and practical possibilities of international technology transfer. The referred Finnish spin-off cases are up to date representative examples of small technology based firms spun off from a research institute.

THE IDEA OF TECHNOLOGY TRANSFER

According to Autio a variety of definitions for technology transfer has been presented since 1970"s. He suggests that the majority of them are to narrow in scope. They refer, for example solely to licensing or to transferring technology between developed and developing countries. Autio himself ends to a definition which views technology transfer largely as a social process (Autio, 1993).

Ayres presented already in 1969 a definition for technology transfer, which is very applicable from practical point of view and broad in its scope so that it can well accommodate also the social dimension in the case an explicit emphasis of that nature is needed. Ayres has defined the technology transfer as follows:"Transfer of technology is the application of a technology in a field outside the one for which it was developed or to which it was first applied (as contrasted with diffusion or penetration)'". The technology diffusion according to Ayres is:"'Diffusion of a new technology is the evolutionary process of replacement of an old technology by a newer one for solving similar problems or accomplishing similar objectives"'. These can be handled as consequent phenomena. Technology transfer brings a new technology to the system and diffusion spreads the technology within the system

(Ayres, 1969). Reflecting the terminology transfer refers to an active goal oriented action while diffusion refers rather to a phenomenon which proceeds on its own. According to the above definition technology can be transferred from one system to another both in lateral and vertical direction. The lateral dimension represents the more common and conventional ideas. It covers the transfer of technology e.g. between the countries, between the industries or between the application fields. Lowell Steele in his textbook recognizes the vertical dimension as essential(Steele, 1988) It includes the transform of scientific results first into generic technologies and further to specific applications i.e. processes and products; and still further to different social systems.

Weinberg in his famous book "Refelections on Big Science"" refers mainly to the vertical dimension as well (Weinberg, 1966). He explicitly interpretates the development of science into technology as technology transfer including even steps from generic technology into specific applications. However, Weinberg does not give any definition for the technology transfer. The recognition of the vertical dimension in technology transfer justifies the statement that nuclear research centers represent the first large scale interdisciplinary attempt ever made to achieve a breakthrough from science into technology.

CHANNELS FOR TECHNOLOGY TRANSFER

In this context the idea of transfer channel is viewed broadly covering a wide range of different means and media for transferring of technology.

Depending on the application and actual situation transfer channels can be identified, built up and utilized accordingly. For example, Robinson has identified altogether 16 different means," channels" to transfer "the technology packages" (Robinson, 1988). That model is suitable for examination of technology transfer between firms.

For the purpose of this study referring mainly transfer between research institutions and industry the following more conventional and generic classification of channels is appropriate (Hull, 1990).

1. Contract research and development
2. Consulting and other expert services
3. Training and continuing education (including provision of professionals)
4. Spin-off companies, licensing', "research exploitation"

The experience has proven that people are the most efficient medium for technology transfer. This underlines the importance of category 3. This category includes also the professionals of the research centers recruited by the industry which is traditionally an important and efficient vehicle for technology transfer. Category 4. covers the spin-off companies which inherently includes technology transfer through people as well. Further spin-off companies demonstrate a visible input of a research institution to the economic environment. According to Hull research exploitation is not client oriented. It typically involves advancement of a technology to the point that a licensing deal, or a corresponding arrangement with the industry can be negotiated. European institutions are not involved in research exploitation in the same extent as the American universities.

NUCLEAR RESEARCH AND TECHNOLOGY TRANSFER

The history of nuclear research centers have distinguished different phases. Along this development much experience has been gained which have contributed in various ways to the development of science in to technology. The development of nuclear energy for military and peaceful applications began in the USA, and the history of nuclear laboratories started there during the second world war.

The joint efforts of nuclear physicists, chemists, material scientists and engineers from various fields, supported by large infrastructure of facilities, were expected to achieve a brake through in the nuclear field first for military purposes, and then afterwards for peaceful use. In order to achieve the said breakthrough new types of laboratories were established, which Weinberg called mission oriented laboratories. The term is well justified since the laboratories were set up to carry out a mission defined and given by the government such as reinforcement of national defense and later on production of cheap energy. The mission originated outside science, science was a tool for realizing of nonscientific, politically defined goals. Whilst traditional basic university research is disciplinary oriented and science driven.

The UK, the USSR and France followed the USA in establishing nuclear laboratories. A few years later several other industrialized countries followed, even some

developing countries e.g .; China, Yugoslavia and India.The main motives in investing nuclear research were the military application and keeping abreast of technological development. In those times nuclear technology was seen as definite high technology. Further it was considered to be a generic core for a vast variety of applications in different fields e.g.; energy, medicine, agriculture an industry. From the mission point of view the initial period was successful. The military applications succeeded in all countries, which started early and which had enough political and military power as combined with adequate resources. The large scale peaceful application via. nuclear power as energy source succeeded as well. First demonstration power reactors were constructed already in fifties and in early sixties the large commercial breakthrough took place. After the commercialization of the nuclear power the nuclear industry took the major role from research centers, signifying that part of the mission was successfully completed (Weinberg, 1966). Filling the mission was also a demonstration of transfer of science into technology on a multi disciplinary way and in a scale which was not seen before. further the centers played a key role in introducing nuclear technology into other fields of application e.g. medicine, agriculture etc. as well as in the very successful internationalization of nuclear science and technology.

Filling the main mission was the beginning of the decline of the nuclear centers first in their status and little by little also as regard to the resources allocated for their use. It is true that the decline with reference to the resources was slow and new centers were still established through the sixties even in seventies in developing countries. After the industry assumed the leading role in commercialized nuclear power, the research centers were left in a supportive position. They executed research on the basis of governmental funding and industrial contracts. They never got any new mission to replace the introduction of nuclear power but a variety of different kind of smaller tasks mainly related to nuclear safety and waste management (Heinonen, 1987).

After loosing the mission the second constrain the centers had to cope was the world wide reduction of nuclear power programs. The centers enjoyed in their early times exceptional resource allocations. As result of the changes in the environment the centers tried to compensate the losses in their original role and domain in diversifying into different fields, mostly related to energy and environment. Education was included as an essential part of the original mission. Most of the centers were closely connected to the universities. Power companies and safety authorities recruited their expert staff largely among those trained in research centers. As proven by experience an efficient technology transfer can be realized in transferring qualified people (Steele, 1988).

THE FINNISH CASES

Finland was a late starter in experimental nuclear research and she never established a nuclear research center in an ordinary sense. The Helsinki University of Technology received a zero power reactor in late 50"s and a Triga Mark II Research Reactor became critical in 1962. At this time an independent research laboratory was established, the Reactor Laboratory. It was funded directly by the government but administered by the Helsinki University of Technology. The main purpose of the unit was to educate specialists for the Finnish nuclear power program. Applied research first for educational and scientific reasons was performed which soon extended from nuclear physics to various multidiciplinary activities to support science, technology, medicine and industry in Finland. These activities were supposed to be funded by research contracts, although this was realized to larger extend only in late 70's and early 80"s.

In 1971 the Reactor Laboratory was incorporated into the Technical Research Center of Finland (VTT). a multi disciplinary research center employing nearly 3000 people at its maximum, in 1990. In the early 80's the Reactor laboratory reached its peak in activities employing some 100 people. Government funding was decreasing being as low as 26% in 1986. Active marketing and the needs of nuclear industry gave a secure outside funding. The closer dependence on the industry and business environment showed that a governmental institute is not very efficient organization to sell services. In late 1980's VTT assumed a policy to encourage entrepreneurship and creation of spinn-offs. Consequently some research groups left the Reactor Laboratory to continue their activities as independent private business enterprises. A part of radioactive tracer group established a firm in 1986 and started to compete with the Reactor Laboratory. The new company was successful it overtook the market

and managed to increase the volume of business significantly, while the Reactor Laboratory gave up almost all it's activities in this field. The budgeted turnover of the company, Indmeas Ky, for 1994 is 3,5 mill. FIM and it employs 8 people. Some 80% of the business is export.
The next group to leave was the one doing environmental modeling in lake and sea ecosystems in 1990. Also that business ceased at the Reactor Laboratory, while the new company increased the volume of the business. The turnover for 1994 will be some 3 mill. FIM and the staff consists of 8 professionals. There is a strongly increasing demand in the market also from abroad. The company, SYVA Oy, has intentionally chosen the strategy of slow growth.
The third sector to leave in a form of a spin-off company in 1991, was the production of radioactive pharmaceuticals. It was an activity which over the years since late 60's had required substantial investments. The total return has been less than the inputs. The new company has started off very well and further success would pay back well the investments to the society. The turnover in 1993 was 6,3 mill. FIM and the personnel 16. The company MAP-Medical Technology Ltd., is the market leader in Finland and the share of the growing export is at present some 30% (Ahola, 1994; Heinonen, 1995)

DISCUSSION

In a world wide perspective the nuclear research centers never developed into the kind of centers of excellence as originally anticipated influencing significantly the overall technological and economic development of the respective countries. As a rule the number of spin-off companies originating from nuclear research establishments is relatively small. However, nuclear science and technology as well as the research centers together demonstrated technology transfer in a scale and efficiency, which was not seen before. This refers both the vertical and lateral dimensions: development of science into technology with large scale commercial applications; and dissemination of technology globally and in different application fields.
In spite of its specificity the Finnish case demonstrates certain features which are worldwidely characteristic to nuclear research. The Reactor Laboratory used to be the most central institution for experimental nuclear research in Finland. It produced during 30 years of operation only three spinn-off companies. The small number is somewhat related to the cultural environment. The total number of spin-offs from VTT was 52 by 1992. most of them were created only in late 80's and early 90's. Thus the Reactor Laboratory is even above the average. Their size, technology orientation and internationalization is clearly above the average among the VTT based spinn-offs. Most important is the fact that the business of all the three firms is prospering, in spite of the very severe recession in the country.
The Reactor Laboratory contributed to technology transfer through three distinquished phases as follows.

1. While building up the capacity on nuclear research, education and expertise advantage of the newly established international collaboration was taken i.e. technology transfer from abroad.
2. Besides education and expertise for the Finnish nuclear power program the mission of the Reactor Laboratory included to study and develop nuclear based technologies applicaple in different fields of science and industry, i.e. the vertical technology transfer.
3. This phase represents the transfer of the skills and know-how from the research institution to the environment. It covered primarily education of experts and research services so that the spin-off companies represent only a small share of the whole. All the three companies are based on the the technology developed at the Reactor Laboratory up to the proven commercial phase. MAP is the only firm, which officially bought the rights to the technology and products. Indmeas and MAP have developed further their technologies including new commercial applications. The active international operations include the technology transfer component as well - this time from Finland to abroad.

Reactor Laboratory has been a contributor to international technology transfer already since 1960's via collaboration programs. As already mentioned the most efficient medium for technology transfer consists of people. The staff members of the Reactor Laboratory have worked as visiting scientists and experts in many countries and organizations including developing countries and United Nations"

specialized Agencies. The laboratory has organized training courses and accomodated trainees and visiting foreign scientists. Thus the international nature of nuclear technology has constantly reflected to the activities of the Reactor Laboratory. That is why the international operations have been natural and significant in the business of all the three firms as well.

These three firms are representative examples on the idea of spin-off companies as agents of technology transfer and qualified high-tech contributors to the business environment.

Further the Finnish case as a whole is an example of an efficient utilization of technology transfer. The basic approach was to apply thin organizations and exploit existing resources at universities, research institutions and industry. Advantage was taken of the newly established international nuclear collaboration (Heinonen, 1987).. The assmilation of the transfered technology into the recipient"s technology environment i.e. the degree of completeness of transferered technology is an essential dimension to be examined. A convincing indicator on completness of technology assimilation is when the recipient is able significatly to develope the technology further and act as transfering party itself. In this regard the Finnish case is a representative - but in no means a unique, example.

THE POTENTIAL OF SPIN-OFF COMPANIES

The nuclear technology with all the established major application fields e.g. nuclear power, nuclear medicine, industrial applications, agriculture and different research tools has reached a certain stage of maturity in its development. Therefore new inventions and innovations are not so frequent any more. It further means that the overall potential for spin-off firms is smaller than in the days when the technology was developing rapidly and was spreading to new fields of application. In particular, the manufacturing product related spin-off companies are not typical in the mature stage of technology. The Finnish cases do not have the statistical justification. Nevertheless, two of the three firms represented services. On line with the present stage of development the sophisticated services would be the most natural and suitable domain for the firms to spin-off from nuclear research institutions. There are a vast amount of experience and examples of firms offering different kind of services in the nuclear field including waste management. Management of radioactive wastes is still a very potential area, in particular, in the global market as we have realized in Europe, even in Finland. There is also the possibility to extend the services beyond radioactive wastes to other hazardous wastes.

Power plant maintenance is another area still having potential for development. Two of the writer's students established a firm a couple of years ago which does certain maintenance work in the context of annual revisons of nuclear power plants. They cannot accept all the work they are offered to in this stage.

Education and training, in particular, professional continuing education is a field, where nuclear experts could play a more active entrepreneurial role. Closely related to that is the further development of consultancy business, where certain traditional strengths of nuclear field could be exploited; such as; national and international networking, internationalization, multidisplinary approach, proximity to basic and fundamental scientific research even theoretical developments.

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CASE STUDY: COMMERCIALIZATION OF THE INNOVATIVE IN SITU VITRIFICATION TECHNOLOGY

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ABSTRACT

Geosafe Corporation is a development stage company that is commercializing the innovative In Situ Vitrification (ISV) technology. The ISV technology was invented and initially developed by Battelle, Pacific Northwest Laboratory (PNL) for the U.S. Department of Energy (DOE). The technology involves the in situ electric melting of earthen materials for purposes of site remediation and/or waste treatment. The technology holds significant promise for highly effective treatment of sites contaminated with hazardous (organic and inorganic), radioactive, and mixed wastes, including sites with buried waste and debris.

Geosafe holds exclusive license rights to the commercial application of the DOE-patented technology. Geosafe initiated the commercialization process in 1988. The commercialization process has been challenging as various barriers and difficulties associated with bringing the new technology to the commercial marketplace. This case study reviews the chronological development of two concurrent and closely related pathways of events: 1) those associated with the ISV technology development, and 2) commercialization efforts. Positive and negative influences on both pathways are reviewed and a lessons learned conclusion is presented.

INTRODUCTION

The purpose of this paper is to present the major barriers experienced by Geosafe during its ongoing efforts to commercialize the ISV technology. Whereas this paper is one of several in a session on "Business Opportunities and Barriers", it is not the purpose of this paper to identify all the types of barriers that an innovative technology undergoing commercialization may encounter; rather this paper presents Geosafe's experience as a case study that may or may not be typical of other technologies and commercializing companies.

The following summary description of the ISV technology is presented to help the reader relate to the type of technology being commercialized. The ISV technology has been recognized by environmental technologists and regulators as one possessing strong potential for significant contribution to the national problem of hazardous, radioactive, and mixed waste site remediation. The technology involves the in situ electric melting of earthen materials for several possible purposes, including: 1) thermal destruction/ removal of hazardous organic and inorganic compounds, 2) permanent immobilization of hazardous and radioactive heavy metals within a high integrity vitrified product form, 3) conversion of waste materials into recycled products, and 4) production of high integrity vitrified masses for various civil engineering applications.

The ISV technology is a variation of the joule-heated glass melting technology that was adapted to waste management applications by PNL starting in the early 1970s. The technology differs from standard melter technology in that the earthen media itself serves as the containment for the melt. In ISV, an array of electrodes (usually 4) is placed a few inches into the media to be treated, and a graphite and glass frit starter path material is placed between the electrodes. The starter path carries the initial electrical current until the adjacent media is melted. Once melted, the earthen media (e.g., natural soil) becomes electrically conductive and becomes the process heating element. Joule heating occurs within the molten mass. Heat is conducted outward into adjacent unmelted media, causing it to melt, thereby enlarging the size of the melt. This process continues until the desired melt size and shape are attained (to 1,200 tons maximum).

The process is operated with a melt temperature in the range of 1,600 to 2,000 degrees Celsius for most silica-based media (e.g., soil, sludge, sediment, mine tailings). The high temperature of the melt causes the pyrolytic destruction and

vapor-phase removal of organic contaminants present in the treatment volume. Most metal oxides become part of the melt and are incorporated within the glass and crystalline structure that results upon cooling. The process results in a large volume reduction due to the elimination of void volume within the particulate media (e.g., 20 to 50% for most soils). The vitrified product typically possesses outstanding physical, weathering, leaching, and biotoxicity properties. The ISV process and its attendant vitrified product may be the most technically preferred and cost effective technology alternative in cases where: 1) a combination of organic and heavy metal, or hazardous and radioactive, contaminants must be treated, 2) onsite and in situ treatment is highly desired, 3) the site contains non-homogeneous soil conditions and/or debris, 4) maximum treatment effectiveness and permanence are required, and/or 5) high volume reduction is desired.

CASE STUDY CHRONOLOGIES

The chronology of the ISV technology development and commercialization efforts can best be understood in relation to two key timelines: technology development and commercialization. The timelines are presented separately below; however, it should be noted that they are closely related and somewhat interdependent. It should also be noted that two separate technology development efforts, DOE's and Geosafe's, have been pursued since Geosafe's establishment. The DOE program is mentioned in the discussion below since, although separate from Geosafe's efforts, it had major influence of the perception of ISV in the marketplace.

Technology Development Chronology

Figure 1 presents the ISV technology development timeline. This figure identifies key milestones related to the technical development, test and demonstration, and application of the technology. The timeline starts with the invention and proof-of-principle testing in 1980. The following two years focused on initial process exploration and small-scale test and development work, all of which continued to indicate the promise of the technology. There was a strong desire to quickly scale the technology up; thus starting in 1983, the program focus moved away from process and technology development to a pilot-scale (e.g., 10-ton melts) equipment design and demonstration focus. The scope of DOE application interest at that time was limited to TRU-contaminated soil. Pilot-scale equipment was designed and fabricated, and a major milestone was achieved in 1983 by the successful completion of a demonstration test on TRU-contaminated soil.

This success resulted in rapid movement of the program toward design, fabrication, and demonstration of a large-scale ISV equipment system capable of treating 100 ton/day of contaminated soil, and designed to be trailer mounted for over-the-road mobility. Some equipment development work was also performed in the areas of electrode design (for extended life) and electronic depth monitoring equipment (for determining the depth of melt in real-time). The multi-million dollar equipment system was completed in 1984, an amazingly short four years after inception of the technology concept. Four large-scale Operational Acceptance Tests (OATs) were performed in uncontaminated soil during the following 1-1/2 years. The equipment was then applied in 1986 to the Large-Scale Radioactive Test (LSRT) which treated a portion of a TRU-contaminated drain field within Hanford's 200 Area. This test was largely successful; however, some equipment difficulties were encountered, and the depth attained was less than desired. It was learned that the depth had been limited by the presence of a cobble rock layer that served as a thermal barrier to melt growth. This finding was very important to the future development of ISV. The large-scale equipment then sat idle for about four years.

During the meantime, the promise of the technology became infectious and several development and test programs were established within DOE for various types of applications, including: 1) liquid seepage disposal basins at ORNL (started in 1985), 2) buried waste at INEL (started in 1987), and 3) underground tank remediation at Hanford, ORNL, and other DOE sites (started in 1989). In addition, in 1984, PNL started performing small scale tests and demonstrations on many types of hazardous chemical wastes for non-DOE clients. The technology was found to be applicable for the effective treatment of a broad range of earthen media and contaminant types. In 1989 Geosafe procured and initiated testing of a large-scale ISV machine for its commercial use.

In 1991, the technology was applied to another large-scale demonstration at Hanford involving a liquid disposal crib (wooden timber type). While quite successful in most areas, this demonstration too, like the LSRT, came up short in the area of

depth attainment. These two large-scale tests had identified the inability to minimize melt width as needed to allow attainment of desired melt depth. Many ideas for resolving the problem were identified by PNL staff; however, they were unable to obtain the funding needed to resolve this problem. This was the apparent nature of the DOE program at that time (i.e., demonstrations were readily funded, R&D was not).

Fig. 1.

Fig. 1. ISV Technology Development Milestone Timeline.

At this point, within about one year of each other, several ISV tests experienced notable difficulties. First, during pilot-scale testing of ISV on buried wastes at INEL, PNL found that the flow of vapors and gases through the melt was sometimes sufficient to disrupt and splatter the melt, making process control difficult and causing damage to the equipment. Second, during very aggressive testing of its new large-scale equipment, Geosafe explored new limits of application, including buried drums containing water, and experienced a melt splattering event that severely damaged the experimental off-gas hood that was being employed. Third, during a large-scale PNL demonstration test on a 22,000 liter underground tank, a vent tube failed midway through the melt, resulting in a thermal excursion, expulsion of melt, and damage to the off-gas hood. These three events clearly revealed an area of technical difficulty that had not been a concern for simple contaminated soil applications, that being the need for process control and equipment design features capable of handling the volumes of gases that would be generated during such applications.

These testing events resulted in the severe curtailment of the ISV technology development and application program within DOE. DOE's Office of Technology Development took the position that the ISV technology was sufficiently developed for contaminated soil type applications, required little or no further development work, and should be considered as an "available technology" for DOE remediation needs. DOE also terminated its ISV Integrated Program which was established to investigate the underlying technical principles that were needed by the various applications. DOE took the position that further development of "advanced" ISV applications (e.g., buried wastes and underground tanks) would wait until ISV became an established technology for contaminated soil applications. The DOE buried waste and underground tank ISV programs essentially terminated at that point; however, several relatively small test and development projects related to contaminated soil applications at ORNL, SRL, and Hanford were maintained.

Geosafe proceeded to investigate the cause of its testing event and developed a significant advancement in understanding of the generation and disposition of gases and vapors during ISV processing. Geosafe established new application limits and process control standards, and designed a new metal off-gas hood capable of withstanding higher off-gas temperatures. Geosafe resumed testing of its equipment early in 1993 and initiated its first commercial remediation project during the Summer of 1993. Whereas the early stages of the DOE ISV development program focused on proof-of-concept, equipment design and development, and test, development and demonstrations of interest to DOE sites, Geosafe's efforts have focused on process dynamics, process control, equipment refinement, expansion of application types, and developing the know-how required to make the technology operate on a commercial basis. These efforts have resulted in a greatly increased ISV knowledge base being established since 1991.

Geosafe's first commercial project was at the Parsons Chemical Superfund Site in Grand Ledge, MI. That project involved the treatment of 4,800 tons of soil contaminated with pesticides, mercury, and low levels of dioxin and arsenic. The project involved some site and soil conditions that had never been tested at large-scale before; and Geosafe had to respond with several adaptations to the equipment and process. The challenges were successfully met, and a highly successful EPA SITE Demonstration Program demonstration and evaluation was performed. Geosafe then performed a National TSCA Demonstration at a private site in Spokane, WA. That project involved the treatment of 3,100 tons of soil and debris (drums, concrete, asphalt) contaminated with PCBs to a maximum concentration of 17,000 ppm. That project was very successful with no significant process or equipment challenges being encountered. Geosafe's large-scale equipment is now operating at the Wasatch Chemical Superfund Site in Salt Lake City, where it is processing 6,000 tons of soil and debris contaminated with dioxin,

pentachlorophenol, pesticides, and a variety of other organics.
Both DOE and Geosafe continue with ISV technology development and demonstration programs. The current DOE program includes: 1) preparation for a large-scale treatability study on an inactive radioactive liquid waste seepage trench at ORNL, 2) preparation for a "spot melting" field demonstration at PNL, 3) development related to application of ISV to low-alkali soils at SRL, and 4) investigation of possible ISV application to buried wastes at ORNL. Geosafe's program includes performance of tests and development work for pending non-DOE applications within the U.S., and for overseas projects including contaminated soil, LLW, and waste treatment applications in Japan, and a major buried waste application at the Maralinga Test Range in South Australia.

Commercialization Chronology

Figure 2 illustrates the ISV commercialization chronology. PNL scientists and engineers invented the technology in 1980. A patent application was filed in 1981, and a very fundamental patent was granted to DOE in 1983. Similar patents were granted by major industrialized countries throughout the world. DOE granted a license to Battelle Memorial Institute (operator of PNL) in 1986. Battelle initiated marketplace promotion and regulatory awareness efforts on a privately funded basis. These efforts resulted in the selection of ISV for remediation use at three U.S. sites, including: 1) the Pristine Superfund (Remedial) Site in Cincinnati, OH, 2) the Griener Lagoon Superfund (Removal) Site also in Ohio, and 3) a private RCRA closure site in North Carolina. Battelle defined a strategic commercialization plan that involved creation of Geosafe Corporation in 1988 to fulfill the needs of these selections. Geosafe procured its first large-scale equipment system later that year.

The Company suffered a major setback within one year after its initiation as all three of the site selections were withdrawn. The Pristine Site Record of Decision (ROD) was reopened after a surprising turn of events. EPA had issued this ROD and many others in response to Congress' mandate (SARA/1986) to show site selection preference to innovative technologies that involved treatment, reduced toxicity, mobility, and volume, and could be performed onsite and in situ. That mandate essentially specified technologies like ISV for many difficult sites. In response, EPA issued many innovative technology RODs without first having performed treatability tests on the selected technologies on actual wastes from the sites. Congress' Office of Technology Assessment (OTA), in one of its twice yearly critiques of EPA, took EPA seriously to task for having done this. In addition, OTA questioned the wisdom of EPA in making the ISV selection because a competitor of Geosafe's had made claims (unsubstantiated) that "ISV doesn't work". In Geosafe's opinion, OTA's challenge to EPA resulted in significant damage to the advancement of innovative technologies, in direct conflict with Congress' mandate.

The Griener Lagoon site selection disappeared on a similarly unanticipated basis. EPA added lime to the site to make the wet soil easier to handle during staging of soil in preparation for ISV treatment. After the site was ready to treat, EPA performed sampling and analysis to confirm the PCB content of the soil. To their surprise, the PCBs had apparently disappeared. Similar results were noted at other PCB sites. This finding was termed EPA's "lime discovery". The ISV site selection was removed while the discovery was further investigated. Some years later it was determined that the addition of lime to the sites did not really destroy the PCBs, rather it may have removed some by volatilization due to the heat generated from the addition of lime, and the presence of lime was found to mask the PCBs making them difficult to detect and measure by the conventional analytical methods then employed.

The private RCRA site selection was lost when Geosafe determined it was not possible to negotiate an acceptable contract with the site owner.

Geosafe was then faced with reestablishing a queue for its ISV test and remediation services. A significant number of treatability tests were performed in support of Superfund RI/FS work. Within a few months the State of Michigan decided to fund an ISV demonstration at one of its sites. In collaboration with the State, it was determined that EPA would fund the demonstration at the Parsons Chemical Superfund Site under the existing ERCS contract. Geosafe was told to be ready to mobilize to the site within two months. This timing was unfortunate in that EPA administrators shortly thereafter determined that the ERCS contracting provisions should not be used for such projects, wherein a subcontractor (Geosafe) would be doing most of the

work. They proceeded to use a competitive contracting approach that stretched the procurement period from the original two months, by a factor of 11 times, to 22 months. Geosafe finally obtained that contract, and another with a private client, during the Fall of 1990.
During these 22 months, Geosafe's marketing efforts yielded significant fruit, with the addition of 10 site (preferred remedy) selections for the ISV technology. These selections involved Superfund RODs and other comparable decision statements by DOD and private parties. At this point Geosafe was vigorously looking for an investment partner to fund the equipment and working capital that would be required to support this anticipated level of demand.
Fig. 2.
Fig. 2. ISV Commercialization Milestone Timeline.
In 1990, Geosafe hired and trained a full operating crew for its new large-scale system. This crew would perform two large-scale melts as part of its training prior to mobilizing to the first project. The first melt went well. The second employed particularly aggressive operating conditions and resulted in the incident reported in the technology chronology above. Geosafe determined that the incident was significant enough that large-scale commercial operations should be put on hold (suspended) while the cause was investigated. It took about one and one half years for Geosafe to fully evaluate the cause and to determine a recovery plan. An additional half year was consumed designing and fabricating a new metal hood. Geosafe resumed testing early in 1993, and commercial operations were initiated shortly thereafter.
During the suspension of large-scale operations, a number of the site selections understandably went away. Two RODs were reopened and revised due to the uncertainty as to how long the suspension would last. Other "multiple choice" selections which allowed alternatives to ISV resulted in decisions against ISV during this time. Two DOD demonstration selections were withdrawn because of budgetary constraints and Geosafe's inability to meet desired mobilization schedules. At the end of the suspension, three of the original selections remained, and two more had been added. Geosafe then initiated commercial operations in 1993 with five intended selections for site remediation by ISV.
Geosafe is currently working at the third of these sites. The attainment of commercial operations status has been helpful in restoring awareness by the marketplace and regulatory community that ISV has returned after the suspension. At this time ISV is being seriously considered for many possible private and government sites.
BARRIERS ENCOUNTERED
Technology Readiness
At the time of Geosafe's inception, the prior ISV test record had led to the belief that the ISV technology was very capable and fully ready for commercial application to soil remediation projects. The technology had been developed and demonstrated at full-scale in an unusually brief time. It was to be realized later, as the technology was tested under more aggressive conditions and for much more complex applications, that the basic underlying technical principles governing the technology had not yet been fully explored and understood, primarily in two areas: 1) melt width control (which impacted attainable depth), and 2) generation and disposition of gases and vapors during processing. Whereas there had been ample funding for equipment and demonstration projects, funding for underlying principles exploration was not forthcoming from DOE. EPA also declined to fund experimental and test work because "ISV was DOE's technology".
These factors, in addition to the fact that no two sites are exactly the same, caused Geosafe to undertake additional efforts in order to understand and offer a viable technology. It also required Geosafe to establish a relatively tight "applications envelope" wherein the underlying principles were understood for applications within the envelope. Other applications would require further testing and development. In this sense, then, Geosafe considered the initial state of technology readiness as a type of barrier (or limitation) to full attainment of the technology's commercial potential.
Volatility of Site Selections
It may take many years for a site to move through the process from initial discovery and characterization to final remediation. For purposes of predicting firm demand on which to base company decisions, the technology-linked remediation vendor may

look at contracts in hand, which is the firmest indicator, or possibly at official site selections specifying use of the vendor's technology. Since ISV is a sole source technology, Geosafe assumed that site selections specifying the use of ISV were pretty firm. Part of the learned reality was that site selections are not inviolable, and it is common for them to be reopened, overturned, or otherwise made highly uncertain. It was also found that typical remediation contracts allow for the client to terminate the contract without cause, further adding to the difficulty of making credible projections.

Difficulty Establishing Current Awareness

It is a major challenge for an innovative technology company to establish and maintain a current awareness of its technology within the marketplace, including the potential client, regulatory, and environmental engineering segments. Many different organizations (e.g., EPA, DOD, DOE) attempt to assemble and publish standardized data on available technologies on a periodic basis. However, in most cases, the information is seriously out of date (usually at least two years) before it finally gets published and into the hands of users. This problem is particularly important to innovative technology vendors wherein the technology status is usually changing quite rapidly. Significant improvements have been made in this area by EPA's VISITT database and SITE Demonstration Program. However, even now, Geosafe routinely encounters decision makers that are using outdated information on ISV. In an effort to maintain a better current awareness level within the remediation marketplace, Geosafe initiated a periodic publication called the In Situ Vitrification Technology Update. This newsletter has received very good reviews and is obviously much more timely than Government published information. The challenge remains to get the information into the hands of the appropriate decision makers.

Lack of Understanding Regarding Small Commercial Business Needs

Innovative technologies are mostly developed and offered by small business concerns with limited capital resources. Geosafe believes that the needs of such companies are not well understood by the Government organizations attempting to deal with them. For example, EPA's SITE Demonstration Program requires the technology vendor to fund its own demonstration, while the SITE Program will fund data acquisition and evaluation efforts. Such large-scale demonstrations can cost many hundreds of thousands of dollars; small innovative technology companies cannot handle that level of "marketing" cost. The alternatives exist for the developer to find a client to fund the demonstration, or for the Government to provide a demonstration site. In the search for an ISV demonstration site, EPA's SITE Program formally requested DOE to consider providing a site where a demonstration could be performed. For unknown reasons, a suitable site could not be found from the hundreds possessed by DOE. Largely because of this situation, the SITE Program evaluation of Geosafe's ISV technology occurred many years after the technology was accepted into the SITE Program; and even then, the site employed was not the ideal site to satisfy the objectives of the SITE Program.

In similar manner, innovative technology companies are faced with the need to fund large-scale equipment without the existence of firm market demand. The companies are also expected to maintain staff during periods of contracting or regulatory delays of various types. The developer typically is expected to withstand these costs without assistance. The developer may also be expected to provide special low pricing as an incentive to obtain a demonstration opportunity. Such costs can be staggering to a small technology innovative company, and should be recognized by those organizations charged with encouraging the development and implementation of innovative technologies. Such costs also mitigate against profitability, which makes it difficult for companies to raise additional investment capital.

Actions of Detractors

Geosafe considers the loss of the Pristine ROD largely to be the result of the competitor's claims against ISV that contributed to OTA's harsh criticism of EPA for the ISV selection. The competitor's claims have since been shown to be unfounded; however, the damage was done. Other actions by competitors and other uninformed persons resulted in the spread of erroneous information regarding ISV. Geosafe has had to counter many myths, including the following beliefs: 1) that all vapors formed during ISV moved into adjacent soil (the myth of "vapor retreat"), 2) that ISV only worked in sandy soils, 3) that ISV would not work in wet soils, 4) that ISV is extremely energy intensive, and 5) that ISV is extremely costly. Geosafe has had to expend significant resources combatting the damage such myths cause in the

marketplace and regulatory community.

Loss of Momentum

The series of unexpected testing incidents, and Geosafe's temporary suspension of large-scale operations, had a very negative impact on the momentum and enthusiasm that had been established for the technology. The message went out within the EPA community that ISV was no longer available. Nearly the same effect occurred within the DOE community, as ISV went from being hailed as one of Admiral Watkins' "flagship technologies" and a well funded segment of DOE's technology development program, to a technology that was almost put on standby until it was demonstrated as a commercial success on contaminated soil applications. DOE's disappointment even resulted in "bad press" from the technology's previous number one supporter. This loss of momentum has been and continues to be a major challenge to Geosafe, as it tries to reestablish market awareness of the current commercial status of ISV.

It is worth noting that, while DOE's response to the testing incidents was one of significant withdrawal of support, EPA, in approximately the same time frame, recognized the need to accept and expect some reasonable difficulties with innovative technology testing if their efforts to implement innovative technologies were to be successful. EPA issued a policy to this effect that not only recognized the risk of some failures as important lessons learned, but actually encouraging the selection of innovative technologies despite such risks.

Contracting Problems

The difficulty posed to contractors by the Government contracting process was epitomized by Geosafe's experience wherein a two month procurement stretched into 22 months. The location of the procuring office, and the personnel involved, changed several times during that time. Geosafe found there to be a significant disconnect between the operations side of many Government client organizations, and the legal/contracting side. The standard wisdom is for vendors to work both sides of the house at the same time. However, our experience indicated that the legal/contracting side would not usually get functionally involved with a project until it was time to negotiate a contract. This situation not only caused problems for Geosafe, but for the operations side of the client organization itself. It should be noted that at the time of this experience (1990), many of the procurement practices for remediation projects were in the fairly early stages of development, and were subject to many changes. Geosafe has not noticed the extent of such problems when dealing with private clients; however, as a general rule, it does seem to be more difficult in terms of time and effort to negotiate a contract than to perform the technical work associated with project development.

Geosafe also experienced a case wherein the Government client was always right, so to speak, when it came to interpreting contract provisions. The Government held firm to its "promise" that there would be no change orders during the fixed price contract, regardless of whether or not they were justified. Geosafe was obligated to accept the cost of unforseen technical challenges and site discoveries during this project. Even extending the duration of the contract extracted a significant additional financial toll. There exists an obvious dichotomy within the Government wherein, on the one hand they are charged with supporting the introduction of innovative technologies, and on the other hand they are charged with being as hard as possible in the area of contract enforcement. Typical development stage innovative technology companies are seldom positioned to withstand such costly "support" from the Government.

Difficult Capital Market

At the time of Geosafe's incorporation, the enthusiasm of investors for new environmental technology companies had just peaked and was on its way down. It is general knowledge today that raising significant capital (e.g., millions of dollars) for environmental ventures is very difficult until the company has passed through the proverbial "valley of death". An occasional exception may be successful at raising significant capital if it can be packaged and sold before any downside difficulties unfold. This situation poses a significant challenge to small venture companies since the environmental remediation marketplace, with its relatively large dollar volume projects, and the long procurement and payable times, typically requires very costly equipment and large working capital reserves. Geosafe has been fortunate to have a committed and experienced parent company (Battelle) that has enabled the firm to survive such challenges of commercialization.

SUPPORT ENCOUNTERED

Geosafe acknowledges the importance of the DOE technology development program, and the strong initial support directed to the transfer of the ISV technology for commercial application. Geosafe also notes the good support of ISV and other innovative technologies provided by many organizations within EPA and various state regulatory organizations (e.g., Michigan, Washington, Utah). EPA's Technology Innovation Office, SITE Demonstration Program, and Risk Reduction Engineering Laboratory, have been most helpful in supporting the introduction of ISV. EPA was also instrumental in developing independently qualified data on the ISV process that put to rest the unsubstantiated competitor claims mentioned earlier. In addition, the industry media, with the exception of the few who have an agenda beyond providing public information, have been helpful in developing marketplace and regulatory community awareness of the technology. Geosafe is also appreciative of the various "ISV champions" that have emerged within various industry organizations; for without these champions, few innovative technologies would attain commercial success.

LESSONS LEARNED/CONCLUSIONS

Geosafe has encountered and overcome a number of difficult challenges in the process of commercializing the ISV technology. The Company has now operated the technology on a commercial basis for about two years. The future of the technology is very promising. With "perfect 20:20 hindsight", Geosafe suggests the following principles should be considered strongly by parties commercializing innovative technologies:

1) Establish a conservative applications envelope for commercial applications of the technology; and do not take projects that involve opening the envelope further without first ensuring all parties understand the experimental and uncertain nature of the outcome.

2) Do not risk your company's ability to perform profitable commercial operations on experimental, envelope-widening projects that could damage that capability.

3) Make great effort not to let various difficulties destroy the company's marketplace momentum.

4) Do not make major commitments to personnel staffing and large-scale field equipment until firm contracts are in hand.

5) Develop marketplace and regulatory awareness as broadly as possible, taking care not to oversell the capabilities of the technology. This requires some means of periodic information distribution, and means to keep the awareness current as the technology status changes.

6) Recognize that detractors will probably attempt to block your marketplace efforts, and that resources will have to be spent countering their efforts. A strong information distribution program, including publication of performance results, is a good defense against detractors.

7) Take great care, even at extra expense, to negotiate contracts that are appropriate for the degree of uncertainty involved in a project. It may be better not to have a contract than to have one that is one-sided, particularly where there exist uncertainties regarding the technical aspects of the project. Ensure that cost-reimbursible standby provisions are included against the possibility of client- or regulatory-imposed delays.

8) Recognize that the guidance followed by Government contracting officers will be stringently followed regardless of the spirit of compromise or support indicated by non-contracts type personnel. If tolerance of unforseen difficulties is to be accepted, such should be defined in the contract.

9) Be financially prepared for commercialization time and cost estimates to stretch out beyond seemingly reasonable estimates. This requires experienced investors with a long-term outlook and capability.

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COMPETITION BETWEEN WORLD MARKETS VERSUS SOCIAL NEEDS AT HOME: THE DICHOTOMY OF PERFORMANCE

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ABSTRACT
The press, public, government and industry have been hearing, talking about technology transfer and defense conversion for at least the past five years. But where is it?
The national laboratories have tried but, except for an increase in the number of cooperative agreements (CRADAs), little progress has been made moving their behemoth weapons capabilities forward to establish the 'sustainable" economic foundation we will need for tackling the next century. There is almost universal agreement that our national laboratories, with some 40,000 scientists and engineers, could be the basis of a comprehensive strategic planning effort to develop technology that would return the U.S. to future world leadership role while providing a major stimulus for economic development. But where is the vision? It is there largely because their raison d'etre was to provide the nation with a nuclear weapons capability, building a competitive, highly demanding, excellent and imaginative scientific complex.
Now budgets are being directed to competitive programs away from defense. We have to re-emphasize the need for competitive technology development. But to do this, the labs must focus on smaller, multiple projects. In order to keep the levels of employment, the kind of research the labs are doing must be directed to both applied and basic research for market applications. This country needs to have a rapid response for market driven products. This means the labs need to direct efforts to high volume projects with rapid return on investment while devoting a percentage of their time to basic research, proliferation intelligences and scientific curiosity.
This paper intends to examine the need for an approach that implants in the structure of the national labs a capability and expertise to develop technologies at a competitive cost, and within a time frame designed to meet market demands. There is a certain skepticism that has been documented in several journals and press accounts questioning whether the laboratories have the capability to achieve this desired result. The intellectual capital is there, the question that the skeptics raise is "do they have the market understanding, economic discipline and chameleon capability to change? Views will be offered on how DOE, current and future Administrations, and Congress can develop missions and initiate a "seed" program that would include market milestones, along with financial incentives, for those labs undertaking technology development programs to meet future market demands, in competition with low cost third world nations, and maintain U.S. competitiveness.
INTRODUCTION
In 1991 at Los Alamos co-author Leo Duffy, as Assistant Secretary of Energy, gave a lecture series on the need for innovative curiositythe search for innovation through the scientific inherent curiosity. The series of lectures was given to the three weapons labs in anticipation of the end of the weapons era. The message was that significant reductions were anticipated as a result of defense reductions and the new market was world class competitive technology in the environmental market.
The social issue was based on the fact that the world will have the same surface area for our foreseeable future. The amount of potable water will be less than one percent of the total water supply and the population of the world will double every 20 years. An environmental, ecological scientific delight! The first seminar was at the Los Alamos lab and recommended that the Los Alamos scientific resource redirect it's effort and future mission to the total economic scientific system. The suggestion was greeted with the expression on the faces of the august body, "Will somebody get the butterfly net."
A similar presentation was given to the Sandia management team and received an even greater receptive response like, "Is he dangerous when out of the asylum?" Finally, a modified version was given to Lawrence Livermore's management team. Updated and modified to"Innovative Curativity." (That's right curiosity plus creativity equals "curativity.") This series completed the circuit of former Secretary of Energy Admiral Watkins "Jewels in the Crown" with the message that the light at the end of the weapons era was our opportunity to join the commercial world of reality or the world of high anxiety. The series of seminars was not received with the same appreciation as the "Sermon on the Mount", and it did not emphasize or espouse the principle that "the meek will inherit the budget".
What it did espouse was "innovation combined with curiosity would produce creativity" but based on a market driven strategy. The authors believe that cost conscious business partnerships between intelligent engineers and scientists and

their cohorts in the private sector will produce a viable survival in the post weapons era. This considers that all the research centers have been amortized and, with unique business partner incorporations, could accelerate the application of the labs to compete in the new economic opportunities.
The Pavlovian Weapons blanket is still the preferred mother nurture at the weapons laboratories. But as some of the articles in the press indicate progress is slow but it is occurring and it is still progress.
It is interesting that inside the Labs (weapons and research) and in many areas of congress there is a religious fervor that believes that within this great potential hides a "roaring tornado of technological solutions trying to burst the seams of bureaucratic control. This appears to be contrary to a 1992 GAO Study of Federal R&D and the Barriers to Technology transfer. The study indicated that the 10 year expenditure of R&D funds by 21 agencies was .5 trillion dollars and resulted in $37 million in Royalties. Thirty three million dollars to the National Institute of Health for an Aids test kit. The study included 21 agencies, 700 labs and 177.000 scientists hardly an endorsement of a raging technological tornado.
But the good news is that as budgets get tight the survival mode "Innovation Incubator" could form within the Department of Energy and other Federal Labs.
It is continually inferred that the National Labs are the source of technology transfer. It appears that this is a legend in their own minds. This is not to be critical but to be pragmatic. Thirty-seven million dollars of Royalties for a $.5 trillion in research and development is not a whirlwind return on investment.
The ideal concept that is promoted is that within these institutions of complex intra science disciplines are hidden, in the neurones and synapses of the most formidable minds, technical and scientific discoveries that are bursting at the Labs Seams, just waiting to be marketable products.
An excerpt from John Ulman's article in Society Magazine of June 1993 states defense conversion means doing something "new". "Many firms that attempt to get out of government work have been sadly unsuccessful." The cause in short is typical of trying to transform government style research and production systems with their cost plus mentality to compete in a cost competitive market. Without recognition of the need for fundamental changes in the culture of their procedure, they will fail. It appears that government contractor firms and the facilities they operate are unable to function in the competitive market environment. The notion that government research, leads to civilian economic production flies in the face of reality.
Ulman states that conversion of government facilities is necessary to relieve congress from the pressure to continue unnecessary operation in the name of a job preservation program.
Based on the real market metrics, the development of market products must be functional, economically viable and better than the products that currently exist. In order to accomplish this, the Labs must know what the market wants and, more important, what will the market pay.
The culture of a commercial enterprise also flies in the face of conversion of government facilities into technology implementors. Contrary to Federal Facilities, Commercial enterprises do not in most cases receive progress payments. The job is paid for when finished, and performance is key to repetitive business a unique requirement contrary to the current mode of Federal operation.
Where can DOE labs "curativity" fit into a solution? It appears, based on the culture of the government labs, with the past mountain of high cost secrecy mentality and uncertainty in market understanding, market driven schedules and cost control, that the Mohammed of the private sector industry should climb the government lab mountain.
In order to demonstrate success the labs should preferably start with a known quantity, in dire need for improvements, with a clear data base to compare with the commercial performance and demonstrate an immediate short term indication of success. Peter Drucker says that accounting costs models rarely provide the cost of non-productivity. In fact, they usually conceal it. With the temperature of the new climate in Congress, the heat to cut costs and get more bang for the buck will make global warming look like a deep freeze to future budgets.
What area would be the easiest to accomplish? It is the authors opinion from interviews that no lab director, M&O president, or Operations Manager wants to spend time nurturing Administrative support, i.e., Budget Accounting Services, property management, facility preservation, maintenance service and environmental support

services. But that is where the most continuous internal and external criticism exists. Administrative lack of control in the above areas has been a continuous criticism by GAO. Is it a Pavlovian reaction? If you don't like doing something, you don't do it well.

Why not separate out these functions and approach the local community for fixed price proposals with unit price service related capabilities. It would provide all administrative, maintenance, and facility services on fixed price basis based on regional comparative cost, new data reporting systems that compare performance with similar regional industries and premium unit price for unique administrative or technical lab required services to function in the weapons, energy or esoteric research field.

This will provide a local firm to perform a measurable administrative and service support system function based on regional commercial rates and comparable across similar local facilities with commercially performed services and have an initial regional ownership of a straight forward competitive service.

The Federal agency should then define the core government required mission for the lab or facility. Contrary to the Galvin Committee, core missions have not been established. There is a need to isolate the core mission, isolate the lab resource and assure cost budget related requirements rather than facility pro rata costs. The mission should bear its full cost. The core mission, special equipment and special services should be clearly separated and budgeted for their unique needs. Thus with local initiatives, small business curativity and competitive innovators, a privatized example can catalyze the core neurons of the labs to imitate and initiate a local, regional covenant for competitive success with a mutually developed strategic business plan and regionally competitive costs of doing business.

Next identify the cream science and technology innovators who exist in the lab and have the most interest in achieving market product capability. Then make them available to competitive private enterprise recruiting. Most facilities have been written off (amortized) when built and are not amortized on a depreciation basis. These facilities could be turned over to a business development contractor on a 5-10 cents on the dollar basis, similar to the Rouge River Plant , after World War II, to develop a marketable business within a 2-5 year basis. Take a unique approach. As Tom Peters in his new book says, "crazy times require crazy solutions." Invite business in to evaluate the lab facilities, lab personnel and lab costs and ask industry to submit the most productive use of the labs they can dream up. Don't write a detailed government RFP. Let industry state what it will do and select the best proposal guarantee for the government. They must be willing and able to support the core mission on demand, work with academia and meet the intent of DOE orders and procedures-not the letter of the written word. They must be willing to show a profitable operation in a specific number of years while transitioning from the current mode of operation while still providing first priority to the core mission. The bids from local, regional or national entrepreneurs would be based on the most financially attractive proposals that would utilize the laboratory human and physical resource and provide the greatest societal economic development for the locality and the region with respect to business development, not just technology transfer. Examples of complimentary dual research projects that can be developed are: materials development of products that are offshoots of waste stabilization; chemical extraction from soil, sediment or sludge cleanup applied to water treatment or chemical process firms; or analytical laboratory services; data processing and image processing; and software development to provide initial entry of DOE into the data highway. These capabilities represent one-half billion dollars of DOE services per year that are all immediate applications for privatization.

The next segment of the bid would be based on research and production meeting regional needs such as agriculture, mining, lumber, and transportation. This should be developed in conjunction with local, regional governments, business and universities, not just all-knowing government bureaucrats. In fact, minimize national government interaction and show today's current congressionally espoused notion of state and regional direction and support.

Now to make it more interesting provide the first year budget on the basis of historical past, second year budget will require 20 percent of operational cost will come from private sales including royalty payments, third year 40 percent from private sales, fourth year 50 percent and fifth year 60 percent from private sales and the core mission will still be provided by the prime lab. Services will be

provided by private contractors on a fixed price plus unit price options for unique selective services. At the end of five years renegotiate the contracts with expectation of moves to off site locations in 10 years to private facilities in the regional area developing economic growth and stability of newly developed industries. Combining economic stability with local and regional university research can produce a microcosm for a national strategy that would implement a tactical national plan that would make the labs, world competitive and innovative to support commercial economics of the future.

The objective is to establish commercial entities with social responsibility (locally and regionally) within each Lab for the end purpose of providing the core mission, business development, with the main objective of reducing the government financial burden of the facility, growing local businesses business utilizing the existing human resource and maintaining the economic viability of the locality by providing technology and products for the local community growth, resource development and international markets.

The most novel curativity suggestion to help achieve innovation is baseball's "Barry Bond's model". Barry Bond a famous Pittsburgh Pirate was entrepreneurially purloined by the San Francisco Giants for 40 million dollars. For a significantly lower sum, an entrepreneurial firm would offer contracts to the best and the brightest talent at the lab. They would offer employment contracts and profit contracts based on the success of the venture. The company would offer stock on initial value potential redeemable in five years. Stock options could be offered to local communities and shares would be based on marketing and financial or tax incentive contributions obtained for the venture. The offerings to the key individuals would be based on their estimated worth to the new venture. It would entail high personal risk but high potential return. Localities would receive stock percentages and would utilize their portion through a holding company. Profits from their stock would be based on time from initial stock sale to a recommended time to sell based on holding company recommendation.

The lab would maintain its core mission and, contrary to the Galvin Committee, initiate a major mission of business development supported from reduced custodial requirements for support services, because major facility costs would eventually be maintained by the private entities using the facilities.

The profit incentive will be the catalyst to change the perceived cavalier attitude toward costs in government work. This attitude has resulted in a lack of many scientists and engineers in the field of training and capacity for cost efficiency. The government Labs have lost the major portion of their original mission in many cases, but they represent the regional economy of towns like Richland, Aiken, Oak Ridge, Idaho Falls and would result in drastic economic hardship if closed. Yet they try to maintain the continuation of high price minimum return projects like the National Ignition Test facility, and the Advanced Neutron Source Reactor and Fusion. The lack of new missions will result in the loss of the best and the brightest the soonest. The under utilization of the 40,000 DOE scientists and engineers has resulted in frustration in wondering what happened to their skills, as they consider the paper factory morass in which they find themselves mired.

Retraining is not "serendipitous" but it is a fact of life for the high-tech technician, engineer or scientist in the DOE Labs. It's part of his or her historical career development.

The development of a Regional and National competitive boutique of industrial incubators can be the economic developer for the local and national markets. The diversity of the Environmental cleanup technology in water purification is just one potential. Composite materials developments, sensor development and data highway imaging systems, data screening, materials recycling and new fabrication businesses can be the light at the end of the tunnel and avoid the oncoming train.

It is time for a pragmatic introspection. "Multi-splendored" competitive business development as realistic alternatives or "clear cut" local communities who will start a 12,000 person reduction-in-force in the third quarter of FY'95. Without a major change in the labs a new era of Leadvilles, Buttes, Scrantons and the steel and coal towns of Pennsylvania, Ohio and West Virginia will be initiated.

The objective is to establish market-driven businesses and technical boutiques or service companies that would develop a major portion of the facility utilizing those human resources that were interested and capable of surviving in a commercial environment. This should have been the imitative at the beginning of Start II talks

in 1991.
The situation is even more dynamic than just the laboratories not having a mission. There is considerable discussion of combining the Energy Department function with other agencies. One discussion is that the relation in technology principles, combined with economic competitiveness is to combine Energy and Commerce. The basic competitive cost to most industries is the cost of energy.
The major intellectual capital resource of the industrial science community-if properly focused-are the forty thousand scientists in the Department of Energy Labs. President Clinton, Vice President Gore and Secretary of Commerce Brown in their discussions on sustainable technology development did not include DOE labs, but outlined the expansion of the National Institute of Science and Technology from a research budget of $70 million in 1993 to $800-900 million in 1998. This equates to a need of 9,000 scientists and engineers assuming $150,000 cost per engineer, per year. Even the most aggressive business economists would identify this a an unrealistic goal. The objective of the new Congress is privatization, development of government assets, the reduction of government support and the transfer of Washington centralized control of assets to the states for direct control and cost savings. The Western labs of INEL, Lawrence, Berkeley, PNL, LLNL, LANL and Sandia have 12,000 scientists and engineers. The development of local, regional technology could be initiated through the Western Governors Association to utilize this resource rather than NIST. The western DOE labs represent $6 billion a year which, when applied to local regional and national needs through the private sector development, is the substance of competitive economics. The physical and capital resource is directly applicable to the current congressional concept. The Western Governors can be the prototype that illustrates how local direction will provide the guidelines of the new decentralized direction of technology development combines energy science and technology through the application of existing resources. The major western laboratories-INEL, PNL, Sandia, Berkeley, Livermore, Los Alamos, and NREL-could be within two years of implementing a competitive nucleus of a national strategy for commerce and international competitiveness. Private development of national assets and the data management capabilities of the National Laboratory will provide the basic and applied technology for economically developed competitive products.
The key to success is local, regional involvement in the strategic and tactical development of local and regional market needs. The national needs of agriculture, diverse manufacturing and production competitiveness, water purification, regional and national transportation technology will be part of the national strategic plan. The concept is "back to the future" where grass roots provide the direction.
The lack of alternative missions will result in drastically reduced budgets and a forced high cost closure, a loss of valuable physical and human resources in the locality and the traumatic community transition. The lack of a mission analysis for major facilities in DOE and other agencies presents a high anxiety environment for the best and brightest.
Intellectual curiosity, with compelling technical vision of dynamic private business grass roots initiatives and implementing complimentary dual path technology will prevent the traumatic alternative and there will still be a viable option for the skim milk of a defense mission.
The delegations from the western states must be willing to concede that the defense mission will not be sustained by the general public built around the laboratories as the arsenal's of nuclear bombs. When GM lays off 50,000 people, IBM 100,000, Boeing 40,000, and California has a ten percent unemployment burden, a better utilization of the $6 billion is needed. To allow the dismantlement of the focused resource of the 12,000 scientists when world competition is at an increasing demand would be shortsighted at best, criminal at worst.
The mission of weapons must be modified to incorporate a social contract, a more robust technology application to competitive positioning in the world economy.
In Chemical and Engineering News (Oct. 5, 1992), Will Lepowski reported that "the breadth of interest with changing world events has caused a loss of coherence thereby reducing their overall effectiveness in responding to their traditional mission as well as new national initiatives and has led to an overall loss of direction for DOE as a whole." This was the report of the evaluation of Secretary Watkin's advisory board. The Galvin Committee recommends limiting the labs to non-commercial enterprises-a total desert of imagination.

The gauntlet has been thrown down. It's up to the labs to take the challenge or attrit to a single purpose diminishing role in the national and international competitive technology market.

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ALTERNATIVE STRATEGIES FOR CONTROLLING COSTS AND AVOIDING DISPUTES ON ENVIRONMENTAL REMEDIATION PROJECTS
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ABSTRACT
As the focus of this nation's massive environmental initiative shifts from investigation and assessment to remediation, it is critical that all participants in this effort develop comprehensive contracting strategies aimed at controlling remediation costs and eliminating protracted performance disputes. Due to the many unknown factors inherent in remediation projects, contractors and owners alike face tremendous challenges and risks in undertaking such work. In developing appropriate contracting strategies, the parties must recognize the unique challenges which confront them and define their shared project goals.
Initially, greater emphasis should be placed on negotiated, cost-type contracts which are more flexible than competitively-bid, fixed-price contracts. Moreover, the contract instrument must be carefully drafted to eliminate errors and ambiguities, and must fairly address the allocation of risk between contractor and owner. Only a contract which achieves equitable risk sharing, rather than wholesale risk shifting, can form the foundation for a cooperative working relationship where the common goal of successfully completing the project is the primary focus of the contractual relationship.
Additionally, the participants in the remediation process should commit to the use of "partnering" as a framework for contract performance, as well as alternative dispute resolution techniques to swiftly settle disagreements. Through partnering, the owner and contractor seek to erect a contract mechanism to focus resources on resolving job-site contingencies, disagreements, and contract interpretation issues before they escalate into disputes requiring the intervention of third parties. Relying on an extension of the partnering concept to the resolution of performance disputes, alternative dispute resolution ("ADR") techniques are designed to lead to the negotiated resolution of disputes swiftly and equitably without jeopardizing the successful completion of the project and at the lowest cost to the contracting parties.
INTRODUCTION
Site owners often turn to well-established contracting methods in retaining a contractor to perform remediation sitework without giving much consideration to the overall effect of such methods on the cost and efficacy of the remediation. Such an approach to an environmental remediation project, however, is a prescription for disaster because it fails to recognize the unique contracting challenges which these projects present.
These challenges, and their attendant risks, bear on all aspects of the remediation contracting process, including selection of the procurement and contract mechanisms, development of the contract risk allocation scheme, performance of the site work, and resolution of performance disputes. With a commitment to apply innovative contracting strategies to the remediation process, the parties can forge a cooperative relationship which equitably allocates risk and ensures that the common goal of successfully completing the remediation is the primary focus of the contractual relationship.
In addition to placing greater emphasis on negotiated, cost-type contracts and modifying contract terms to achieve an equitable allocation of risk, the participants also must focus on project performance. The use of "partnering" as a framework for contract performance, and reliance on various alternative dispute resolution ("ADR") techniques to facilitate settlement of disputes, will go a long way toward achieving this goal.
PROCUREMENT STRATEGIES
The procurement strategies adopted should recognize that it may be impossible to accurately define the nature and scope of environmental contamination in the assessment and design stages. To accommodate this reality, flexibility is required

in all stages of the remediation process, from contract formation through to completion and closeout of the work. Similarly, the contract should establish a reasonable allocation of risk. To achieve these goals, greater emphasis must be placed on negotiated, cost-type contracts which are inherently more flexible and allocate risk more equitably than do competitively-bid, fixed-price contracts. Negotiated procurement methods provide the owner the greatest flexibility in discussing appropriate alternatives for the remediation and in selecting the remediation contractor best-suited for the project. Similarly, the contract mechanism employed must be flexible enough to equitably account for changes in the scope of work arising during performance. A cost-reimbursable contract can readily accommodate a changing scope of work as well as a changing design. With proper cost and project management oversight, contracting on a cost basis can lead to the elimination of unnecessary work and to overall cost savings to the owner.

RISK ALLOCATION

Considerable attention also must be given to the terms of the contract instrument and the allocation of risk between contractor and owner which it achieves. Only a contract through which the parties share risk can form the foundation for a cooperative working relationship. The inequitable allocation of risk which characterizes most construction contracts has plagued both owners and contractors alike for some time. It is undeniable that the extensive use of risk-shifting provisions in construction contracts has contributed to the proliferation of claims and disputes in the industry. In recent years, there have been a number of studies examining this phenomenon and proposing realistic solutions, including studies by the Business Roundtable's Construction Industry Cost Effectiveness (CICE) Project (1) and the Construction Industry Institute's (CII) Contracts Task Force (2). These reports conclude that the owners' practice of shifting performance risk to the contractor undermines the relationship of the parties, and undermines the ability to control costs and timely achieve the contract's performance objectives. Relying on the contract to insulate it from risk which it should otherwise bear, the owner assures that an adversarial relationship will exist on the project.

To remedy this problem, the contracting parties should adopt a realistic and equitable allocation of risk. Therefore, rather than seeking to transfer liability to the contractor for all contingencies, the owner and contractor should agree on contract language which places performance risks on the party best able to avoid or mitigate the problem and requires the parties to share the risk associated with uncontrollable events.

PARTNERING PROJECT PERFORMANCE

Once the remediation begins, considerable effort must be made to resist the temptation to view the other contracting party as the enemy and the project as a series of conflicts. Although nothing constructive is accomplished by adopting this approach, it continues to be the norm for construction contracting projects in this country. Rather than perpetuate this self-destructive means of conducting business, the contracting parties should seek to achieve a non-adversarial relationship. This relationship is made possible in the first instance by a flexible contracting scheme which relies on an equitable allocation of risk. Through the "partnering" of projects, the owner and contractor seek to control costs and avoid disputes by maintaining a cooperative working relationship throughout performance. Critical to the success of this approach is a commitment to avoid protracted disputes.

The Construction Industry Institute conducted a study of the partnering process in which it acknowledged the many benefits inherent in its application (3). The project participants maintain constant communication throughout performance, jointly evaluate the progress and quality of the work, share resources to devise solutions to performance problems, avoid adversarial posturing, and develop strategies for attaining their shared goals.

Where there is a strong commitment by both parties to work together, they can strive to prevent disagreements from becoming disputes; when that is not possible, the parties will be in a position to resolve disputes in the course of performance through the involvement of management and the application of alternative dispute resolution techniques.

The mechanics of the partnering process are described in various industry publications, including ones issued by the Associated General Contractors of America (4) and the U.S. Army Corps of Engineers (5). However, the nature and complexity of each project will dictate the precise partnering strategy employed by the

contracting parties.
Recognizing the potential of partnering in the remediation industry, the Hazardous Waste Action Coalition ("HWAC") and the U.S. Army Corps of Engineers entered into a partnering agreement "to enhance the delivery of timely, cost-effective, and quality services on Superfund and defense cleanups" (6). This agreement falls under the umbrella partnering agreement between the American Consulting Engineers Council and the Corps, and addresses issues such as risk sharing, dispute resolution, safety, use of innovative technologies, and small business opportunities.

RESOLVING PERFORMANCE DISPUTES

In the event the parties cannot resolve performance disputes, mechanisms must exist which promote and encourage their negotiated settlement. The parties should commit at the outset of project performance to pursue alternative strategies for dispatching with lingering, unresolved performance problems. Relying on an extension of the partnering concept to the resolution of performance disputes, such alternative dispute resolution ("ADR") techniques seek to remove disputes from traditional forums and place them in settings conducive to prompt and equitable settlements.
The most effective ADR systems rely on continuing and open communication during performance, and the involvement of field personnel. Of particular benefit is the use of a "dispute escalation network" which defines the parties' internal mechanisms for approaching the resolution of disputes. The precise makeup of a particular dispute resolution system depends on the nature of the project; the goal of all such processes, however, is the low-cost, negotiated resolution of disputes during performance before adversarial positions harden and litigation becomes unavoidable.
Given the potential that the parties may be unable to resolve performance disputes among themselves, a mechanism to involve third-party mediators to assist in the resolution of such disputes should be in place at the outset of the project. One of the most promising alternatives in this regard is the use of a "standing neutral," such as a single project advisor or a more formal Dispute Review Board, which is available throughout performance to evaluate and resolve disputes. The standing neutral concept is flexible, and is designed to effect a swift resolution of disputes which survive the partnering process.
Where initial efforts at dispute resolution are unsuccessful, moving to the next level implicates various alternatives which have enjoyed considerable success in all types of contractual relationships (7). Among the most common ADR techniques used following contract performance are mediation, mini-trials and arbitration.
Recognizing the benefits inherent in the widespread use of various alternative dispute resolution techniques, the U.S. Congress passed the Administrative Dispute Resolution Act (8) in November 1990, authorizing and encouraging the use of ADR by federal agencies. Contractors performing remediation work for the government therefore are encouraged to pursue the use of appropriate ADR techniques to resolve contract disputes, regardless of whether the dispute is in the early stages of agency review or is the subject of protracted litigation.

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BARRIERS TO INNOVATION IN THE MARKET FOR ENVIRONMENTAL REMEDIATION TECHNOLOGY: A MODEL, A CASE STUDY, AND PRELIMINARY IMPLICATIONS FOR POLICY

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ABSTRACT

The U.S. faces extensive and varied waste management problems, both remediating existing contamination and preventing further pollution. The perception exists that there are many unnecessary barriers to the development and deployment of innovative environmental technologies which might help address these waste management problems. However, many of these barriers are intricately tied to otherwise beneficial policies, and thus can not simply be excised. Furthermore, several innovative technologies have been developed and deployed, thus providing indications that these barriers are not always insurmountable.

This paper describes two complementary efforts: development of a model for examining barriers to environmental technology development in the context of rational policy decisions, and a case study of a successful innovative environmental technology firm. The case study has provided examples to test the model. The initial sections of this paper describe the development of the barriers model, and present the case data gathered for the successful innovative technology company. The case data include descriptions of the company, Terra Vac, Inc., and its soil vapor extraction system, but the primary focus of the case data is how the company has overcome or avoided barriers to development of soil vapor extraction systems. The paper then describes the use of the model and case data to develop general policy options for eliminating or reducing barriers. One detailed example and an overall summary of the analyses are provided. Finally, conclusions regarding the applicability of the model are drawn, and areas needing further research are highlighted.

INTRODUCTION

Most studies of barriers to development of innovative environmental technologies have focused on the deleterious effects of these barriers on various technologies and on developers and purveyors of technologies. These same studies have tended to ignore the existence of some successful innovative technologies whose success argues that the barriers are not universally insurmountable. The study described in this paper seeks to develop a model which explains the presence of barriers in the environmental technology market and which can be used to develop and evaluate various policy options for removing or otherwise mitigating these barriers. The study also includes a case study of a successful environmental technology company. The case study provides examples for testing and using the model.

The first section of this paper describes the development of the barriers model. Subsequent sections discuss the methodology used for the case study, present brief descriptions of the selected environmental technology and environmental technology company, and present the case data which show how the company has dealt with the barriers it has faced. A subsequent section then describes the use of the model, combined with the case data, to develop general policy options for eliminating or reducing barriers. One detailed example and an overall summary of the analysis are provided in this section. Finally, conclusions regarding the applicability of the model are drawn, and areas needing further research are highlighted.

BARRIERS MODEL DEVELOPMENT

This study is not intended to simply provide another list of barriers to the development and deployment of innovative environmental technologies. Numerous such lists, developed by a variety of working groups, government agencies, industry associations, and individuals, already exist.(1-7) However, compilation, paired with subsequent analysis, of barriers was required for this effort, both to provide a framework for the subsequent case study and to provide a basis for the resulting policy suggestions.

A simple aggregation of the various lists of barriers contained in the references cited above would have been unwieldy. The source documents present and discuss barriers at various levels of sophistication. Some, such as the DOIT report,1 are intended to be detailed discussions of barriers with extensive back-up material,

which could be used to help define appropriate actions for the future. Others (2) are intended to provide general overviews for the purposes of familiarizing those outside actual environmental practice with the obstacles present in this market. Still other reports address causes and/or effects of barriers.

Since merely enumerating a diverse list of barriers was insufficient for this study, a model has been proposed to facilitate analysis of barriers in the environmental technology arena. In general, a model is a simplified representation of some aspect of the real world, and is intended to be a useful and intentional reduction of a mass of information to a manageable size and shape. (8) This model provides a structure which serves two main purposes:

The model provides guidance for grouping and understanding disparate barriers, and for eliminating from consideration those factors and circumstances which may not be true barriers, and

The model facilitates understanding of the interplay between barriers and desired policy goals and thus can serve as an aid to developing policy options to address barriers.

The model, which is presented graphically in Fig. 1, assumes that barriers in the environmental technology market are unintended results or secondary effects of rational policy decisions which have been implemented to meet distinct environmental policy goals. This assumption leads to a corollary: that few, if any, barriers have arisen spontaneously, or have been put in place with deliberate obstructionist intent. If these assumptions hold, they have a major implication: it usually will not be viable to remove or streamline barriers simply by removing the policies which have given rise to the barriers. Rather, it will be necessary to carefully design ways to provide continued attainment of the original policy goals, while reducing or removing the barriers desired to be addressed.

Figure 2 shows the structure which results when the model is applied to the lists of barriers assimilated from the many documents referenced above. The left-most column of boxes contains a set of generally accepted environmental policy goals. The middle group of boxes represents policy choices which have been made to enable one or more of the policy goals to be achieved. The lines connecting the two sets of boxes identify which choice(s) support which goal(s). Finally, the right-most set of boxes depicts the barriers which are unintentional results of the policy choices listed in the middle column of boxes. Again, lines show the ties, this time between the choices and the barriers they engender. Comparing Figs. 1 and 2, it can be seen that the sequence from policy goals to policy choices to barriers which is depicted across the top of Fig. 2 is the same sequence depicted across the top of Fig. 1. Since this effort is primarily focused on barriers, however, the "feedback" loop which appears in Fig. 1, and which connects the desired results of the policy choices to the policy goals, has been omitted from Fig. 2.

METHODOLOGY FOR INITIAL CASE STUDY

As discussed in previous sections, substantial effort has been devoted to examining barriers within the environmental remediation technology market. However, previous efforts generally have focused on the barriers themselves, and have emphasized the negative impacts these barriers have had on specific technologies, or even specific projects. Such analysis can be flawed: the assumption has been made that the technologies studied are in fact "good" technologies and should have been able to surmount all barriers.

As a result, the initial case study carried out for this project was chosen to be a case study of a successful innovative environmental remediation technology, and in particular, a case study of the individual(s) and/or company which introduced this technology in the marketplace. A two step process was used to select the technology and company to be studied. First, a successful technology was selected from a number of possible alternatives, and second, the pioneering entity marketing this technology was examined to determine whether it would be suitable for this study. No ranking or endorsement is intended in the selection of technology and company to study; rather the first pair of technology and pioneering company which met the criteria was selected.

Table I details the criteria used to determine whether a technology was appropriate for study. Several technologies were considered at this stage, including vitrification technologies, bioremediation technologies, and extraction technologies. A tentative selection of soil vapor extraction as the technology for the case study was made. The company which pioneered the technology in the

marketplace, Terra Vac, Inc., then was examined to determine whether it met the criteria for suitability of study enumerated in Table II. Terra Vac did meet these criteria, and thus soil vapor extraction and Terra Vac were selected for this study. A two part process also was used to carry out the case study. In the first part of the process, publicly available information concerning both the technology (soil vapor extraction) and the company (Terra Vac, Inc.) was gathered. This information, together with the barriers framework developed in the preceding section, was used to develop a list of areas which required additional information. This list of areas was used to provide topics for discussion during the second part of the process: interviews with employees of the company, including the company founder, Mr. James Malot, and with USEPA personnel who were familiar with the company. Interviews focused primarily on how Terra Vac had addressed barriers, but also included discussion of other subjects, including history of the technology, history of the company, and barriers which might arise in the future.

DESCRIPTION OF SELECTED TECHNOLOGY

The concept behind soil vapor extraction is fairly simple. A vacuum is applied to soil that contains volatile contaminants. The flow induced by the vacuum is drawn from wells drilled into the soil with screened intervals placed at depths corresponding to areas of high contaminant levels. The vacuum induces air flow through the soil and the volatile contaminants evaporate into the passing air stream. Depending on the types and concentrations of contaminants, the air stream, which now contains some concentrations of contaminants, then may either be released to the air or subjected to appropriate post-treatment. Possible types of post-treatment include collection of contaminants by a filter (e.g., activated carbon), conversion of contaminants to non-harmful compounds via catalytic oxidation, or extraction of contaminants from the air via some sort of air stripping technology. Soil vapor extraction systems lead innovative remediation technologies in frequency of use, and have been selected in more than 30% of Superfund records of decision. (9) A schematic view of a typical soil vapor extraction system is presented in Fig. 3. (10) The specific soil vapor extraction technology chosen for this study is the in-situ vacuum extraction system originally developed and patented by Mr. James Malot of Dorado, Puerto Rico and Mr. Melvin Visser of Portage, Michigan.

DESCRIPTION OF SELECTED COMPANY: TERRA VAC, INC.

Technology Development

Vacuum extraction technology was developed by Mr. Malot and Mr. Visser in the early 1980's. Much of the work was done while Mr. Malot was employed as an engineer with Upjohn Corporation at the facility located in Barceloneta, Puerto Rico. In response to a release of carbon tetrachloride at this facility, Mr. Malot proposed the use of a vacuum extraction system to remediate contaminated soils. Most of the equipment and techniques (such as vertically and horizontally drilled wells, vacuum pumps, and carbon filters) needed for a vacuum extraction remediation system already had been developed for other applications, such as oil and gas extraction. However the process of in-situ vacuum extraction of volatile contaminants in soil was new, and was deemed patentable in 1986. Although the initial patent was assigned to Upjohn,11 Mr. Malot now owns exclusive rights to this technology, and to an extension of this technology which allows simultaneous soil and groundwater cleanup. Mr. Malot established Terra Vac, Inc. in 1984 as a licensee of these technologies.(12)

Initial Deployment

Vacuum extraction was first deployed as a remediation technology at an Upjohn Corporation site in Barceloneta, Puerto Rico. The Upjohn facility located in Barceloneta (hereafter referred to as Site 1) was an active pharmaceutical manufacturing plant on the north coast of the island of Puerto Rico. Site 1 was located in a sparsely populated portion of the island near the towns of Tiburones and Garrochales. The total population of the area was approximately 3,000. During August 1982, approximately 15,300 gallons of liquid waste were spilled from an underground storage tank at Site 1. The waste was estimated to be comprised of 65% carbon tetrachloride (CCl_4) and 35% acetonitrile. CCl_4 is a volatile organic compound (VOC) and is a suspected human carcinogen.13 CCl_4 is regulated as a hazardous substance under CERCLA, RCRA, the Occupational Safety and Health Act (OSHA), and Puerto Rico's Regulation for the Control of Hazardous and Non-Hazardous Solid Wastes. Table III presents the chronology for Site 1, including relevant

regulatory events.
In January, 1983, Upjohn began an investigation of soils on and around Site 1. The investigation revealed levels of CCl4 in soils of up to 2,200 parts per million (ppm). Also during January, 1983, a pilot vacuum extraction system was installed at Site 1. The system consisted of three vacuum extraction wells, four vacuum monitoring wells, one vacuum pump, a cold water condenser, and a collection tank.(14) By late 1984, Terra Vac and Upjohn had expanded the system to full scale operation with a total of 19 vacuum extraction wells using six vacuum pumps. Although the system initially used the cold water condenser to remove contaminants from the extracted gas, it was found that significantly more CCl4 was recovered in the exhaust stack, so the condenser was removed for further operation.
The Terra Vac system operated at Site 1 from January, 1983, until March, 1988, when the criterion established by USEPA Region II for closure, that the exhaust stack of the system show "non-detect" levels of CCl4 in its emissions for three consecutive months, was met. At that point the Terra Vac system had removed a total of 17,781 gallons of CCl4 from soils at Site 1. This is almost double the amount of CCl4 that reportedly spilled during August, 1982.
Participation in the USEPA SITE Program
Soil vapor extraction was one of two technologies that were incorporated into the USEPA Superfund Innovative Technology Evaluation (SITE) program at its inception in 1986.(15) As part of the SITE program, Terra Vac deployed a vacuum extraction system at the Groveland Wells Superfund site in Groveland, Massachusetts during December 1987. The specific location for the demonstration of the vacuum extraction system was the Valley Manufacturing Company (VMC) site. VMC was a manufacturer of various metal and plastic parts.
The former waste disposal system associated with the VMC facility had dispensed liquid wastes to buried leachfields. Between 1964 and 1970, up to 20 gallons per month of hazardous materials, such as cutting oils, VOC's, and acid bath wastes, were disposed of in this manner. In 1979, two of Groveland's municipal wells were shut down due to high levels of trichloroethylene (TCE). The entire watershed and aquifer (including the Valley Manufacturing site) was placed on the NPL on September 8,1983. Remedies selected for the site included groundwater treatment and soil vapor extraction.(16)
The demonstration phase of the project ran for 56 days from December 1987 through April 1988. Approximately 1,300 pounds of VOC's (mainly TCE) were removed from soils at the site. Conclusions from the demonstration report stated that "the process achieved non-detectable levels of VOC's at some locations and reduced the VOC concentration in soil gas by 95%... the technology successfully remediated soils contaminated with TCE." (17)
Commercialization
Approximately concurrent with the incorporation of vacuum extraction technology into the SITE program and initiation of the SITE demonstration, Terra Vac became involved with remedial actions at two other Superfund sites: the Tyson's site in Montgomery, Pennsylvania, and the Verona Well Field site in Battle Creek, Michigan. Terra Vac had also installed systems at several non-Superfund contaminated sites dating back to 1984. By June 1989, more than 60 Terra Vac systems had been installed, and by May, 1993, Terra Vac had more than 400 successful application sites worldwide.12 Included in these numbers are 23 Superfund sites at which Terra Vac has deployed a vacuum extraction system. These 23 sites account for more than 25% of Superfund sites at which a soil vapor extraction type technology has been deployed.
CASE STUDY RESULTS
Interviews were conducted with Mr. James Malot, president of Terra Vac, Inc., and with Mr. Joe Pezzullo and Mr. Steve Johnson, also of Terra Vac. Interviews were also conducted with USEPA personnel familiar with Terra Vac, including Ms. Mary Stinson, and Mr. Jim Cummings. Additional information was gathered from newspaper articles and trade publications. The primary focus of this case study was to examine how Terra Vac met and addressed barriers to design, development, and implementation of soil vapor extraction systems. The interviews and other information indicated that the company had generally developed creative ways of addressing the barriers it faced. For example, Terra Vac obtained large amounts of operating data to support permit applications by initially using its technology, not for regulated remediation activities, but for monitoring applications, which were subject to less regulation. There were also some barriers which Terra Vac did not

have to face, usually as a result of the company's early entry into the market, or because Terra Vac's headquarters are located in Puerto Rico. The following sections provide a barrier-by-barrier discussion.

Splintered Markets

Terra Vac has found the splintered markets resulting from multiple levels of regulation to be a barrier both dealing with regulators and dealing with individual clients. Both groups have been reluctant to accept demonstrations conducted elsewhere as applicable to their particular jurisdiction or site. However, Terra Vac has been able to address this barrier by making use of its extensive collection of data on soil vapor extraction system performance with different types of contaminants and in different soil types. Terra Vac also has had some success overcoming this barrier by using an incremental approach with some clients and regulators, that is, by setting up a pilot system, and then, as favorable results are obtained, gradually expanding the scope of the pilot system until full scale implementation is achieved. Finally, Mr. Malot indicated that approximately 50% of the U.S. remediation market is found in only five states. Recognizing this concentration has helped Terra Vac address its efforts towards less splintered markets.(18)

Because Terra Vac has mostly been financed internally and with standard banking procedures,(12) rather than using venture capital funding, the company has not experienced one of the more commonly described results of splintered and difficult to verify markets, difficulty in obtaining venture capital funding. Mr. Malot did not believe that splintered markets would have been a major issue if he had chosen to vigorously pursue venture capital funding. However this belief remains untested.(18)

Technology Lock-In

Technology lock-in has not been a barrier for Terra Vac. In fact, since soil vapor extraction was one of the earliest innovative technologies to reach full-scale implementation, soil vapor extraction has sometimes benefited by being the technology which is locked-in.(18)

Loss of Intellectual Property Rights

Intellectual property concerns have been a major barrier for Terra Vac.(12,18) A particular source of difficulty has been problems with patenting a process, rather than a physical object, such as a piece of equipment. Terra Vac now addresses this barrier primarily by maintaining a position as a technology leader rather than by relying on patent infringement concerns to protect Terra Vac's soil vapor extraction technologies. In other words, Terra Vac continually pursues improved performance and added capability from their soil vapor extraction systems. Another major intellectual property problem for Terra Vac was its participation in the initial SITE program. Mr. Malot feels that the extensive data Terra Vac had to release in SITE program reports were very beneficial to competitors, yet provided little benefit to Terra Vac in terms of increased business.(19) Malot does not feel that Terra Vac would participate again in a similar program without significant changes in the requirements for public disclosure of intellectual property.(18)

Lengthy and Expensive Development Process

Although Terra Vac's initial implementation was relatively rapid, many subsequent implementations have been slow. Although Mr. Malot attributes this barrier to regulatory hurdles,(12) he believes the hurdles are due less to the regulations themselves and more to the inadvertent incentives built into the current system. For example, potential clients frequently wish to postpone expenditures, and are often able to do so by taking time for additional studies and planning. Although this approach may raise the total cost of the remedial action, it can postpone immediate outlays of funds, and without strong regulatory incentives for completing cleanups, postponement of expenditures be very attractive. Terra Vac was able to overcome this barrier at its initial implementation site of Barceloneta because it had a pro-active client (Upjohn) and a single local regulatory agency (the JCA) who were willing to try the then relatively new technique of soil vapor extraction so long as they were kept well informed of progress. Terra Vac has continued to use this approach to address this barrier by searching for pro-active clients and by involving applicable regulatory agencies early in the implementation process.

In other cases Terra Vac has found the self-interest of consultants and attorneys to be a barrier to implementation, since consultants' and attorneys' roles are much reduced once studies are completed and treatment is underway. Again, Terra Vac

tries to overcome this barrier by finding and encouraging pro-active clients.
Terra Vac has found that the need to collect extensive amounts of data to substantiate permit applications is a barrier which contributes to the expense of the development process. Although Terra Vac needed the same large amounts of data as any other innovative technology developer, Terra Vac was able to overcome this barrier by indirectly benefiting from a 1984 Puerto Rican law which required companies with underground storage tanks to be able to detect leaks from those tanks at the moment they occurred. By initially marketing soil vapor extraction as a monitoring technology, Terra Vac was able to partially defray the costs of gathering needed data.
Terra Vac has also found that the expense of obtaining funding, especially in a market that is not conducive to venture capital funding, can be a barrier. Terra Vac has addressed this barrier in two ways. First, since Terra Vac is incorporated in the Commonwealth of Puerto Rico, Terra Vac has been able to take advantage of Section 936 funding, a pool of funds "trapped" in Puerto Rico due to restrictions on transferring these funds to the 50 states. Second, again due to its incorporation in Puerto Rico, Terra Vac has been able to reinvest high proportions of its profits due to extremely advantageous federal and commonwealth tax codes.(18)

Public Opposition

Terra Vac did not face public opposition to its initial implementation. There were no active stakeholder groups other than the client and the applicable government agencies. The Barceloneta site is remote and there was little interest on the part of local residents or workers.(18)

PRELIMINARY IMPLICATIONS FOR POLICY

General Methodology

This section describes how the model can be used to examine the strategies used by successful environmental technology companies to overcome barriers, and to derive, from these strategies, more generally applicable policy options to address barriers in the environmental technology industry. As depicted in Fig. 4, each strategy is examined to derive one or more generally applicable policy options if possible. It should be noted that, in some few cases when a strategy is very specific to a technology or company, it may not be possible to derive any general policy options. The model is then used to identify, for each option, those policy choices and policy goals which may be affected if the option is implemented. The probable effect of the option on the goal is then evaluated to determine whether the effect is likely to abrogate the original intent of the policy goal. These further steps are also shown in Fig. 4.
This procedure can, and should, be carried out at more than one level. For example, at an early, conceptual stage of policy option development, the model can be used to identify policy goals which might be impacted, and which should thus be areas of concern during more detailed development of policy options. At a later, more detailed level of development, the model can be used to provide a yes/no type answer to the question, "Is this policy option likely to unacceptably weaken the implementation any policy goals?"

Application to Case Study: Terra Vac

The following sections of this paper use the model at the initial, more conceptual, level to examine possible policy options which might be derived from approaches used by the case study company, Terra Vac, to overcome barriers to development and implementation of soil vapor extraction technology. An example, which examines policy options to address splintered markets, has been traced in detail in the next section, and is depicted pictorially in Fig. 5. Since such detailed descriptions quickly become lengthy and cumbersome, however, the results for other barriers have been summarized following the detailed example.

Detailed Example: Preliminary Recommendations to Address Splintered Markets

Initially Terra Vac was able to use an extensive collection of data, much of which was gathered in a non-remedial application (monitoring of underground storage tanks), to overcome the concerns of regulators and clients about the applicability of the soil vapor extraction process at different sites in different regulatory environments. Since there will rarely be corresponding non-remedial applications available to facilitate gathering such large amounts of data, it will be necessary to develop a more general approach if other technology developers are to be able to use this strategy. One policy option which might facilitate gathering performance data is modifying regulations and/or the regulatory process to make it easier to get

permits for development and demonstration testing. Another policy option which might also facilitate gathering performance data is to make government remediation sites more available for development and demonstration testing. Both of these policy options would be intended to reduce the impediments caused by splintered markets, and in a detailed treatment both should be more completely examined. However, the rest of this section will only examine one policy option, that of modification of regulations and/or the regulatory process to make it easier to obtain permits for development and demonstration testing.

By using the model to track "backwards" from the barrier of splintered markets, it can be seen that the policy choices most likely to be affected by implementation of this option are the multiple levels of regulation, the complex and exacting nature of regulations, and the opportunity for public input to cleanup plans. Modifying the regulations and/or the regulatory process to make it easier to get permits for development and demonstration testing would definitely affect the complex and exacting nature of regulations. Depending on how these modifications are enacted, they might also affect the multiple levels of regulation and the opportunities for public input. As a result, it is not possible to eliminate any of the policy choices from consideration at this level of definition. However, this discussion will examine only the policy choice of complex and exacting regulations.

The model then is used again to trace backwards from this policy choice to the policy goals which might be affected. The goals which might be affected by changes to the complex and exacting nature of regulations are the need to reflect priorities of local, regional, and national stakeholders, the need to choose appropriate methods, the need to ensure appropriate implementation, and the need to reduce risks associated with cleanup. These further steps are also shown in Fig. 5. It appears, at this rather low level of definition, that the policy option of modifying the regulations and/or the regulatory process to make it easier to get permits for development and demonstration testing could adversely affect any or all of these policy goals. Thus all need to be retained for consideration during more detailed definition of this option.

Although it is not within the scope or capability of this paper to provide high-level definition of policy options to address barriers, two contrasting examples should illustrate how this further definition can occur and how the model can assist in this part of the process as well. In the first example, one can envision that one way to ease the process of getting development and demonstration testing permits would be simply to eliminate any regulatory requirements for such permits. But the model indicates that one of the policy goals met by requiring development and demonstration permits is reducing the risk of cleanup , and simply eliminating the need for permits might raise, rather than reduce the risk of cleanup by allowing poorly conceived demonstrations to proceed without adequate forethought, preparation, or review. Possible ill-effects of a poorly conceived demonstration could include further spread of contamination, perhaps by release from a poorly designed reactor, or increased difficulty of cleanup, perhaps by a poorly understood chemical transformation of contaminants into compounds that are less volatile or that are bound more tightly to soil. Thus, without some safeguards in place, risks might increase rather than decrease, and simply eliminating the need for development and demonstration permits is unlikely to provide satisfactory results.

A second policy option, however, intended to accomplish the same purpose, might be more promising. Most regulations already contain some provisions for less stringent requirements for obtaining development and demonstration testing permits. These provisions often go unexercised, however, either due to ignorance about or unwillingness to use these provisions.20 Thus an educational program intended to familiarize and better train regulators in the writing of development and demonstration testing permits would most likely facilitate gathering of needed performance data for innovative environmental technologies. Careful examination of this policy option shows that it seems unlikely to adversely affect any of the original goals (the need to reflect priorities of local, regional, and national stakeholders, the need to choose appropriate methods, the need to ensure appropriate implementation, and the need to reduce risks associated with cleanup), and thus this policy option shows more promise than the previous approach.

It should be noted that this model does not provide a methodology for evaluating and comparing the likely performance of the options. Rather the model provides a

context for making sure that any options considered do not themselves have unintended deleterious side effects.

Summary for Other Barriers

The model has been used to examine the strategies which Terra Vac has used to overcome barriers to development of soil vapor extraction technologies. The results of these analyses are presented, barrier-by-barrier, in Table IVA, B, C, D. The specific barrier being considered is identified in column 1, and columns 2 and 3 respectively identify all policy choices and policy goals which the model indicates may be affected by policy options chosen to address the barrier described in column 1. Column 4 identifies specific strategies which Terra Vac has successfully used to address the barrier in column 1, and general policy options derived from the specific strategies are described in column 5. Finally, a brief discussion of the likely effects of the policy option on the goals is presented in column 6. As indicated in the previous detailed example, it is not within the scope or capability of this paper to provide high-level definition of policy options to address barriers, so the discussion of likely effects indicates potential areas of concern, rather than specific recommendations.

Close examination of Table IV shows that there are two barriers, technology lock-in and public opposition, for which no policy options are presented. This is because the company studied, Terra Vac, has not had major problems with these barriers, and thus has not developed strategies to address these barriers. As a result, the methodology can not be applied to these two barriers. This does not indicate that these barriers are not real, nor does it indicate that there are no policy options to mitigate these barriers. Rather it simply indicates that, with the limited data set of one company, there is no basis for developing options or making recommendations. Close examination also shows two Terra Vac strategies for which there are no corresponding policy options. These two strategies are specific to Terra Vac, and although other companies might choose to adopt similar strategies, no policy actions would be required to enable other companies to pursue these approaches.

CONCLUSIONS

A model has been presented which treats barriers to development of environmental remediation technologies as unintended results of rational policy choices made to meet specific policy goals. The model has been successfully used to examine a lengthy list of barriers drawn from trade publications, industry reports, and anecdotal evidence. Because the model emphasizes that barriers are closely tied to policy choices and policy goals, the model shows that it will normally not be possible to simply remove the causes of barriers, because those policy choices which result in barriers normally also have intended, beneficial, effects which support desired policy goals. However, because the model clarifies the ties between barriers and policy goals and choices, it can be used to evaluate the broader impacts of policy options which might be used to address barriers. More specifically, at an initial stage, when approaches to eliminate or lower barriers are being conceptualized, the model can be used to highlight desired policy goals which might be affected by the various options. And, at later stages, when detailed development of options to address barriers is ongoing, the model can be used to determine whether an approach will unsatisfactorily impact desired policy goals.

Use of the model to develop top-level policy options to address barriers to development of environmental remediation technologies has been demonstrated for one case, the successful environmental technology company, Terra Vac, Inc. Terra Vac was the first company in the market with soil vapor extraction technologies, and has successfully overcome many barriers to continue developing and growing since its inception in 1984. As part of this study, strategies used by Terra Vac have been generalized into policy options whose viability has been evaluated using the model. Since resources have limited this effort to one case study of a single company, the effort has resulted in an incomplete, preliminary, list of policy options to address barriers. However, this effort has successfully demonstrated the use of the model for examining policy options to deal with barriers in the larger context.

The model does have some limits. First, an implicit assumption is made that the policy goals which drive the policy choices are valid. Any attempt to evaluate desirability or validity of top-level policy goals is well beyond the scope of this paper. Second, the model does not address deliberately obstructionist actions by those who may wish to maintain existing barriers to serve as barriers to entry by

newcomers. Although barriers may inhibit the process of development and implementation of new environmental technologies, those already in the environmental technology market, especially those who have invested significant resources to develop approaches to overcome barriers, may actually have interest in seeing these barriers maintained due to their inhibitory effect on newcomers. Finally, the model does not provide any basis for prioritizing or ranking policy options, but is limited to evaluating whether policy options will have unacceptable impacts on desired policy goals.

Additional research is needed. The scope of the study has been limited to one study of a successful company, and thus use of the model has been limited to addressing those barriers which were met and overcome by that company. This means that the model has seen only limited testing. Additional case studies of successful companies would broaden the scope over which the model has been tested. Interesting results might also be expected from case studies of unsuccessful, or marginally successful companies, marketing what appear to be promising technologies, but having significant problems with one or more of the barriers described in the model. Finally, there are some technologies which have been developed and then abandoned within the private sector, only to be picked up for further development and possible use by the federal government. Applying the model to this type of situation might assist in determining whether the initial developers abandoned promising technologies due to barriers which they were unable to surmount, but which the federal government was better able to address, or whether the federal government is considering flawed technologies whose development was halted for valid reasons.

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32-1

RECYCLING OF RADIOACTIVE SCRAP METAL IN DOE AND A CONCEPT FOR OVERCOMING THE CRITICAL IMPEDIMENT

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ABSTRACT

This paper addresses the need for recycling radioactive scrap metal in U.S. Department of Energy's (DOE) Environmental Restoration (ER) Program and potential actions to enhance recycling. The quantities of these metals currently on hand within the DOE complex and large quantities expected to be generated by the ER Program are described and discussed in terms of their asset value rather than merely as waste to be disposed of. The DOE has an opportunity for realizing substantial cost recovery utilizing secondary metals markets, with emphasis on long term savings, through decontamination and recycling of radioactive scrap metals. Also discussed is the critical impediment to achieving worthwhile recycling of these metals arising from the lack of release and decontamination standards established by regulatory agencies that would allow metal to be processed and released for unrestricted use. To aid in overcoming the critical impediment, a new concept utilizing a multi-tiered approach for returning these metals to the general metals markets is suggested. The tiered approach has a potential for not only realizing a direct monetary return, but that also could minimize the need for extensive decontamination of some radioactive scrap metals prior to recycling.

INTRODUCTION

The Environmental Restoration Program in the Department of Energy (DOE) was formalized and begun in earnest only a few years ago, and is still in its infancy. However, the pace is quickening as the program advances from the assessment phase into the remediation phase, and will accelerate as the Department continues the phase-out of its old plant facilities that are no longer needed. Dismantling and removing those facilities will result in massive quantities of materials requiring disposal, especially radioactive scrap metals. Historically, DOE production sites generated substantial quantities of radioactive scrap metal that were simply held in inventory rather than being disposed of, and many of those inventories are still on hand, awaiting a disposal decision.

All too often, considerations for disposal of radioactive scrap metals are limited to long-term storage or burial, especially the latter. Seldom is disposal of these

unwanted metals considered in terms of returning them to the stream of raw materials flowing into production processes, that is, recycling. However, recycling can solve several problems at once. In addition to returning the metals to the raw materials stream, recycling also will conserve natural resources by reducing the mining of virgin materials, and recycling can reduce demand for energy needed for processing virgin materials that would be used if recycled metals were not available.

There are many compelling reasons for disposing of radioactive scrap metal by recycling. Foremost are the volumes of metallic wastes on hand and the significant amounts expected from future remediation work. Any large quantities should be viewed as an incentive and a great opportunity for recycling. This is not to say that recycling the metals would be easy, particularly where volumetric contamination is concerned. Radioactive scrap metals present an excellent opportunity for technology development and application, from two viewpoints. One is the need for faster, more efficient techniques for surveying scrap metal in a wide variety of shapes to determine the levels of radioactive contamination. The other is the need for faster, more efficient technology for decontaminating the metal so it can be returned to productive use.

MAGNITUDE OF THE DOE SCRAP METAL PROBLEM

Precise quantity data are not available for the current DOE inventory of contaminated materials, and a survey is currently underway that it is hoped will put some boundaries around the problem of establishing reliable inventory estimates. However, existing estimates range from about 400,000 tons to over 1.5 million tons of radioactive scrap metal in storage at various DOE sites. The high end of this range resulted from an informal (and disputed) survey a few years ago. The resulting estimate broke down the inventory as follows: aluminum, 195,000 tons; copper, 38,000 tons; lead, 3,000 tons; nickel, 245,000 tons; and steel, 1,300,000 tons. If the same ratios were to hold for the low end of the range, the breakdown would be: aluminum, 43,797 tons; copper, 8,535 tons; lead, 674 tons; nickel, 55,027 tons; and steel, 291,980 tons.

Similarly, there is no current estimate of the annual generation rate of scrap metals within the DOE complex, but as the pace of major DOE facility decontamination, decommissioning and dismantling increases in the Environmental Restoration Program, the levels of scrap metal inventories are expected to increase substantially. Preliminary estimates indicate that the program will add at least 1.0 million tons of radioactive scrap metal to the DOE inventory. This is expected to be a low estimate. The conceptual planning for decommissioning the Oak Ridge Gaseous Diffusion Plant (ORGDP) indicate that this plant alone may generate 775,000 tons of radioactive scrap metal over a 10-year period. Similar quantities could occur at other major DOE sites.

The great majority of DOE radioactive scrap metal bears low levels of radioactive contamination. Thus, much of it is or will be available for decontamination and release for unrestricted reuse. The remainder could be at least partially decontaminated and put into a variety of industrial or nuclear uses where the material would not come in contact with workers or the public, or would do so in restricted ways.

COST TRADE-OFFS

The potential value of DOE scrap metal can be understood by reviewing a few metals, their typical secondary market prices, and the extrapolation of scrap-metal market values from the identified quantities as previously discussed.

METAL	MARKET PRICE	QUANTITY	MARKET VALUES (LOW-END EST.)	
Nickel	$6,000 per ton	55,027 tons		$ 330,162,000
Steel	107 per ton	291,980 tons	31,242,000	
Aluminum	600 per ton	43,797 tons	26,278,000	
Copper	1,500 per ton	8,535 tons	12,802,000	

These prices are displayed as nominal amounts because scrap metal prices change on a daily basis due to market factors. These prices are typical of levels in the American Metal Market 1991 series (1). Nonetheless, these data provide insight into the potential direct dollar value of the DOE scrap metal inventory. Also, these values may support development of appropriate expenditures on research and development for processing and decontamination methodology that could result in

alternative waste management approaches.
Market prices, however, are not the only important aspect of scrap metal recycling. Anyone considering the relative merits of recycling versus burial must also account for the costs of not recycling the scrap metal. The additional aspects include:

The value of natural resources conserved from not mining and processing materials represented in the scrap metal;

Environmental preservation and cost avoidance from not building or expanding scrap metal disposal facilities;

Avoidance of the costs of long-term storage of scrap metal wastes and maintenance of disposal facilities;

Avoidance of the costs of environmental restoration of subsequently contaminated storage areas.

Aluminum provides an example of the cost benefits available from recycling and the conservation of natural resources. Approximately four tons of raw bauxite ore are needed to produce one ton of aluminum and the nominal value for the raw ore is $20 per ton. On that basis, the DOE scrap aluminum inventory (low-end estimate) represents a potential resource conservation of 175,000 tons of bauxite ore at a cost avoidance savings of about $3.5 million. Additionally, there is a significant energy savings because processing scrap aluminum requires only 5% of the energy required for smelting and handling raw bauxite ore. Aluminum industry specialists have calculated that 15,000 kilowatt hours are used in producing one ton of aluminum from raw ore. Thus, use of scrap aluminum would save 14,250 kilowatt hours per ton or $6.24 million for the 43,797 tons of aluminum scrap metal, with power conservatively priced at $0.01 per kilowatt hour. Viewed from this perspective, the cost of not recycling DOE scrap aluminum is comprised of the scrap market value loss of $26.3 million plus the costs of raw bauxite ore ($3.5 million) and energy in production ($6.24 million), bringing the total to some $36 million. These costs do not consider the environmental impact of mining 175,000 tons of ore, nor the aluminum's replacement value if purchased new at new-metal market prices.

While such costs are substantial, they must also be considered relative to the alternative option of disposing of the metals as low level radioactive waste. There are two aspects to consider. Burial disposal costs for low level radioactive wastes are within a range from $60 up to $160 per cubic foot. Therefore, disposal costs for a scrap metal inventory of approximately 400,000 tons, assuming roughly 400 pounds per cubic foot after some volume consolidation, can be calculated in to be in a range from $120 million up to $320 million. Less efficient volume reduction will increase the costs. In the case of burial disposal of the 43,797 tons of aluminum scrap metal, assuming a density of roughly 200 pounds per cubic foot, disposal costs can be calculated in to be in a range from $26 million up to $70 million.

Note however, that this does not consider the potential cost impact if any of the radioactive scrap metal must be considered mixed waste due to contamination by any hazardous substances defined by the Resource Conservation and Recovery Act (RCRA). If this were the case, the cost for disposal would be likely to increase significantly, because the Federal Facilities Compliance Act requires that mixed waste be treated before land disposal. Alternatively, it is possible that the nature of the decontamination process and in particular, a melting process, would result in the waste no longer being required to be considered hazardous under RCRA and permissible to be disposed of as low-level radioactive waste. This would be the result of the melting process removing or oxidizing many possible RCRA constituents and the resultant waste may no longer meet the requirements of a RCRA characteristic waste. Consequently, this is a RCRA compliance benefit completely incidental to the scrap-metal melting process.

There are still further costs to be considered. These are the costs for long-term surveillance and maintenance at radioactive waste material disposal sites. Over the period of radioactive decay for even short half-life elements, these costs can be significant. The important point here is that these long-term costs can be completely avoided by recycling the metals.

The final key point that must be recognized on the subject of costs is that involving technology development for decontamination and recycling. When the types of costs described and discussed above are accounted-for (e.g., the aluminum example above), their total provides an indication of the amount that could be spent on technology development.

THE CRITICAL IMPEDIMENT: LACK OF STANDARDS

Given that recycling may be economically and ecologically attractive, the question naturally arises as to why recycling is not being implemented on a large scale. The most critical impediment confronting recycling of radioactive scrap metal is lack of definitive release standards. Radioactivity level standards are essential for both surface and volumetric contamination, to qualify the metal for release into the marketplace or to some restricted use. This problem requires action by the appropriate regulatory agencies (the Nuclear Regulatory Commission and the Environmental Protection Agency) before recycling of the scrap metals can implemented on a large scale in a publicly acceptable way. The cut-off often has been defined by concerns of institutional liability or adverse public reaction rather than direct risk reduction.

The U.S. Nuclear Regulatory Commission (NRC) has established release limits for surface contamination and has published them in its Regulatory Guide 1.86. These guidelines incorporate the "As Low As Reasonably Achievable" (ALARA) concept, which states that decontamination efforts should not stop when the stated release limits have been reached, but should proceed to decontaminate as far below those limits as is technically and economically feasible. Decontamination efforts that achieve such ALARA levels (below release limits) constitute successful decontamination. The result is that the material is considered to be safe for any use.

The DOE has adopted the NRC guidelines and incorporated them into DOE Order 5400.5, "Radiation Protection of the Public and the Environment." The NRC guidelines are not technically enforceable standards, although many entities (especially NRC licensees) treat them as enforceable standards. Faced with the uncertainties, especially the potential liability, associated with the lack of standards, even those who might be in favor of decontamination and recycling are opting for immediate and ultimate disposal, that is, burial.

Surface contamination is relatively easy to remove from metals, because it is limited to a few microns in depth. The shapes of individual pieces, however, can make surface decontamination a time-consuming task, and the more complex the shapes, the more tedious and time-consuming the task. Therefore, there is a need for more efficient technology. This is also true for the measurement of extremely low level contamination, both the capability and speed for practical application.

Unlike the case of NUREG 1.86 Guidelines for surface contamination, there are no guidelines or standards for volumetric contamination. Neither the NRC nor the Environmental Protection Agency (EPA) have resolved this issue and this lack effectively has precluded volumetric decontamination for recycling in this country. A few other countries have developed at least limited volumetric contamination standards (2)(3), but the lack of consistency, together with the lack of U.S. standards, has precluded the decontamination and recycle of radioactive scrap metals for trade on international markets. The fear is that metals imported into a country would not be sufficiently decontaminated before being converted into products that would then be exported from that country for unrestricted use. Worse would be the importation of presumably clean metal subsequently discovered to be contaminated.

It must be noted that both the NRC and the EPA are continuing their studies of the standards problem. However, no definitive answers are anticipated in the near future.

WASTE MINIMIZATION OPPORTUNITY

It is inescapable that some quantities of radioactive scrap metal will not be reusable and will have to be permanently disposed of, most likely by burial in engineered facilities. Our continuing objective should be to reduce to the absolute minimum the metal quantities that require permanent disposal. Active pursuit of this objective will aid in minimizing both the cost and number of disposal facilities that will be required. If we - all of us - do the job correctly, there will be only two types of scrap metal waste requiring permanent disposal. One type will be characterized by slag, dross, small shavings and grinding dust from scrap metal recycling processes. The other will be quantities of contaminated metal that are too small for cost-effective processing/recycling, even for re-entry into nuclear use.

The challenge of achieving this minimum scrap metal waste for disposal will require a significant departure from conventional wisdom for managing radioactive scrap metals. It will require that we implement a concept that will provide flexibility in reusing the metals and which recognizes two key factors: 1) there are great

quantities of metals that can be safely decontaminated and returned to productive use without undue restrictions; and 2) there are many industrial applications for metals that would require only partial decontamination and which would pose no threat to the public. One such concept is presented below.

THE TIERED CONCEPT FOR RECYCLING RADIOACTIVE SCRAP METALS

Radioactive scrap metals, because they are metals, could be recycled and utilized in many applications and products. Not all of these would require that the metals be decontaminated to levels that would permit release for unrestricted use. Therefore, it appears that a "tiered concept" for decontaminating for recycling would be very useful. Three basic tiers are suggested:

TIER 1: This would be a "below regulatory concern" level of contamination, i.e., a level below which the metals can be released for reuse without radiological restrictions. This means that the material is considered to be safe for any use. There would be no need for tracking or record keeping after the verification survey or sample measurements are completed and unrestricted release is authorized. Manufacturers would be free to utilize the metals in any products, including consumer products - household appliances, toys, tools, furniture, automobiles, etc., with no concern for liability for radioactivity.

TIER 2: This would be a non-regulated, "limited use" level of contamination which would allow the metals to be used in industrial or other pre-established specific applications. The standard could be set by the critical exposure group, most likely the workers, to assure their safety. However, certain worst-case scenarios assumed in Tier 1 would be eliminated by the nature of the limited use restrictions and the pre-established applications, and thus allow a higher level of contaminant concentration without risk of increased public exposure. In particular, consideration would be given to the remelting of the scrap after release as Tier 2 material to the specified user, such as a maker of construction rebar. Other examples of potential applications are 1) use of radioactive nickel as a catalyst in petroleum or other chemical refining; 2) use of the nickel in alloys for induced draft fans for hazardous material incinerators or power plants; 3) use of steel to make railroad rails or rolls for steel rolling mills ; 4) use of copper in selected electric power substation equipment, for example, transformers; 5) use of steel in pipe designated for cross-country petroleum pipelines; and, 6) use of various metals in heavy-duty industrial or other structural applications which go unattended for long periods e.g., bridges. The point is that they are all uses where people (workers or the public) would have little or no opportunity for exposure to the radioactivity. The list of such uses is limited only by the imagination of the potential user. It should be noted that the NRC has, through its licensing process, already exempted from regulatory disposal controls the use of uranium in fire detection units and thorium in finished aircraft engine parts, on the basis that the impact of the contamination is minimal and the cost of maintaining regulatory controls would be disproportionate. In general, however, this level would require that records be maintained as to the specific first use and application of the contaminated metals. The key to this tier is the recognition that certain conservative assumptions in Tier 1 can be easily overcome with limited restriction, that is, the first use of the material would be such that the otherwise unusable material can be reused in specific practical, but non-regulated ways. Tracking and record keeping would be required for the contaminated metal until it goes into the decontamination process, but no records would need to be kept after the residual contamination level has been verified and the end-use specified.

TIER 3: This would be a regulated, but non-controlled "nuclear industry use only" level, and would allow the direct reuse of the metals by either government or industry in nuclear applications, without restriction within the nuclear site boundary. Within this tier, metals would be restricted from off-site release as any suspect material is from a nuclear site, but available for use within the site without restriction, that is, outside the controlled areas. One example would be the manufacture of new waste containers which could be stored anywhere on site until loaded with radioactive waste. Other uses might be structures, piping, etc., i.e., direct re-use without further processing. Records would have to be maintained to document the on-site existence of the material, especially if sent off-site to a licensed melting, re-forming, or decontamination facility for re-working prior to the re-use, but would not be required within the site boundary.

One issue that needs to be resolved is the potential use of dilution processes, that

is, allowing the mixing of radioactive scrap metal with uncontaminated metal (new or scrap) in melting processes to reduce the final contamination level of the melt to either the unrestricted-use level or the limited-use level, i.e., Tier 1 or Tier 2. However, there appears to be no reason for not allowing - in fact, even promoting - the use of dilution to reduce contaminant concentrations in recyclable metals to acceptable levels.

SUMMARY AND CONCLUSIONS

The institutionalization of the proposed Tiered Concept rather quickly could lead to a strong, ongoing program of decontaminating and recycling DOE's existing and anticipated inventories of radioactive scrap metal. Its extension into the international arena, accompanied by agreement on equivalent regulations among all concerned nations, also could resolve the dilemma of international trade in contaminated metals. This approach is also consistent with the European approach of obtaining licenses for project or site specific applications, as opposed to the establishment of general national standards. In this regard, generators and scrap metal processing facilities could obtain licenses for specific recycle operations that would permit metal recycling at one or more of the approved Tiers. Licensing the operations should result in the ability to remove unwarranted conservatism in the development of generic standards.

There are a number of points to be remembered when considering the potential for recycling radioactive scrap metal. It has been shown that there is an opportunity for substantial cost savings when all costs, both near-term and long-term, are taken into account. There is a significant opportunity to introduce new technology for decontaminating the metals, especially to address volumetric contamination, whereby the metallic waste that would require permanent disposal could be decreased to an irreducible minimum. Achievement of this minimum is a desirable national objective that could be enhanced materially through implementation of tiered standards governing residual contamination.

Before making a decision to send radioactive scrap metal to a burial ground burial rather than recycling, there is a fundamental question that must be addressed: "Is the permanent disposal of large volumes of low-level contaminated metal in the nation's best interest?"

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THE DEPARTMENT OF ENERGY APPROVAL PROCESS FOR FREE RELEASE of VOLUMETRICALLY CONTAMINATED RADIOACTIVE SCRAP METAL

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ABSTRACT
Volumetrically contaminated metals are those metals which, through either neutron activation, accelerator activation, or the melting of surface contaminated metals, have radioactive contamination distributed throughout their volume. Unlike materials with only surface contamination, no Federal standards apply to the disposition of these materials by any means other than disposal as a radioactive waste. Department of Energy Order 5400.5, Radiation Protection of the Public and the Environment, provides for the disposition of such materials by other means in specific cases approved by the Assistant Secretary for Environment, Safety and Health.
This paper reviews three requests from DOE field organizations for approval to dispose of materials containing small quantities of radioactive volumetric contamination. One request utilized the format of an Environmental Assessment with the addition of an ALARA analysis. This request has been reviewed and approved by the Assistant Secretary, permitting the future sale of the material for unrestricted use.
This paper discusses the type of information included in the approved request and is not to be considered as formal guidance.
CURRENT REGULATORY POLICY FOR RECYCLING, REUSE, AND DISPOSAL OF NON-RADIOACTIVE WASTES
On July 3, 1990 the Nuclear Regulatory Commission (NRC) issued a policy statement on the disposition of materials with levels of contamination identified as "Below Regulatory Concern," or BRC. This BRC policy met with substantial opposition from both environmental groups and industry and was eventually withdrawn. In the Energy Policy Act of 1992 (1), Congress directed NRC to refocus its efforts and stated that "The policy statements of July 3, 1990 and August 29, 1986 relating to radioactive waste below regulatory concern, shall have no effect after the date of enactment of this act." NRC has undertaken an enhanced participatory rulemaking to develop cleanup standards for NRC-licensed sites. This standard will not address the release of materials in the same manner as the original BRC standard was intended. On January 26, 1994, the NRC issued their "Draft Radiological Criteria for Decommissioning.(2)" While these criteria do not apply to the release of contaminated materials, the Commission recognized that for decommissioned sites the possibility exists that materials left on site could subsequently be recycled. In its response to comments on the draft criteria during the public hearings the Commission states that:
"The Commission plans to consider separately the issues of how to deal with cases where the licensee proposes to release material containing residual radioactivity intentionally for reuse or recycle either as a part of decommissioning or ongoing operations. In the interim the Commission will continue to be (sic) review such actions on a case-by-case basis."
Although EPA stated publicly in 1987 that "We intend to establish criteria for identifying wastes with sufficiently low levels of radioactivity to qualify as `Below Regulatory Concern' (3)," to date no such standard has been developed. Current activities of EPA are focused on assessing the scope of the recycle metals market, the economics of recycling, identifying recycling options, and the risks and impacts from those options. Once these questions are answered then other issues regarding the acceptability and implementability of a standard will be answered.
In the absence of Federal standards governing the disposition of materials containing low levels of volumetric radioactive contamination, DOE must make decisions regarding recycling, reuse, and disposal of these materials in facilities not licensed to accept radioactive materials.
INTERNATIONAL POLICY
Several European countries have either adopted standards established by the International Atomic Energy Agency (IAEA) or have developed their own for recycling contaminated scrap metal in their countries. These various regulations are presented in Table I. The IAEA proposed to call such standards "clearance levels"

for releasing materials from regulatory control.

DOE POLICY

Unlike scrap materials with only radioactive surface contamination, for which Federal free release limits exist, no corresponding release limits exist when the contamination is an integral part of the material. Volumetric contamination may result by either activation of atoms in the material by irradiation in a neutron flux or by other particles in an accelerator, or when metals with radioactively contaminated surfaces are melted.

DOE policy on the release of materials containing "volume contamination", contained in DOE Order 5400.5 (4), permits such materials to be released if criteria and survey techniques are approved by the Assistant Secretary for Environment, Safety and Health. However, formal guidance on the approval process has not been issued. Figures 1 and 2 show what happens when metals of various thickness, having surface contamination at the maximum levels allowable under DOE Order 5400.5 (in disintegrations per second per 100 square centimeters), are melted, and how the resulting volumetric contamination (in picocuries per gram) compare to the proposed IAEA standards.

Since DOE Order 5400.5 was last revised in 1990 a limited number of requests for EH review and approval of activities that would lead to the release of radioactively contaminated materials have been submitted. These materials resulted from DOE research and development, and D&D activities.

Recently, the DOE Oakland Operations Office requested and received permission from the State of California Department of Health to dispose of limited quantities of copper, concrete and rebar, and oil that contained low levels of radioactive contamination. Although the concrete and rebar was disposed in a local landfill, OAK was informed that disposition of the copper and oil would have to wait pending review and approval by DOE Headquarters. These and other requests are described below.

Fig. 1.

Fig. 2.

Oakland Operations Office

On August 2, 1994, the Assistant Secretary for Environment, Safety and Health, approved a request to disposition 140 metric tons of high-purity (99.99%) copper by selling the material to a scrap metal dealer. The copper was used as coil windings in the 184-inch Cyclotron at Lawrence Berkeley Laboratory which was dismantled several years ago (5). It is currently stored in an outdoor storage area in wooden crates in seven foot long strips, four inches wide and a quarter inch thick. The copper contains 60Co (a gamma emitter) at concentrations ranging from 0.1 pCi/g to approximately 20 pCi/g, with an average concentration of 3 pCi/g. The copper also contains a small quantity of 63Ni at a concentration of 1.5 pCi/g. 63Ni is a beta-emitter and in the metallic form does not contribute to external dose. An additional 150 tons of auxiliary coil copper was found to have concentrations of 60Co greater than 20 pCi/g along with traces of 106mAg, and was disposed of as a radioactive waste at Hanford.

A second request, involving the proposed recycle of waste oil contaminated with tritium at a concentration less than the 20,000 pCi/g limit for drinking water, has been withdrawn.

Chicago Operations Office

In February, 1993, a document entitled "Radioactivity Release Criteria for Materials and Equipment" developed by Fermi National Accelerator Laboratory (Fermilab) was submitted to DOE for approval. This document does not address any specific material, but generically identifies the release limits that could be applied for volumetrically contaminated materials and the instrumentation necessary to assure compliance. Specifically, the document states that:

"Studies conducted with radioactive materials and calibrated g-emitting radioactive sources with energies between 0.5 and 1.3 MeV have indicated that 2000 counts (per minute) above background on such an instrument corresponds to exposure rates between 0.006 and 0.012 mR/h with an average of about 0.01 mR/h. Reasonably long-lived radioactive isotopes produced at high-energy accelerators (such as those at Fermilab) that constitute the nuclides in the equipment and material released are mostly 3H, 7Be, 22Na, 54Mn and 60Co. Except for 3H these isotopes are gamma emitters with energies between 0.478 keV and 1.33 MeV. The Bicron survey meter mentioned above is particularly sensitive to such energies of emission."

"The detection criterion stated here results in a minimum detectable specific activity of between 20 and 50 pCi/g (depending on the gamma ray energy) in bulk materials such as aluminum, concrete, or iron if irradiated uniformly."
This document is currently under review by DOE.

Paducah Gaseous Diffusion Plant

Another request is for the approval to sell of 9,300 tons of contaminated high-purity nickel from the Paducah Gaseous Diffusion Plant. One of the options being evaluated is to decontaminate this material in the United States at a NRC-licensed smelting facility for sale overseas for the production of stainless steel.
This request is presently under review by DOE.

OBTAINING DOE APPROVAL FOR RELEASE OF RADIOACTIVELY CONTAMINATED SCRAP

This section provides information on the documentation prepared to support the request for recycling of high purity copper. The intent is to review the type of information which was provided and a sample format for the request document.
It should be noted that while the basic documentation package was an environmental assessment (EA) (5), there is no formal requirement that an EA be prepared. However, because the NEPA process is well defined, it allows for a consistent methodology for both preparer and reviewer of these requests.
This EA was later supplemented with an ALARA Assessment (ALARA stands for As Low As Reasonably Achievable.) The content of these documents follows the format of a standard NEPA environmental assessment and the actual ALARA Assessment submitted. Additional information in italics is based on guidance provided by the DOE Office of NEPA Oversight (9).

Environmental Assessment Format

Introduction:

The introduction provides a short overview of what is included in the document. It identifies why it is being written; for compliance with the National Environmental Compliance Act of 1969 (42 U.S.C. 4321-43470 and DOE Order 5400.5, Radiation Protection of the Public and the Environment. It provides a short general description of the proposed action, the types of impacts, a general discussion of the alternatives, and other DOE policies considered in the evaluation, such as waste minimization.

Background:

This section contains a discussion of the origin of the material, how it became radioactive, what the concentration of the radionuclides are. The information provided in the text is general in nature with details provided in an appendix.

Purpose and Need:

The purpose of the proposed action and any additional benefits of that action are described. (NOTE: Under guidance provided by the Office of NEPA Oversight this section should not address the purpose of the proposed action. The section on Purpose and Need should discuss why there is a need to take any action and what the purpose of that action is. For example, the purpose and need of an action would be to disposition scrap metal that contains low concentrations of radioactive contamination by the most cost effective means. The purpose was not to recycle scrap metal.)

Description of the Proposed Action and Alternatives:

A subsection is devoted to the detailed description of each alternatives, starting with the proposed action. The description of the proposed action is slightly more detailed than the others. In this case the proposed action was to sell the copper to a local scrap metal dealer who would probably resell the material to a processor. Because of the sensitive nature of the material the EA identified the possible options available to the reprocessor; reuse as is, melt and refine with or without being added to other material, or possibly ship overseas for reprocessing. As with all EAs and EISs a discussion of the No Action Alternative was also included.

Description of the Affected Environment

This section contains a discussion of the existing environmental conditions relative to the general area in which the copper was stored, generally restricted to human health issues, and a discussion of the environment that could be affected by the proposed action and each of the alternatives. With respect to the environment, topics include:

Air Emissions - an estimate was made of the particulate emissions that would be associated with reprocessing the copper by melting.

Water Quality - Water use in cooling towers is the only water use identified associated with copper reprocessing but no specific releases were identified in the blowdown.

Energy Use - Energy use savings greater than 90 percent were identified as being realized by reprocessing as compared to mining and processing new ore into products.

Traffic - The number of shipments, seven, needed to transport the material to any one of three scrap dealers within ten miles of the site was identified. The risks associated with this number of shipments was reported to "have been analyzed and found to present an acceptable low level of risk."

Geology, Soils, Hydrology, Aesthetics, Waste - The discussion in this section was directed towards activities associated with the mining of new copper ore. These topics are not relevant with respect to the proposed action or any of the alternatives.

Natural Resources - A comparison was made between the quantity of copper to be recycled (140 metric tons) and the total quantity of copper produced in the United States annually and mined world wide annually.

Land Use - Land use as it relates to landfill space, should the material be disposed of in a local landfill, was identified as part of the affected environment.

Existing conditions relating to human health were the potential exposures from the 60Co in the copper and from background radiation and other sources of radiation. Dose rates were estimated based on the quantity of 60Co if taken as a point source. Dose equivalents are calculated at 1, 30 and 50 years depending on the alternative (actual doses were in the impacts section). Self shielding considerations were evaluated.

The discussion on Background Radiation and Doses is very complete and provides valuable information which was used for comparative purposes.

Environmental Consequences of the Proposed Action and Alternatives

As stated in the opening paragraph of this section in the EA, "This section addresses the potential environmental and human health effects of the proposed action and the five alternatives. Direct impacts, as well as indirect impacts (such as averting air emissions from mining and smelting an equivalent quantity of copper) are assessed." Normally, only the impacts associated with the proposed action are assessed in an EA, and a comparison of impacts of the alternatives is performed in an EIS.

(NOTE: The Office of NEPA Oversight has stated in their "Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements", May 1993, that:

"Unlike EISs, where generally the depth of quantitative analysis is approximately the same for each alternative analyzed in the range of reasonable alternatives, EAs often can focus the quantitative analysis on the proposed action; that is, discussions of alternatives in EAs generally can be qualitative. However, certain EAs may need to identify and analyze more alternatives in the range of reasonable alternatives and provide more in-depth analysis than usual (e.g., greater quantitative analysis). These EAs generally deal with proposals where there is heightened technical controversy surrounding potential impacts from the proposed action or where there is otherwise greater potential for significant environmental impacts from the proposed action.")

Until applicable standards are promulgated which are specific to volumetrically contaminated materials, EAs prepared for the purpose of obtaining DOE approval for the recycling of such materials will probably be considered as falling into this category of "heightened technical controversy." (See above quoted section.)

In addition to the seven categories of affected environment (Air Emissions, Water, etc.) for which the impacts of each alternative are assessed, assessments are made of the impacts for each alternative for the following topics:

Maximum Lifetime Individual Worker Excess Fatal Cancer Risk
Maximum Collective Transport Worker Excess Fatal Cancer Risk
Maximum Lifetime Individual Public Excess Fatal Cancer Risk
Maximum Collective Public Excess Fatal Cancer Risk
Maximum Collective Public Excess Fatal Cancer Risk from Transportation

Both the exposure, in person-rem, and the risk, expressed as excess fatal cancers, are identified. Exposures were calculated using the NRC IMPACTS Code. Doses to the public were calculated assuming that the copper would be used in home electrical wiring and plumbing.

ALARA Considerations

The EA contains a section on ALARA in which an estimate of the "Total ALARA Cost" is made. This "cost" is defined as follows:

"The ALARA Cost is the estimated value of the sum of the collective dose equivalents using the DOE recommended value of $10,000/person-rem. This value allows comparison between the `cost' of the radiation exposure and other costs and benefits."

In the case of the copper the maximum dose associated with the proposed action is 0.072 person-rem to the public, for an ALARA Cost of $720. (NOTE: In the licensing of commercial power plants, NRC used an unofficial ALARA cost of $1,000 per man-rem. This means that if the cost of reducing the population dose equivalent by 1 manrem/yr was less that $1,000 it was considered cost effective and should be incorporated in the design. If, on the other hand, that 1 manrem/yr dose reduction cost more than $1,000 it was not cost effective.)

In response to the EA, DOE prepared an ALARA analysis in which they discussed the costs, or cost savings, associated with each alternative. While the proposed action has the highest exposures to both the public and to workers, both exposures are so small as to have no consequence in terms of health effects. Their analysis stated: "The relative insignificance of the collective dose for all options in this example eliminates the health effects as a significant factor in deciding on a course of action." The proposed action results in significant cost savings compared to the alternatives. The DOE analysis states that cost should not be a consideration in selecting the preferred alternative or proposed action unless the impacts are assessed as essentially being equal amongst the alternatives.

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32-3

ESTABLISHING REUSE POLICIES FOR RADIOACTIVE SCRAP METAL

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ABSTRACT

Radioactive scrap metal (RSM) is a waste management issue with implications for several industrial and governmental organizations. Establishing effective and acceptable reuse policies for RSM may require a stakeholder focused involvement process. Two examples of stakeholder involvement include the Nuclear Regulatory Commission's (NRC) enhanced participatory rulemaking, and the Department of Energy's (DOE) Recycle 2000 Initiative. Both incorporated important stakeholders early in the

developmental stages of policy making.

INTRODUCTION

The end of the Cold War has fueled tremendous forces of change within DOE. "With the shift in the defense mission, a large number of new sites and facilities will be added to the inventory of surplus facilities and inactive sites during the next 20 to 30 years" (1).

In 1989, the Environmental Management (EM) division of DOE was created with a goal of fostering more environmental responsibility (1). A growing cultural change is brewing within DOE. After over 40 years of primarily a military mission, the nations nuclear weapons complex needs to be cleaned up. How to effectively do so is a task of enormous complexity.

An important element of DOE's environmental restoration and waste management program is the recycle and reuse of RSM. RSM presents many technical uncertainties. For example, although the question, "how clean is clean" in regards to radioactive contamination generates many answers, there is little consensus among experts. Moreover, there is a lack of adequate health risk standards to guide the development of acceptable RSM policies.

Even more important than the technical unknowns may be the political uncertainties. In the 1980s, the NRC issued a policy statement known as "below regulatory concern" or simply BRC.

Section 10 of the Low-level Radioactive Waste Policy Amendments Act of 1985 requires the NRC to develop standards and procedures for expeditious handling of petitions for rulemaking to exempt disposal of radioactive waste determined to be below regulatory concern (2).

The BRC policy met with vocal opposition from a coalition of interest groups, and concerned members of Congress. Both the Environmental Protection Agency (EPA) and DOE expressed concern about the BRC policy in comments submitted to the NRC. The BRC initiative eventually was revoked by Congress in the Energy Policy Act of 1992.

This paper first describes the politically charged atmosphere that surrounds complex and controversial issues such as radioactive waste. Like other areas of radioactive waste management, the establishment of effective and acceptable RSM reuse policies likely will be the focus of intense political debate. The second part of this paper describes stakeholder involvement in policy decisions. In the search for effective RSM policies, decision makers should consider more proactive methods of incorporating stakeholder input into the policy making process.

METHODS

In 1992, the Energy, Environment, and Resources Center (EERC) commenced a research project on RSM, with a particular focus on stakeholders. Several reports and articles have been written during this period, providing a foundation for our current effort. Our methods for acquiring information include a review of the relevant literature, interviews with important policy actors, a survey of executives in the steel and scrap metal industries, and attendance at numerous conferences, forums, and symposiums.

SCRAP METAL AND RADIATION

In this paper, RSM is defined as those metals that either emit measurable levels of radioactivity or that potentially could be radioactively contaminated. We feel it is prudent to classify potentially contaminated metal as RSM until the extent of contamination is proven otherwise

RSM is generated from a number of sources. In accordance with the DOE Weapons Complex Scrap Metal Inventory Report, DOE controls roughly 400,000 tons of RSM. An additional 600,000 tons of RSM is anticipated from the dismantlement of the Department's three gaseous diffusion plants (3). Figures presented at various conferences and in a number reports suggest that there is some disagreement over the amount of RSM that actually exists at DOE sites. However, even the most conservative figures indicate that an enormous amount of RSM already is stockpiled with much more likely to accumulate through ongoing and future decommissioning and decontamination (D & D) operations at federal facilities.

The nuclear power industry also produces a certain amount of RSM during routine maintenance and retrofit operations. Roughly 5,000 tons of RSM from nuclear power plants is recycled annually (4). Many nuclear power plants are scheduled for decommissioning in the coming years. Should widespread D & D of nuclear power facilities commence, a significant amount of RSM could become available.

A third source of RSM results from naturally occurring radioactive materials (NORM).

The petroleum industry is a major generator of NORM contaminated materials, primarily piping and oil extraction equipment. A study conducted for the Petroleum Environmental Research Forum (PERF) focused on the potential advantages and disadvantages of recycling NORM-contaminated equipment (5). During the 1980s, the oil and gas industry recognized with growing alarm that elevated levels of radiation could be detected on piping and oil processing equipment. "In response to such concerns, facilities in the United States have been characterizing the nature and extent of NORM in oil and gas production equipment, evaluating potential exposures to workers, and developing methods for properly managing such low specific activity wastes" (5). No clear numbers are available on the total volume of NORM contaminated metals.

The scrap metal recycling and metal manufacturing industries have a different perspective on RSM. Representatives from these industries have expressed concern over radiation in the scrap metal stream. Radiation potentially presents serious health risks to a facility's workers. Moreover, several facilities have smelted a source of radiation, resulting in multi-million dollar decontamination operations. The Institute of Scrap Recycling Industries (ISRI) and the Steel Manufacturers Association (SMA) have lobbied for tougher controls on radiation sources.

Several waste management companies view RSM as an opportunity. Scientific Ecology Group (SEG), American Ecology, Manufacturing Sciences Corporation (MSC), and Alaron Corporation have contracted with DOE in projects involving RSM. A trade association representing these and other industries is currently in the early stages of formation.

THE POLITICAL ENVIRONMENT

The development of public policy in the United States was never meant to be quick and easy. The Constitution outlined a republican democracy through which the tumult of public opinion, public participation, and politics would be canalized. The output hopefully would be good public policy.

The process of public policy development has intrigued many observers of the American political system. Walter A. Rosenbaum offers one model of the public policy process in his excellent work, Energy, Politics, and Public Policy. In his view public policy develops through a series of stages: agenda setting, legitimation, implementation, evaluation, and termination.

Policy implementation is "the manner in which a policy is translated from the language of law into action" (6). The implementation stage of the public policy process is arguably the most difficult. The number of policy actors involved (including nongovernmental groups), confusing or vague legislation, fiscal constraints, political demands, and overlapping authority between regulatory institutions are just a few of the hurdles that must be overcome before implementation of a policy can proceed.

The next few sections describe in more detail, developments that are affecting the public policy process. In the latter part of the twentieth century the establishment of effective radioactive waste management policies has become an increasingly difficult endeavor.

GROWTH OF GOVERNMENT

On the opening page of his book, American Public Policy: Promise and Performance, B. Guy Peters discusses the growth of government.

> Government in the United States has grown from a small, simple "night watchman state" providing defense, police protection, tax collection, and some education into an immense network of organizations and institutions affecting the daily lives of all citizens (7).

The explosive growth of government is primarily a post World War II phenomena. New demands for social programs, health and safety standards, as well as energy and environmental concerns resulted in legislative and regulatory initiatives. Further, new governmental institutions at the local, state, and federal levels evolved from these demands. Moreover, the growth of government has strained the nation's fiscal resources and has often limited programmatic success.

The growth in government and certain governmental reform efforts have led to a "diffusion of political authority." Several federal agencies including NRC, EPA, DOE, the Department of Defense (DoD), and the Department of Transportation (DOT) maintain certain responsibilities in the radioactive waste policy arena. Moreover, state agencies are responsible for overseeing radioactive wastes. The number of regulatory institutions has led to concurrent and sometimes overlapping

jurisdiction.

EXPANSION OF POLITICAL ACTORS

Another development of the past three decades is an expansion of political actors. The constitution produced a "benign political climate" in which citizens were free to form organized groups and petition the government (6). Over the course of the last thirty years citizens have done just that in incredible numbers.

The explosion of environmental advocacy organizations, and antinuclear groups has added a new dimension to policies related to nuclear technology. Public Citizen, Inc., and a coalition of groups led the charge against BRC. The coalition levelled a lawsuit against NRC in 1990 claiming failure to conform with several national laws. The case was eventually deemed not ripe for review; however within two years BRC implementation was suspended.

CRISIS OF LEGITIMACY

Possibly the most troublesome changes in the American political environment are the low levels of trust citizens express towards their government. Polls have indicated that distrust of the federal government is widely diffused among cross-sections of American citizens (8). More specifically public confidence in science and technology has gradually declined.

During the last 15 years the relationship between the general public and those who create, manage, and profit from science and technology has gradually soured. A litany of ecological disasters and technical failures - from Love Canal to the space shuttle Challenger explosion to the Chernobyl nuclear meltdown- has turned many Americans away from their technological optimism. More important, Americans have lost their faith that those who control technology will do so competently and in the public interest (9).

STAKEHOLDER INVOLVEMENT AND THE NEW POLITICAL ENVIRONMENT

The new dynamics of the public policy process, almost certainly ensure that the development of public policy in complex and controversial areas such as radioactive waste will be exceedingly difficult. Thus, successful development and implementation of RSM policies may require a more proactive means of involving interested parties. A stakeholder is defined as "any group or individual who can affect or is affected by the achievement of an organization's objectives"(10). In this paper we adopt this definition with clarification. Two statements elevate "stakeholder" to a more practical level for the RSM issue.

First, a stakeholder must have a legitimate interest in the development of RSM reuse policies. For example, scrap metal brokers have an important economic interest. Radiation in the scrap recycling stream can result in enormous financial difficulties for scrap dealers. Also, citizens living near a decontamination facility are another set of stakeholders with a legitimate interest.

A second criteria suggests that a potential stakeholder must have knowledge and experience with an issue. For example, the International Atomic Energy Agency has recommended safety standards for the recycle and reuse of radioactive materials. Some environmental groups were adamant opponents of the Nuclear Regulatory Commission's "below regulatory commission" policy. Moreover many of these groups have been active participants in the enhanced participatory rulemaking process for site decommissioning standards.

In this paper as with previous reports we loosely identify three general categories of stakeholders for RSM reuse. Table I outlines the stakeholders.

As Table I illustrates, stakeholders for RSM reuse policies incorporate a diverse array of organizations, institutions, and industries. Moreover, the listing above incorporates those stakeholders with a legitimate interest and relevant experience. Several of the stakeholder categories identified above are general descriptions of a wide array of individuals and organizations. For instance, "state and local radiation protection bureaus" could be represented by not only individuals from particular states but also delegations from the Conference of Radiation Control Protection Directors (CRCPD) or the Organization of Agreement States. Scrap metal companies are represented by the Institute of Scrap Recycling Industries (ISRI). Trade associations also represent the different manufacturers of metals. A decision maker should carefully identify all relevant stakeholders before proceeding with an involvement process.

Stakeholder involvement procedures have been utilized in a number of ways for the development of different policies. Two examples are particularly relevant to the establishment of RSM reuse policies: the NRC's enhanced participatory rulemaking

and, the Recycle 2000 Initiative.

ENHANCED PARTICIPATORY RULEMAKING

In 1992, following the demise of BRC and a short lived consensus building effort, the NRC initiated an enhanced participatory rulemaking (EPR) process. The EPR was designed to facilitate the development of site decommissioning criteria for NRC licensed facilities.

The objective of the rulemaking is to enhance the participation of affected interests in the rulemaking by soliciting commentary from these interests on the rulemaking issues before the NRC staff develops the draft proposed rule ... the Commission believes that this will be an effective method for illuminating the decision making process on complex and controversial public health and safety issues (11).

NRC sponsored seven workshops around the United States and the meetings were facilitated by the Keystone Center. Participants included representatives from environmental groups, public interest organizations, local community groups, industrial and trade association representatives, state and local governments, the medical community, Native American tribal representatives, officials from federal agencies, and labor unions.

RECYCLE 2000 INITIATIVE

At the July 1994 Workshop on Radioactively Contaminated Scrap Metal hosted by the Energy, Environment, and Resources Center and Oak Ridge National Laboratory, the Recycle 2000 Initiative was unveiled.

The Department of Energy is considering a policy which would encourage the reuse of Office of Environmental Management (EM) generated radioactively contaminated materials within the EM program (12).

A stakeholder workshop was held in Denver, Colorado during December, 1994. DOE invited participants from a number of different stakeholder organizations. The final participants list includes representatives from environmental and public interest organizations, the metals industry, management and operation (M & O) contractors, academia, the medical community, decontamination industries, labor unions, and federal and state governmental agencies. With the Recycle 2000 Initiative, DOE has radically shifted its decision making process. Instead of interested parties reviewing a virtually completed policy, stakeholders were given opportunity to comment and help develop the Recycle 2000 Initiative from conception.

CONCLUSIONS

Cyril L. Comar eloquently expresses the gulf that often exists between risk analysis and risk acceptance:

Society is becoming increasingly well informed and anxiety-prone about technology-associated risks which leads to desire for their elimination. The logical and traditional approach is first to estimate the risk, a scientific task. Then comes the issue of risk acceptance, a most difficult step -moving from the world of facts to the world of values (13).

Establishing effective RSM reuse policies is as much a scientific and a technical process as it is a political undertaking. History has shown the difficulty that regulatory agencies can face when trying to establish effective policies for managing radioactive waste.

Proactive involvement of affected interests is one means of combating the difficulties of policy implementation especially those involving highly complex and exceptionally technical issues. However this statement must be viewed in light of certain caveats. A proactive participatory environment likely will entail greater resource expenditures in time and money. Moreover, the debates over reuse policies for RSM could run into communication problems. Nuclear technologists and scientists work in a complex field and converse in a highly specialized language. The possibility of communication problems is very real. Finally, BRC is still on the radar screen of many nongovernmental environmental organizations. RSM reuse policies will inevitably stimulate comparisons with the failed BRC policy.

Stakeholder involvement processes offer no guarantees of success. However, given the past record of regulatory attempts to establish radioactive waste management programs, stakeholder involvement methods may offer the best means of establishing effective and acceptable RSM reuse policies.

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32-4

DEVELOPMENT OF NATIONAL STANDARDS FOR COMMERCIAL REUSE OF BUILDING CONSTRUCTION PRODUCTS AND RECYCLED METALS FROM DECOMMISSIONED DOE FACILITIES

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ABSTRACT

The American Society for Testing and Materials (ASTM), with it's membership open to all organizations and individuals, is a leader in the development of voluntary, consensus standards on characteristics and performance of materials, products, systems, and services. ASTM administers the U.S. Technical Advisory Group (TAG) to the International Organization for Standardization (ISO) Technical Committee 207 on Environmental Management. ASTM is currently developing a number of national standards on environmental management likely to affect DOE facilities: assessment, waste management, biological effects, environmental fate, and pollution prevention, reuse, recycling, reclamation and environmental efficiency measurements, and lifecycle analysis.

In accordance with the 1993 Executive Order, recycling by Federal facilities is encouraged in response to the Pollution Prevention Act of 1990. This paper focuses on development of ASTM standards for recycling and reuse of radioactive contaminated materials and facilities. ASTM Committees interested in DOE facilities decontamination and decommissioning have targeted low-level contaminated and potentially nonhazardous commercial reuse of building construction products and recycled metals for the first standards to be developed with open participation of government, public, and academia via the ASTM process. Among the participating committees developing these standards include:

ASTM E 50 on Environmental Assessment. Subcommittee E 50.03 on Pollution Prevention, Reuse, Recycling and Environmental Efficiency;

ASTM E 10 on Nuclear Technology and Applications. Subcommittee on Radiological Protection for Decontamination & Decommissioning of Nuclear Facilities and Components;

ASTM D 34 on Waste Management. Subcommittee D 34.07 on Mixed Waste; and

ASTM C 26 on Nuclear Fuel Cycle. Subcommittees C 26.05 and C 26.07 Methods of Test and Waste Materials.

INTRODUCTION

Cooperation among nongovernment technical standards setting organizations, both national and international, offers the American Society for Testing and Materials

(ASTM) opportunity to show leadership in world standards with DOE. Technical and management standards packages to support federal facilities represent an immediate opportunity. Members of an ASTM Inter-Committee Task Force (ITF) met on October 7, 1994 with the Department of Energy's (DOE) Technical Standards Program. A possible new leadership role for ASTM in this regard was favorably received and the task force has held subsequent meetings to initiate these activities.

APPROACH TO DEVELOPMENT OF ASTM/DOE TECHNICAL STANDARDS

The overall goal of the newly formed ASTM Inter-Committee Task Force (ITF) is to provide DOE with nongovernment standards packages that integrate related technical and management standards targeted to high priority DOE issues/problems.

This goal is just beginning to be accomplished by various ASTM committees working together to meet identified DOE standards needs. The first proposed project for the newly formed ASTM ITF is "commercial reuse of building construction products and recycled metals from decommissioned DOE facilities"; this group of ASTM standards for DOE may serve as a model for ASTM "Teams" to meet needs of other federal organizations. This first project is being developed using the following phases:

Phase IStandards Project and Product Definition;
Phase II"Test Case" DOE Standards Packages;
Phase IIIBuilding ASTM-DOE Partnerships; and
Phase IVParticipation by Other Federal Agencies.

Phase I - Standards Project and Product Definition

In Phase I, the ITF has begun to define the approach, interfaces, specific work within and the format for the product standards packages. Phase I includes planning and further development of the model approach to implementing this activity. The ITF has begun the following activities:

Working with DOE-TSP to identify DOE program needs.
Working with DOE-TSP to define an interdependent process for developing the standards packages.

Development of a recommended methodology to evaluate the applicability of non-government standards (e.g. ASTM, ANSI, ISO) and DOE standards using consistent decision criteria. Discriminators such as technical, economic, legal, and availability will be structured into a decision-focused spreadsheet.

Interface with ASTM regarding efforts to identify possible inclusion of non-ASTM standards in the packages.

Coordinating with the ASTM Publications, Marketing and Information Technology Division the "design" of a format for the product standards packages.

Phase II - Providing "Test Case" DOE Standards Packages

In Phase II, ASTM standards packages for additional "test case" activities identified by DOE may be developed. Inter-committee teams will combine existing ASTM technical standards with existing or new management standards into standards packages specifically designed to meet identified "priority" DOE needs. Each team will include at least one ITF member. The "drafting" teams will likely be DOE/contractor employees who are members of the involved ASTM committees. DOE may be requested to identify a programmatic point of contact for the team leader and, if necessary, additional non-ASTM members to participate. To accomplish this, the teams will:

Select "test cases" to be pursued.
Develop process descriptions, including diagrams if applicable, which identifies specific steps within the target activities where standards may be an asset.
Review standards for application and develop abstract information.
Recommend accelerated standards development where there are gaps.
Facilitate DOE decision-making about standards applicability and the need for standards development.
Manage and participate in ASTM standards development including accelerated standards.
Identify additional standards development with external standards organizations.
Assemble standards packages for the target activities.

Phase III - Building ASTM-DOE Partnerships

In partnership with DOE TSP, ASTM task forces may in the future expand this approach to general application throughout DOE. The ITF and DOE-TSP may work together to update the model as a result of the Phase II activities. With a proven, functioning model in place, ASTM and DOE may in the future develop funded mandates and/or working arrangements for continued application throughout DOE. Additional

activities that may be required to finalize the model might include:
Developing detailed descriptions of the process including roles and responsibilities and protocols for interface among ASTM, involved committees, and DOE.
Defining the means for matching DOE "project" needs with appropriate committees.
Determining the best options for project coordination and communication.
Institutionalizing of any contractual and financial protocols by DOE and ASTM.
Identifying training needs and developing training programs for DOE/contractor staff. Development would be coordinated with ASTM Standards Technology Training.
At this point, general application of the model would be expected to be implemented throughout DOE. The extent of this implementation will be in direct response to DOE program needs, interest, and (where required) funding. The ITF will be supported by ASTM committee members and identified DOE/contractor staff.

Participation by Other Federal Agencies

If the ASTM/DOE standards packages are successful, ASTM intercommittee task forces may extend the development of standards packages to other federal agencies. This may require adapting or customizing the model for other identified agencies.

Project Management

Don Sandberg, Committee Chairman of C-26 (and a DOE contractor) is currently coordinating our ITF. In this role, Mr. Sandberg has met with DOE's Technical Standards Program staff to identify and prioritize DOE needs. The ITF will be supported by Pat Barr, ASTM Headquarters staff, in identifying ASTM Committees and resources to review standards packaging and accelerate standards where important gaps are identified. Members of the Inter-Committee Task Force are:

Gwen Eklund, ASTM E-50 Vice Chair (a DOE contractor), provides liaison on any environmental management standards that may be appropriate and provide input into pollution prevention/recycle reuse planning for DOE sites and facilities.

Bill Gulledge, ASTM E-50 and E-51 Executive Committee member and Board member ISR (former DOE employee), who participates in ISO and other national standards organizations provides technical and strategic advice on coordination of Task Force activities internal and external to ASTM and the involved committees.

Dick Meservy, ASTM E-10.03 Subcommittee Chairman (a DOE contractor) represents specialized knowledge of a DOE Site Contractor and specialized knowledge of nuclear applications that can be expected to be targeted by DOE as high priorities.

Jim Satterfield, ASTM E-51 Chairman who provides liaison on environmental risk management issues, and, as needed, serve as an "on-site" liaison to ASTM headquarters including attendance at meetings to represent the Task Force in Philadelphia.

Nancy Trahey, Past Chairman of ASTM Board of Directors provides senior technical review of process and liaison with existing ASTM governing bodies.

Srini Venkatesh, ASTM D-34.07 Subcommittee Chairman (a DOE contractor) represents specialized knowledge of a DOE Site Contractor and specialized knowledge of mixed nu clear/toxic waste management issues that have already been acknowledged by DOE as high priorities.

This ITF is combines ASTM leadership with DOE/contractor staff to achieve the proper liaison between non-government standards organizations and those with the internal knowledge of DOE's needs. This task force approach also combines quality and cost effectiveness with strategic focus on priority DOE activities.

STATUS OF STANDARDS DEVELOPMENT FOR COMMERCIAL REUSE OF BUILDING CONSTRUCTION PRODUCTS AND RECYCLED METALS FROM DECOMMISSIONED DOE FACILITIES

Thus far, five ASTM Committees have agreed to participate in the development of standards development for commercial reuse of building construction products and recycled metals from decommissioned DOE facilities. Figure 1 illustrates some of the likely technical standards areas for defining necessary technical standards. In some cases, these initiatives will require only the grouping of existing standards into the new DOE format. Table I, B shows examples of some of the environmental, nuclear, and decommissioning ASTM standards that already exist that may be applicable to this package. Table II, B shows examples of ASTM standards under development that may be applicable to the recycled metals and reuse of building construction products. ASTM E 50.03 on Pollution Prevention, Reuse, Recycling and Environmental Efficiency has begun to develop industry end use requirements for reused and recycled materials; these standards also will have to be reviewed by the committees on building construction materials.

Fig. 1.

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EFFORTS TO STANDARDIZE LOW LEVEL RADIOACTIVE WASTE CONTAINERS TO BE MADE FROM RECYCLED RADIOACTIVE SCRAP METAL

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ABSTRACT

Containers used by the U.S. Department of Energy (DOE) for transportation, storage and disposal of low level radioactive waste are widely varied in design and cost. Optimization and standardization of these products will yield a broad range of cost savings, including bulk procurement costs, reduction of redundant design costs, improved handling procedures and equipment, and minimization of disposal void volume. The current move away from plywood strong-tight packages into metal Type A packages, coupled with efforts to recycle radioactively contaminated scrap into metal packages for use within DOE provides an opportunity to introduce refinements. Use of a metric volume container is preferred for bookkeeping purposes and also to comply with DOE Order 5900.2, which requires system-wide adoption of metric units. Historically, low level waste containers have been 4' x 4' x 8' dimensions (with variations). Existing DOE facility capacity for recycling metal includes rolling mills that have 42" wide capacity. If plate were produced on such a mill, a typical container based on a four foot wide dimension would require an additional section to be welded onto plate stock to meet requirements. While greater than 90 percent of most radioactive material can be removed during the recycling process, the potential for residual contamination to exist in material produced from recycled radioactive scrap will dictate the degree of personnel protection required during welding and fabrication to assure that exposure is "As Low As Reasonably Achievable" (ALARA). Designing packages to a metric standard (39.37"=1 meter) would minimize the need to weld plate sections together for fabrication, thus reducing ALARA concerns. Design of products that have less void volume associated with access points for handling will minimize the final disposed volume of the waste package. These concerns are being considered by a working group with representatives from several DOE sites. Implementation options currently underway include development of one or more standardized containers and revision of waste acceptance criteria to ensure the widest use of such containers. This paper discusses the issues raised concerning implementation of recycling radioactive scrap metal into standardized containers and the proposed solutions posed by the working group.

DISCUSSION

If a nation were to prosecute a war by arming its warriors with perhaps a dozen different kinds of rifles, supply, and repair of these weapons would become an unmanageable quagmire. If each branch of the armed services and each unit procured their weapons according to their own specification of "rifle, anti-personnel," such a quagmire might come to pass.

Waste generation, shipment, and disposal bears no fundamental similarity to fighting a war other than the fact that the government pays for it. Within the Department of Energy, a variety of wastes are generated at its 49 sites. The diversity of material being disposed is phenomenal. Similarly, the diversity of packages being used to store, transport, and dispose of these materials is phenomenal. The single largest disposal site in the DOE system has documentation for receipt of 27 different packages. The fundamental question arises as to whether the number of packages is clearly consistent with the number of waste types.

If one were to survey the packages, it becomes obvious that essentially, there are drums and there are boxes. Drums are generally 22-1/4" diameter by 33-1/2" height, presumably an artifact of some oil company collusion in the early days of near monopoly.

Boxes, on the other hand, are 48" x 54" x 96" or 84" or 72" or 28" x 48" x 96." Some are plywood while some are steel. Some are intended to meet DOT Type A, and some are "strong, tight."

Designs are often vendor-dependent, and, of course vendors are bid-dependent, so the same specification may be filled by one vendor in one procurement and by another in the next procurement with different designs.

A variety of people within the DOE system have observed that there is little

justification for this degree of diversity. If drums can be standard, why not boxes? In an ideal world, all sites could use one product to ship, store and dispose their wastes. Most low level waste is low density material whose longest dimension does not exceed six feet.

The Westinghouse GOCO Waste Minimization committee led by Dave Zigelman of Savannah River began work on trying to unify the procurement of boxes under one specification in 1993. This committee was composed of waste minimization personnel from Savannah River, Hanford, West Valley and Westinghouse Idaho Nuclear Company (WINCO).

The GOCO group produced a specification for a nominal B-25 type DOT 7A metal box with 48" x 54" x 72" dimensions. This container will be procured for Savannah River and other Westinghouse GOCO operations as desired. Efforts to approve a specification for a cargo container (also known as a Sea-Land) 8' x 8' x 20' were also pursued. At the time of this writing, the Sea-Land specification was not completed.

In parallel with the GOCO work, the Westinghouse Hanford Waste Management group designed a family of metal DOT 7A containers. This "family" procurement would allow waste generators to buy boxes of several different dimensions without going through the approval process for each design. This family included nominal B-25 dimensions, as well as long narrow boxes to accommodate 20 to 40 foot long tank lances. The family specification is not a product specification, and the vendor may offer the closure design of his choice, for example.

The decontamination and decommissioning (D&D) group of EG&G Idaho funded the efforts of Don Kenoyer and Dale Wells to develop a new generation of containers. The Idaho design is intended to replace the current plywood 4' x 4' x 8' boxes. Idaho has several types of waste that are stored rather than disposed, so this metal box will be coated with a thermoplastic layer to protect it from the elements.

The Idaho work became broader when the idea of complex-wide standardization was suggested in early 1994. The WINCO Metal Recycle group expressed the idea that metal recycle could be benefitted by standardizing a product that could be made using recycled contaminated scrap. Bruce Becker of Nevada Test Site and Ron Pope of Oak Ridge confirmed that there was a broad-based interest in such a concept. An April 1994 meeting was held in Salt Lake City to test the waters of fleshing out this support. Ultimately, 4 meetings were held in 1994, attended by people from Fernald, Hanford, Idaho, Los Alamos, Rocky Flats, Sandia, Savannah River.

The outcome of these discussions was that although each site has its own interest in the box that they use, at least 85% of the waste generated throughout the complex could be stored, transported and disposed in a single design. Existing designs should be revised to minimize or eliminate the amount of void space that the container represents to maximize effective use of the disposal site.

With respect to structural strength, there are some discrepancies. Some sites stack boxes 4 high. Some sites stack 5 high. If the Savannah River design is loaded to its 5,000 lb capacity, the bottom box in a 5 high stack must support 25,000 lbs, although it is tested to 20,000 lbs. Idaho National Engineering Laboratory Radioactive Waste Management Complex (RWMC) uses the standard of 4,000 lbs per square foot as a strength limit. A plywood RWMC box should meet a 12,800 lb load. This means RWMC would need to revise their operations if they were to adopt the Savannah River design. The 12,000 lb limit used at the RWMC would crush the SRS B-25 if used in a 5 high stack.

The plywood box used at INEL and some other sites suffers from another of the flaws that were identified. It has significant amounts of unusable void space. The internal bracing of 2 x 4 studs consumes 10 cubic feet, but more on the order of 26 cubic feet of the volume is unusable because the space between bracing cannot usually be filled with waste because the liner does not allow it to reach that space. Coupled with the 3/4 inch plywood walls, the double floor, and the 4 inch risers that it sits on, a nominal 128 cubic foot package can only serviceably hold approximately 80 cubic feet of waste. If Nevada Test Site received 1,000 such boxes, 48,000 cubic feet of void space at seven dollars per cubic foot would equal $336,000 of disposal cost that could largely be avoided by adoption of a more space efficient container. Add the low fire rating, and it is easy to conclude that the plywood box is an undesirable package.

Other void-generating culprits are closures and risers. The closure on a metal box is often a bolt lip that extends out 3 inches in most cases. The riser is usually a 4" high block that allows forklift access to the bottom of the box. On a 4' x 4' x

6' box, the bolt lip represents a 20 cubic foot void while the riser is 8 cubic feet per box. This is not to say that all voids can be eliminated, but there are improvements that can be made. Nevada Test Site can handle packages without risers, and RWMC is working on designs that can be handled by a lifting fixture, eliminating the riser.

All of the voids that consume valuable disposal sites also enter into the volume that can be transported. More efficient designs would reduce the number of trips to disposal sites, ultimately improving safety for the public as well as saving in transportation costs.

The suggestion that a new paradigm be developed led to the thinking that a metric package would serve the D&D groups by having a lower lift height while having nearly the same useful waste volume. It could also accommodate production of plate using the existing rolling mills that are in use in the DOE system. A 39.37" box could be made with a single piece of plate from a 42" mill more easily than a 48" dimension box.

The ultimate outcome of the discussions from the several sites and groups is that a standard should be set. There is clear disagreement as to what that standard should finally be. The standard could be mandated from DOE headquarters, but adoption could be complicated in the field. The most clear cut solution that has been advanced to date is that the disposal sites determine the best solution in a dialogue with the waste generators and transporters, and be empowered to buy the containers and supply them to the generators. This could be enforced by provisions in each site's waste acceptance criteria documents that would exclude non-approved, non-standard containers.

The emphasis must be maintained that this development be supported by the several affected components of DOE Headquarters; Environmental Restoration (EM-40), Waste Management (EM-30), and Transportation Management Division. The dialogue must involve the field personnel who are most familiar with system problems, and it must involve the full range of those who use and handle the containers. In this way, integration can save money and waste disposal volume.

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RADIOLOGICAL RELEASE CRITERIA AT THE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT THEORY AND PRACTICE

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ABSTRACT

As environmental restoration activities progress at the DOE's Fernald site, and across the country, large volumes of radioactive scrap metal (RSM) are being generated. Despite the existence of "free-release" guidelines from DOE, the strategy of onsite decontamination and release of RSM for unrestricted use has been generally overlooked in recent years. A pilot project was completed at Fernald in which 120 tons of RSM were decontaminated onsite and released for unrestricted use. This paper compares that strategy to more traditional DOE RSM management practices.

BACKGROUND

In managing the Fernald Environmental Management Project (FEMP) for DOE, FERMCO has established an organization founded on the EPA concept of Operable Units (OUs). For the Fernald site, five distinct OUs have been defined, each with a corresponding management organization, known as a CERCLA/RCRA Unit (CRU). CRU3 is responsible for managing the cleanup of OU3, which covers 136 acres and consists mainly of the former uranium processing area, including buildings, equipment, stored wastes, and associated materials. A map of the site showing the boundaries of the various OUs is shown in Fig. 1.

Fig. 1.

The FERMCO Recycling Department was established to support the CRUs, particularly CRU3, in their remediation efforts. Specifically, Recycling provides CRU3 with information to evaluate the viability of recycling or beneficial reuse as compared to other management options for various materials and waste streams. When recycling is selected as the preferred approach, the Recycling Department then generates the plans, procedures, contracts, etc. to implement the strategy.

DOE Order 5400.5 "Radiological Protection of the Public and the Environment," was

issued in February, 1990. Based on the requirements of this order, personnel and equipment have routinely been "free-released" from the OU3 radiologically controlled areas. However, the release of equipment has been strictly limited to relatively small-scale items, such as tools, trailers, and vehicles. The free-release of large amounts of RSM was considered too ambitious (almost "taboo"). FERMCO Recycling sought to demonstrate the viability of free-releasing RSM, after onsite decontamination, by performing the Material Release Facility (MRF) Pilot Project.

DOE GUIDELINES AND REQUIREMENTS

DOE Order 5820.2A "Radioactive Waste Management," issued in 1988, stresses waste minimization. It states "...LLW shall be managed on a systematic basis using the most appropriate combination of waste generation reduction, segregation, treatment, and disposal practices...." This requires DOE contractors to utilize a portfolio of strategies in managing their LLW. For RSM, one element in the portfolio is decontamination and free-release.

In 1990, DOE Order 5400.5 "Radiological Protection of the Public and the Environment," established the radiological contamination guidelines for the free-release of materials and equipment. These guidelines include numerical limits for residual surface contamination (see Table I) and a qualitative evaluation of the effectiveness of segregation and treatment efforts (ALARA evaluation).

At Fernald, the predominant nuclides of concern are U-natural, U-235, and U-238. The corresponding surface contamination limits from DOE Order 5400.5 are well within the capabilities of swiping and hand-held surveying techniques routinely utilized at Fernald and other DOE sites. Theoretically, material with measurable surface contamination, but which was below the 5400.5 limits, could be released without restriction on end use.

MATERIAL RELEASE FACILITY (MRF) PILOT PROJECT: INTRODUCTION

A multi-disciplinary team of FERMCO personnel was assembled to plan the free-release project. Team members were selected based on experience, area of responsibility, and organizational affiliation, so that all FERMCO divisions with a stake in project outcome were represented. Management empowered each team member with the authority to make decisions which impacted project outcome.

The team's first task was to develop a strategy for conducting the project. Presented in Fig. 2 is the basic approach devised by the team. Based on this flowchart, a work plan was developed to decontaminate and free-release 120 tons of RSM.

Fig. 2.

The team capitalized on several factors which combined to give a strong indication that the Pilot Project would be successful. First, a building with adequate decontamination and material handling capabilities was readily available. The FEMP's Building 78 was originally designed to decontaminate vehicles and process vessels used in the production of uranium. By the time construction was completed, production operations had ceased and Fernald's mission had become one of environmental restoration. This brand new facility basically sat idle, until the Pilot Project was conducted. Building 78 became known as the Material Release Facility, or "MRF."

Another factor which bode well for the project was the abundance of seemingly ideal RSM available in OU3. Several lots of material were targeted for decontamination and free-release, including stacks of unused structural steel (50 tons), sheets of steel deck plate cut from a decommissioned U.S. battleship (45 tons), and unused furnace pots (120 tons). Although none of these items were ever used in the production of uranium, they had all accumulated significant surface contamination through years of storage in radiologically controlled areas of OU3. The 120 tons of furnace pots were ultimately chosen as the target for the Pilot Project.

The third factor which pointed to a successful project was the existence of a fairly thorough infrastructure of SOPs at Fernald. Although there was no "How to Free-release RSM" procedure, nearly every step of the MRF Process Flow Diagram (see Fig. 2) was covered by an existing SOP. The MRF Work Plan was generated to coordinate the existing SOPs and to fill in any gaps. The MRF Work Plan served as the overall SOP for the Pilot Project.

Quality Assurance (QA) is an integral part of all activities at Fernald. Since this was a somewhat unprecedented undertaking, QA was especially rigorous for the MRF Pilot Project. FERMCO QA was involved throughout the project, from the earliest planning sessions through project completion. Through this QA involvement, the need

for extensive documentation was identified as a means to control the system.

MRF PILOT PROJECT: IMPLEMENTATION

With QA input, a new form called the "MRF-100" (see Fig. 3) was implemented. This form became the foundation for the documentation package required to free-release material. The MRF-100 served as a reservoir for all information and documentation generated during each step of the MRF process. It was designed to provide QA with a comprehensive, straightforward summary of all the steps taken to justify the free-release of the furnace pots. The MRF-100 greatly simplified QA's job of certifying that the material and documentation met all requirements for free-release.

Fig. 3.

To move 120 tons of furnace pots (600 individual items weighing 400 lbs. each) completely through the MRF process took about two-and-a-half months. The main decontamination technique was steam/detergent spraying, although some grinding, scraping, and torch cutting were required for small areas of particularly stubborn surface contamination. All material handling and decontamination tasks were performed by FERMCO hourly (union) employees.

For the final, comprehensive radiological characterization, a conservative approach was adopted. As discussed above, DOE Order 5400.5 theoretically permits the free-release of items with measurable surface contamination, assuming ALARA requirements are satisfied. The approach adopted for the MRF Pilot Project required that no detectable activity be found on any item to be free-released. Again, the radiological surveying techniques used were capable of measuring contamination levels well below the 5400.5 limits. By adopting the "less than minimum detectable activity (<MDA)" limit, a safety net was built into the system and the ALARA process was greatly simplified.

The total cost of release activities for the project was about $72,500. This figure includes ALL costs associated with the onsite processing of the furnace pots: transportation to and from the MRF, material handling at the MRF, flame cutting of stubborn contamination spots from some pots, decontamination activities, radiological surveying, technical, clerical, supervisory, and management support, and supplies. See Table II for a task-by-task breakdown of the total cost on a percentage basis.

As illustrated in Table II, only 31% of the total cost actually went to release activities (material handling, transportation, and decontamination), while radiological surveying ate up 42% of the budget. This figure was probably driven up by the difficult shape of the pots and by the adoption of the "<MDA" limit. Nevertheless, a seemingly inordinate chunk of the budget was consumed by this activity. Although MRF processing is a very cost-competitive option for processing some material types (as illustrated below), a great opportunity to improve the system exists in the area of radiological surveying.

MRF COST VS. BURIAL AT NTS

Traditionally, disposition of items like these furnace pots is accomplished by burial at NTS. Fernald has a great deal of experience shipping LLW to NTS, and so an accurate estimate of the cost to disposition 120 tons of furnace pots at NTS was readily available through the FERMCO Estimating Services Dept. This estimate includes the material and labor for packaging, transporting, and burying the material. As depicted in Table III, MRF processing is a very economical option, costing slightly more than half the price for NTS disposal.

PLANNED ENHANCEMENTS

Because of the success of the Pilot Project, and the abundance of material remaining in OU3 which is ideal for MRF processing, the work plan under which the trial was conducted has been upgraded to an SOP. Funding has been secured for FY-95 to decontaminate and free-release RSM at a rate of 50 tons per month, using a production line approach.

Additional decontamination technologies are also envisioned for FY-95. Start-up of the MRF's high pressure (2,500 psi) and ultra-high pressure (35,000 psi) water spraying systems is anticipated by April. Also this spring, a vacuum grit blaster will be purchased and installed at the MRF (thanks to funding received through EM-334). Procurement of automated radiological surveying equipment is also being considered, as is leasing a mobile, self-contained decontamination facility for specific material lots. With these enhancements, an ambitious goal of 800 tons per year may be achievable.

SUMMARY
Many options exist for managing RSM. DOE orders dictate that contractors demonstrate flexibility in utilizing a combination of techniques to optimize the benefits of waste management activiites. The FERMCO Recycling Department led an effort to provide their customer with an economical alternative to the traditional approach of burying contaminated metal as LLW, based on established DOE free-release guidelines.
Fernald's MRF Pilot Project demonstrated that onsite decontamination and free-release is a viable option for managing RSM in the DOE complex. In developing an overall environmental restoration plan, this approach should be included in the portfolio of strategies to be considered. At Fernald, decontamination and free-release of RSM is becoming a routine operation, part of everyday life.

32-7
POTENTIAL IMPACTS OF PENDING RESIDUAL RADIOACTIVITY RULES

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ABSTRACT
The purpose of this paper is to present an overview of pending rules governing residual radioactive release criteria and radioactive waste management, and the potential impact of these rules on the Fernald Scrap Metal program. More than 300,000 cubic meters of radioactively contaminated waste will be generated during the dismantlement of three complexes at the Fernald Site over the next year and a half. Under current regulations, as much as 70% (5,000 tons) of steel will be either recycled or re-used in controlled applications. Depending on regulatory developments, the ratios of recycling to burial will range from 100% burial to recycling more than 90% of the waste.
The absence of federal rules and regulations for classification of permissible levels of residual radioactivity is one of the most troublesome issues in the nuclear industry. The issue is growing in importance with the approaching end of useful life for many nuclear power generating stations and the planned remediation of the DOE nuclear weapons complex. Federal regulators have been involved in the "Enhanced rulemaking" process for over two years. The DOE Fernald site offers a good opportunity for understanding the potential impacts of the pending residual radioactivity regulations due to the maturity of the planned D&D activities, aggressive recycling program, and simple nature of contamination. The Fernald experience may offer a point of departure for many facilities engaged in D&D and waste management.

BACKGROUND
The Fernald Site is a former uranium metal production facility which was utilized for the conversion of UF6 to uranium metal and other applications within the Department of Energy. The production mission commenced in 1952, and proceeded through 1989. In 1989, the Department of Energy made a decision to end the production mission at the Fernald facility and began the remedial action dedicated to the cleanup of the former production facility.
As a result of production activities, uranium contamination was dispersed throughout the 80 acre production area. Two major areas being addressed within the complex include the former production facility and the waste pit area west of the production facility used for land placement of various process generated from the beginning of operations until 1985.
In 1985, land burial at the facility was ended and process waste were either stockpiled or packaged for transport and burial at the Nevada Test Site (NTS).
The mission was very straight forward with respect to the operations at the Fernald facility. The primary contaminants associated with all areas at Fernald are uranium, and thorium. No reprocessed fuel was used at the facility, therefore, fission products and activation products are not suspected at Fernald.
Various forms of Uranium were produced during the life of the project, which included a depleted uranium metal, normal distribution metal, and also low enriched uranium up to approximately 2% Uranium 235. The goal of the remedial action at the Fernald site is to excavate and stabilize the waste that was previously placed in the ground, remove contamination from an aquifer which underlies the entire

facility, and to take to grade or demolish all of the production facilities formerly used for uranium production.
As a result of the remedial actions, a large quantity of radioactive waste will be generated. Table I depicts the volumes of the major categories of waste. As can be seen, the total is nearly 3,000,000 cubic meters of waste, two-thirds of which will be soil and clay. The next major contributor to the volume of waste requiring remediation is 600,000 cubic meters of waste pit contents, and the remaining portion, nearly 300,000 cubic meters, will involve the management of the construction debris from the dismantlement.

SCRAP METAL MANAGEMENT

Historical Practice

The historical practice for the management of radioactive scrap metal generated during the production mission was to stockpile the metal in the northeast section of the production area. By 1989, more than 6,000 short tons of radioactive scrap metal had been placed in the scrap metal storage area. An aggressive project was initiated in 1991 to clean up this area and plans were developed for the management of the radioactive scrap metal.
Nearly 4,000 tons of the radioactive scrap metal were packaged into large 8 foot x 8 foot x 20 foot containers and transhipped to the Nevada Test Site for burial. A project was initiated in 1991 to recycle or beneficially reuse the remaining metal stockpiled at the scrap metal storage facility.
The radioactive scrap metal destined for recycling or beneficial reuse consisted of both ferrous and non-ferrous metals (primarily ferrous metals) with a nominal contamination level of 50,000 dpm per 100 centimeters squared or 8.3 becquerel per centimeter squared with natural uranium.
A turnkey project was initiated to hire a subcontractor to provide characterization, size reduction, packaging, transportation, surface decontamination, metal melt, and secondary waste disposition. The end product for the action was the fabrication of shield blocks which would be transhipped to the Department of Energy for use as shielding in accelerator projects within the medium energy physics program. At the completion of the project, 90% of the material by weight had been beneficially reused or recycled.
A cost assessment was performed for the activity in which the recycle and reuse contracts was compared to the historical practice of disposal at the NTS. Disposal of the 2,210 tons of scrap metal would have cost approximately $4 million as compared to the expenditure of nearly $4.8 million to contract the services for beneficial reuse.
A net cost advantage was realized, given that the DOE avoided the expenditure of more than $1.7 million for the purchase of virgin metal shield block for the medium energy physics program. The benefit equated to nearly $1 million savings within the DOE.

Future RSM Management

At the completion of the initial scrap metal recycling project the management at Fernald reviewed the performance of the contract and the methodologies employed. A primary consideration was made to further segregate any future generated radioactive scrap metal. The segregation would occur primarily based on physical form, with the distinction being made on not only the radiological characteristics, but the presentation of the substrate.

METAL CLASSIFICATIONS

Metal waste at the FEMP is divided into two categories: refuse and recoverable. Recoverable metal (scrap metal) is further divided into two subcategories; recyclable and reusable. The distinctions are based on the physical and radiological characteristics of the metal form. Disposition of these materials can only be identified once the materials are appropriately categorized. Appropriate segregation into these categories will facilitate the most cost effective and timely final disposition of metal waste. The following are descriptions of the categories:

1. REFUSE - Refuse metal waste is metal which is radiologically contaminated or suspected of being radiologically contaminated. The physical form of the metal is such that is excessively oxidized or a bimaterial form where separation of the metal from the other materials is not cost effective. Evaluation of cost effectiveness requires a comparison of the cost of managing the material as refuse considering the regulatory status of the material as a waste (a specific material may be

cost effective to recover if it would be regulated as mixed waste, whereas it may not be cost effective to recover if it would be regulated as low level radioactive waste).

2. RECOVERABLE - Recoverable metal is metal which is radiologically contaminated and can be processed for unrestricted release or controlled reuse. Generally, this category includes all metal which does not have the refuse characteristics.
   A. Unrestricted Release metal is metal which can be decontaminated and all potentially contaminated areas are accessible for direct contamination survey. Generally, unrestricted release scrap metal has a low surface area to mass ratio. Examples of reusable scrap metal are structural steel, tanks and decking. Metal forms may be considered for unrestricted release even if there are minor portions which cannot be cleaned or monitored if that portion can be effectively removed from the form.
   B. Restricted Release scrap metal is metal which cannot be decontaminated or surveyed to verify that the release limits have been met. Generally, restricted release metal is light gauge or has inaccessible areas where contamination may be present, such as ductwork, cabinets, machinery, and odd sized forms. Restricted release scrap metal may include unrestricted release metal when it is determined that the restricted end-use is more cost effective.

Fernald will be generating large quantities of radioactive scrap metal. It is anticipated that during the demolition of the former production area, more than 50,000 tons of radioactive scrap metal will be generated. Nearly one-third of this will fit into the category of Unrestricted Release Recoverable metal, while the remainder will be considered Restricted Release Material not conducive to free-release in accordance with existing surface radioactivity guidance.

Plant 7

As a result of the demolition of the building of Plant 7, 710 tons of structural steel and deck plate has been generated. All of this material has been containerized into reusable containers, and is awaiting shipment to an offsite facility for surface decontamination and free-release. The contamination level of the structural steel is a nominal 30,000 dpm per 100 centimeters squared or 4.51 becquerel per centimeter squared.

Only depleted uranium was processed at this facility, and measurements were taken to determine the thickness of lead base paint on the members. 8 mils of lead base paint were discovered to be on the surfaces. A contract has been let for the transportation, surface decontamination, survey, release, and secondary waste disposal of the 710 tons of scrap metal. The end product will be recycled scrap metal with no restrictions, and will be sold to a commercial vendor. It is believed that 95% by weight will be recycled.

The cost of the activity is approximately $1.4 million, as compared to a disposal cost of $2.6 million for this material. An important consideration in conducting cost comparisons between recycle and reuse options versus disposal is an understanding of the packaging efficiency for this type of material. Previous experience at Fernald has indicated that a density of 16 lbs/cubic foot can be obtained without exhaustive size reduction actions. Given that no automated or methodized size reduction capabilities exist at Fernald, it is appropriate to use this density in the disposal analysis.

Material Release Facility

Another project initiated at the Fernald site is the utilization of a previously unused facility as a Material Release Facility. The purpose of this facility is to provide the necessary quality assurance, survey and decontamination operations to release metal from the radiologically controlled area. The candidate material identified for processing through this facility is in general heavy gauge, lightly contaminated material that is suspected of not requiring exhorbant decontamination technologies. In fact, the only decontamination techniques which are employed are dry vacuuming, scrubbing, scraping and low pressure steam with detergent additives. It is anticipated in the future that additional decontamination technologies (i.e. grit blast, close circuit grit blast) will be employed but will not be complex from the perspective of either capital investment or technology.

Through the first five months of the project nearly 180 tons of metals have been

released and sold to local scrap dealers for nominal scrap value. This facility operation will continue through the life of the remedial project. As long as activities are ongoing in the radiologically controlled area, there will exist a need for the controlled survey and release of items that may become potentially contaminated.

Fernald believes that it will process approximately 600 tons of material through this facility annually.

Waste Management Approach

The approach at Fernald is to develop a portfolio of disposition options for the waste generated as a result of the remedial action. At different times within the life of the project, various needs will become priority. Most notably, the needs will consist of economic evaluations and scheduler concerns. It is felt that with a portfolio of options for the management of the various types of radioactive waste the most responsible disposition will be able to be utilized.

Regulations

All of these management techniques have been developed to conform to the currently existing regulations. Changes are anticipated in the regulation of radioactive waste treatment storage and disposal. Most notably in the definition of radioactive material itself and also recycling radioactive scrap metal criteria.

At this point, the existing regulations only allow for the release of material which can be demonstrated to conform to surface radioactivity guidance. No regulatory foundation exists for the release of volumetric contamination or material that has inaccessible contamination for surveying.

As a summary to the pending regulations within the United States, Table II is offered to depict the activity. When cleaning a facility, it is easily visualized that there are four modes of releasing contaminants into the environment which could result in potential exposures. Of the four exposure pathways two are extremely well regulated. Air emissions resulting from the operation and decommission of a facility are well regulated under the Clean Air Act. Additionally, any water effluent associated with a facility are well regulated under the Clean Water Act and the Safe Drinking Water Act. The direct exposure associated with the facility, and the exposure associated with the solid waste generated at the facility are less well regulated.

There are two pending regulations for the control of direct exposures as a result of a facility being remediated. One regulation, "Radiological Criteria for Decommissioning" issued by the United States Nuclear Regulatory Commission (NRC), is designed to regulate facilities which operate under an NRC license. A parallel regulation issued by United States Environmental Protection Agency entitled "Radiation Site Cleanup Standards" is being developed for implementation at facilities other than NRC license facilities, such as federal facilities.

By definition, if the air, water, and direct exposure routes are regulated to certain levels, this will dictate a certain amount of solid waste be generated to conform to these standards. The industry is in great need of regulations which will adequately address the issue of solid waste, of which recycling regulations would be a subset. The U.S. EPA is developing regulations for solid waste. The overall program originally titled "Radiation Waste Management" has been developed and will continue to be worked on for the next several years. As a subset of this, a specific regulation will be developed for the management of materials which may be recycled out of this solid waste.

DESCRIPTION OF PENDING REGULATIONS

The U.S. EPA Decommissioning Staff Draft

The scope includes setting standards for the remediation of soil, groundwater, surface water, and structures at Federal facilities. A staff draft is in review and comment resolution. Pathway analysis and modeling are in progress, the most mature of which are the soil regulations. The major element of this regulation is the establishment of a 15 millirem per year effective dose equivalent exposure to the reasonably maximumly exposed individual. If this level is met, the facility may be abandoned with no restrictions based on its future use.

The 15 millirem per year value includes a four millirem per year component dedicated to the groundwater associated with the facility. The basis for this regulation has been developed from the International Atomic Energy Agency, the International Council on Radiation Protection, and the National Council on Radiation Protection recommendations. The risk based levels are consistent with the CERCLA requirements

for an excess cancer rate of 10-4 to 10-6. In actuality, the 15 millirem per year dose equates to a 3 x 10-4, which is considered to be within the range described by the U.S. EPA.

United States Nuclear Regulatory Commission Radiation Site Draft Cleanup Staff Draft

The scope includes specific radiological criteria for decommissioning of soils and structures at NRC license facilities. A final rule is anticipated in May 1995. The major element of this regulation is the establishment of a 15 millirem/year total effective dose equivalent, distinguishable from background and with ALARA considerations. The basis is the International Council on Radiation Protection and the National Council on Radiation Protection recommendations for individual dose.

Solid Radioactive Waste Regulations

The scope of regulations under being developed are, however, not fully determined. They may include source material, special nuclear material, byproducts, high-level waste, mixed waste, transuranic waste, and low-level waste. An issue paper and a preproposal draft have been developed. The proposed draft, Environmental Radiation Protection Standards for the Management, Storage and Disposal of Low Level Radioactive Waste (40 CFR 193), was issued in December 1994. The major elements will include requirements for treatment, storage and disposal of radioactive waste.

The most significant issue being discussed in the preliminary development of this regulation is the inadequacy of current waste classification systems. In essence, it may be appropriate for regulators to come up with a new classification system based on hazard rather than the generating process. In some cases, low-level waste are more hazardous than some forms of high-level waste, as well as some forms of NORM waste being more hazardous than mixed waste.

U.S. EPA Radioactive Material Recycling

The scope for recycling radioactive material rules has not been determined, but may include both restricted and unrestricted scenarios for regulation and implementation. The current status is that an issue paper is being developed to initiate the discussions and identify the need for any future regulations. Presumably, the regulation will rely on recommendations issued by the IAEA, OECD, ICRP and NCRP. The IAEA has issued a document on exemption criteria for radioactivity and the NCRP has authorized the formation of a committee to make recommendations on clearance levels.

Impact of Potential Pending Regulations

It is premature to address the impact of the pending regulations on the current waste management practices at the Fernald site. The desire is to have consistent and accepted rules governing the activities associated with radioactive waste management. At the Fernald site the question is extremely significant because the issuance of any of these rules will occur during the implementation of the Fernald cleanup. Therefore, adjustments will have to be made as the work is conducted. At the Fernald site it is not possible to wait for resolution of these issues and issuance of these regulations.

Recycling radioactive scrap metal will continue to play an important role in the remediation of the Fernald site.

Risk Based Regulations are welcome, and it is felt that they can be implemented at the site with little concern. The impact of any regulation will primarily be associated with the cost of dispositioning the material. In the absence of adequate regulation, or with regulations that result in clearance levels which are indistinguishable from the background radiation, Fernald may elect to provide timely land burial versus recycling or reuse.

SUMMARY

The Fernald site is an ongoing project. We are on the verge of implementing large scale activities which will result in a generation of large quantities of radioactive waste, including radioactive scrap metal. Under the current regulations, Fernald is able to recycle a portion of the radioactive scrap metal being generated at a cost which is comparable to other viable options such as land burial.

The pending regulations will be issued during the life of the Fernald project, and may have a severe impact to the ability of Fernald to continue beneficial reuse or recycling of its radioactive scrap metal, and may result in the burial of this material along with the contaminated soils and other radioactive waste residues. At this time, however, it is premature to speculate on these impacts given the lack of scope definition and lack of confidence in the ability to develop a widely accepted regulation concerning release of radioactive scrap metal.

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R.F. Williams, WTA
33-1

NAVY TECHNICAL RISK MANAGEMENT:
A TEN YEAR STRUGGLE

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ABSTRACT

Poor Naval weapon system reliability in the late 1970s led to a DoD manual, DoD 4245.7-M, which identified areas of risk in selected design, test and manufacturing topics, called templates, and provided outlines of ways to reduce these technical risks. Subject to differences in Government and industry interpretation, most assessments of these risks remained qualitative and dependent on the experience of the assessor, in spite of Navy awareness/education efforts. The absence of metrics and the inability to quantify and communicate technical risk led the Navy in 1993 to begin an effort to collect, organize and publish specific metrics for the Design, Test and Production templates. The result, Methods & Metrics for Product Success, is currently under review by selected industry and government representatives. Today, the Navy and DoD are beginning to focus on technical risk at defense acquisition program milestones. The lengthy evolution of this methodology and the cultural changes that were necessary for success are the subjects of this paper.

BACKGROUND

"In the beginning was Apollo." The manned lunar landings and lunar surface operations culminated the most massive technological buildup since the Manhattan Project. Beginning in the 1950s with sometimes futile attempts to place small satellites in orbit, and spurred on by such early Russian feats as Sputnik (October 1957) and Yuri Gagarin's orbital flight (April 1962), the technology to minimize technical risk gradually evolved through the Mercury and Gemini missions, reaching its apogee on Apollo's first lunar landing mission in 1969.

Early in Apollo's history, efforts were still being made to meet reliability requirements using math models based on piece part failure rates and mission functional and environmental timelines. The Apollo spacecraft fire in 1967 probably did as much as anything to prove the futility of statistics-based reliability math models and the need for technical excellence (not to mention common sense) in design and manufacturing. Probability and statistics were finally discarded and the focus turned to leaving no design or manufacturing stone unturned.

Little new technology was developed-rather, the importance of the old was rediscovered. For example, the extreme influence of stress derating on failure potential was underscored. Everything was made redundant that could be, and alternative modes or workarounds were developed for nearly everything else. Design of redundant elements tried to ensure that a single failure mode could not wipe out the whole function. For example, redundant relays might be oriented at different angles or located in different spaces. The designs of the few items that could not be made redundant were more heavily derated and tested. In the factory, rigorous manufacturing process specifications and controls such as those defining acceptable techniques for soldering were put in place, and everything at all levels from piece parts to end items was subjected to test environments designed to stimulate and detect latent defects for immediate corrective action. Rigorous environmental qualification and certification tests were keystones of the manned space program. Everything was "flown" to the maximum extent possible on the ground, and many special facilities such as huge thermal vacuum chambers were constructed to simulate mission environmental conditions as nearly as possible.

NASA put in place a very large reliability and quality organization at Headquarters and the three cognizant Centers (Houston, Huntsville, and Cape Kennedy). These engineers were largely technical specialists who came to disdain statistical approaches to R&QA. Each an expert on a particular subsystem, they consulted daily with their contractor counterparts during design and manufacturing of an end item, and visited the contractor frequently to review documentation and to get answers to

any technical issues requiring resolution. Everything was questioned. Every "untoward event," however seemingly small, was documented for review and satisfactory closeout, resulting in an exhaustive audit trail. Only unresolved technical issues were discussed at milestone decision points, and decisions were based on satisfactory plans to resolve these issues before the next milestone.

DEPARTMENT OF DEFENSE

In the 1970s, the Navy was experiencing problems with Fleet readiness. Sophisticated weapon systems were seriously compromised by low reliability and quality. The high rate of equipment failure required too many spare parts. The maintenance overload was worsened by the continuing difficulty in recruiting, training and retaining specialists for increasingly complex systems. The cost of parts and man-hours, aggravated by inflation, was taking its toll on the budget.

Encouraged by the dramatic success of the manned space program, the Navy turned to NASA experience in 1973 for new ideas to make weapon systems reliable. At the conclusion of the last manned lunar landing mission, the Naval Material Command invited Mr. Willoughby, then director of Apollo reliability and quality, to come aboard to aid in improving the readiness of Navy weapon systems based on Apollo experience. The Navy didn't need space technology per se-only the approach to lower technical risk.

This new Navy office initiated space-program-oriented technical reviews of individual programs, reorienting them where necessary to correct the most flagrant problems. Changes were directed at shifting emphasis from statistics to engineering; from measuring reliability to designing it in. At the risk of oversimplification, this approach could be said to focus on (1) designing to reduce stress, (2) testing under simulated field conditions to find design shortfalls, (3) using manufacturing processes that minimize defects, and (4) stress screening to stimulate and find remaining manufacturing defects.

These get-well efforts were faced with difficult obstacles, including a hostile working-level organization which was fearful of change, a competitive defense acquisition environment, an oppressive burden of unnecessary and cost-driving specifications and standards, budget constraints, political influence, program manager turnover, military bureaucracy, and even tradition. Without an infrastructure of appropriate directives and instructions to replace the existing faulty ones, this seat-of-the-pants approach worked only because the Navy delegated program approval authority to the new office, short-circuiting the bureaucracy. In effect, top Navy management decided to put reliability and quality first, ahead of cost and schedule. It may have been a risky decision for the Navy but it worked.

DEVELOPMENT OF THE TEMPLATES

By the early 1980s, the need for expanding the attack on unreliable weapon systems was beginning to be recognized by the Department of Defense. The many specific design and manufacturing fundamentals and disciplines were seen as components of a transition process from development to production. And reliability from a global perspective was more appropriately thought of in terms of risk. The transition from development to production had proven to be a major source of defense system acquisition risk-risk of cost escalation, schedule slippage, failure in service, and reduced readiness. Consequently, in 1982, the Under Secretary of Defense (Research and Engineering) commissioned the Defense Science Board (DSB) to organize a task force to examine the transition from development to production from a technical risk viewpoint, and Mr. Willoughby was asked to be the task force Chairman. Membership included corporate officers from the technical operations of Bell Labs, Boeing, General Dynamics, Gould, Honeywell, Hughes, Litton, Lockheed, McDonnell Douglas, Northrop, Raytheon, Texas Instruments, and Westinghouse, as well as DoD officials concerned with material acquisition.

The task force stated that "This industrial process of weapon system acquisition demands a better understanding and implementation of basic engineering and manufacturing disciplines. Once rigorous, disciplined engineering practices are employed and institutionalized, both the risk of deploying unsuitable weapon systems and the time in the acquisition cycle associated with design, test and production will be reduced." Drawing on the experience of their respective companies, this task force documented a subset of specific technical issues in the classic product design and manufacturing cycle which have the greatest influence on risk. It called this carefully identified subset the "minimum technical baseline." The task force document was published in 1985 by the Secretary of Defense as DoD Manual 4245.7-M,

Transition From Development to Production.
This "Transition Document" addresses specific technical issues which, based on industry experience with many programs, are revealed as having the greatest influence on acquisition risk, accounting for virtually all cost, schedule, and performance shortfalls in current systems. These technical issues are described in "templates" because each outlines a pattern for minimizing risk, reflecting best practices that defense systems contractors should follow in both new and ongoing programs.
To satisfy a need for more detailed guidance, the Navy enlisted technical consultants from several of the companies represented on the DSB task force to identify proven best practices for the minimum technical baseline defined in the Transition Document. Their knowledge of industrial technology, combined with Navy experience involving several hundred material acquisition programs, resulted in a second document, Best Practices: How to Avoid Surprises in the World's Most Complicated Technical Process, NAVSO P-6071, which describes the best practices for reducing technical risk that are used in industry today. Each template in the Transition Document is amplified in this "Best Practices Manual," published in 1986.
To be useful predictively, technical risks must be identified well before they become actual problems. An assessment of risk areas early in the development process can provide a systematic foundation for further analysis and revision as a system moves through acquisition. But an assessment based, for example, on tests conducted just prior to production does not assess the risk that the problems will occur. It uncovers the fact that problems have already occurred. The General Accounting Office, in its April 1986 report on the status of DoD technical risk assessment efforts, defined five requirements for technical risk assessment:

- Prospective assessment
- Planned procedures
- Explicit attention to technical risk
- Documentation
- Reassessment in each acquisition phase

If an assessment is to be called a "technical risk" assessment, all five of these criteria must be present.

THE NEED FOR EDUCATION

Efforts to implement widespread use of these best practices were hampered by inconsistent understanding or total lack of understanding in many subject areas. References for the risk areas defined by the templates were scattered, often existing as technical reports, standard practices in corporate manuals, and informal documentation. Specific references for some were nonexistent. Table I lists technical documents that the Navy's Product Integrity Directorate sponsored or supported to fill some of the most critical voids. Some risk areas concern basic engineering fundamentals, emphasizing design approaches which have seldom been included in academic curricula. Others involve the practical applications of those fundamentals, approaches which have been proven to reduce or eliminate risk through their use in successful programs. Still others are based simply on time-tested industry experience. As templates were tailored to specific programs, and as more experience was gained in their application and use, they were frequently modified and refined, which further complicated the references and resources picture. As a result, the risk areas were subject to differences in specific interpretation. Government and industry implementation of templates for reducing risk was not consistent and often did little to reduce risk.
Most government people lacked enough hands-on industry experience to utilize the templates effectively. The practical engineering and manufacturing know-how evolved by industry on-the-job and embodied in the templates was not taught in formal engineering curricula. Industry generally retains a greater range and depth of corporate knowledge and technical experience than the government because design and manufacturing are its stock in trade, while government programs are affected by an oversight mentality, limited hiring flexibility, engineers without industry experience, and program managers with little appreciation for design and manufacturing fundamentals. The irony in all of this is that industry still looks to the government for guidance because the government is the customer, and their lack of technical experience is too often considered irrelevant. There were other factors involved, of course, such as affordability, politics, competition and procurement legislation. Results, therefore, were largely dependent on individual knowledge and

experience.
It was obvious that an awareness/education effort would be required. Unfortunately, there were no courses available which teach the technical processes of defense acquisition. To solve this problem, the Navy's Product Integrity Directorate called on industry to assist in the implementation of educational programs to indoctrinate the material acquisition community-industry and government alike-in the fundamentals of best practices for technical risk management. This effort began in early 1988 with a contract to AT&T to prepare comprehensive technical reference guides for the templates and to develop the classroom education program. The objectives of the program were explicitly to

Increase awareness and understanding of design, test, production and logistics fundamentals

Provide a technique for assessing technical risk in accordance with DoDD 5000.1, DoDI 5000.2, and DoD-5000.2-M [DoD's top-level defense acquisition policies and procedures]

Over the period of a year beginning in mid-1990, pilot courses using technical reference guides as they were completed by AT&T were presented to some 200 government and contractor personnel. Later, completed course modules were presented to a similar number of personnel in the Navy acquisition community. At the same time, McGraw-Hill edited and published the AT&T technical reference guides as a set of four textbooks on design, test, production, and logistics. The Defense Systems Management College eventually incorporated the series into its education program for program manager designates. But at the working level, implementation of a DoD-wide or even Navy-wide education program for the defense acquisition community proved elusive-top management commitment was missing and adequate resources for staffing and funding could never be identified, especially in light of the end of the Cold War.

METHODS & METRICS FOR PRODUCT SUCCESS

Since DoD Directive 5000.1 requires technical risk assessment and management in major acquisition programs, DoD program managers are required to perform technical risk assessments prior to major program decision milestones and to develop aggressive risk management plans in order to obtain approval to proceed into the next program phase. However, technical risk assessment has continued to be primarily qualitative and subject to the experience of the assessor. In virtually all approaches, experts are asked for subjective judgments of what the risk elements were, as well as the likelihood of their occurrence. Ratings are based on various sources of information, such as expert judgment, test or simulation results, and published technical reports on similar systems.

We realized that managing technical risk requires the ability to identify, measure and communicate it, and that the Navy lacked the metrics to measure it; consequently, poor communication was impeding implementation efforts. Starting in 1993, we began to collect and organize specific metrics for the design, test and production templates. Technical risk occurs where there are differences between the technical practices of design, test and production, as implemented, and corresponding proven best practices. Metrics provide a means to quantify the deviation of implemented practices from best practices in order to minimize subjectivity and provide a sound and consistent basis for program management decisions.

The results are documented in our Methods & Metrics for Product Success handbook. They include both risk management methods (process identification, process baselining, risk mitigation and reporting) and risk assessment metrics. A typical risk assessment summary chart for a defense acquisition program is illustrated in Fig. 1. The colors (green, yellow, red) signify the degree of compliance between actual and best practices as determined using the metrics for each template appropriate to the program phase. This approach is especially good at assessing changes in technical risk over time, as a result of mitigating actions. The Navy's Product Integrity Directorate has used this methodology successfully in diagnosing technical risk and recommending appropriate mitigating actions in several major defense acquisition programs to date. A small printing of the handbook has been distributed to various government and contractor people to get feedback before we commit to full publication. We are encouraged by early reports of its use.

FIG. 1.

IN RETROSPECT

This approach to defense acquisition has required a cultural change in the acquisition community-from reliance on specifications and standards; legal, financial and contractual regulations; reorganizations; latest fads, whims and slogans; and armies of auditors-to best design and manufacturing practices and technical risk management. Over the ten year struggle since Transition from Development to Production was published, progress is seen in the top-level DoD defense acquisition policies and procedures which now focus on design and manufacturing, and require technical risk assessment. Instances of successful risk assessment and management, not only by Navy program offices but also other services as well, are increasing. With DoD acquisition reform phasing out military specifications and standards, we are convinced that focus on best design and manufacturing practices and processes, using technical risk management techniques such as those in Methods & Metrics for Product Success, offers the best opportunity for continued product success.

33-2
STRIPPING AWAY THE BUREAUCRACY:
A RETURN TO BASIC ENGINEERING PRINCIPLES
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ABSTRACT
Many of the problems in Government programs today are the result of a lack of engineering and scientific product and process engineering. We have moved away from running programs based on sound engineering and scientific decisions. Instead, we are making decisions based on bureaucratic considerations.
To help overcome these problems, CSC developed a series of expert systems and associated tools to provide acquisition personnel with Knowledge, Insight, and Experience into the world's most complicated technical process. This system, the Program Managers' WorkStation (PMWS), was created for the Best Manufacturing Practices (BMP) Program and is available to anyone in the U.S. industrial base.
A key PMWS tool is the Technical Risk Identification & Mitigation System (TRIMS). TRIMS establishes a technical risk management system utilizing a knowledge-based approach. The PMWS and TRIMS are fully compliant with the DoD 5000 series directives; yet, they are not cumbersome. These tools are regularly used by commercial industry throughout the U.S., Canada, and the United Kingdom.
A number of large programs are successfully using the PMWS tools. Some of these programs are CAS (Aircraft Automated Test Set), U.S./U.K. Surface Ship Torpedo Defense (SSTD), Photonic Mast, and P3 Upgrade.
There are many companies which have adopted PMWS as a standard. One such company is McDonnell Douglas. Still others, such as Texas Instruments, credit BMP/PMWS as a major contributor to their winning the Malcolm Baldridge award. A DOE facility, Martin Marietta Energy Systems (Oak Ridge Centers for Manufacturing Technology), operates as a BMP Regional Support Center. They are using PMWS tools to evaluate and improve their internal operations, as well as support regional businesses and educational institutions.
EXPEDIENCY OR ENGINEERING?
For the past few years, acquisition programs, and engineering in general, have been under attack by the troops from "buzz word land," e.g., TQM, QFD, Re-Engineering, CALS, CE, IPD, and many more. While these "buzz words" all represent potentially useful processes, in many cases we have become enamored with their infrastructure and philosophy and have lost sight of the goal -- improved profits, quality, and market share.
The engineering process needs to return full circle, back to the basic engineering principles of the past. However, the process will need to add a modern day slant through the use of COMPUTERS. Even though this will be a return to basic engineering principles, employing current computer technology, specifically expert systems, can improve the efficiency of the systems engineering process by orders of magnitude.
If the problems of today's Government acquisition community are to be solved, we must first understand how they got there or we will end up right back here at some future date. A key problem that has been identified is a lack of engineering knowledge on the part of senior decision makers. Today, MBAs and accountants rule the world of engineering; directives are written mandating the solution of technical problems (erroneously thinking this will magically solve the problem). Key front end

engineering practices are ignored because the benefits cannot be shown using limited or inappropriate accounting systems. Another factor is the past DoD budgets. These budgets were robust and made it possible to not worry about how well a system worked. The system could always be "fixed in the field," regardless of how inefficient or expensive that might have been.

As program failures increased, the oversight organizations (made up mostly of non-engineers) started adding "wickets" to the Defense Acquisition Process. These 'wickets" were added to insure program success, or so it was thought. In DoD, they are embodied in the current DoD 5000 series directives. At this point, it must be made clear that the writer is not against the DoD 5000 series, rather, against their use as an acquisition or engineering model. An effective acquisition program cannot follow a DoD 5000 model. Engineering should be based on a good Concurrent Engineering (CE) model and the results of this activity used as input to oversight reports, completed in a DoD 5000 format. This problem is further complicated by the fact that the DoD 5000 is a highly political document. Hence, it is subject to change at any time, routinely changing several times throughout the life of a program. Good engineering, on the other hand, remains essentially constant, unchanged by political events.

Problem

Engineering processes are evolutionary and technology breakthroughs cannot be directed. Good engineering cannot be based on bureaucratic considerations.

Solution

Decouple engineering from reporting and oversight models.

Results

Better, less expensive, products and more timely oversight data.

A SOLUTION: AN ELECTRONIC ENGINEERING CONSULTANT

What program managers' need is a super smart, experienced, fast thinking "Consultant" -- with a perfect memory. This "Consultant" would not only provide them with rapid access to sound engineering practices, but would lead them right to what they should be doing each day. The Program Managers' WorkStation (PMWS) shown in Fig.1 is the program manager's electronic "Consultant."

Fig. 1.

The PMWS was developed and is maintained by CSC for the Navy's Best Manufacturing Practices (BMP) Program, under the direction of Ernie Renner. The PMWS tools have been used successfully on many programs and are now available for use by anyone in the U.S. industrial base!

The PMWS consists of a series of expert systems and decision assistance tools. It provides knowledge, insight, and experience, in addition to an extensive network of information and software resources all easily accessible via PC. PMWS tools are centered on the systems engineering and acquisition process itself. If all engineering processes are understood, appropriate, and under control, the results will be as good as the state-of-the-art equipment will allow.

Workload reduction is a top priority of the PMWS. Typical project management tools, based on cost and schedule, use the graphical power of the computer to show numerous items on the screen at once. While these tools have their place and "master schedules" are critical for long lead planning and coordinating between organizations, there are many acquisition programs "failing" despite the use of these tools. Their power to display and "scroll" through data is enormous. The graphical interface makes it very easy for the user to maneuver through the data. However, they do little to reduce the heavy manager/engineer task loads; in fact, studies have shown that displaying large amounts of data tends to confuse the user and distract from focusing on the critical items that need attention. Rather then displaying potentially thousands of things involved in a particular program, or even the entire critical path, the PMWS typically shows the user the one to five most critical items that he or she should be concerned with NOW!

The goal of the PMWS is to do for acquisition personnel what the CAD terminal did for designers. CAD terminals allow designers to concentrate on designing instead of paperwork; likewise, the PMWS will allow program management personnel to concentrate on running their program(s).

THE POWER OF PMWS

Workload reduction is provided by the PMWS combination of knowledge, insight, experience, and communications features.

KNOWLEDGE through KnowHow, an automated and intelligent information access system

that speeds a search for required information by up to 95%. Typically, the information needed is on the screen in less than three minutes. This tool has several unique features. One is a high speed fuzzy-logic text search, capable of searching several inches of documention in only a few seconds. The system can be automatically tailored to users' needs. Both DOS programs and cut-and-paste in Windows features are supported.

INSIGHT through the Technical Risk Identification & Mitigation Systems (TRIMS), a technical risk management system illustrated in Figure 2 that identifies when corrective actions are required and who is responsible. This system can be tailored to a program's unique requirements. TRIMS currently has two knowledge bases for Systems Engineering and Software Development. TRIMS is compliant with the DoD 5000 series, NAVSO P-6071, DoD 4245.7-M, and the Software Engineering Institute's Software Risk Evaluation Methodology model.

Fig. 2.

Cost and schedule are downstream (sometimes way downstream) indicators of technical problems. This is why TRIMS monitors technical processes. It identifies engineering problems at the earliest possible point. In this way, problems can be addressed and mitigated before cost and schedule problems are indicated.

For example, if TRIMS indicates that a program has/is incurring a risk due to a Design Reference Mission Profile (DRMP) not being conducted; the user can go directly to the corporate knowledge base, KnowHow, and get full details on why a DRMP is needed and how to develop one, including metrics and examples. This type of problem would typically be identified by the PMWS early in the design process. However, cost and schedule indicators of this problem usually will not surface until the operational test and evaluation process; when the system did not perform properly because the design environments did not reflect current life cycle operational environments. This is just one simple example of how the PMWS can support the program manager.

EXPERIENCE through the Best Practices Database; proven and verified solutions from industry, government, and the academic community can be applied to a program. Table I presents a sample database output.

TABLE I

COMMUNICATIONS through BMPnet. Currently, BMPnet is used to facilitate communication between DoD components - DOE, DOC, FAA - and commercial companies. As shown in Fig. 3, program managers can now simply and easily communicate with each other when solving problems, gaining insight from each other's experiences.

Fig. 3.

BMPnet provides several productivity-enhancing tools. As an example, E-mail is available with several unique functions. The system not only records each user's name, but, also cross-references each user to any Special Interest Group (SIG) in which he or she has expertise/interest. In this way, a user can send a message, for example, to the welding SIG (via BMPnet or Internet) and every user who signed up as an expert in welding will get the message - even though the sender does not know them personally. The ability to "CHAT" with other users online is also provided by BMPnet. Nine "conference rooms" are available to hold online conferences, of up to six people, for problem solving that requires a quick turnaround. The system will even take notes automatically.

All PMWS programs can be downloaded from BMPnet for use on local computers/Networks.

CONCLUSION

Many programs are presently reaping the benefits that the PMWS has to offer. Currently, more than 10,000 copies of the PMWS software have been distributed, with the BMPnet answering more than 50,000 calls per year. PMWS success stories include: the Navy CASS program ($1.2 billion and failing - now exceeding requirements); McDonald Douglas (adopting PMWS company-wide); U.S./U.K. Surface Ship Torpedo Defense Program ["best IPS Risk Management Annex D ever seen" by ASN(RDA)]; Texas Instruments (significantly helped in winning the Malcolm Baldridge award); P3 Upgrade program (mandated use); and the Photonic Mast and Universal Modular Mast programs (required TRIMS assessment with proposals).

Through the use of current PMWS tools and the expansion of this methodology in both breadth and depth, acquisition managers and systems engineers will have an ever-increasing knowledge base to draw from. Future enhancements to the PMWS are currently underway. These enhancements include: fuzzy logic text searches for the BMP database; a performance specification generator based on the proven KnowHow

technology (with editing and formatting capabilities); expanded Internet capabilities (WWW, TCP/IP remote control of BMPnet sessions); Windows versions; distribution on CD-ROM; lexical analysis links between PMWS programs; and much more. While pleased with the current PMWS product, both CSC and the BMP Program Office are firmly committed to continuous process improvement. People are asked, with an open mind, to put the PMWS tools to use; they are then encouraged to call or E-mail comments to the PMWS Help Desk personnel (703-538-7253). The PMWS is the user's tool. It would not be what it is today without the quality feedback that has been received from its users.

You are invited to experience the expanding benefits of the PMWS. Exercise sound engineering-based decisions while you advance the competitiveness of the U.S. industrial base and the Nation.

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33-3

ACQUISITION AND CONSOLIDATION FOR SUCCESS AND SURVIVAL

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ABSTRACT

The steadily declining and changing nature of the defense business in the 1990's requires dramatic action on the part of defense contractors. The ability to effectively and efficiently face this competitive challenge is a matter of survival. In August 1992, the Hughes Aircraft Company acquired the Air Defense Systems Division and the Convair Divisions of General Dynamics and on January 1, 1993 merged the acquired activities with its Hughes Missile Systems Group to form the Hughes Missile Systems Company (HMSC). The physical consolidation of the HMSC activities in the Hughes Tucson, Arizona plant resulted in one of the largest and most ambitious industrial defense consolidations in US history. The Manufacturing consolidation was planned and completed within eighteen months and the Engineering consolidation within twenty-four months of the acquisition.

Consolidation planning methods were developed based on critical element definition, risk assessment and risk management. Transition and consolidation planning techniques were established based on an adaptation of the "Willoughby Templates" and a planning review process established which involved customer representatives. Six major in-production programs, seven key associated manufacturing processes and nine technology-based engineering transition teams were formed. A real time system of lessons learned feedback was employed to refine and modify the planning and implementation process as new transition teams were established. The major lesson learned during the merger and consolidation was that personnel issues were the most complex, demanding and important issue addressed. Transition planning and implementation examples will be presented along with a discussion of what worked particularly well, and what did not.

DISCUSSION

Fiscal Year Defense Planning (FYDP) in the 80's was typically overly optimistic in terms of availability of defense spending dollars. During the late 80's and on into the early 90's, it was becoming more apparent that defense spending was to be drastically curtailed (See Fig. 1). To survive in this financial environment, it became obvious that contractors would have to obtain 'critical mass' through liquidation, restructuring and/or consolidation, buying, selling or merging, or some combination of these activities. Companies were forced to respond to unpredictable, accelerating change and many contractors were experiencing an excess of facilities, equipment, real estate and personnel. In the face of reduced business volume,

overhead rates were soaring while compliance with procurement procedures remained unchanged.
Fig. 1.
Hughes responded to this crisis by initially committing to remain in the tactical missile business. Then, in an effort to obtain 'critical mass', Hughes acquired the General Dynamics missile businesses. It established the Hughes Missile Systems Company as a subsidiary of GMH Electronics by merging the Hughes Missile Systems Group and the General Dynamics missile businesses into a single unit. Further, it was decided to consolidate all manufacturing and engineering assets in Tucson, Arizona. In the end, real estate was trimmed from 9.0 M to 3.5 M square feet, employment reduced from 14,000 employees to 8,000 of which approximately 3,000 employees were relocated to Tucson.
In order to accomplish the consolidation and relocation efforts, a significant planning exercise was undertaken to determine the methodology for doing it. No data was available relating to an effort of this magnitude. During this review process, it was decided to utilize the "Willoughby Template" approach because of the risk assessment/risk management philosophy embraced by these guidelines and the rigorous discipline imbedded in the implementation process. This uniquely developed planning process was utilized for each transition team. This process is shown pictorially in Fig. 2.
Fig. 2.
Each transition team was required to receive the same training in terms of format followed with particular emphasis placed on risk identification, risk assessment and risk mitigation techniques. As each team progressed through its planning cycle (typically on the order of 6-8 weeks), lessons learned from earlier teams were transferred to the next team in the process. Each transition team was comprised of cross-functional organization representatives who were responsible for determining the planning parameters for their particular area of responsibility. Figure 3 illustrates a typical example of this representation.
Fig. 3.
The first transition team to conclude its planning activities and physically complete its relocation and consolidation was the Tomahawk program in January of 1993. Within a year of its transition, Tomahawk all-up round assembly time had not only duplicated prior performance levels in San Diego, but total assembly time had been reduced from an average of approximately 100 hours to 55 hours. This was primarily attributed to a new workforce looking at assembly procedures with "new eyes" and not constrained by traditional approaches. A comment made by one of the manufacturing managers was that "...a brand new workforce isn't necessarily bad", and in this case was a positive influence on the program. In September of 1994, it was announced that HMSC had won the Tomahawk single-up bidding against Mcdonnell Douglas, a contract worth approximately $1 billion over the next five years. The successful transition of this program, and of course, many other factors contributed to this success.
Results on other programs have been equally dramatic. Although the AMRAAM program was resident in Tucson during the consolidation process, improvements implemented during this period in the Seeker assembly area resulted in a:

- 41% productivity increase
- 36% WIP decrease
- 68% area layout decrease
- 28% lead time decrease

Stinger guidance section and seeker test yields were 5% higher than average results at General Dynamics/California operations. TOW test yields exceeded the 99% 1994 goal.
By mid-year 1994, all production programs had transitioned to Tucson on schedule, the last being the Stinger program. All engineering transition activities were completed by year-end. Four engineering organizations were combined into one; engineering managers were reduced from 240 to 70 and layers of management were reduced from four and five to three. Total relocated personnel to Tucson from the various off-site locations was approximately 3,000. In general, reduced operating costs resulted from efficient use of facilities, assets and a very skilled workforce.
A number of factors were responsible for the successful transition of the overall effort. HMSC constantly appraised its customers of transition status and involved

them in the teaming activities. Dedicated transition team leaders were selected for close-out activities and were not tied to new organizational structures. All production systems and processes were moved "as-is" so as not to introduce any new variables during the transition process. Key to the entire effort was the detailed and disciplined up-front program transition planning that was imposed on every team. Most important, we sustained financial performance and achieved all major transition milestone dates during the consolidation process.

To be expected with a consolidation of this magnitude, some things did not go as expected. The impact of the consolidated procurement and dock-to-stock transition overwhelmed the system in Tucson. For example, requirements increased from about 4,000 to 50,000 parts per month. The engineering transition decision, which occurred after the operations transition started, resulted in a change of receiving inspection location and delayed its timely installation. There were difficulties encountered in combining five different MRP systems into one. The sale of the Camden, Arkansas CCA facility fell through near the end of transition resulting in loss of job security for its personnel and the lost of a number of key personnel. Transferred kits were inaccurate with parts shortages, and combined with the planned reduced area capacity, resulted in a 9 month - 1 year delay in getting back up to rate production.

There were situations that arose that could not be avoided as a result of the relocation/consolidation decision. Our customers felt at risk until success could be demonstrated; the best people in the organization had the greatest ability to leave the company. Systems, procedures and terminology differences between sites caused confusion and delay and the housing/rental market differences between California and Tucson caused considerable consternation.

As a result, there were a number of considerations that were noted to be used as suggestions for improvement for the 'next time'. On September 12, 1994, Hughes announced the details of the new Hughes Aerospace and Electronics (HAEC) company. HAEC headquarters would be based in Washington D.C.; activities would be consolidated into four business units (Radar, Communications, Electro-Optical, and Weapon Systems); 3M square feet of office and manufacturing space will be vacated over the next 18 months, primarily affecting the Hughes Fullerton operation in California. The Fullerton plant (about 286 acres of real estate), would be basically closed down, transferring operations to El Segundo and Long Beach, California and several Southeast sites with an announcement that approximately 4,000 employees would be laid off by year-end 1995.

Things that needed improvement were deemed to be the following:

- Sell the consolidation/transition strategy to the employees
- Define the new organization and leaders early
- Identify process owners up-front
- Accelerate the decision making process
- Execute the program transition per the developed plan
- Communicate better and more frequently to all affected employees
- Reduce the time span for consolidation
- Make the tough staff reduction decisions and do it
- Facilitate empowerment
- More realistically consider the volume and capacity of affected systems

SUMMARY AND CONCLUSION

One of the most important elements in the success of the Tucson consolidation was the early involvement and participation of all Hughes customers. It was critical to sell them on the benefits of the plan and get 'buy-in'. Essential to the plan was the early selection of organizational and process leaders who could assist in the staff reduction decisions and selling of the business strategy on a one-to-one employee basis. It was recognized that once the implementation phase of the plan began, that it was important to move rapidly to sustain the momentum of the previous intense planning efforts. Tough decisions had to be made in not changing any more than was really necessary to consolidate and that after the consolidation was completed to then implement improved processes and procedures, and of course, to continually address staff size and performance.

Although the consolidation effort wasn't perfect, HMSC successfully completed a very challenging consolidation involving differing cultures, people and products. It was found that management attention was equally important at both the sending and receiving facilities; it was easy to neglect or give only passing attention to the

facilities to be shut down. Problems with differing processes and systems cannot be over-emphasized. If there was only one major lesson learned throughout this entire process, it was that successful handling of people and people related issues was dominant in ensuring consolidation success. This knowledge was utilized again in the formation and consolidation of the HAEC organization which predominantly affected the Hughes Fullerton operation. This facility will be, with minor exceptions, closed down and selected employees will be given the opportunity to continue employment with other Hughes sectors.
The process of becoming (and remaining) competitive . . . is never ending, and as the Hughes organization continues to change over time, all the lessons learned to help assure success will be recorded and utilized.

33-5
TECHNOLOGY TRANSFER:
A PERFORMANCE-BASED SYSTEM
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ABSTRACT
The paper briefly describes the performance-based system used to assess technology transfer activities at the Department of Energy facilities in Oak Ridge, Tennessee, managed by Martin Marietta Energy Systems, Inc. The progress of program activities is measured and evaluated by selected indicators. Results are discussed and benchmarked against other leaders in the field. Efforts are under way to achieve a step-function additional improvement in performance, with several new emphasis areas discussed.
INTRODUCTION
Technology transfer projects need to be performance based to 1) effectively analyze risk and benefits before initiation, 2) effectively negotiate the terms and conditions of the project with the collaborating partners, 3) maintain cost and performance tracking during the project, 4) measure the "outcome" of the project after its completion, and 5) determine if the achievements of the project can be applied elsewhere to important needs, thereby leveraging the taxpayer's and the private sector's investments.
The primary focus of this discussion will be on number (4), methodologies used to measure and evaluate the success of our technology transfer programs.
PERFORMANCE MODEL
The foundation for a performance-based system is a performance plan (including strategic and operational elements) to give direction and to establish goals that must be accomplished. Achievement needs to be measurable by key performance indicators appropriate to the activity involved. The performance model we are using is shaped after the Deming Cycle (some know it as the Plan, Do, Check, Actor PDCACycle) set forth by Dr. Edwards Deming (Fig. 1). A strategic plan was developed in which 11 key strategic goals were established. The planning horizon was the year 2000 and beyond. Our vision for technology transfer is that the commercialization of Oak Ridge technologies will have national impact on the economy, thus creating high-value private sector jobs.
Fig. 1.
This discussion will concentrate primarily on the "evaluation" portion of this model, with brief comments on the "execution" and "improvement" elements to illustrate specific points.
Performance evaluation is key to understanding and improving our technology transfer processes. Internal evaluations of federal technology transfer have traditionally focused on easy-to-count items such as the numbers of licenses and numbers of Cooperative Research and Development Agreements (CRADAs) signed. Much government- and academic-sponsored researchby recognized experts such as Barry Bozemen and David Roessner of Georgia Institute of Technologyis being conducted to explore methodology useful in the systematic assessments of the technology transfer processes. However, measuring the effectiveness is complicated, and no consensus exists to date

regarding a best method.
Technology transfer includes many different technologies and is accomplished through many different transfer mechanisms:
- licensing,
- cooperative research and development,
- industrial assistance and consulting,
- guest workers and personnel exchanges,
- use of facilities,
- contract research, and
- alliances and partnerships.

Usually, one thinks of licenses or CRADAs with the ultimate metric being jobs, invested capital, or new products and services. However, no universal measure of success is available. Technology transfer depends on the particular partner's need, technology area, and the mechanism that is employed.
The objective of our technology transfer program is commercialization of technology, with the ultimate impact being to strengthen and grow the economy. Within this context, time is an important element in the overall process. In particular, the time requirements for commercialization vary significantly (Fig. 2).
Commercialization of some technologies occurs faster than others; for example, software can take a 6- to 12-month cycle compared with advanced materials, which may
Fig. 2.
have a 10-year cycle. This time factor is important to consider when measuring and evaluating private sector benefits from a particular technology transfer program. This time element can also mean one may have to rely on process measures to evaluate performance in the early stages, followed by such indicators as capital investment, research and development spending, and jobs created as the technology moves along the commercialization cycle. In addition, one must recognize that these ultimate measures are long-term in nature, with statistical viability not being realized for 5 to 10 years.

EVALUATION

Using widely recognized process metrics, FY 1994 data indicate a strong technology transfer performance from Oak Ridge (Table I). These performance levels showed a significant increase over FY 1993 levels. However, one cannot escape the fact that performance can be affected by external factors, for example, financial stability, market dynamics, luck, and the home run hit! If you look just at the numbers from our licensing program, we are achieving good success for a 10-year-old program (Fig. 3): 23 licenses in FY 1994, with a cumulative total of 113 at the end of FY 1994. These licenses, which generated royalty revenues of $520K in FY 1994 alone, have generated a cumulative total of $2.8M over the life of the program. When comparing our performance with other Department of Energy (DOE) federal laboratory systems, Energy Systems is one of the leaders. When comparing our performance with top universitiessuch as the multiple campuses of the University of California, Massachusetts Institute of Technology, and Stanfordwe are lower; however, ages of our programs differ, as do the types of technologies being transferred.
Fig. 3.
Our Energy Systems technology transfer organization is participating in one of six DOE Reinventing Government pilot projects. The focus of this technology transfer pilot is emphasizing planning, measurement, and evaluation efforts. Program elements at DOE Headquarters and at the Operations Office as well as selected contractors are involved. Four of our strategic goals were selected to measure, evaluate, and report on performance during the life of the pilot (Table II). Each of these goals has three to six performance indicators that are tracked and evaluated. Results after the first year demonstrated performance above expected levels for three of our four goals. Expectations are that at the end of the pilot phase (1995-1996), the lessons learned will be shared throughout the DOE technology transfer community, thus resulting in adoption of more consistent measurement methods.
The technology transfer program at Oak Ridge is in transition, striving to attain a step-function improvement in performance. We are moving from the "numbers game" to a more focused attention on private sector impact aimed at improving regional and national industrial competitiveness as a result of our programmatic technology transfer efforts (Fig. 4). Our activities will place more emphasis on 1) technology commercialization via licensing; 2) proactive industrial regional assistance such as Technopolis, the 21st Century Jobs Initiative, to assist the region in utilizing

DOE-developed technologies; and 3) technical assistance to small- and medium-sized businesses.
Self-assessment and benchmarking efforts are leading to changes to improve and strengthen our licensing program. Changes are targeted to improve 1) interfacing with corporate Business Development leaders, who can be sources of future deals; 2) nurturing after completion of a deal; 3) increasing the experience base of internal staff through lessons learned workshops, intermediaries, and collaborative efforts; 4) developing improved licensing strategy methods; and 5) encouraging the entrepreneurial development of new ventures by way of new company starts.
Fig. 4.
Customer feedback is an important source of information to measure the success of any business effort. In actuality, it is the ultimate test because without satisfied and repeat customers, one will not be successful. We are making an active, consistent, and sustained effort to survey our customer base to measure and evaluate their satisfaction with our technology transfer interactions. At the same time, our organization is attempting to avoid typical traps of 1) "pestering" our industrial partners for data or 2) requiring the disclosure of competitive information considered to be proprietary in nature. This feedback is improving our understanding of the impact these interactions are having on their customers' businesses; in addition, we can use this information to drive change. Conducting customer feedback surveys also serves to market our services as being truly customer oriented.
INCENTIVES
Incentives are an integral part of any performance-based system. Three incentive systems particularly important to the technology transfer program are 1) award fee; 2) royalty sharing, including both inventors and technology maturation; and 3) recognition and reward. These incentives impact all levels of our business from the corporate level to the scientist or principal investigator. Technology transfer has high visibility with our DOE sponsor and with Congress. For this reason, technology transfer performance is designated a "special emphasis area" in the Martin Marietta Energy Systems award fee contract with DOE. Our inventors and support personnel receive 25% of the royalty funds from licensed inventions annually. This amount has reached approximately $700K over the life of the program. In addition, 25% of royalty revenues go back to the technical organizations to support further maturation of selected research and development projects. This money helps ensure the continued support of the technical community and helps provide future sources of technology for licensing and commercialization. An additional incentive is the recognition and rewards received for accomplishments from technology transfer activities. These accomplishments are integrated into our performance reviews and our reward systems and are used to promote and encourage involvement in the program. One of our strongest incentives and motivations is to improve the benefits of the program so that the value of technology transfer programs is recognized from a cost-benefit perspective. We are attempting to shift our paradigm. The goal is to move from a 100% government-funded effort to a program that is fully supported by royalty revenues. This change will make us be driven by a bottom line and will move us closer to the business environment of our private sector partners.
CONCLUSION
These are challenging and exciting times for the federal laboratory technology transfer community. Given the changing political climate and tight federal budgets, increased pressures will exist to demonstrate the contribution programs make to the national economy. Significant progress has been made over the life of the program (the past 10 years). The future challenge is to double orin some casestriple these performance levels in half the time: a difficult job . . . yes, but we believe we know the path to success.

33-6
IMPLEMENTING AN INTEGRATED STANDARDS-BASED MANAGEMENT SYSTEM TO ENSURE COMPLIANCE AT

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ABSTRACT
Los Alamos National Laboratory (LANL or the Laboratory) is developing and implementing a comprehensive, Integrated Standards-Based Management System (ISBMS) to enhance environmental, safety, and health (ESH) compliance efforts and streamline management of ESH throughout the Laboratory. The Laboratory recognizes that to be competitive in today's business environment and attractive to potential partnerships, Laboratory operations must be efficient and cost-effective. The Laboratory also realizes potential growth opportunities for developing ESH as a strength in providing new or improved services to its customers. Overall, the Laboratory desires to establish and build upon an ESH management system which ensures continuous improvement in protecting public health and safety and the environment and which fosters a working relationship with stakeholders. A team of process experts from the LANL Environmental Management (EM) Program Office, worked with management system consultants, and the Department of Energy (DOE) to develop an ESH management systems process to compare current LANL ESH management systems and programs against leading industry standards. The process enabled the Laboratory to gauge its performance in each of the following areas: Planning and Policy Setting; Systems and Procedures; Implementation and Education; and Monitoring and Reporting. The information gathered on ESH management systems enabled LANL to pinpoint and prioritize opportunities for improvement in the provision of ESH services throughout the Laboratory and ultimately overall ESH compliance.
Preliminary results of the ESH management systems self-assessment were analyzed to determine trends, and compliance management system gaps and redundant activities. Based on the study results, LANL identified short-term and long-term goals for improving the ESH compliance process throughout the Laboratory, and options for achieving the goals. The Laboratory is using this baseline of ESH management systems performance to establish a framework for identifying, developing, evaluating and implementing ESH compliance requirements. By assessing its baseline performance and utilizing benchmarking data to reeingineer the ESH compliance management process, LANL is laying the foundation in its effort to establish a Laboratory-wide ISBMS.
BACKGROUND - ESH MANAGEMENT SYSTEM SELF-ASSESSMENT
The LANL EM Program Office initiated the effort to improve its formalized ESH management system. To be successful, an ISBMS must: establish management commitment to ESH compliance; assure compliance with external and internal requirements; focus on prevention and continuous ESH improvement instead of "fire fighting;" reduce costs of compliance and increase competitive advantage; and identify related operational and ESH costs including short and long-term liabilities, thereby revealing hidden costs and potential benefits.
To accomplish this task, LANL's EM Program Office contracted Environmental Resources Management (ERM) to jointly conduct an ESH management systems self-assessment and to provide a process and framework for enhancing existing management systems. ERM's approach addresses three fundamental questions:
Where is the Laboratory Now? To determine the current performance of the Laboratory's ESH management systems.
Where Does the Laboratory Want To Be? To identify a set of goals and measurable objectives for improving ESH management systems throughout the Laboratory.
How Does the Laboratory Get There? To design and implement an Integrated ESH Standards-Base Management System to facilitate the achievement of Laboratory goals and objectives.
The scope of the self-assessment covered Laboratory-wide management systems for compliance, monitoring and prevention programs for air quality, water quality, solid, hazardous, radioactive and mixed wastes, environmental restoration, occupational safety and health, training, emergency preparedness and response, recordkeeping and reporting, measurement and improvement, hazardous and nuclear materials management, and process safety. The self-assessment addressed ESH programs at the Division, Group, and facility levels, focusing on both Laboratory research and operations.
Assessment Methodology

A proven private sector methodology, the GEMI ESAP (Global Environmental Management Initiative's Environmental Self-Assessment Program) coupled with established "best-in-class" management system elements was used to benchmark Laboratory's ESH management systems. The ESAP is a management tool designed to assess an organization's progress in meeting the intentions of the International Chamber of Commerce (ICC) Business Charter for Sustainable Development. The ESAP tool was customized to address the unique characteristics of the Laboratory and was also supplemented to include elements of the International Standard Organization's draft Environmental Management Standard (ISO 14000), as well as the Malcolm Baldrige National Quality Award criteria. Together, these standards define the "best-in-class" ESH management system characteristics and provide the Laboratory with a pathway to achieve improved performance in ESH management.

The modified ESAP is comprised of sixteen principles for effective and sound ESH management. Each principle contains a series of elements which describe the management systems and activities necessary to implement the principle. Each element contains four successively comprehensive performance levels, which allows respondents to rate the Laboratory's performance from one to four. A score of "one" indicates a baseline level of compliance where management systems are reactive and informal and regulatory compliance is not always achieved. Level "two" indicates that formal management systems are in place to maintain compliance. A score of "three" indicates the presence of an integrated ESH management system that ensures the facility always meets or exceeds compliance. Level "four" indicates that a fully implemented, integrated ESH management system with total quality management is in place that allows the facility to always exceed compliance. The protocol also requires the respondents to weight the importance of each element to arrive at a weighted average score for each principle. Figure 1 presents LANL's self-assessment results for one of the management principles - Process of Improvement. The Laboratory achieved a level of 1.5 based on the status of ESH management systems processes of improvement. System elements necessary to improve performance to successive performance levels are described at each increasing level of compliance.

Fig. 1

To conduct the assessment, representatives from 39 Laboratory organizations, plus Johnson Controls (LANL's facility support contractor), and DOE were interviewed using the modified ESAP protocol. Interviewees comprised "top-to-bottom" LANL personnel including senior and mid-level Laboratory management, facility managers, line staff, and ESH coordinators. These interviewees provided substantive insights regarding the performance of ESH management systems throughout the Laboratory. ERM also reviewed documentation of numerous Laboratory and DOE policies, strategic plans, procedures, requirements, and reports to supplement the understanding of Laboratory organizations and activities.

COMMON THEMES FROM THE SELF-ASSESSMENT

The self-assessment results were presented to the interviewees and other Laboratory personnel with ESH-related responsibilities. LANL's results are illustrated in Fig. 2. LANL's self-assessment results can be grouped into the following topics: Laboratory ESH Management Systems; Accountability, Priority, and Goals; Integration; and ESH Issues Management. The following discussion summarizes self-assessment observations on these topics.

Fig. 2.

Laboratory ESH Management System

Basic components of an ESH management system include: Strategic Planning; Organizational Roles and Responsibilities; Policies and Standards of Performance; Programs and Implementation;

Training and Communication; and Measurement of Performance. Overall, the assessment revealed that basic ESH management systems are in place for the majority of issues identified in the assessment protocol. However, many of these management systems are ad hoc, fragmented or not Laboratory-wide.

Due to fragmentation and inconsistent application, some ESH management systems are perceived as ineffective and are, therefore, not utilized by many of the respondent organizations. Once systems are established, however, they become entrenched and are not systematically evaluated for improvement opportunities.

Although an overwhelming array of ESH policies and procedures, administrative requirements and internal Laboratory standards define "what is required," a lack of

"how to" guidance for operations and facilities impedes implementation. The requirements do not assign responsibility, or define criteria for measuring performance. Consequently, without supporting management systems and limited tools to interpret, customize and implement requirements, compliance cannot be sustained.

Accountability, Priority, and Goals

Accountability is critical at each level of the organization to ensure that individual responsibilities for ESH management are defined, communicated, understood, and fulfilled. Accountability starts with top management commitment through identifying ESH priorities and goals and establishing policies and standards of performance. It also occurs through such measures as effective oversight and monitoring, routine reinforcement of positive performance, written job descriptions, written performance standards, and when performance lags, disciplinary action.

The self-assessment results indicate a lack of accountability for ESH performance throughout Laboratory organizations. Although staff and management interviewees voiced a personal commitment to ESH performance, most respondents indicated that ESH priorities, goals, and expectations are undefined, and it appears that effective systems are not in place to hold management accountable. Respondents were quick to point out that in an environment of competing priorities and resources, responsibility for ESH compliance is often shifted to individual ad hoc initiatives versus an organized, systematic, integrated effort.

Integration

As discussed in the Assessment Methodology section, Laboratory ESH management systems were benchmarked against performance criteria contained in the ICC Business Charter, ISO 14000, and Baldrige National Quality Award. A fundamental principle of these standards is the integration of ESH systems with mainstream operating and business activities of an organization. The most effective and efficient ESH management occurs by identifying and addressing ESH issues routinely within the context of day-to-day management, operating, and research functions.

The self-assessment results indicate that ESH roles and responsibilities are not consistently integrated with position descriptions, ESH procedures are not integrated with standard operating practices, and communication of ESH priorities is not generally integrated with existing forums and mechanisms for communicating Laboratory-wide initiatives and priorities.

The Laboratory has undertaken efforts to integrate ESH management as evidenced by recent ESH re-engineering activities, and development and implementation of a Facility Management Model, which will streamline and coordinate building management functions, including ESH. These are important steps in moving the Laboratory from a reactive, crisis management-mode of operation to preventive, cost-effective consideration and attention to ESH issues.

ESH Issues Management

ESH issues management is a critical function in an effective ESH management system. An issues management system facilitates the identification, tracking, evaluation, and response to new information, such as upcoming regulations or advances in technology, allowing an organization to proactively manage ESH issues.

At the Laboratory, ESH issues typically are recognized when they pose a potential problem. Extensive efforts are then made to characterize and assess the issue, often with no attempt to match dedicated resources to the seriousness of the problem, and without assigning "ownership" to ensure the issue is resolved. Extensive resources are also directed at collecting ESH data; however, little attempt is made to determine which data is important for operations, and should be funneled to the operating groups, and which data is for DOE consumption or external public relations. While these efforts to collect and disseminate information and data are necessary, it is equally important to synthesize and prioritize information for the end user. As a result, critical ESH issues may not get the attention they deserve, while other less significant problems may reap ample resources which may not be warranted. This is especially evident with regard to DOE Orders. Interviewees noted that continually shifting priorities from DOE leads the Laboratory to inefficient issues management.

Similarly, tremendous resources are expended in preparing for and responding to continuous and simultaneous ESH-related audits (by DOE, New Mexico Environment Department, US Environmental Protection Agency, and internal auditors). Audit prioritization and rationale are often not communicated, and corrective actions responding to audit findings do not focus on improving the management systems which

are, in many cases, the root cause of the deficiencies.

INTEGRATED STANDARDS-BASED MANAGEMENT SYSTEM

Results of the ESAP self-assessment, and preliminary improvement goals and objectives were presented to senior Laboratory management (the Operations Working Group) in early August 1994. Coincidentally, DOE Headquarters had begun a number of new initiatives designed to move the Department and its subcontractors from a compliance-based approach to regulatory requirements to a performance-based approach. Inherent in these initiatives are efforts to reduce DOE Orders and establish standards for documenting expectations of how work will be performed. To address this initiative, the Operations Working Group established a Standards Committee in early Summer 1994 to review current Laboratory standards and identify a process to develop and implement one consistent set of Laboratory standards. Additionally, the Operations Working Group established the Compliance Process Team (CPT) to work with the Standards Committee to evaluate specific ESH compliance-related requirements and ongoing activities throughout the Laboratory to address the requirements.

Subsequently, the Operations Working Group directed that the ESH management system project and the Standards Committee be merged with the initiative to move the Laboratory toward an ISBMS in accordance with Criteria for DOE's Standards Program. This entire effort is currently under the direction of the CPT. Key elements of the Integrated Standards-Based Management System as it has been defined by the CPT are depicted in Fig. 3.

Fig. 3.

The ESAP self-assessment established a performance baseline, including system gaps and overlaps, for broad, Laboratory-wide ESH issues. To supplement the assessment results, additional information was gathered by the CPT from throughout the Laboratory to identify and better understand specific program initiatives, efforts, and institutional functions and organizations addressing ESH compliance. This effort confirmed the preliminary ESAP self-assessment results, that redundant efforts to develop compliance standards, collect performance data and conduct audits are routinely occurring without coordination, integration, or communication. Additionally, systems are not in place to ensure implementation of standards, measure and improve compliance performance, or to hold employees accountable.

The findings were presented to an expanded group of representatives from numerous Laboratory Divisions, Groups, facilities, and levels of management at workshops conducted by the CPT in December 1994 and January 1995. The goals of the workshops were to: verify and enhance the identified list of redundant ESH compliance activities and system gaps; evaluate options for eliminating redundant activities and closing system gaps; and determine preliminary goals and actions to move the Laboratory towards an ISBMS.

To identify redundant ESH compliance activities and system gaps, workshop participants defined the "As Is" process flow of each management system element in Fig. 3. Workshop participants then defined how each system element process flow "Should Be," following the ISBMS approach advocated by DOE. This effort to define the "Should Be" process flow included establishing detailed procedures and organizational roles and responsibilities for identifying applicable ESH requirements, developing standards, implementing standards, and measuring and improving performance for each management system element in Fig. 3.

The Laboratory's ISBMS will address customer (DOE) requirements, prioritize compliance activities, define areas to be measured, and establish a continuous improvement process. The preliminary ISBMS is under review by DOE Headquarters and senior Laboratory management. The Laboratory CPT is proposing to "jump start" the system by developing preliminary Laboratory-wide standards by mid-June 1995 and finalizing the standards by November 1995. Additionally, the CPT is proposing to test the ISBMS by modifying and implementing a Laboratory standard to address radiation worker protection requirements of 10 CFR 865 as a pilot ESH compliance project.

SUMMARY

The implementation of an ISBMS will allow the Laboratory to align current efforts for assuring ESH compliance while eliminating system gaps and overlaps. By identifying goals, objectives, responsibilities, accountability, and measurable endpoints, the Laboratory can achieve its goal of cost-effective compliance assurance.

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33-7

SYSTEMS MANAGEMENT - A CLASSIC WAY TO APPLY SYSTEMS CONCEPTS AND SYSTEMS ENGINEERING PROCESSES FOR LARGE, COMPLEX PROGRAMS

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ABSTRACT

Classic systems management represents an alternate way to use conventional systems engineering (SE) techniques, particularly when dealing with large complex programs. SE has tended to emphasize specific techniques for analyses and drifted away from thoughtful consideration of the implications of work in complex systems. This technique-over-art focus is powerful because of its narrow focus. The narrow focus is clear in terms of what it depicts, but what is depicted is not necessarily complete; this condition leads to less successful systems.

Classic systems management, as articulated by practitioners in the mid-60s and early 70s (1), provides relief from this overly narrow focus. These practitioners directed attention to the broad, holistic nature of the real environment that makes up our systems. They recognized that complex undertakings involve many more influences, both external and internal. This complexity and fuzzy definition means that the systems developed to respond to complex undertakings must be able to evolve. Both external and internal influences are integral parts of the system. External influences, or externalitites, are such considerations as societal values, public policy, or markets. Internal influences, or internalities, are such factors as organizational values or the design system. In classic systems management any major human activity exists in and produces systems that are holistic, open, and dynamic. The term holistic includes factors that influence a system and creates more complex, less well-defined systems than envisioned by conventional SE. This type of system means that the use of SE methodologies is even more important. However, SE has difficulty with such systems because they are fuzzy both in terms of defining boundaries and in dealing with change. Systems problems are pared and defined to fit technique capabilities rather than the reality of the environment. We can free ourselves from the pitfall of rigid adherence to technique by using the classic systems management vision of open, dynamic systems. Such systems can integrate influences that may not be obvious interfaces in conventional systems.

The Hanford Site Systems Engineering effort is an example of the holistic approach

of classical system management. The set of system functions from this effort includes Program Management, Stakeholder Acceptance, and Acquiring Capability. These nonconventional functions are provided equal weight by the decision makers with the more conventional function of "Remedy Unsafe or Unacceptable Conditions." The overall system being created to meet these functions can be more realistic because it responds to critical influences of the real environment. This kind of system is unconventional but realistic given the scale and context of the mega-sized mission at Hanford.

Conventional systems engineering methodologies are essential tools to use in this process, but need a modified paradigm in terms of defining the environment for large complex undertakings. Classic system management provides that modified paradigm and bridges the gap left by using conventional SE on the unconventional environment for such undertakings.

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Session 34 -- LLW/Mixed Waste Treatment; Alternatives to Incineration
Co-chairs: Leon Borduin, LANL;
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34-1

THE ISSUES OF EMOTIONAL FEARS AND TECHNICAL REALITY--INCINERATION WORKS

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ABSTRACT

In the public outcry over the risks of hazardous waste incineration, it's important to fight fire with facts. Misperceptions and bad information can be overcome, but how much is the federal governments reaction adding to public reluctance to accept this effective form of waste disposal and reduction?

NO SECOND CHANCE FOR A FIRST IMPRESSION

In the arena of public reputation, how issues arc perceived is usually more important than what those issues actually are. Politics is a great example of this. But so, unfortunately, is the business of waste management--especially when it comes to hazardous waste incineration.

By way of an example, one can imagine oneself in the role of an ordinary citizen, and picture whatever comes immediately to mind when the word: "Furnace" is used. The image most people come up with is the friendly little machine in the basement that keeps them from being cold during the winter months. It's usually a pleasant association.

But something happens when the word "Incinerator" is used. All of a sudden, the mental image of the little heating machine turns into that of a bigger, more menacing, apparatus--usually either belching smoke or barely containing the fire within it. The word incinerator is widely taken to imply destruction. And this is not wrong; whatever is in the incinerator is going to be destroyed. But the implied threat to humans in that destructive force is hard to get past.

The situation is even worse if the image is a amplified to: "Hazardous Waste Incinerator." The mind's eye has now added a dimension to this picture that no amount of P.R. yet has overcome. The mental association with danger has shifted from implied to explicit--there may be radioactivity, toxic gases, noxious smoke, or invisible fumes. Psychologically we have crossed from the territory of fear into the realm of potential death.

Is this just a word game? A battle of competing images that the hazardous wastes industry has lost? I believe it is more than that. We need to use facts to overcome fear, and to learn that rhetoric can be turned into an effective tool rather than the media cudgel it has become.

FACTS FIGHT FICTION

There is a way to combat the negative images, but it takes a united and concerted effort.(1) The facts relating to hazardous waste and mixed waste incineration must be made widely and thoroughly known. For example:

Fact: incineration is the best method currently available for destroying--that's right, destroying hazardous organic compounds.
Fact: incineration reduces the volume of refractory contaminants so they can be recovered if useful or stored if not.
Fact: incineration is not perfect, but it works; there is no comparable alternative.
Dempsey and Oppelt point out: "Of all the 'permanent' treatment technologies, properly designed incineration systems are capable of the highest overall degree of destruction and control for the broadest range of hazardous waste streams. Substantial design and operating experience exists in these areas and a wide variety of commercial systems are available.''(2)
Now this is not to say that there are no problems associated with incineration of hazardous wastes. There are, of course, both technical problems and real risks. By taking a technical approach to the problems and a comparative risk-analysis to the dangers, however, it can be shown that incineration not only makes sense as the best way to deal with much of our hazardous waste materials, but in some cases incineration should be considered the primary or even sole method of disposing of them.
According to one Environmental Protection Agency estimate, some 150 million tons of what is classified as hazardous waste are generated each year. And that figure is increasing over time.
More than 75 percent of hazardous waste is treated. 20 percent is placed in landfills, injection wells, or other similar methods that may well be neither permanent nor dependable. Poor containment, flooding, leaching, or other disturbances can allow substances once labeled "disposed" to become new or different disposal problems.
Less than one percent of our hazardous waste is recovered and used again.
And less than two percent of what remains of our annual output of hazardous waste--some two-and-a-half-million tons--is burned. Approximately half that amount is burned as waste in incinerators, the rest is used a fuel in boilers, cement kilns, or industrial furnaces.
The figures are just for hazardous waste. The amount of mixed waste being burned is even smaller, with more than 80 percent of commercial mixed waste being treated in boilers, cement kilns, and industrial furnaces. For some types of mixed wastes--like those generated by medical facilities using low-level radioactive materials in their diagnostic procedures--incineration is the only method available for treatment. The labs that generate this kind of mixed waste have no other means available to them for disposal.
This potpourri of percentages adds up to one simple sum: at this time, very little hazardous waste is being incinerated, and very little of that is mixed hazardous waste. Is this because incineration is a poor way to treat hazardous waste? Is this because of some flaw in the incineration process? If we look to the facts, the answer is no.
Fact: incinerators are theoretically capable of complete and permanent destruction or removal of organic constituents. In practice, the performance is almost that good--good enough that well-operated incinerators routinely exceed the 99.99 percent efficiency required by regulations.
Fact: while the remaining solid waste may still contain a fraction that is hazardous--and mixed waste will still contain a fraction that is radioactive--the volume of waste that comes out of an incinerator is usually smaller by orders of magnitude than the volume of waste that goes in. What's more, the residue is much easier to contain, transport, and store.
Fact: even when incinerators are not operating at maximum design efficiency, the combustion process is so efficient in and of itself that even an incinerator burning at far less than its 99.99 percent ideal will not cause significant health risks. No matter how a fire gets started, once it's underway it's pretty hard to put out until whatever it's burning is destroyed. Thus also with incinerators.
One real problem with incineration is emissions, and there is no getting around the fact that once an emission makes it out of the stack, the ability to control it ends. This is probably the main issue that incinerator critics can point to, because these emissions can pose a health risk to people who live nearby.
But we can also fight fears of hazardous waste incineration and emissions by emphasizing the facts behind the comparative risks of its alternatives.
The following discussion skews the available data to the disadvantage of the

incinerators--to assume that the data show incineration to be safer than it actually is--to assume that scrubbers don't work as well as we know they do, and that destruction efficiencies have been dramatically overstated.
The federal government's basic unit of measurement in calculating the health hazard posed by hazardous wastes is called the incremental lifetime cancer risk--the chance that an exposed individual will develop cancer over a 70-year life span due to exposure to emissions. The threshold value for most hazardous air pollutants is one chance in one hundred thousand for exposure to air containing one microgram of contaminant per cubic meter of air. This method of risk measurement would result in a threshold value of an additional 2500 cancers per lifetime--or 36 per year--for the population of the United States.
According to the EPA, the risks from emissions from hazardous waste incinerators is 21 lifetime cancers--or 0.3 annual cancers--as a result of emissions from the incineration of hazardous waste, less than one percent of the threshold value given above (1)
What are the risks for some common hazards? According to the EPA, air pollution from motor vehicles results in 1,115 additional cancers per year in the U.S. population as the result of exposure to organic emissions--more than 78,000 lifetime cancers. Even if this estimated figure is three times as high as the real risk, it still makes auto exhaust two orders of magnitude more dangerous than incinerator emissions.
Here is another case: the incremental risk of disposing of hazardous waste in a Treatment, Storage, or Disposal Facility--or TSDF--rather than burning it in an incinerator.
According to the EPA, emissions of dioxins from TSDFs are responsible for more than 6,600 lifetime cancers, or 94.5 incremental cancers per year in the U.S. population. As our understanding of the long-term effects of dioxin improves, this risk may be revised upward. To make the worst possible case for incineration, I'll assume that incinerator emissions are three times as hazardous as EPA's figures, and that the risk from TSDFs is only one-third of EPA's estimate. Even with that adjustment, Treatment, Storage, and Disposal Facilities represent 35 times the risk of incinerators.
How about a comparison with other processes involving combustion? Is an incinerator more dangerous than burning municipal waste? No--statistics show dioxin emissions from burning municipal waste represent 20 times the cancer risk.
Smelters? Incinerators beat the formaldehyde emitted by smelters five-to-one.
That cozy wood fire in the fireplace? The products of incomplete combustion (PICs) coming out of the chimney are responsible for 100 times the annual cancers than would result from a hazardous waste incinerator running at one-third efficiency.
These are the facts. And even though incineration poses a relatively small risk when compared to more common hazards, public opposition is intense and seems to be growing. But those people who are unwilling to accept the very small risks associated with incineration often tolerate other, greater dangers willingly.
Because another fact is that, as individuals and as a society, we voluntarily accept risks on a daily basis--sometimes knowing the odds, frequently disregarding them.
For example; automobiles kill more than 40,000 of us each year and maim an additional quarter of a million of us. Yet most of us--even those of us who live, work, and play in major cities with excellent mass transit systems--would not think of giving up the family car.
Sometimes the risks we accept are tradeoffs for other risks. The clinical trials of the drag Tamoxifen are an example of this. The drag appears to be a breakthrough for treatment of breast cancer. But at the same time it may triple the chance that a woman taking it will develop endometrial cancer. Why accept such a risk? Because the statistical likelihood is that the drag will save more lives than it threatens. In this case, the increased risk of endometrial cancer is offset by the benefit of preventing a greater number of cases of another form of the disease.
But frequently our own risk acceptance is based more on personal preference than on a careful risk analysis. Actor and bodybuilder Arnold Schwartzenegger--former President Bush's advisor on physical fitness--is seldom seen off-camera without a cigar in his mouth. Baseball players who need to be in passable physical shape--when we have a baseball season--frequently chew wads of carcinogenic tobacco. Women who talk over a glass of wine or a cup of coffee about starting their families may be drinking the very substances which decrease their fertility.

Our current methods of storing hazardous wastes must include the possibility that fences will not keep out vandals, that liners will not prevent leaching, and that containers or treatment systems will fail or perform badly. So in a nation where our cars and our fireplaces pose a greater and more immediate threat to our health than the incinerators which would reduce or eliminate vast amounts of stockpiled hazardous wastes, why are we as a society reluctant to move toward the technology that could assist us?
Some negative public reaction is to images and not reality.
THE PACE OF PROGRESS SLOWED
When the government agency set up to regulate and oversee how we as a nation deal with the toxins that we as a nation produce takes a public stand that waste combustion policies need to be re-examined, the logical result is that people are going to wonder if the old rules were somehow lax or "wrong." And when that agency says it's decided to decrease reliance on incineration and instead pursue the course of reducing waste generation, it makes incineration look either unnecessary or undesirable, or both.
This is just what the EPA has done over the past couple of years.
But waste reduction and incineration are not an either/or proposition. We are generating those 150 million tons of hazardous waste every year. Those 150 million tons generated every year and not disposed of before the next years wastes are generated. And we're only incinerating less than two percent of that waste. So that means the leftover wastes, the wastes that were not buried or encased are with us. Waiting.
Yet in a Combustion Strategy memorandum this past May, the EPA called for increased inspection of existing incinerators, and a tightening of permitting rules for proposed incinerators.(3) In addition, EPA says it's giving "low priority" to issuing those permits for new
incinerators. Some of the agency's most recent statistics show this policy to be well-documented. The total number of incinerators in the U.S.--commercial and non-commercial, demilitarized and remedial--is 164. Those are the ones either with permits or on interim status, awaiting final permits. By the end of fiscal year 1995, EPA estimates there may be an additional 14 incinerators permitted. By the end of FY 1996, 15 more. This is not the sign of a booming incinerator industry.
The statistics for boilers and industrial furnaces are also less than encouraging. In a universe of 142 permitted and interim status facilities, them may be another 10 BIFs by the end of fiscal year 1995, and by the end of FY 1996, 31 more permitted boilers and industrial furnaces. These are the facilities that use hazardous waste and mixed waste as fuels.
Waste reduction is an important goal. The creation of less waste materials will mean lowered risks from whatever the source. But what is not openly taken into account is that unless we treat more waste, we are not decreasing our risks in any meaningful way. The stockpile stays put. The risk remains about the same.
And the EPA recognizes that. In that same May 1994 memo, EPA Administrator Carol Browner is quoted as observing that "remediation wastes present a different circumstance than newly generated wastes and, given the finite set of options for dealing with historic cleanup sites, combustion may be the most appropriate remedy. In addition, waste minimization opportunities at cleanup sites are usually severely limited." (2)
The EPA memo goes on to delineate agency guidelines which slow that "most appropriate remedy" by requiring site-specific risk assessment for incinerators, site-specific emission levels for furans and dioxin, unspecified assessment methods which the EPA says should "ensure that an incinerator can be operated in a manner protective of human health and the environment," and a vague call for "effective public participation" in the process of developing proposed incinerator sites.
While the EPA is saying that for existing hazardous wastes--especially at sites on the Superfund National Priorities List--incineration may be the most appropriate means of reducing the stockpile, the agency is at the same time promoting an internal policy of painstaking assessment that is resulting in delay of permitting for some incinerators, and complete abandonment of other proposed incinerator plans.
And how is the public concern about the safety of incinerators addressed by this policy? In essence, it isn't. A provision for reciprocal exchange of real information that might allay anxiety about the incineration process. The voice that's left out of the EPA's policy is the voice that can cite the facts about the

risks, explain the technology behind the process--the industry that builds and operates incineration facilities across the country and around the world.

CONFRONTING THE 'NIMBY' MENTALITY

Apprehension cannot be overcome by repetition. Encouraging individuals or groups to express and repeat their fears does not change those fears. So the EPA's public comment approach is lacking a key element--dialogue.

Hearts and minds can be changed. One good example of this is the growing number of American communities that embrace and endorse recycling. Ten years ago--or 20-- it would have seemed ludicrous that would all be picking through and separating our trash on a weekly basis. American industry has adopted a plastics numbering system to help distinguish containers.

Recycling has come to pass because we as individuals and communities have come to understand that the efforts we make are an accommodation for the greater good. Plenty of consumer education went into our "greening" and plenty of industry adaptation accompanied it. Both sides learned to see the benefit of cooperation.

The same can be with incineration. But we need to see the carrot--not the same old stick. Communities have grown used to fighting facilities which are seen as society's necessary evils--like prisons, landfills, and even highways. The rallying cry often is "Not In My Back Yard"--the zoning acronym NIMBY.

The NIMBY mentality needs to be seen as a luxury no community can afford. For any one group of people--through affluence, obstinacy, or whatever force--to deny its role in the common scheme of society is wrong. If we need the prisons, they must go somewhere. Likewise the landfills, the highways, and incinerators.

The federal government may have a role to play here, as will local governments and community groups and industry representatives. But the role should not be one of asking whether a particular place "wants" a hazardous waste incinerator; who would answer such a question in the affirmative? The approach should instead be one of consensus-building. If a particular area is producing a concentration of hazardous materials, the questions need to be: where should an incinerator be located to pose the lowest possible threat to health and public safety? what kind of incinerator will best perform the needed task? and what will be done with any products of incomplete combustion?

If the infamous WTI incinerator in East Liverpool, Ohio, had not been built within sight of an elementary school, the community reaction to it would have been appreciably different.4 East Liverpool is a worst-case example of how site selection cannot be performed in a vacuum, and how dialogue before the fact could have averted a huge community outcry and much hostility.

The fact is that incinerators are not risk-free, but they bear up well by comparison to other methods of hazardous waste disposal and other socially-accepted risks. The current level of suspicion and anxiety regarding incinerators can be reduced through the sharing of expert information about the need for and process of hazardous waste combustion, and early involvement of community and industry representatives even before a particular incinerator site is chosen.

CONCLUSIONS

The facts about hazardous waste incineration are supportive of using this important technology for hazardous and mixed waste disposal.

The facts are that incineration works. It destroys organic contaminants, reduces the amount of remaining waste that needs to be stored, and can be shown to pose a much smaller risk than any alternative treatment developed thus far.

While the EPA's recent focus on risk assessment may be beneficial in theory, the agency needs to reexamine what it is saying to a fearful public about incineration and slowing the permitting process with layers of site-specific requirements. If the government will begin to promote dialogue among all concerned parties, perhaps we will all start to agree on how we can move forward together to address the hazardous waste problem through the judicious use of incinerators.

And finally, we need--as neighborhoods and individuals--to get over our insular approach to our communities. The problems of hazardous waste belong to us all, and must be confronted as a communal challenge. The days of the NIMBY mind set are past. For our common good, we must work together to resolve the conflict over where, not whether, hazardous waste incineration is in our future.

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34-3
RADIOACTIVE AND MIXED WASTE INCINERATION: AN UNCERTAIN SOLUTION

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ABSTRACT

There is a sound basis for many criticisms of waste incineration. Incinerators emit a wide range of toxic substances, many with poorly understood toxic properties. While well-maintained, state-of-the-art incinerators can keep emissions to low levels, U.S. regulations have tended to provide inadequate assurances of health and environmental protection for two principal reasons: 1) they do not ensure that low emissions are maintained during routine operations through frequent monitoring of key pollutants, especially toxic metals; and 2) emission limits are based on risk assessments that involve a high degree of uncertainty and often fail to consider important factors. In making decisions about the appropriateness of incineration, many key issues are often poorly considered, such as the variability of waste feeds, the viability of alternatives on a waste-specific basis, and equity issues. Incineration has too often been used as a catch-all treatment method, while other technologies that may be more appropriate for specific waste streams have not been thoroughly investigated.

INTRODUCTION

Incineration has come under increasing criticism from environmentalists and citizens impacted by these facilities. There is a factual, analytical, scientifically sound basis for much of this criticism, especially with respect to incineration of radioactive and mixed waste streams. In this paper, we provide examples of policy decisions and technical issues which underlie much of our criticism. In preparing this paper, we have drawn upon our experience reviewing an incinerator under construction at the Department of Energy's (DOE) Savannah River Site, known as the Consolidated Incineration Facility, as well as information gleaned from a review of incineration technology completed by the Institute for Energy and Environmental Research in 1993.

We do not, however, attempt to provide a complete catalogue of the criticisms that have been raised about incinerators. Many issues are specific to individual facilities or to the relationship between facility operators and local communities. An effective effort to address concerns about a specific incinerator should include a review of site-specific information and involve local communities and facility managers.

Incineration may be an effective treatment for many waste types, but the tendency to adopt incineration as a catch-all treatment technology for widely varied waste streams is not supported by the current state of the technology and existing regulatory controls. Successful incineration of waste material depends on relatively uniform and consistent waste feed...In practical applications considerable variation in feed materials may be present," as the U.S. Environmental Protection Agency (EPA) noted in a 1991 document on incineration of radioactive and mixed wastes (1).

While high temperature can destroy toxic organic compounds with high efficiency, it can lead to volatilization and potential release of metals and radionuclides. Yet real-time (continuous) monitoring of some individual toxic emissions of concern such as heavy metals and dioxins has not been demonstrated. Indeed, given limitations inherent in the combustion process, incineration may not be at all suitable for wastes containing significant quantities of volatile metals or long-lived radionuclides.

In the sections below, we touch briefly on each of the following issues: potential emissions and uncertainties associated with those emissions; the need for more frequent, more comprehensive monitoring; pitfalls of risk-based regulation; and the importance of public participation in waste management decision making.

WHAT DO INCINERATORS RELEASE INTO THE ENVIRONMENT?

Incinerators release a wide variety of substances; the toxicity of many are poorly understood. Some of these emissions are highly carcinogenic or cause other health effects at low concentrations, are long-lived, and can accumulate in the tissue of animals and humans.

Incinerators can reduce the volume and total toxicity of the waste burned. By reducing the volume of solid and liquid materials, incineration can reduce releases following disposal for some wastes. Incinerators cannot, however, break down all of the complex chemicals fed into them. Some portion of these chemicals, generally a very small percentage, escapes the combustion process unaltered or partly burned. By-products of combustion also are formed from the recombination of compounds following combustion, as with all combustion processes. Some of these "products of incomplete combustion" (PICs) are more toxic per unit of weight than the original waste; the toxic properties of many PICs are not well-understood, while others have not even been well-identified. The EPA has described the problem as follows (2):

> ... small quantities of "new" organic compounds may form from the breakdown and recombination of the original compounds. These products are called "products of incomplete combustion" (PICs)...Among the types of compounds found in various analyses of PICs are some that are considered toxic, including dioxins and dibenzofurans.

The total quantity and toxicity of PICs created in full-scale incinerators is uncertain (3). Data are limited in part because monitoring is done only on an infrequent basis during trial burns when incinerator maintenance and operator performance are at their peak (4). Further, the EPA has found that at most 60 percent of PICs emitted during past test burns have been chemically identified (5). While the most toxic PICs known, dioxins and dioxin-like compounds, apparently can form during any combustion process, there is a clear relationship between chlorine content of the waste and dioxin formation. For instance, in one study commissioned by Germany's Environmental Protection Administration, Umweltbundesamt, researchers found that incineration of chlorinated products contributed disproportionately to dioxin formation (6). Other characteristics of the waste can confound this relationship however (e.g., the presence of sulfur in the waste).

Even though many air pollutants are emitted in very small quantities compared to the volume and weight of the original waste, their high chemical toxicity makes pollution control particularly crucial. The health effects of many of these compounds (e.g., dioxins), even when present in small quantities, are the focus of much of the controversy surrounding incineration. Some data suggest that the background environmental concentration of dioxin is near levels that could cause adverse effects on the human immune and reproductive systems (7). Yet, the focus of regulation is not on such effects, but on cancer risk.

Incineration does not destroy metals or reduce radioactivity of wastes. Radionuclides and metals either leave the incinerator in the stack gas, or are left behind in ash, filters, or waste water. Radioactive waste incinerators, when equipped with well-maintained, state-of-the-art pollution control systems, can capture all but a very small fraction of the radionuclides and metals in the waste. Radioactive particles that do escape, however, are small in diameter and can be carried by winds over long distances. Due to their small size, fine particles (radioactive or otherwise) can be inhaled more easily than larger particles, and become lodged in the lungs.

At Lawrence Livermore National Laboratory (LLNL), an internal review panel raised some of these issues when recommending against a proposed mixed waste incinerator in the increasingly urban Livermore, California community (8):

> We have never been comfortable with the EPA's position that incineration of mixed waste to eliminate its chemical toxicity should be the first procedural step and burial of its radioactive residuals the second step. This approach commits to the volatilization of important radionuclides, including tritium, carbon-14 and several isotopes of iodine. Furthermore, the incineration of non-volatile nuclides, including those of uranium and plutonium, leads to a finite, although exceedingly small, probability of radioactivity being emitted from the incinerator's stack.

We view incineration as a violation of the cardinal principal of radioactive waste management; namely, containing radioactivity rather than spreading it. The LLNL panel, nevertheless, went on to recommend incinerating the waste in a more remote location, pointing out that an incinerator at the Idaho National Engineering Laboratory, "is allowed to, and currently willing to, process [Livermore] Laboratory's mixed- and low-chlorine-containing waste into a form suitable for burial (9)." While the remoteness of the Idaho incinerator may reduce the number of exposed people, it creates additional transportation risks and does not resolve the inconsistency between incineration and the principle of containing radioactive waste. It also does not resolve equity issues associated with urban centers transferring risks of disposal to rural communities.

Air Pollution Control Devices

Incinerators require careful operation and maintenance to ensure optimal performance. They rely on complex, often fragile, pollution control systems to capture volatile metals, radionuclides, unburned toxic organic compounds, and by-products discharged from the combustion chamber. Emissions can be expected to increase as a result of poor maintenance, carelessness or error on the part of the operator, and facility aging (10). If pollution control systems are improperly maintained, emissions can increase substantially. For example, in a compliance test at one new municipal waste incinerator, lead emissions increased by roughly two orders of magnitude due to a bag filter coming loose in a baghouse system composed of several hundred individual bags (11)(see Table I). Radionuclide filtration systems are especially vulnerable to failure and leakage. The High Efficiency Particulate Air (HEPA) filters most commonly used for fine particle capture "can sustain structural damage relatively easily under conditions of higher-than-designed-for rates of airflow, shock waves (for example, as a result of incinerator explosions), higher-than-designed-for temperatures, excess humidity, and excess particulate deposits (12)."

Secondary Wastes

Incineration produces secondary wastes in the form of wastewater from wet scrubbers and ash. The more efficient the pollution control system, the more PICs and metals that are concentrated in secondary wastes--primarily ash. Although typically at very low concentrations, some pollutants in treated wastewater may accumulate to potentially harmful levels if the receiving waterway is already polluted. For example, polychlorinated biphenyls (PCBs) show significant patterns of accumulation in fish (13). Some epidemiological evidence suggests that PCB levels in some people could already be at body burdens that are producing developmental and behavioral effects in offspring (14).

The majority of toxic substances leaving the combustion chamber end up in the ash. These residues can leach into groundwater and contribute to off-site exposures. On one hand, the greater surface-to-volume ratio of ash relative to the original waste form tends to increase leaching for a given chemical solubility. On the other hand, incineration or post-incineration treatment may reduce leachability of the ash (e.g., by converting some constituents of the ash to less soluble oxide forms). The net effect will depend on specific circumstances and must be evaluated on a case-by-case basis.

REGULATORY INADEQUACIES

Federal regulations on emissions from hazardous and mixed waste incineration in the U.S. are inadequate for three principal reasons. First, they fail to require continuous, or even frequent, monitoring of key emissions. Second, they are based on risk assessments that involve a high degree of uncertainty and fail to consider important factors. Third, they are focused primarily on cancer risk, though other risks, such as reproductive risks, may in specific instances be far more important.

Emissions Monitoring

Frequent emissions monitoring of individual toxins is critical to evaluate the health and environmental impacts of incinerators for two key reasons:

- combustion products can vary markedly depending on waste inputs and operating conditions, and
- pollution control systems can deteriorate or fail during normal operations.

While incinerator performance tests are designed to stress a facility by requiring operation under non-optimal conditions, the EPA has stated that "many incinerators run very efficiently during a trial burn...it may not be possible to achieve this

high efficiency [referring to carbon monoxide levels--one indicator of PIC formation] on a consistent basis (15)." The EPA has further stated that, "In addition to differences in waste feed compositions, incinerators may have uneven temperature distributions, poor mixing conditions, or low temperature areas, which might lead to poor combustion conditions that result in an increased likelihood for the formation of PICs (16)."

These fluctuations have occurred frequently at some facilities. In a survey of records from 29 incinerators burning hazardous waste (two of which were DOE mixed waste incinerators), the EPA and the U.S. Occupational Health and Safety Administration found that over 40 percent of the facilities required emergency feed cutoffs more than a hundred times in during a 30-day study period (17). Waste feed cutoffs indicate that permitted limits on carbon monoxide or other indicators of combustion efficiency were exceeded. Describing the formation of PICs during these and other "upset" (or non-optimal) conditions, the EPA's Science Advisory Board noted that "the amount and composition of PICs will depend in a complex and unpredictable way on the nature of the perturbation."[emphasis added](18). In tests at one full-scale hazardous waste incinerator, researchers found that formation of some toxic organic compounds increased 10-fold or more during observed upset conditions (19). In compliance tests at a state-of-the-art municipal incinerator, measured total dioxins and furans at the stack were 70 times higher during "unstable operation" than under "normal conditions (20)."

Federal regulations in the U.S. rely on carbon monoxide (CO) and total hydrocarbon (THC) limits to determine incinerator performance during day-to-day operations. However, low levels of these compounds do not necessarily indicate low levels of products of incomplete combustion. According to the EPA's Science Advisory Board, "The correlation of CO, THC, or other parameters with combustion efficiency and PIC emission is weak for some conditions and some combustion devices (21)."

Analysis of known toxic PICs such as individual dioxins and furans is only conducted on a limited basis, at most annually, due to the time-intensiveness and high cost of the process. A system has yet to be developed that can continuously, or near-continuously, monitor even the most well-studied PICs emitted from incinerators, let alone the poorly understood components of the combustion gas. Analysis of toxic metals is also costly, but more feasible on a near continuous basis. Germany requires some incinerators to sample for a range of individual metals once a week. For example, the Karlsruhe Nuclear Research Center's incinerator in Germany is required to sample weekly for 13 individual metals (22)(see Table II). At the 3M corporate incinerator in St. Paul, the company has been using an innovative metals sampling technique on a weekly basis for several years (though the procedure has some limitations)(23). At one hazardous waste incinerator in Biebesheim, Germany, toxic metals are sampled daily (24).

Continuous or near continuous monitoring of key pollutants could contribute to improved compliance and enforcement at incinerators and build public confidence in the efficacy of air pollution control. Improved public access to this data could also build public confidence and contribute to better enforcement of emissions limits. Data could be transmitted directly to a public venue, in the same way that some states now require "telemetering", or direct transmission of basic emissions data such as CO, to the regulatory agency (25).

Risk Assessment

The EPA has adopted a strategy of setting facility specific limits on individual pollutants based on risk analysis focused on cancer risk.* Assessing the health and environmental risks of incineration, however, involves a high degree of scientific uncertainty. Assumptions must be made where data are absent or highly variable. While risk assessments generally include many highly conservative assumptions, the overall conclusions may not be conservative if key variables are omitted or underestimated. For example, risk assessments of exposure to dioxin emissions from incinerators that have assumed only the inhalation pathway have been found to dramatically under represent the exposures of some individuals. Researchers have estimated that food chain exposure can exceed doses from inhalation by one to three orders of magnitude (26)

Although food chain risks of dioxins and furans are now being incorporated into permitting and compliance decisions, many potentially toxic compounds released from incinerators have not been adequately characterized. As noted previously, a large portion of PICs emitted during test burns have not been identified. These unknown

compounds could be more toxic than identified compounds or could have synergistic effects when combined with other substances. Similarly, releases from other facilities could produce synergistic effects when combined with incinerator emissions. These factors introduce large uncertainties into risk assessments.
Risk assessment procedures typically focus exclusively on cancer-risk potential when evaluating carcinogenic substances, although other effects may be of greater significance at low exposure levels or for some pollutants. The EPA has concluded, for example, that the effects of dioxin on reproduction and development are of greater concern than dioxin's ability to cause cancer (27). Summarizing key findings of the EPA's dioxin reassessment panel, EPA's Assistant Administrator for Research and Development, Erich W. Bretthauer, stated in a late 1992 memorandum to the EPA's Administrator, "Some data suggest that [non-cancer] effects may be occurring in people at body burden levels that can result from exposure at, or near, current background (28)."
Further, risk assessments often fail to quantify background levels of pollutants that can amplify risks; the cumulative effects of releases from various sources over time; the existing body burdens of toxic and carcinogenic pollutants in the local population; or significant past exposures to toxins. For example, children already exposed to toxic levels of lead from paint or other sources may be harmed by quantities released from incinerators that on their own would not be likely to cause health effects. According to the American Academy of Pediatrics, 2 to 4 million children already have blood lead levels sufficient to slow growth and delay development (29).
Finally, risk-based regulation does not reflect many values important in decision making. For example, equity is a legitimate consideration when reviewing who benefits from waste generation and who is affected by waste treatment and disposal. If a community has already been disproportionately impacted by activities designed to benefit the society as whole, society has an obligation to redistribute additional costs or at least recognize that equity concerns justify the financial and regulatory commitment necessary to minimize risks to the lowest level technically feasible. Risk-based regulation also fails to address economic and cultural impacts on communities near incinerators. These and other values, such as urban-rural equity, are vital to democratic decision making and underline the limitations of using risk assessment alone for decisions potentially affecting entire communities. In addition, regulatory officials and facility managers must recognize that small risks, if involuntarily imposed, are much less tolerable than higher risks one voluntarily chooses to face.
PUBLIC PARTICIPATION: THE CIF EXAMPLE
The decision-making process leading to the construction of the Consolidated Incineration Facility (CIF) at the Savannah River Site illustrates the need for more thorough analysis of specific waste streams, treatment options, and greater stakeholder participation. The CIF started in the mid-1980's as a plan for combining three incinerator projects into a single, consolidated facility. That was certainly a reasonable idea which should, at least theoretically, have reduced construction and operating costs. In subsequent years, however, the capacity needs changed due to changes in activities generating waste. Rather than reevaluating the CIF in light of these circumstances, the DOE chose to proceed with construction.
It was only after extensive public pressure that the DOE agreed to review the CIF. The agency initially identified compaction as a viable alternative for perhaps 50 percent of the projected waste feed. Also as part of the CIF review, the DOE was encouraged to reconsider supercritical water oxidation as an alternative for specific wastes (30). This closed-system is being used on a commercial scale for some hazardous wastes and a demonstration plant for mixed wastes is expected to operate in 1995 (31). The DOE has recently proposed vitrification as a potential alternative to incineration for much of the waste, noting that this option may put secondary wastes in a more stable form for disposal. Public participation encouraged the DOE to reevaluate the CIF incinerator. By expanding public input in decision making, the DOE may have identified a more effective slate of treatment options for the Savannah River Site.
CONCLUSIONS
Though incineration has crucial limitations, incinerators should not necessarily be rejected as waste treatment facilities. Rather, the appropriate role of incineration should be better defined, and the regulatory system should be changed

to reflect valid concerns about the technology. Specifically, we conclude:

There is a need for proper segmentation and control of the waste feeds to incinerators, in order to reduce health and environmental risks from their operation.

Given variability in incinerator performance, continuous, or near continuous monitoring of emissions is necessary to ensure compliance with regulatory limits on specific compounds. Frequent monitoring would enable and encourage better operating performance and build public confidence in the efficacy of pollution control at incinerators.

All reasonable efforts must be made to minimize involuntary risks to local citizens from waste incineration.

The highly uncertain nature of risks from incineration dictates special efforts to include democratic participation in decision making and to thoroughly investigate alternative waste management solutions on a waste-specific basis.

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INCINERATION AND NON-INCINERATION OPTIONS FOR TREATING A BROAD SPECTRUM OF HAZARDOUS AND RADIOACTIVE WASTES

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ABSTRACT

The mixed low-level hazardous waste in the U.S. Department of Energy (DOE) complex consists of organic and inorganic solids and liquids comprising a wide variety of materials contaminated with radioactive substances. Treatment systems are needed to destroy the hazardous organic materials and to immobilize any hazardous inorganic materials, along with the treatment residues and radionuclides, in order to comply with the regulations promulgated by the Environmental Protection Agency (EPA), state agencies, and the DOE. This paper briefly describes 19 fully integrated systems for the treatment of mixed low-level waste that are being evaluated by DOE in a comprehensive Integrated Thermal Treatment Systems Study.

INTRODUCTION

DOE's Environmental Management Office of Technology Development has commissioned an Integrated Thermal Treatment Systems (ITTS) Study to assess alternative systems for treating contact-handled, alpha and non-alpha mixed low-level radioactive waste (MLLW). The ITTS study is a systematic engineering evaluation of a variety of integrated MLLW treatment systems. The emphasis is of the total system -- from the receiving dock through disposal of the treated wastes. A total system approach is critical since the selection of one treatment technology over another can impact other parts of the system, such as waste sorting and sizing or immobilization of the residues.

Some innovative technologies under development within DOE and industry are being proposed as alternatives to incineration. However, they may not be a direct substitute for an incinerator as they may require substantially more or fewer support facilities because of different waste input requirements (e.g., for composition, physical size, or both). They may also have different requirements for air pollution control (APC), residue stabilization, etc. Therefore, before any technology is selected as an alternative to something else, an evaluation must be performed to determine the system changes associated with the use of the new technology. Is one set of problems replaced with another? Can the same wastes be treated or will some waste types be excluded? Will the technology produce the same or a different quantity of residuals? What auxiliary systems will be necessary to

treat the portion of the waste stream not treated by the alternative technology? Will the residuals have better or poorer long term stability in the environment? This study has attempted to address these and other related questions for nineteen systems using different combinations of technologies.

The technologies included in this study consist of both conventional and innovative treatment concepts combined in ways to examine the effect of the primary treatment system, air pollution control, type of oxidant, type of residue stabilization, type of energy (methane vs electricity), and other subsystems. The systems selected not necessarily represent concepts that would be advocated by the investigators, but the selections all have one or more attributes that appeared at the outset of the study to merit evaluation in the system context.

A major constraint in this study was the requirement for each system to treat the same waste profile and at the same mass throughput. If a particular primary unit operation could not accept a certain waste type, an additional treatment subsystem was added to accommodate the excluded waste type. The study developed a set of consistent functional requirements and then proceeded to develop preconceptual designs and relatively rigorous mass and energy balances. The preconceptual designs consisted of functional diagrams, facility layouts, equipment lists, and construction and operational cost estimates. Each system had the same waste processing throughput, treated the same waste profile, and was required to produce a waste form from the residuals that was believed to be able to pass EPA Land Disposal Restrictions (LDR) under the Resource Conservation and Recovery Act (RCRA) and DOE Performance Assessment Objectives identified in DOE Order 5820.2A. Since this was a conceptual study, only engineering judgment could be used to determine acceptability for the final waste forms. Similarly, air pollution control systems were configured such that the systems would represent state-of-the-art concepts producing air emissions at least an order of magnitude lower than current requirements.

STATUS OF THE STUDY

Two publications from the first phase of the study are currently available: 1) a description of the waste profile used for the analysis of the mass and material balance, and 2) the documentation describing the design of the first ten systems analyzed (1,2). The study is nearing completion and the final results will be soon be published (3).

WASTE PROFILE

The waste profile used in this study (1) was derived from a DOE report (4) that was compiled from inputs from 49 DOE sites. The data were known to be imprecise but represented the best possible inventory that could be developed at the time. The waste in storage at the DOE sites comes from operations that span almost 40 years -- long before requirements for good record keeping, waste treatment, and disposal existed. Consequently, the source data on physical matrices generally do not have a significant level of detail. Nevertheless, an effort had to be made to define the DOE MLLW in terms of physical matrices in order to get some estimate of the chemical composition of the wastes for use in performing mass and energy balances for each of the treatment system concepts. After assigning a chemical composition to the physical matrices and combining the data, the final waste profile was developed for the mass balance calculations as shown in Table I.

TREATMENT REQUIREMENTS

The waste consists of many physical matrices contaminated with RCRA-listed or RCRA-characteristic hazardous wastes. The wastes include significant quantities of mercury and lead, neither of which should be subjected to thermal treatment since, because of their volatility, they will pass through a thermal treatment system and pose a challenge to the air pollution control system. Thus, a requirement for segregation of the bulk lead and mercury was imposed on the system design. These metals are treated in separate subsystems. Also, two separate ferrous metal treatment subsystems were included in the baseline concept -- one for removal of surface contamination and one for bulk metal melting. A basic premise of the system concept was that real-time radiography could be used to examine all the containers, that containers having lead, mercury, or ferrous metal in bulk quantities could be identified, and that those materials could be removed for processing in the appropriate subsystem. Finally, a "special waste" subsystem was included in the preconceptual design, with building space and equipment allowances provided to accommodate the unknowns.

With some exceptions, the quantity of the RCRA contaminants and radionuclides in the

DOE MLLW can generally be considered at the trace concentration level. Toxic metals and organically contaminated sludges and some organic liquids are some of the notable exceptions. Since the waste inventory was limited to that specified as contact-handled (i.e., <200 mR/h at the surface of the container), radioactive constituents will be present only in the parts per million level. Thus, with few exceptions, the treatment processes must be selected to treat the physical matrices and the process must ensure that the RCRA contaminants are either destroyed or immobilized along with the radionuclides, to a satisfactory level, in the final waste form.

RESIDUE IMMOBILIZATION

As indicated above, the residues from the treatment process are expected to require immobilization to meet EPA LDR requirements for hazardous materials because of their hazardous characteristics or because they are covered under the "derived from" rules. Immobilization is also assumed to be required to meet DOE disposal and performance assessment requirements. Consequently, this study included immobilization as an integral part of the integrated treatment system.

For purposes of this study, it has been assumed that the baseline process for immobilization of the radionuclides would be vitrification (vitrification will be used throughout this paper to describe any process that melts the residues regardless of whether the final product is a glass or a ceramic material). Vitrification was selected as the baseline process because it achieves the maximum volume reduction and provides the most stable long term waste form for radionuclides. Much of the MLLW has alpha emitting radionuclides that remain radioactive for very long times and, thus, should be stabilized for long periods. The study also includes various combinations of cement and polymer stabilization to identify the comparative costs of these processes. Some of the systems evaluated produce only a vitreous waste form from the thermally-treated solids. Systems that produce acid gases have a scrubber in the air pollution control system that produces an aqueous waste stream laden with soluble salts. This stream is processed and dry salt is produced, which is subsequently stabilized with a polymer for disposal because it is presumed that the salt will be contaminated with some radioactive elements that pass through the filter systems. If this were not the case, stabilization of the salt would probably be unnecessary.

TREATMENT SYSTEM DESCRIPTION

The systems being evaluated in this study are summarized in Table II along with some of their major features. Detailed process flowsheets have been developed for all systems (2,4). The APC subsystem for most of the systems consists of an equipment lineup that allows the fly ash to be separated from the scrubber salts [partial wet quench, baghouse, pre-filter, carbon bed filter, HEPA filter, packed bed scrubber, demister, NOx abatement (if required), fan, and stack]. Thus, the scrubber salts are separated from the fly ash and can be immobilized independently in a polymer matrix. The fly ash is returned to the vitrifier for vitrification along with the bottom ash and soil. (Soil is added to the vitrification systems to help produce a leach-resistant final product). Additional benefits to DOE are expected to be derived if the soil used as the additive is contaminated soil from DOE sources. The balance of the paper briefly describes the systems selected for the study and some of their major features.

System A-1, Rotary Kiln with Vitrification of the Ash Residues (Baseline System)

This system uses conventional rotary kiln technology for the treatment of both the combustible and non-combustible waste streams. For this study, it is assumed that the poorly characterized DOE waste has some RCRA contaminants and that it is less costly to treat everything than to attempt to sample the heterogeneous waste to determine if RCRA-listed or -characteristic hazardous materials are present. All wastes will be shredded prior to feeding to the kiln to enhance the uniformity of the feed and to reduce the chance for classical "puffs" caused by excesses of combustibles that cause an oxygen demand beyond that to which the incinerator control system can quickly respond.

In addition to the wastes, the soils for vitrification are also sent through the rotary kiln. It has been observed in tests at the Idaho National Engineering Laboratory that if soils are pretreated at high temperatures prior to vitrification, the subsequent process is easier to control and less material carryover occurs. The pretreatment of soil drives off moisture and decomposes any carbonates that may be present. A side benefit of the soil in rotary kiln operation is the reduced

requirement for excess cooling air in the kiln. The vitrification system receives both the bottom ash and the fly ash recovered from the baghouse operation. As noted earlier, the baseline APC system uses a scrubber for chlorine removal down stream of all filter equipment. Unlike some commercial systems that use similar equipment lineups, the scrubber water is sent to a waste water treatment system rather than being recirculated to the quench where the fly ash becomes commingled with the scrubber salts.

System A-2, Rotary Kiln with Vitrification of the Ash Residues and Enriched Oxygen for Combustion

Except for the use of oxygen instead of air for oxidation, this system is nearly identical to system A-1. The reduced use of air (air in-leakage is still assumed) substantially reduces the total gaseous mass flow through the system, thus allowing a smaller air pollution control system to be used. To maintain the proper operating temperature, cooling water is injected directly to the kiln. Much less quench water is necessary to drop the temperature before the baghouse. All other parts of the system are identical to System A-1.

System A-3, Rotary Kiln with Vitrification of the Ash Residues and Wet APC System

This system is identical to System A-1 except that a wet APC system replaces the wet/dry system described above. In this system, the gas leaving the secondary combustor is quenched to below saturation. The gas then is filtered by a hydrosonic scrubber, sent to a packed tower for neutralization, passed through a mist eliminator, reheated, HEPA-filtered, treated to reduce NOx (if required), and released to the stack.

System A-4, Rotary Kiln with Vitrification of the Ash Residues and Retention of CO2 Emissions

This system is similar to System A-2; a subsystem to capture CO2 has been added and 90% of the inert gases are recirculated through the kiln. A small gas bleed system is used to prevent inert gas accumulation. Air in-leakage is assumed, as for the other systems. CO2 retention is employed to delay the release of the off gas until such time that it can be sampled to verify that there are no unacceptable contaminants present. A fluidized bed of CaO is used to adsorb the CO2, forming CaCO3. After the CaCO3 is determined to be clean, it is sent to a calciner to drive off the CO2 and the CaO is reused. It is assumed that the CaO can be used 10 times, after which the discarded CaCO3 is polymer stabilized. The packed bed scrubber is eliminated from this system since any acid gases in the system will react with the CaO to form salts in the fluidized bed.

System A-5, Rotary Kiln with Polymer Stabilization of the Ash Residues

This system is similar to System A-1, except that instead of using vitrification for the primary stabilization system, a polymer such as polyethylene is used for stabilization of the bottom and fly ash and the scrubber salts. This equipment lineup greatly simplifies the system by eliminating the vitrification system and the associated air pollution system, but at the expense of producing a much larger disposal volume. The rotary kiln and its APC are identical to System A-1.

System A-6, Rotary Kiln with Vitrification of the Ash Residues and Maximum Recycling of Salt Residues

This system is identical to System A-1 except that the scrubber salts are processed to recover HCl and NaOH. This eliminates a waste stream that requires polymer stabilization.

System A-7, Rotary Kiln with Integral Vitrification of the Ash Residues

This system, similar to System A-1, was evaluated in Phase 2 to assess the overall effect of combining combustion and vitrification in one unit. The kiln operates at a higher temperature to accomplish the vitrification. Depending on the waste makeup, fluxing agents may be required to reduce the kiln ash melting temperature to a range where kiln operation and refractory lifetime are acceptable. This system eliminates the need for ash cooling, handling, and storage, and for the separate vitrification system included in the baseline rotary kiln system and all of its variations (Systems A-1 - A-6).

System B-1, Indirectly Heated Pyrolyzer With Vitrification of Ash

This system uses an externally-heated pyrolyzer to destroy the organics in a reducing atmosphere. Offgas from the pyrolyzer is fed to an electrically-heated secondary combustor. The ash is fed to a closely-coupled vitrification unit so that ash handling, cooling, and storage is not required. Oxygen is used instead of air to minimize the total gases in the system. Only combustibles are fed to the

pyrolyzer. The noncombustibles and soil are fed to a blender/dryer and then to the vitrifier. Oxygen is provided to the vitrifier via an oxygen lance to ensure combustion of any hard carbon coming from the pyrolyzer. The APC for this system is the same as used on the baseline system, but it is smaller since the total gas flow through the system is less than 10% of that in the baseline system. A single APC system is provided for the pyrolyzer and vitrifier rather than separate units as were provided for the rotary kiln systems. This system has the second lowest offgas of any system in the study.

System C-1, Plasma Hearth Furnace

This system uses a plasma hearth furnace to destroy the wastes and to vitrify the residues in a single unit. It requires less waste preparation and sizing because entire drums can be fed to the plasma unit (if safety and regulatory practices allow). Soil is fed to the plasma unit in the same quantities as for all other vitrification systems in this study. The APC subsystem is similar to that of the baseline system, but sized for the reduced gas flow. This APC system uses an electrically-heated secondary combustion chamber (SCC) to minimize total system gases. The use of electricity for the plasma and SCC heat source greatly reduces the volume of offgas compared to systems using methane, such as a rotary kiln.

System C-2, Plasma Hearth Furnace with Oxygen for Combustion and CO2 Retention

This system is similar to System C-1, but oxygen replaces air for combustion. The CO2 produced is retained in a holding system identical to that described for System A-4, except it is smaller because no methane is used in the plasma process.

System C-3, Plasma Hearth Furnace with Steam Gasification

This system uses a plasma concept similar to that of System C-1, but the wastes are reacted with high temperature steam rather than air. The endothermic steam gasification reaction produces CO and H2 from the combustibles. If a use for this synthesis gas were readily available within the treatment facility, the energy could be tapped. For this study, however, the gases are passed through a thermal oxidizer to combust with air. The total gaseous effluent is the same in either case. Because the thermal oxidizer is downstream from the APC, the gas mass flow through the APC system is very small, comparable to that of System C-2.

System D-1, Fixed Hearth Pyrolyzer with Oxygen and CO2 Retention

This system uses a fixed hearth incinerator fired with methane for the combustible fraction of the waste. Additional sorting is assumed to separate the combustible wastes. Oxygen is supplied to the incinerator and the secondary combustor. The CO2 in the offgas is treated as in Systems A-4 and C-2. The ash from the incinerator is blended with the dried noncombustibles and the additive soil and sent to a vitrification unit. Polymer is used to stabilize the spent salts from the CaO regeneration unit.

System E-1, Thermal Desorption and Incineration with Grout Stabilization

This system limits the amount of material that is incinerated. RCRA regulations allow debris to be grouted without other treatment. This system provides some additional treatment beyond that required by RCRA, but less than other systems in the study. Combustible solids and liquids are treated in a small rotary kiln, while noncombustibles are treated in a thermal desorption system. The solid effluent from the thermal desorption system is grouted along with the ash from the rotary kiln. The rotary kiln has an APC system identical to the baseline system. Vapor effluent from the thermal desorption system is condensed and destroyed in the SCC unit of the APC.

System F-1, Molten Salt Oxidation with Vitrification of Ash Residues

This system employs a molten salt oxidation (MSO) unit for the combustible fraction of the waste. Molten NaCO3 is used in the bed. Finely shredded organic wastes are injected into the bed along with combustion air. The bed provides the thermal environment for oxidation and neutralizes any acid gases in situ. Due to viscosity limitations, the bed ash content must remain below 20%, and thus a salt overflow system is used for viscosity control. The discharged salt must be recycled or the volume of waste for disposal would be enormous. Therefore, a salt recycle system is used to separate the ash, NaCl, and NaSO4 from the unused NaCO3, which is returned to the MSO reactor. Ash from the salt recycle system is combined with the noncombustibles and soil, dried and blended, and fed to a vitrifier for melting into a vitreous waste form. Waste salts from the salt recycle unit are stabilized with polymer for disposal.

System G-1, Molten Metal Melter

This system uses a molten metal melter for decomposition of organic wastes in a molten steel bath. The bath is operated in a reducing mode such that CO and H2 are the primary gaseous effluents. These gases are eventually reacted in a thermal oxidizer prior to release (similar to System C-3). This system is assumed to accommodate large quantities of metal directly, and thus separate metal melting subsystems are not required. (Methods of feeding this system with bulk metals are under development but not yet demonstrated.) All waste must be finely shredded for injection into the molten metal bath. It is assumed that a method will be developed to provide the fine shredding capability for pretreatment of the DOE waste stream (both combustible and noncombustible). If these key development requirements are met, this system may be as omnivorous as the plasma torch and slagging rotary kiln systems. The system uses an APC fairly similar to that used in the baseline system, but the aqueous quench is replaced by a fluidized bed sand cooler that rapidly drops the offgas temperature to below 200F. Dry gas is filtered in equipment similar to that used for the baseline system. A unique aspect of the molten metal reactor is that CaO is fed into the reactor along with the waste and reacts with the chlorine released from the waste to form CaCl2 in situ. This salt is expected to combine with the slag. Some HCl (10%) is expected to escape unreacted, so a scrubber is provided downstream of the filters to capture the remaining acid gases. Similar to the plasma systems, this reactor makes two separate forms of solid effluent waste in the same reaction vessel: metal and slag.

System H-1, Steam Gasification System with Vitrification of Ash

This system uses an indirectly-heated fluidized bed to gasify the finely shredded combustible waste fraction. Ash from the bed blowdown stream, bed material, and fly ash entrained in the gas flow are separated from the offgas stream by a series of coarse and fine filters and sent to the vitrification unit. Noncombustible wastes are blended with the waste from the gasifier, combined with the soil, and vitrified. Acid gases are separated from the gas stream in a wet scrubber and the salts are polymer stabilized. H2 and CO are sent to a thermal oxidizer, similar to Systems C-3 and G-1. Since heat for gasification is indirect, offgas passing through the system is small and the gas cleanup equipment is small compared to the rotary kiln systems.

System J-1, Joule-Heated Vitrification

This system uses a high-temperature, Joule-heated vitrification system to combust the organic wastes and melt the residues, along with the noncombustible wastes and soils, in a single unit operation similar to the slagging rotary kiln, the three plasma systems, and the molten metal reactor. The waste is shredded and fed to the melter along with the soil; a slag waste form is produced. For Joule-heated melters, it is important to minimize the amount of metal in the melt to prevent short-circuiting of the electrodes, so additional sorting is required for this system. Except for waste preparation steps, this system is similar to the plasma system (C-1). An APC subsystem similar to that of the baseline system is used, but sized for the smaller offgas load (electricity has replaced methane). Air is used for combustion in this system.

It is also appropriate to point out that this system could have been configured with an AC or DC graphite electrode melter instead of the Joule-heated unit. These types of melters are not sensitive to metal content in the feed and require less waste sorting and shredding. The use of one of these melter types would allow elimination of the separate metal melter subsystem that is required for the Joule-heated melter system. Their waste preparation requirements are similar to the pretreatment requirements defined for the rotary kiln in this paper, i.e. shredding.

System K-1, Thermal Desorption and Mediated Electrochemical Oxidation (MEO)

This system is designed to treat shredded waste using only low-to-moderate temperature processes. Thermal desorption using an indirectly-heated desorber is used to heat the combustible and noncombustible solids to vaporize and remove volatile organic compounds (VOCs). The VOCs are condensed and oxidized in a MEO cell. The desorbed solids and residues from the MEO reactor are stabilized in grout.

System L-1, Thermal Desorption and Super Critical Water Oxidation (SCWO)

This system is similar to System K-1. All solids are shredded and fed to a thermal desorption system to remove low-boiling-point VOCs, which are then condensed and oxidized in a SCWO unit operation. The solids from the thermal desorber and residues from the SCWO reactor are solidified in grout.

SUMMARY
The ITTS study has identified a variety of MLLW thermal treatment systems and explored the differences among several core thermal treatment technologies, combustion gases, air pollution control systems, CO2 management methods, and techniques for stabilization of residues. When complete, this study will establish the variations in gaseous effluents, solid residues, and preliminary life cycle cost estimates for these 19 systems. Cost sensitivity analysis will be performed to show the effect of uncertainty on system cost.

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EVALUATION OF ALTERNATIVES TO INCINERATION FOR DOE MIXED WASTE
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ABSTRACT
Through its Mixed Waste Integrated Program (MWIP), the U.S. Department of Energy (DOE) is identifying and evaluating technologies capable of treating DOE hazardous organic mixed wastes without the problems associated with permitting and siting incinerators. Various innovative thermal processes have been evaluated, including supercritical water oxidation, molten salt oxidation, molten metal bath, steam reforming, chem-char oxidation, packed-bed reactor, and electric arc. Some of these technologies will treat only aqueous or gaseous waste, but several show promise for treating the DOE's range of wastes. The MWIP has prepared a Technical Area Status Report for waste destruction technologies that tabulates and discusses available options. This paper discusses various thermal processes as alternatives to incineration, evaluating their advantages, disadvantages, and state of development.

INTRODUCTION
The U.S. Environmental Protection Agency's land disposal restrictions (as set forth in Title 40, Code of Federal Regulations, Part 268 [40 CFR 268]) require that mixed waste be treated prior to land disposal. Mixed waste streams containing hazardous organic compounds, which are typically treated by thermal systems, are a particular problem. The DOE has 247,000 cubic meters of mixed waste to be treated (1), the largest portion of which is aqueous. The aqueous waste is largely wastewaters and can be treated to meet regulatory requirements with available technology. The second largest portion of the DOE's mixed waste is inorganic sludges and particulates. These wastes can be treated by stabilization or thermal technologies. It is the remaining waste, which is a debris consisting of a wide variety of process solids and rubble, that represents the greatest challenge in terms of treatment.
There are conventional and innovative treatment options that can potentially be used to treat the DOE's mixed waste. The MWIP was formed to provide a vehicle for the DOE to identify and support new technologies that could improve cost-effectiveness, environmental safety, process performance, and waste form performance. The MWIP was charged with identifying technologies that could be applied to mixed waste across the entire DOE complex. In pursuit of this goal, the MWIP has supported numerous waste destruction, off-gas treatment, waste characterization, and final-form technology evaluations. The MWIP has focused primarily on two technologiesplasma hearth processing of a wide variety of waste streams and vitrification of mixed

waste sludges and soils.
The MWIP has prepared a Technical Area Status Report for waste destruction technologies that tabulates and discusses available options (2). This report evaluates technologies based on the following criteria: regulatory acceptance and stakeholder concerns, treatment efficiency and effectiveness, volume reduction, and improved environmental safety.
This paper is based on the Technical Area Status Report and discusses various thermal treatment alternatives to incineration. Technologies that will be described in this paper include:
- Supercritical water oxidation
- Molten salt oxidation
- Molten metal bath
- Steam reforming
- Chem-char oxidation
- "Flameless" secondary combustion system
- Electric arc.

Nonthermal alternatives are discussed in a companion paper presented in this conference by Schwinkendorf et al. (3).

TECHNOLOGIES

Incineration is a mature, well-developed hazardous waste treatment technology. Traditional incineration processes, including controlled air, rotary kiln, liquid injection, fluidized bed, and infrared, commonly heat waste to temperatures between 900C and 1,200C, volatilizing and oxidizing organic constituents. Some recent incinerator designs in remediation applications are using enriched or pure oxygen for the oxidation. Incinerator off-gases generally contain entrained particulates, NOx, products of combustion, and excess air. Other off-gas constituents may include SOx, volatile metals, HCl, and volatile metal chlorides. Trace quantities of light organics (residual organic stack emissions) are commonly present, and recombination products such as dioxins and furans can be produced during the off-gas cool-down period.
The thermal processes discussed in this paper use processes other than flame combustion in excess air. These include thermally oxidizing organics in an oxygen-starved or inert environment, or causing an organic material to react with something other than oxygen. In pyrolysis processes, the off-gas will contain combustible gases that are typically treated in an afterburner (4). Systems are also being developed to oxidize wastes in baths of molten salt or to break down organic molecules into elements in baths of molten metal.

Supercritical Water Oxidation

Supercritical water oxidation (SCWO) involves the oxidation of organic wastes in the presence of a high concentration of water at temperatures and pressures above the critical point of water, i.e., 374C and 22 Mpa (705F and 218 atm). Above the critical point, the properties of water change significantly, such that oxygen and most organic compounds become completely miscible with the supercritical fluid (eliminating the usual transport and mixing problems associated with reaction of two or more phases), and above 450C inorganic salts become almost completely insoluble. The organic waste feed may be in the form of an aqueous waste stream containing organics or a pumpable organic sludge. The oxidant may be air, oxygen, or hydrogen peroxide; in some cases, nitrates have been used successfully. Under these conditions, the waste is treated at high-destruction efficiencies of over 99.99%, and the resulting effluents, which consist primarily of H_2O and CO_2, are relatively benign (5).
Advantages:
- SCWO processes achieve high organic destruction efficiencies at lower temperatures without producing NOx (5).
- SCWO can process aqueous wastes with low concentrations of organics.
- Residence times are short, due to extremely good heat and mass transfer.
- The liquid phase medium facilitates recovery of combustion heat to heat the incoming waste stream and minimize auxiliary heating requirements.

Disadvantages:
- No metals should be present. The waste must be sorted.
- As currently developed, solid wastes require pulverizing prior to injection into the reactor.
- Corrosion-resistant alloy construction is required to contain the radioactivity at

the temperature and pressure requirements. Testing has not identified good long-term construction materials for the reaction chambers.

Safety provisions are expected to be extensive to protect against high-pressure releases.

Process thermodynamics and kinetics are not well understood.

Inorganics in the waste are not soluble in superheated water, resulting in plugging the oxidation chambers with inorganic salts.

Required Development:

Scale-up and demonstration of reliability and continuous, long-term operation on a variety of DOE wastes.

Monitoring, control, and determination of the fate of radionuclides in the SCWO process.

Definition of pretreatment requirements and demonstration of pretreatment systems.

Identification of suitable construction material.

Molten Salt Oxidation

Molten salt oxidation (MSO) has been identified as a potential alternative treatment process for low-level radioactive and organic mixed waste streams. The MSO process entails destroying the organic component of the waste at elevated temperature in the presence of the catalytic molten salt. The basic concept is a combined chemical and thermal waste treatment technology that does the following:

1. Feeds wastes and air into a ceramic-lined vessel containing a bed of molten salt at 900C to 1,000C
2. Oxidizes organic wastes using air sparged into the catalytic salt bath
3. Neutralizes acid gases in the bath
4. Discharges the salt for disposal or processing and recycling.

The molten salt acts as a dispersing medium for both the waste being processed and the air or oxygen used in the processing. It enhances the completeness of chemical reactions by providing improved physical contact and a stable heat-transfer medium that resists thermal surges. It also acts as a catalyst for the oxidation reactions to accelerate the destruction of organic material. The melt retains soot and char for more complete reaction and retains most of the radionuclides and other noncombustible materials associated with the waste. Sodium carbonate, or sodium carbonate mixed with other salts, is typical of the salts used in the melt. Tests performed by Rockwell demonstrated >99.9999% destruction of polychlorinated biphenyls, chlordane, trichlorobenzene, and other organics (6). The two-stage MSO unit at Lawrence Livermore National Laboratory has achieved organics destruction efficiencies of 99.99% for organic liquids when the temperature is maintained above 900C in the second stage (7).

Advantages:

MSO can accomplish destruction up to 99.9999% of various organics, acid gas absorption/neutralization, and some particulate and volatile metal removal from a waste stream in a single-unit operation.

This technology may be able to process difficult-to-treat wastes successfully, such as nuclear-grade graphite and tributyl phosphate, and may be suitable for other difficult waste streams containing low-melting metals, such as sodium.

Gas/liquid contact in the melt is intimate, providing high mass and heat-transfer rates and resulting in high destruction efficiency.

Disadvantages:

Fine shredding is required to achieve good solid/liquid contact.

Selection of construction materials for long-term operation is a challenge, as molten halide carbonate salts in an oxidizing atmosphere constitute an extremely corrosive chemical environment for the refractory lining.

No metals are to be present. The waste must be sorted.

Real-time monitoring of molten salt composition and viscosity is necessary to ensure high combustion efficiency. These controls have not been demonstrated.

The salt itself generates particulates in the off-gas, in addition to particulates generated from the waste. This carry-over could cause buildup in the system or pluggage of off-gas ducting.

There is a potential for adverse reactions of certain wastes with the molten salt.

Required Development:

Scale-up and demonstration of reliability and continuous, long-term operation on a variety of DOE wastes.

Monitoring, control, and determination of the fate of radionuclides in the MSO

process.
A simple, cost-effective method for the disposal or recovery of salt.
Molten Metal Bath
Metal-melting and refining technology is a decontamination method in which contaminated steel, copper, or other contaminated waste can be melted to form a molten metal phase and a molten oxide slag. This technology may be used to treat metal-bearing waste to destroy the organic contaminants, achieve waste volume reduction, recycle the contained metal, and produce a compact, leach-resistant waste form. The waste is introduced into a molten metal bath within a refractory-lined vessel, and the process parameters are controlled so that the constituents separate into a metallic layer and a slag layer (oxidized material). Uranium, transuranic elements, and many radionuclides of concern are preferentially oxidized and concentrated in the oxide, or slag, phase. Cooled slag is chemically stable and resistant to leaching. The organics are decomposed by the intense heat of the metal melting operation, producing either gasification products or oxidation products, depending on the mode of operation of the bath.
Advantages:
Destruction and removal efficiencies (DRE) of >99.9999% for injected organic compounds are regularly achieved (8).
Efficient in-bath heat transfer and the benefits of a homogenous catalytic solution can result in lower operating costs.
Metal recovery for recycling is possible.
Disadvantages:
The waste needs to be shredded.
The dissolution of the refractory material by the slag in a hazardous and radioactive environment can result in frequent maintenance in a difficult environment.
The off-gas system solids will contain fine particulates, metals, and radionuclides that are volatilized or otherwise carried from the melt at temperatures around 1,600C.
There may be a potential need for additives to control metal and slag properties, which will decrease the potential for volume reduction.
Keeping the metal in a molten state at all times is necessary to avoid substantial delays incurred in a shutdown.
Required Development:
Scale-up and demonstration of reliability and continuous, long-term operation on a variety of DOE wastes.
Monitoring, control, and determination of the fate of radionuclides in the molten metal bath process.
Effects of organic content of the waste on radionuclide partitioning and generation of particulates.
Steam Reforming
With steam reforming, hazardous organic chemicals can be destroyed by reacting with steam to produce a synthesis gas consisting of CO, CO2, H2, and H2O, in addition to light organics. Several variations of the basic steam-reforming system have been commercialized and are under development or demonstration. A system that has been used to demonstrate treatment of DOE wastes is the Synthetica Technologies system (4). The Synthetica Detoxifier consists of a two-stage destruction process. In the first stage, organics are volatilized by heat and high-temperature steam in a waste-feed evaporator. The hydrocarbons are vaporized within the evaporator upon exposure to superheated steam at 300C to 600C, which begins the steam reforming chemistry to thermally decompose and fragment the organic molecules. These gases generated in the evaporator pass to a noncatalytic, high-temperature steam reforming reactor (or detoxifier) operated at 1,200C, where they are mixed with superheated steam, and the reaction proceeds to completion. For many organic solvents commonly found in mixed wastes, greater than 99.99% destruction efficiencies have been achieved at 1,200C. An alternate process by MTCI uses a fluidized bed of sodium carbonate for reforming and acid gas control.
Advantages:
Greater than 99.99% destruction efficiencies have been achieved for various organic components of the mixed wastes.
The system is suitable for a wide variety of DOE waste streams at various concentrations.

SOx and NOx formation is minimized.
The system, sized for drum quantities of waste, is easily transportable.
A useful energy source (combustible gas) is generated.
Disadvantages:
One of the major limitations of the Synthetica system as it is currently configured is the limits on chlorine content of the waste (9).
The combustible gas stream must be burned for energy recovery, or energy input is required.
Required Development:
Scale-up and demonstration of reliability and continuous, long-term operation on a variety of DOE wastes.
Monitoring, control, and determination of the fate of radionuclides in the steam reforming process.
Identification of a method of treating chlorinated wastes.
Chem-Char Oxidation
Chem-char oxidation takes place in a reaction chamber that consists of a cylinder packed with a triple-reverse burn (TRB) char, which is a low-grade activated carbon product. The process can treat wastes in the form of solids, liquids, sludges, and soils. Liquid wastes are adsorbed on the char, contaminated soils and sludges are mixed with the char, and solid organic wastes are pulverized and mixed with the char. Oxygen is passed through the reactor from the top and flows countercurrent to a flame-front traveling upward. The primary destruction process is reduction of the waste to a combustible synthesis gas consisting of CO2, CO, H2O, H2, CH4, and trace volatile organics, and a dry, inert, carbonaceous solid. Radioactive and heavy metal constituents are retained in the char residue or ash. Organic DREs >99.9999% have routinely been achieved with second-stage combustion, with no detectable dioxins and furans. Tests have shown that the char can retain over 98% lead and over 99% of copper and zinc (10). Additional tests have demonstrated over 98% retention of radionuclides on the char (11).
Advantages:
Organic DREs of greater than 99.9999% have been achieved.
The process can treat a wide variety of wastes, including high ash, high water content, and viscous sludges.
The process requires no energy inputs.
A high percentage of heavy metals and radionuclides are retained in the regenerated char and slag.
Disadvantages:
Wastes must be shredded.
The technology involves a high processing temperature (1,200C), where some ash constituents may melt.
A secondary combustion chamber is required, typically a flame-type device, and potential permitting issues are involved.
The need to purchase or generate TRB char from coal.
Required Development:
Scale-up and demonstration of reliability and continuous, long-term operation on a variety of DOE wastes.
Monitoring, control, and determination of the fate of radionuclides in the chem-char process.
"Flameless" Combustion Systems
The Thermatrix technology is a flameless thermal destruction process for vapor-phase organics, which takes place in a packed-bed reactor (12). In the packed-bed, high-temperature thermal oxidation occurs within a reaction zone in a bed of chemically inert ceramic material, typically operated at 870C to 1,000C. The packed-bed consists of an insulated cylinder, lined with an alumina and silica refractory and containing a ceramic matrix composed of oxides of aluminum and silicon, with low levels of oxides of titanium, calcium, manganese, and magnesium (13). The matrix is immune to moisture and acid, is noncatalytic, and has a temperature rating up to 1,400C. Various test results by Thermatrix have indicated destruction efficiencies greater than 99.99% over a range of volatile organic compounds. The ceramic matrix does not deteriorate during operation.
Advantages:
DREs are >99.99% over a range of volatile organic compounds.
Low NOx formation, typically below 2 ppm.

Low CO formation, typically not detectable.

Low-pressure drop across the system, typically 3 inches water gauge.

Ability for internal heat recovery.

Ability to treat low flows with low concentrations, as well as high flows with high concentrations.

Disadvantages:

The system can process only gaseous feeds with no particulate.

Low organic content gases require supplemental fuel.

The inability of the system to handle certain compounds that may precipitate on the packing and potentially cause sticking and agglomeration of the packing material.

Required Development:

Demonstration on DOE wastes or as second-stage destruction device.

Electric Arc Heating

Arc melters maintain an arc between carbon (graphite) electrode(s) and the processed material. In the vicinity of the arc, a plasma is produced. The electric arc provides the primary energy for heating and melting the target material. This occurs in a sealed chamber, thus reducing the amount of gas produced during pyrolysis and allowing the gas to be removed from the system in a nonoxidizing atmosphere. During operation as a melter, an unopened drum may be fed into a molten bath. The drum melts, and the waste material is subjected to intense radiation from the arc and heat of the molten bath. As a result, organic material vaporizes and decomposes, and inorganic material is melted, producing a slag. The decomposition products are further decomposed by the electric arc. The products are sent to an off-gas system, where volatile or particulate organics are captured and the remaining organics are destroyed.

Advantages:

The system can accept a variety of inputs and waste matrices, including combustible and noncombustible wastes.

The wastes can be treated in their original container in an "as-received" condition with minimum characterization or pretreatment.

The use of electrical energy significantly reduces the volume of off-gas and particulates, along with associated pollution control requirements.

Disadvantages:

Off-gases containing trace organics and volatile metal compounds will require secondary oxidation.

High temperatures may produce high NOx levels.

High temperatures tend to volatilize heavy metals, which require capture in the off-gas system.

Required Development:

Scale-up and demonstration of reliability and continuous long-term operation on a variety of DOE wastes.

Routine electrode replacement.

CONCLUSIONS

The discussed thermal technologies individually have advantages over incineration in specific applications, but they also have several disadvantages and uncertainties. Some are not capable of handling all kinds of waste streams, i.e., solids, liquids, and vapors. A few of the technologies are not capable of handling the large volumes that incineration can handle; other processes may be suited for treating the low volumes of wastes existing at some facilities. Most of the technologies have not been tested in the field, and hence their performance standards and efficiencies are uncertain.

Supercritical water oxidation, in its current state of development, is capable of treating hydrocarbon contaminated aqueous wastes. The flameless secondary combustion system could process gases from a primary thermal treatment system or waste hydrocarbon gases from other processes. Molten salt oxidation, molten metal bath, steam reforming, electric arc treatment, and chem-char are being developed for solid (and liquid) waste processing. Overall process throughput requirements, waste composition, and project timing are factors that could justify application of one of these technologies to a waste destruction. In general, however, none of these processes are developed to the point that they can be accepted as a universal replacement for current incinerators.

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LOW-TEMPERATURE ALTERNATIVES TO INCINERATION FOR DOE MIXED WASTE

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ABSTRACT

Recently the Department of Energy's Mixed Waste Integrated Program initiated an evaluation of technologies capable of treating DOE mixed wastes without the problems associated with permitting and siting incinerators. Various categories of non-thermal technologies were evaluated including direct chemical oxidation, radiation induced oxidation, biological treatment, dehalogenation, and gas phase destruction. Although most of these technologies cannot directly replace incineration, many can destroy organics prior to separation and/or concentration and subsequent incineration. Three technologies appear to be relatively versatile, able to treat a variety of waste streams and contaminants including some solids, and are near field demonstration. These include the DETOXSM, the electrochemical oxidation, and the acid digestion processes. However, integrated systems must be demonstrated for secondary waste recovery and/or treatment before field implementation.

INTRODUCTION

The United States Department of Energy (DOE) is identifying, evaluating, and developing appropriate waste destruction and stabilization technologies for a variety of mixed waste streams. The DOE has approximately 250,000 m3 of mixed low-level waste (MLLW) to be treated. Aqueous liquids (containing <1% organics) comprise the bulk of these wastes (approximately 39%), inorganic sludges and particulates comprise approximately 33% of these wastes, and the remainder is

organic liquids, soils, debris, and other wastes. Although the concentrated organic material requiring destruction is not large, hazardous organic compounds and radionuclides are spread throughout these matrices in various concentrations presenting a challenge in terms of treatment.

Mixed waste containing hazardous organic compounds, which is typically treated by incineration, is a particular problem. Although incineration is the EPA BDAT for many organically contaminated wastes and is an omnivorous treatment process (i.e., it can treat a variety of wastes and can tolerate changes in the waste stream in terms of content, concentration, and feed rate), DOE is concerned about the current difficulty of permitting and siting incinerators. Public acceptance and perceptions associated with air emissions of toxic metals and organics are major issues (1).

Recently the Department of Energy's Mixed Waste Integrated Program (MWIP) initiated an evaluation of alternatives to incineration to identify technologies capable of treating DOE mixed wastes and which may be more easily permitted. It was expected that several non-thermal technologies would alleviate stakeholder concerns by decreasing off-gas volumes and eliminating the emission of particulates, volatilized metals and radionuclides, PICs, NOx, SOx, and recombination products (dioxins and furans).

Technologies are being evaluated based on the following criteria: regulatory acceptance and stakeholder concerns; treatment efficiency and effectiveness; volume reduction; and improved environmental health and safety. Ideally the technology would be versatile, relatively omnivorous, capable of treating a variety of wastes with varying constituents with minimal pretreatment or characterization, easy to implement, and would produce secondary waste stream volumes significantly smaller than the original waste stream.

The low-temperature alternatives to incineration were grouped into the categories shown in Table I. This paper will discuss these technologies with respect to the above stated criteria.

TABLE I

TECHNOLOGIES

Although the effectiveness of most non-thermal technologies increases at elevated temperatures, for the purposes of these evaluations a low-temperature technology is defined as one that is capable of destroying or decomposing organic molecules at temperatures 350C.

These processes may provide a direct or indirect substitution for incineration. That is, certain incinerable wastes may be directly treated to destroy the organic contaminant in lieu of incineration. For example, most direct chemical oxidation systems can treat some organic solids and organic liquids that are legacy wastes, or condensed from a thermal desorber carrier gas or from an air stripping or soil vapor extraction operation. Application of gas phase destruction techniques can eliminate the condensation process and treat the organic vapors directly. Waste streams that cannot be incinerated, such as dilute aqueous wastes, may be treated to destroy the organic contaminants thereby eliminating the need for separation processes (e.g., oil/water separators, activated carbon beds, etc.) and indirectly eliminate the need for subsequent incineration of the concentrated organic waste stream.

Most of these technologies have minimal off-gas (typically CO2, water vapor, CO, HCl, and in some casesVOCs and NOx), metals are not vaporized and dioxins are not produced, and it may be easier to gain public acceptance and permitting. However, because these technologies will not accept bulk wastes, except possibly acid digestion, care is required in the pre-treatment (size reduction) and metering of waste into the process, and most of the technologies are limited in waste type and matrix for which they are effective.

Except for wet air oxidation, the direct chemical oxidation processes are the most versatile of the low temperature technologies, and are effective in treating organic liquids, sludges, and some solids if adequately reduced in size. Other technologies are limited to specific waste streams such as halogenated organic liquids, organic vapors, or dilute organics in an aqueous matrix. Many of these technologies involve relatively weak oxidation processes so that long residence times may be required for adequate destruction, or rate limiting reaction byproducts may be produced requiring post-processing. In almost all cases destruction and removal efficiencies (DREs) and reaction byproducts depend on the contaminants in the waste stream and residence time in the reactor. These technologies are discussed in somewhat more detail in the following sections. Greater detail and additional references are provided in Ref. 2.

Direct Chemical Oxidation Processes

Direct chemical oxidation processes use a chemical oxidizer (oxygen, nitric acid, or oxidizing cation) to decompose and oxidize organic contaminants that are either in solution or suspension in the reacting medium. These reactions are most effective on soluble organic compounds, although some solids and immiscible liquids may also be treated. However, because the reaction rates depend on surface area, solids require size reduction and liquids require mixing for the reaction to take place in a reasonable period of time.Wet air oxidation (WAO) is a commercialized process that involves the aqueous-phase oxidation of dissolved or suspended organic substances with oxygen. Typical reactor temperatures range from 150C to 325C, and pressures of 2,069 kPa to 20,690 kPa are maintained to control evaporation and maintain the liquid state. However, high pressures in a radioactive environment introduce complexity and additional hazards to the operation.

WAO is not a highly predictable process, and the design is usually based on bench-scale tests for specific compounds so that a thorough knowledge of the waste to be treated is necessary. The degree of oxidation of organics and inorganics is limited by the solubility of oxygen in water. Up to 95% to 99% destruction in 15 to 60 minutes can be achieved; however, only 60% to 70% destruction is achievable with more resistant compounds (3). Although WAO has a high DRE for many organic compounds, they are rarely completely mineralized and some hazardous compounds can be produced as products of oxidation. This oxidized effluent, which contains residual organics, suspended solids, and soluble metal salts, generally requires further treatment to meet EPA treatment standards. Oxidized sulfur and nitrogen species are retained in the liquid phase; thus, SOx and NOx do not pose air pollution problems.

Catalyzed chemical oxidation refers to the class of reactions that use oxidizing agents with a catalyst to oxidize hazardous organic material to CO2, H2O, mineral acids, and less toxic organic materials. An example is the use of Fe(III) as an oxidizing agent in an HCl solution in the DETOXSM process being developed by Delphi Research, Inc. Such oxidizing agents are stronger oxidants than O2 and overcome wet-air oxidation process limitations caused by the limited solubility of oxygen in water. Cocatalysts increase the oxidation rate for organics, and Fe(II), formed in the organic oxidation process, is oxidized back to Fe(III) by a second catalyzed reaction with oxygen (4). The process typically takes place at 100C to 300C, depending on the waste form and composition, and at pressures of 240 kPa to 1480 kPa. The low temperature process produces no NOx, SOx, dioxins, or furans; however, elevated pressures in a radioactive environment introduce complexity and operational hazards.

This process is capable of treating dilute organic solutions, concentrated organic liquids, and certain organic solids up to 1 cm in size, and is highly tolerant of waste composition, form, water content, and particle size. However, care must be taken to meter in concentrated organics to prevent excessive temperature buildup and a runaway reaction, and to meter in aqueous wastes to prevent dilution of the acid bath and allow excess water to evaporate. In addition to oxidizing waste organics, most heavy metals and radionuclides are solvated and concentrated in the acidic catalyst solution. Thus, organic and inorganic contaminants may be removed from various matrices such as soils and debris.

The process is currently at the bench-scale level of development, although a pilot-scale demonstration is scheduled for FY95. Tests have been performed on various organic compounds, heavy metals, and radionuclide surrogates. Organic DREs >99.5% have been achieved in several hours, although faster destruction rates are expected in a well-mixed reactor with higher organic loadings. Small quantities (mg/liter to g/liter) of residual volatile and semivolatile organic compounds, including chlorinated hydrocarbons, have been found in the condensate and process solution (5). The secondary wastes produced by the process are primarily residual solids, heavy metal salts precipitated from the reaction solution, condensate, and exhausted reaction solution. The condensate containing HCl is typically returned to the DETOXSM solution or used to rinse filtered residual solids before being returned to the reaction chamber. The offgases, principally CO2 and O2, are recirculated to aid in agitation of the solution and to oxidize remaining volatile organics.

Acid digestion uses nitric acid as the oxidizing agent. Two acid digestion processes were investigated: the sulfuric acid process developed by Westinghouse Hanford Company (WHC) in the 1970s and 1980s, and the nitric-phosphoric acid oxidation

process being developed by Westinghouse Savannah River Company (WSRC). Both systems operate at temperatures of 200C to 250C and at atmospheric pressure.
Acid digestion systems are able to treat a wide variety of waste streams containing hazardous organics. The process works especially well for cellulosic materials and is suitable for polymers and non-volatile chlorinated hydrocarbons such as PCBs. Low volatility liquid organics may be treated by injection beneath the liquid surface to prevent flash-off and to increase residence time. Waste streams with high ash or metal content may produce large secondary wastes, and the offgas from both systems consists of HCl from chlorine-containing waste, and NOx released from solution as the nitric acid is depleted. The off-gas from the WHC system also contains SOx. Sulfuric acid and phosphoric acid act as dehydrating agents, but phosphoric acid raises the temperature at which nitric acid remains liquid from 150C to 190C. Preliminary results indicate that a 30C increase in temperature increases reaction rates by an order of magnitude. Tests at WHC on ion exchange resins, decontamination rags, HEPA filter frames, wooden structural materials, high plutonium bearing sludge, and plastic bottles were successful and demonstrated the removal of Pu from the treatment residue, and overall organic destruction efficiencies of 99.99%. However, the system was shut down in the mid-1980s due to lack of funding. The SRS system has been tested on combustibles such as plastics, neoprene, cellulose, and tributylphosphate with good results; however, destruction efficiencies are not yet available.
Electrochemical oxidation processes are similar to the two preceding processes in that an oxidizing cation (usually Ag(II)) in a nitric acid solution is used to produce highly reactive OH and NO3 radicals. The cation and free radicals attack organic compounds converting most of them to CO2, water, and inorganic ions. The resulting Ag(I) is recycled to Ag(II) at the anode of an electrochemical cell to maintain a supply of oxidant and minimize consumption of Ag. The Ag(II) migrates back into the bulk electrolyte to continue the oxidation process. A microporous membrane is usually placed between the electrodes to prevent the oxidizer produced at the anode from being reduced at the cathode. The oxidizers and acid also dissolve inorganic compounds that may contain radionuclides, including PuO2 which is not soluble in nitric acid alone.
Experiments at Lawrence Livermore National Laboratory (LLNL) achieved nearly 100% destruction of Trimsol and reagent-grade cellulose. Tests at Pacific Northwest Laboratory (PNL) have achieved destruction efficiencies for oil of >99% of the organic carbon at temperatures in the range of 60C to 90C. An ultrasonic generator was used to produce a micro-emulsion of the immiscible organics in water to increase the surface area, and the oxidation rate, of the organic material (6). PNL has also tested cerium as an oxidizing agent in HNO3 and, at higher temperatures (~100C), found it to provide the same or better oxidation rates for Trimsol as did silver without precipitation as a halide (7). Work in the United Kingdom (8), using the same Ag/HNO3 based electrolyte, has shown that a variety of organic materials can be destroyed including aliphatic and aromatic hydrocarbons, phenols, organophosphorous compounds, organosulfur compounds, and chlorinated aliphatic and aromatic compounds, including PCBs. Several ion exchange resins were also destroyed with the oxidation of the organic component nearly complete; however, comparative trials with flaked polyethylene showed almost no reaction.
Problems with the Ag(II)/HNO3 process include precipitation of Ag as a halide salt when treating halogenated organics; leakage of cations and anions through the electrode separator; degradation of polymeric membranes; excessive corrosion of the anode in the presence of acids; and generation of HNO2 and NOx at the cathode. The process also typically operates with low concentrations (<5 wt %) of oxidizable materials (9).
All direct chemical oxidation processes operate with highly reactive media making corrosion and compatibility with materials of construction major engineering and safety issues. Secondary wastes include organic reaction by-products or products of incomplete reaction, inorganic sludges, and depleted acids. Thus, these systems tend to be complex due to the need for an integrated system to recover and/or treat acids, and recover dissolved metals either for reuse or for disposal. An off-gas system will be required to insure no acid fumes are vented with the CO2 and CO produced in the destruction process, and to regenerate nitric or sulfuric acid from the off-gases.
Radiation Enhanced Oxidation

It is generally inappropriate to treat large-volume, dilute streams with incineration or direct oxidation processes designed for higher concentration organics. Radiation induced oxidation technologies destroy dilute organics in water eliminating the need for separation processes and subsequent treatment of concentrated wastes. Radiation is used to generate OH radicals in an aqueous medium which in turn attack the oxidizable materials. Various radiation sources have been used (such as ultraviolet radiation, ultrasound, electron beams, and x-rays), along with oxidation agents such as H2O2 and catalysts (such as metal ions), as a means to generate hydroxyl radicals.

These systems generally operate with low organic concentrations (typically <5%), and are applicable to existing waste and groundwater, and effluent from soil washing and decontamination operations. Destruction efficiencies are affected by the presence of OH scavengers in the water (such as carbonates and bicarbonates) so that characterization and pre-processing to remove these scavengers may be required. Room temperature and ambient pressure operation, and the lack of corrosive media, minimize safety hazards associated with these systems.

UV light induced oxidation of organic contaminants in aqueous waste streams has been commercialized for several years and is under investigation at several DOE facilities to determine its applicability to specific wastes. The principle of UV/oxidation technologies is the generation of hydroxyl radicals through UV photolysis of various oxidants such as H2O2, O3, or Fenton's reagent. Because the reaction rate is directly proportional to light intensity, the waste stream must be low in turbidity and solids, and the glass window of the reactor must remain clear to allow penetration by the UV light.

Although some organic species are easily destroyed by UV oxidation processes, including water soluble cellulosic material, they are rarely converted completely to carbon dioxide and water, and hazardous compounds can result. In addition, the applicability of these technologies to complex mixtures is unknown. Systems developed for specific waste streams that have unvarying species and concentrations lack operational flexibility to accommodate the changing stream characteristics in many waste treatment facilities. Thus, waste streams must be characterized and the process tailored for the contaminants to achieve optimum destruction and minimize generation of toxic end-products.

Ultrasound affects organic oxidation primarily through cavitation, which is the growth and catastrophic collapse of a gas- or vapor-filled bubble in a liquid. The high localized transient temperatures (thousands of degrees Kelvin) and pressures (hundreds of atmospheres) in the center of the bubble cause direct pyrolysis of organic vapors within the collapsing bubbles, and decomposition of water molecules into free radicals that propagate outward into the surrounding fluid to react with organic material. As with the common ultrasonic cleaning technology, ultrasound can be used in conjunction with soil washing to remove organic contaminants from soil and destroy the organics that migrate into the aqueous matrix.

The technical feasibility of this process has been demonstrated in small batch units at Argonne National Laboratory. This process can be used to treat a wide variety of organics, due to the non-specific nature of the process, and should be able to handle aqueous streams with moderate amounts of suspended solids and low light transmissivity. However, this is a new and emerging technology and will require extensive development and testing before it can be applied with confidence in the field. Work has not been done to determine the nature of the end-products of this process, and application to the destruction of a variety of solid organic materials needs to be determined.

Electron-beam processing involves exposing material to high-speed electrons produced by electron accelerators. The process of irradiation in aqueous solutions produces sizable quantities of the free radicals eaq-, H., OH., and the more stable oxidant H2O2 that react with organic contaminants to break the chemical bonds and produce CO2, H2O, and salts. Various studies have been performed, and e-beam systems tested within and outside the DOE complex on dilute (in the hundreds of ppm range) organic/aqueous waste streams. These include high-voltage (1.5 Mev to 2.0 Mev) systems tested at Los Alamos National Laboratory (LANL) and at Florida International University (FIU).

Typical DREs >99% for most common solvents have been achieved. Although some removal efficiencies are less than desirable (e.g., 60% for 1,2-dichloroethane and 77% for methylene chloride), these are based on a single pass through the system. Since this

is a flow-through system, there is essentially no retention timeonce the electrons enter the water, treatment occurs in a fraction of a second. Therefore, the waste stream can be recycled two or three times to achieve the required removal efficiency or to destroy stable intermediate reaction products. This technology can handle variations in the waste stream and is not sensitive to the treatability of the contaminants, the particular target species, or the water quality. However, the system is currently in the development stage and is applicable only to dilute organics in aqueous waste streams.

X-ray and gamma ray treatment of organically contaminated soil and water products is based on the in-depth deposition of ionizing radiation. Collisions of energetic photons with moisture in most waste streams generates a shower of energetic secondary electrons within the contaminated waste material. These secondary electrons produce both highly oxidizing radicals and highly reducing aqueous electrons similar to the electron beam process. While hazardous by-products may form during treatment, the complete conversion of contaminants and by-products is claimed to be achieved at sufficiently high dose levels and residence times without undesirable waste residuals or air pollution (10). Because of the range of x-rays, large volumes can be treated and standard container walls will not absorb a significant fraction of the ionizing radiation so that waste contained in vented disposal barrels can be treated. As with the electron beam process, the major hazard associated with this technology is high voltage and radiation, which can be mitigated by adequate shielding.

Biological Treatment

Biological treatment (or biodegradation) refers to the degradation and detoxification of wastes using microorganisms like bacteria and fungi. Biodegradation is directly related to environmental conditions and, when implemented successfully, is simpler, less expensive, and less disruptive compared to conventional physical/chemical or thermal treatment methods, and not as much of a regulatory hurdle. Although destruction rates are slower compared to thermal treatment methods and maintenance of the microbial culture is required, biological treatment is perceived as more acceptable by the public and other stakeholders. Biotreatment processes operate at near-ambient temperature and pressure and neutral pH, reducing the risk and severity of accidental releases due to process failure. In general, the contaminant is most available to the microorganisms if it is in an aqueous environment. Aqueous or organic liquid waste streams and soils have the most promise for biological treatment, although biodegradation of neat organics (pure non-aqueous phase) is generally not practical. Microorganisms can also initiate oxidation-reduction reactions that can cause metals to precipitate from solution and facilitate their separation. The end result of biological treatment of organic waste is the conversion of the contaminant into biomass (more microorganisms) and by-products (water and carbon dioxide in the case of complete aerobic mineralization). Typically 40% to 50% of aerobically degraded waste is converted to biomass, which becomes a secondary waste sludge that must be disposed or treated. Biomass sludge in anaerobic environments may be only 5% to 10% of the mass of the organics degraded (the remainder of mass going to carbon dioxide, methane, and water). Metals, radionuclides, and other refractory compounds that are not degraded will either remain in aqueous solution/suspension and exit the process with the treated effluent, or may sorb to the biomass and exit with the biosludge.

The DRE from biological treatment will require evaluation on a case-by-case basis, but when implemented successfully can be very high and comparable to other physical/chemical treatment technologies. However, even under optimal conditions, microorganisms may not be able to reduce contaminant concentrations to the required health-based limits due to inadequate sustenance of the microbial populations at very low concentrations. In these cases, post-treatment may be needed to polish the final treatment products to meet governing regulatory limits. The challenges to more widespread use of biological systems for the treatment of hazardous and mixed wastes include the lack of successful full-scale examples, and the ability to translate results from controlled laboratory conditions to field conditions. Large-scale demonstrations of anaerobic systems have been rare, in part due to the sensitivity of the anaerobic cultures to upsets or variations in the process, and the lengthy recovery periods required once the balance of the system is upset.

Dehalogenation

Dehalogenation processes remove halogens from the molecular structure, rendering the

compound less- or non-hazardous. Reaction by-products, some of which may be toxic, may require post-treatment to meet discharge requirements. The technology is most effective for wastes contaminated with semivolatile halogenated organic compounds and pesticides. These technologies include chemical- and radiation-induced dehalogenation, and DREs >99% have been achieved. However, some contaminants may require extensive residence time in the reactor vessel to complete dehalogenation. The two most common types of chemical dehalogenation processes are KPEG (potassium hydroxide with polyethylene glycol), and APEG (alkaline polyethylene glycol). In these processes, wastes are mixed with the dehalogenation reagent and heated to replace halogens with polyethylene glycol, rendering the compound nonhazardous and generating a chlorinated liquid secondary waste stream. The Base-Catalyzed Decomposition Process (BCDP) destroys PCBs and produces aliphatic hydrocarbons, biphenyls, and sodium chloride. In the Birch process, soil contaminated with PCBs is slurried with liquid NH3 and Ca/NH3 at ambient temperatures to produce NH3 (which is flashed-off upon completion), calcium chloride, calcium hydroxide, and biphenyls. Photo-dehalogenation processes use UV light combined with a reducing environment to dehalogenate organics. The Light Activated Reduction of Chemicals (LARC) dehalogenation reactions are normally conducted in a basic water, alcohol, or hydrocarbon solvent. The Reductive Photo-Dechlorination (RPD) technology uses ultraviolet light in a reducing atmosphere and elevated temperature to remove chlorine atoms from vaporized organochlorine compounds. Typical reaction products from the LARC process include biphenyl and sodium chloride; by-products from the RPD process include HCl and hydrocarbons such as ethane, acetylene, ethylene, and methane. No oxygenated derivatives, such as dioxins or furans, have been observed.

Gas Phase Destruction

There are a variety of gas or vapor phase organic destruction technologies that require a pre-treatment process to vaporize the organic material such as thermal desorption, vapor phase extraction of VOCs from soils, etc. The low-temperature technologies in this category include electron beams, cold plasmas, and photolysis, and involve direct destruction of the molecular bonds or generation of radicals that attack the contaminants. Application of these technologies eliminates the need for condensation or carbon adsorption of the organic vapors and subsequent incineration. Electron-beam and cold plasma discharges, although different in concept, use similar chemical processes to achieve organic destruction. Electron-beam technologies use accelerators to produce medium energy electron beams (in the 200 kev range) to directly break the chemical bonds of the organic compounds or produce free radicals from oxygen in the carrier gas. Cold electric discharge plasmas, generated by an alternating electric field established between two electrodes, produce lower energy electrons (in the 10 ev range) that react with the organic contaminants and carrier gas much the same way as the higher energy electrons.

In the silent discharge plasma (SDP) device developed at LANL, small electrical discharges occur in the space between two opposing insulated planar electrodes connected to a source of alternating high voltage. A large number of micro-discharges of short lifetime but high instantaneous current are uniformly distributed over the discharge space. In the case where the annular space between a pair of cylindrical electrodes is filled with dielectric pellets, the high-strength electric fields developed in the interstitial spaces between the pellets form multiple corona sites which generate high-energy electrons. This configuration is termed a high-energy corona (HEC) process and is being developed at PNL. Destruction and removal efficiencies greater that 99% have been achieved. However, a variety of reaction by-products have been detected in the effluent from these processes including phosgene and small quantities of dioxins and furans from the SDP process. Tests with gas phase photolysis have demonstrated destruction and removal efficiencies of >99% in seconds to hours, depending on the contaminant. Low-temperature and high-temperature photolysis have produced a large number of toxic reaction by-products, depending on the temperature and parent compound.

CONCLUSIONS

Although non-thermal technologies do not have the large volume off-gas problems associated with incineration, they do have several general disadvantages: 1) in some instances, depending on the waste constituents and the technology, the organic material is not completely oxidized and intermediate toxic compounds may be produced that require post-process destruction; 2) many processes cannot treat organic solids, and those that do either require size reduction (sometimes to powder form)

or can treat only a limited class of solids such as ion exchange resins or soluble cellulose; and 3) the inorganic solids in the waste stream results in a sludge that requires additional treatment. For most technologies, thorough characterization of the waste stream is required to choose a particular treatment system, determine optimum processing conditions, and determine the required pre-treatment to remove OH. scavenging species.

Most of these technologies are limited in effectiveness to specific waste streams such as dilute aqueous wastes, concentrated chlorinated liquids, or vapor phase organics. However, these processes can achieve destruction of the target organic species without separation, concentration, and subsequent incineration. Three technologies appear to be relatively versatile, able to treat a variety of waste streams and contaminants including some solids, and are near field demonstration. These include the DETOXSM, the electrochemical oxidation, and the acid digestion processes. However, integrated systems are required for secondary waste and acid recovery and/or treatment. These technologies and their supporting systems require further development before field implementation can be accomplished.

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Session 35 -- High Level Waste: Vitrification and Feed Preparation
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35-1

VITRIFICATION FACILITY HIGH LEVEL WASTE ACCEPTANCE STRATEGIES

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ABSTRACT

The U.S. Department of Energy's Office of Environmental Management (EM) has developed Waste Acceptance Product Specifications (WAPS) for Vitrified High Level

Waste Forms. These WAPS define the technical requirements and documentation requirements that must be satisfied before the producer's vitrified high level waste form can be accepted into the Civilian Radioactive Waste Management System. The waste form producers are required to document their compliance with the WAPS in the Waste Form Compliance Plan (WCP), the Waste Form Qualification Report (WQR), the Production Records, and the Storage and Shipping Records.

This paper describes the compliance methods and strategies being used at the Defense Waste Processing Facility (DWPF) and at the West Valley Demonstration Project (WVDP) to demonstrate compliance with each of the WAPS technical specifications. Several different methods are employed including: laboratory experiments, vitrification pilot plant studies, waste qualification testing, engineering calculations, and radioactive startup. This paper also assesses the compliance methods and strategies and assigns each a technology maturity level. The technology maturity level is an indicator of the degree of complexity in complying with the WAPS specifications. Technology maturity is defined as either High, Moderate, or Low. High Technology maturity means that the methods used to comply with the WAPS is based on long-term testing, well-defined procurement documents, and/or field proven implementing procedures. Moderate technology maturity is assigned to a compliance methodology that is somewhat complex but not entirely a new approach. A low technology maturity is used for a methodology that is being implemented for the first time for this application or is significantly complex.

The conclusion describes the effort being made by the facility operators to get the low and moderate approaches to work successfully. The programs and procedures necessary to implement these strategies/methodologies are approved and in place. Personnel staffing is adequate and the systems needed for waste qualification have been verified as operational.

BACKGROUND

In the late 1970's, DOE was investigating different approaches for immobilizing HLW resulting from its nuclear materials processing operations. A number of options were investigated, and in early 1980 the Savannah River Site (SRS) requested funds for a project to design, construct, and operate a facility to immobilize HLW via a vitrification process (formation of glass in a high-temperature melter). An Environmental Impact Statement (EIS) (1) and Record of Decision for the design, construction and operation of a vitrification facility was issued in 1982. Later that same year, the Department prepared an Environmental Assessment (2) which identified borosilicate glass as the preferred waste form for immobilizing SRS HLW. Shortly thereafter, vitrification was identified as the technology for immobilizing HLW into borosilicate glass at the West Valley Demonstration Project (WVDP). In 1990, the Environmental Protection Agency issued a determination that vitrification was the Best Demonstrated Available Technology for the treatment of high-level mixed waste, which exists at DOE facilities.

Two facilities are being completed by the Department of Energy (DOE) to immobilize HLW using the vitrification process for ultimate disposal in a federal repository. One of the facilities, the WVDP, located in western New York, is being constructed on the site of a former commercial nuclear fuel reprocessing facility. The other facility, the Defense Waste Processing Facility (DWPF), is being completed at the Savannah River Site (SRS) near Aiken, S.C.

INTRODUCTION

As part of the high-level waste acceptance process, the U.S. Department of Energy's Office of Environmental Management has developed Waste Acceptance Product Specifications (WAPS) for vitrified high level waste forms. These WAPS (3) define the technical requirements and documentation requirements that must be satisfied before the producers' vitrified high-level waste form can be accepted into the Civilian Radioactive Waste Management System.

The WAPS, which are based on the system-level requirements defined in the Waste Acceptance System Requirements Document or WASRD (4), are divided into five sections dealing with the borosilicate glass waste form, the canister, the canistered waste form, quality assurance issues, and documentation and other requirements. The DWPF and the WVDP are required to document their compliance with the WAPS in the WCP, the WQR, the Production Records, and the Storage and Shipping Records. The WCP is the waste form producer's plan for demonstrating compliance with waste acceptance product specifications which will assure that the waste forms will be accepted into the Civilian Radioactive Waste Management System (CRWMS) for eventual disposal in a

geologic repository. The WQR is a compilation of the results of the testing and analysis programs identified in the WCP.
This paper describes the methods and strategies by which DWPF and WVDP plan to demonstrate compliance with each technical specification in the WAPS. Several different methods are employed including: laboratory experiments, vitrification pilot plant studies, waste qualification testing, engineering calculations, and radioactive startup. (The quality assurance and documentation specifications, while very important to the waste acceptance process, are not covered in this paper because their compliance strategies are not technical per se).

WASTE FORM COMPLIANCE PLAN

The overall strategy for complying with the WAPS is to assure the quality of the product (i.e., canistered waste form) by a combination of component specifications and process controls. Many of the WAPS specifications require that the canister and waste form be well-characterized before DWPF and WVDP begin production of their actual waste forms. Some specifications are satisfied through a series of research and development activities. Others are satisfied through production scale operations using non-radioactive simulated waste. The remaining WAPS specifications address canistered waste forms produced during radioactive operations. This strategy is intended to demonstrate that the product will be acceptable over the range of anticipated chemical compositions and operating conditions.
At DWPF, a comprehensive test program has been developed covering all aspects of facility startup. Those portions of the DWPF startup test program which demonstrate integrated operation of the facility to produce an acceptable product are called Waste Qualification Runs. During the Waste Qualification Runs, the compliance strategies for many of the WAPS specifications will be demonstrated. At WVDP, the five year series of Functional and Checkout Testing of Systems (FACTS)* runs has provided considerable qualification data. FACTS included fully integrated melter runs. The melter runs had durations ranging up to 45 days. Simulated waste, based on the most recent analyses of waste tank samples with appropriate elements substituting for the radioactive elements, were used during these tests.

DESCRIPTIVE OVERVIEW OF THE METHODS AND STRATEGIES BEING EMPLOYED TO DEMONSTRATE COMPLIANCE WITH THE WAPS SPECIFICATIONS

The methods and strategies being used by the producers to demonstrate compliance with each of the WAPS Specifications are described in Table I. Each of the compliance strategies in Table I is assigned a technology maturity level. The technology maturity level is an indicator of the degree of complexity in complying with the WAPS specifications. Technology maturity is defined as either High, Moderate, or Low. High technology maturity means that the method(s) used to comply with the WAPS is based on long-term testing, well-defined procurement documents, and/or field-proven implementing procedures. Moderate technology maturity is assigned to a compliance methodology that is somewhat complex but not entirely a new approach. A low technology maturity is used for a methodology that is being implemented for the first time for this application or is significantly complex.

EVALUATING THE METHODOLOGIES AND STRATEGIES FOR THEIR TECHNOLOGY MATURITY

The methodologies/compliance strategies being employed are evaluated below for their technology maturity. Each of them is associated with a previtrification development activity or a vitrification process activity. This is depicted in Table I. Except where specifically noted, the compliance methodologies/ strategies discussed are used at both WVDP and DWPF.

Qualification Activities

Feed Batch Preparation: This activity is associated with WAPS specifications 1.1 and 1.3, which verify feed batch composition and project the Product Consistency Test (PCT)[5] results. A blending algorithm that utilizes chemical analyses information is embedded in the control scheme that verifies the composition of the melter batch. The batch (waste plus glass formers) acceptability strategy uses a statistically based process control model. The statistical process control model is complex and requires a cognizant engineer to interpret the model's output which correlates to the durability of the final product. Also, there is uncertainty associated with any model because it is developed from a finite amount of measured data. This approach is rated as low technology maturity.
Canister Fabrication. The canister fabrication activity covers six of the WAPS specifications:

Verify Canister and Filler Metal Composition (WAPS 2.1)

Test canister Integrity (WAPS 2.2)
Verify Unique Canister Label (WAPS 2.3)
Document Canister Dimensions and Welds on As-Build Drawings (WAPS 2.4)
Inspect Canisters for Foreign Materials and Package for Shipment to Facility (WAPS 3.3 and 3.4)

All of the methodologies/strategies for this activity are rated as high technology maturity. This is because they are straightforward compliance measures that have proven to be successful in the past.

Canister Receipt Inspection. This activity involves four WAPS specifications:
Remove Packaging and Inspect for Shipping Damage (WAPS 2.4)
Reinspect Canisters for Foreign Materials (WAPS 3.1, 3.3, 3.4)
Transfer to Shielded Facility and Install Temporary Cover (WAPS 3.3, 3.4)

Each of the methodologies are rated high technology maturity because they are proven, state-of-the-art approaches.

Canister Closure. This activity involves eight WAPS specifications:
Control Glass Fill Height Equivalent to 80% of Canister Volume (WAPS 3.6)
Remove Glass Samples and Analyze Representative Fraction of Samples (WAPS 1.1, 1.2, 1.3, 3.8, 3.9)
Install Lid and Perform Canister Closure (WAPS 2.2, 3.2)

The methodologies/compliance strategies for demonstrating compliance with the Canister Fill Height (3.6) and Sampling of the Glass Product (1.1, 1.3) are rated as low technology maturity. The canister fill height specification is demonstrated (at both DWPF and WVDP) with the Infrared (IR) camera Level Detection System. The IR system is a new application. The IR level detection system is not completely automated. It requires an operator to perform final control actions to ensure the specified fill height is reached. The methodology/compliance strategy for removing glass samples from the melter (at DWPF) and from the canister (at WVDP) use special-design sampling hardware. The technology maturity is low.

The methodology/compliance strategy for demonstrating Canister Weld Closure (2.2) is rated as moderate maturity because it makes use of new equipment and highly qualified operators, but the approach is based on standard welding techniques.

Decontaminate Canister. The methodology/compliance strategy that demonstrates that the filled canister's external surface is decontaminated and free of adhering glass particles (WAPS 3.7) is given a moderate technology rating. The decontamination process is not complex. At DWPF, glass frit slurry blasting is the method used. At WVDP, the method involves immersing the canister in a nitric acid/cerium solution. After the decontamination process, smear tests will be done to show that the alpha and beta/gamma contamination is less than specified levels. The smear test is demonstrated at the time of shipment. Adding some complexity is a remotely-operated needle gun. The needle gun is used to remove any glass particles adhering to the canister's external surface.

Canister Storage. This activity involves monitoring the interim storage cell/building temperature to ensure that it remains below the glass transition temperature of approximately 450o C (WAPS 1.4). The methodology/compliance strategy utilizes a combination of thermal analyses calculations and resistance-temperature-detectors (RTDs). The technology maturity level is High.

Transfer Canisters to CRWMS. This activity involves seven WAPS specifications:
Report the Radionuclide Inventory During Production (WAPS 1.2)
Certify Canister Storage Temperature (WAPS 1.4)
Verify Label Integrity (WAPS 2.2)
Swipe Canister and Repeat Decontamination (WAPS 3.7)
Estimate Heat Generation and Surface Dose Rates (WAPS 3.8, 3.9)
Weigh Canisters and Verify Dimensions and Ability to Stand Upright (WAPS 3.11)

Only one of these compliance strategies is rated low technology maturity. It is the one that demonstrates smearing of a decontaminated canister and counting of the smear's radioactivity with standard instrumentation (WAPS 3.7). The smear survey is carried out remotely. This compliance strategy can only be implemented after radioactive operation begins. The other compliance strategies are rated as High Technology Maturity.

Previtrification Qualification Activity. This activity involves eleven WAPS specifications:
Estimate Crystalline Phases Present in Waste Form Based on Conservative Canister Cooling Rates (WAPS 1.1)

Formulate and Verify Glass Composition Based on PCT Response Model (WAPS 1.3)
Prepare Time-Temperature-Transformation (TTT) Diagram (WAPS 1.4)
Verify Waste Form does not exhibit RCRA Hazardous Characteristics via the Toxicity Characteristic Leaching Procedure (TCLP) (WAPS 1.5)
Show Glass does not Generate Gas and Calculate Radiogenic Gas Generation Rate (WAPS 3.2)
Demonstrate that melting process destroys organics and pyrophoric materials (WAPS 3.3, 3.4)
Show waste form does not lead to canister corrosion (WAPS 3.5)
Demonstrate canistered waste form shall not result in nuclear criticality accident (WAPS 3.10)
Perform canister impact tests (WAPS 3.12)
Design and test canister grapple for use at the repository (WAPS 3.13)

One of the compliance strategies (WAPS 1.3) has a low technology maturity. It deals with verifying the product glass composition and product durability. The compliance strategy makes use of a predictive mathematical model. The model development is based on extensive pilot plant and laboratory testing. The other compliance strategies are High Technology Maturity and are discussed in Table IA, B, C, D.

CONCLUSION

Some of the methods that DWPF and WVDP employ to demonstrate compliance are complex and are being implemented for the first time. A concerted effort is being made on the part of the facility operators to get these to work successfully. Extensive testing and characterization of the waste have occurred. The programs and procedures necessary to implement these strategies/methodologies are approved and in place. Personnel staffing is adequate and the systems needed for waste qualification have been verified as operational.

We believe that sufficient measures are being taken in development, design, testing, procedure formulation, personnel training and Quality Assurance to ensure that DWPF and WVDP will comply with the WAPS and produce satisfactory and acceptable waste forms.

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35-2

HYDROGEN EVOLUTION AND SLUDGE SUSPENSION DURING THE PREPARATION OF THE FIRST BATCH OF SLUDGE AT THE SAVANNAH RIVER SITE

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ABSTRACT

The first batch of High Level Radioactive Sludge for the Defense Waste Processing Facility is being prepared in two 4.9 million liter waste tanks. The preparation involves removing water soluble salts by washing (water addition, sludge suspension, settling and decantation). Sludge suspension is accomplished using long shafted slurry pumps that are mounted on rotating turntables. During the sludge suspension runs in 1993 and 1994, the slurry pumps' cleaning radius was determined to be less than that expected from previous determinations using synthetic sludge in a full size waste tank mockup. Hydrogen concentrations in the tanks' vapor space were monitored during the sludge suspension activities. As expected, the initial agitation of the sludge increased the hydrogen concentration, however, with the controls in place the hydrogen concentration was maintained below seven percent of the lower flammability limit.

INTRODUCTION

Approximately 13.6 million liters of radioactive waste sludge resulting from the chemical separation of uranium, plutonium, and fission products are to be removed from twenty-one high level waste storage tanks, vitrified, and placed in canisters at the Defense Waste Processing Facility (DWPF).
Prior to sending the sludge to the DWPF for vitrification, the sludge is treated in the Extended Sludge Processing Facility (ESP). All of the sludge will undergo sludge washing. The sludge washing process uses a series of water additions and decantations to remove soluble salts from the sludge. Sludge containing high aluminum concentrations is also subjected to aluminum dissolution prior to sludge washing. The gibbsite form of aluminum can be solubilized and washed from the sludge by the addition of sodium hydroxide and heat. The first batch of sludge had previously undergone aluminum dissolution in an In-Tank Demonstration in 1982. This paper discusses the results of recent sludge washing activities in ESP with regard to sludge suspension and hydrogen evolution.
DISCUSSION
Sludge pretreatment is performed in modified type IIIa waste tanks. These tanks are 25.9 meters in diameter and 10 meters high. Each tank contains a center annulus which isolates a concrete roof support column from the tank contents. Also contained in the waste tank are 22 sets of cooling coils extending from 0.3 meter above the bottom of the tank to 0.3 meter below the tank's top. These 7.6 cm diameter carbon steel coils are located on 0.9 meter centers throughout the entire tank's cross-section. The tank's vapor space is continuously force-ventilated with outside air. Approximately 8500 liters per minute of tank vapor is continuously discharged through a HEPA filter and a centrifugal fan. The hydrogen concentration in the tank's vapor space is continuously measured by an on-line analyzer.
Suspension of the sludge lying on the tank floor is accomplished using vertical long shaft (13.7 meter) slurry pumps. This method of sludge suspension was developed using kaolin clay sludge in a full scale model of a waste tank at SRS. Kaolin clay was selected because the clay's rheological properties are similar to high level waste sludge. The slurry pumps have also been successfully used to mobilize radioactive sludge for removal of waste from other waste tanks. Two models of slurry pumps are used for ESP sludge suspension. The standard pump is a centrifugal pump containing two horizontally opposed discharge nozzles. This standard pump is designed to discharge 2300 liters per minute through each of its two 3.8 cm. diameter nozzles at a design speed of 1800 rpm. During testing the pump was able to suspend all of the kaolin clay sludge within a 7.6 meter radius. The ability of the slurry pump to remove sludge is related to the product of the pump's nozzle diameter and the nozzle discharge velocity. This quantity, designated as the VoD, is 1.27 m2/sec for the standard slurry pump.
The second type of slurry pump used in ESP is the Quad Volute (QV) Slurry pump. The QV pump is designed to discharge 7600 liters per minute through each of two 7.6 cm diameter discharge nozzles at a design speed of 2200 rpm. The VoD for the QV pump is 2.11 m2/sec. During testing with kaolin clay sludge, the QV pump was able to suspend all the sludge within a 12.2 meter radius.
A mathematical model was developed by Churnetski (1) to predict the cleaning radius of a pump based on the VoD, the Yield Stress of the sludge (to) and slurry density (SpG). The metric version of this equation is:
Eq. (1)
Where the units are:

m	for	ECR
m	for	D
m/sec	for	Vo
g/cc	for	SpG
dynes/cm2	for	to

The two tanks currently used for processing sludge at SRS are tanks 42 and 51. Four standard slurry pumps are installed in existing risers on tank 42. Four quad-volute slurry pumps are installed in existing risers on tank 51.
The first batch of sludge designated for DWPF was transferred into tanks 42 and 51 in 1987. During 1987 and 1988, four wash cycles were completed in tank 51 and three wash cycles in tank 42. During that time, the effectiveness of the pumps to suspend the sludge was not established. In 1988, the tops of tanks 42 and 51 became contaminated with low activity water that had leaked from the slurry pumps' bearing water columns. Sludge washing was suspended in 1988 and did not resume until

secondary containment enclosures could be designed, fabricated, and installed on each of the slurry pump sections that extended above the tank tops. The sludge has been settling in tanks 42 and 51 since the end of sludge washing operations in 1988.

Methodology

During 1993 and 1994, sludge washing was resumed in ESP with the initiation of the Process Verification Tests (PVT) in tanks 42 and 51. Data was collected during the PVT to determine processing parameters such as; the effective cleaning radius of the slurry pumps, the slurrying time required to suspend the tanks' sludge, and hydrogen releases. The PVT program included operating the slurry pumps for designated intervals at which time the pumps were stopped, the slurry immediately sampled, and the depth of the unsuspended sludge measured at three locations. The process was repeated with increasingly long slurring intervals. The three locations available for measuring the sludge height in tank 51 were located at risers E1, B3 and C3. For tank 42, the available risers were B2, C3 and E1.

The sludge depth is determined by lowering a stainless steel wafer attached to a steel measuring tape into the tank. The tape tension is reduced when the wafer contacts unsuspended sludge. When the tension is reduced, the length of tape below a reference point is recorded. With this information the settled sludge depth can be determined. The method indicates a general trend. However the accuracy of the measurement is dependent on the skill and experience of the operator performing the task.

Hydrogen Released During Sludge Suspension

Hydrogen is continuously generated in the waste tanks as a result of the radiolysis of water. The tank vapor space is continuously ventilated and hydrogen concentrations are nominally <100 ppm which is <0.3 % of LFL.(2) During the initial mixing of the sludge it was expected that hydrogen which had accumulated in the sludge would be released. During the 1993 initial startup of the tank 51 slurry pumps, the hydrogen concentration in the tank vapor space rose from 1% of the lower flammability limit (LFL) to a peak concentration of 6% LFL (0.24 Vol. % hydrogen). The peak concentration occurred approximately two hours after starting the third pump. At the time of peak hydrogen concentration, one slurry pump was operating at 1900 rpm, two other slurry pumps were at 1500 rpm, and the fourth pump was not operating. Due to increasing hydrogen concentration, the pump operating at 1900 rpm was reduced to 1500 rpm. The pumps remained at this speed until a power outage stopped all slurrying activities approximately 5 hours later. The hydrogen concentration and accumulated quantity of hydrogen released from the sludge are shown in Fig. 1. The release of hydrogen continued at elevated concentrations for approximately 50 hours. Included in Fig. 1 is the cumulative hydrogen released based on the hydrogen analyzer data. During the first few hours of operation the tank level dropped 2.3 cm. This drop in volume is equivalent to a release of 22.7 standard cubic meters of hydrogen. Based on hydrogen generation rate calculations for the waste composition in tank 51, the volume of hydrogen released during this initial slurrying activity is approximately 8% of the hydrogen estimated to have been generated in the tank since 1988.

The hydrogen concentration remained below 1% LFL during the remaining wash. After an idle period of five months, no increase in hydrogen concentration was detected during resumption of sludge washing in tank 51.

In May 1994, washing was initiated in tank 42. Each pump was run for four hours before startup of the next pump to minimize the hydrogen release rate. The hydrogen concentration remained below the analyzer's lower detection limit of 1% LFL (0.04 Vol. % H2) throughout testing in tank 42.

Sludge Suspension with Quad Volute Slurry Pumps

At the initiation of the PVT in tank 51, the bottom of the "B1" and "B4" slurry pumps were 1.0 meter above the tank floor, and the bottom of the G and H slurry pumps were 0.2 meter above the tank floor. During the startup of the "B1" slurry pump, it was discovered that the pump's bearing water column could not be supplied sufficient water to maintain column pressure. The high bearing water demand for this slurry pump was suspected to be a result of a failure of the mechanical seal located at the bottom of the pump's bearing housing. Washing continued without using the B1 slurry pump. In addition, the remaining slurry pumps also leaked water into the waste tank at higher than design rates. These slurry pumps were limited in speed in order to reduce bearing water in leakage.

Prior to starting the slurry pumps in tank 51, the sludge depth was approximately

two meters. The measured depths of unsuspended sludge as a function of slurrying time with three pumps are shown in Table I. The results indicate that the quad volute slurry pump's effective cleaning radius (ECR) at 2200 rpm in tank 51 is between 6.2 and 6.9 meters. Using equation 1 and a yield stress of 381 dynes/cm2, the shear stress measured for a tank 42 sludge sample, the predicted ECR is 10.9. The mean cleaning radius, 6.6 meters, is approximately 60 percent of the predicted ECR. A view of the area of the tank floor inside the ECR is shown in Fig. 2. Lowering the slurry pumps that are 1 meter above the tank floor may result in the pumps' producing a greater force on the sludge near the tank bottom and, thereby, increasing the ECR. A test is planned with the pumps in their lowest position, 0.2 meters from the tank bottom.

Sludge Suspension with Standard Slurry Pumps

At initiation of the PVT in tank 42 the sludge depth was approximately 2 meters. Much lower bearing water leak rates were observed than in the tank 51 tests even though both the standard and QV pumps have the same mechanical seals. Two of the four pumps, G and H, were elevated 68 inches off the bottom of the tank while V1 and V2 pumps were 8 inches of the bottom of the tank.

Two of the slurry pumps, G and H, did not draw their expected current during the testing. The G pump reached 40% of its expected amperage and was not operated through most of the testing out of the concern that the pump was cavitating. When the H pump was initially started, the pump amperage increased to the expected range. However, the motor current slowly decreased over the first few hours of operation. When H pump was shutdown for several hours and restarted, the pump would not load. However, if the pump was restarted after being shutdown for several days, the pump would again load to within the expected range but the amperage would slowly decrease over the period of a few hours.

The temperature measured by a thermocouple located beneath the tank floor near the location of H pump rose when the H slurry pump was operated. It is postulated that operating the H slurry pump only affected the sludge in the close vicinity of the pump's discharges. In this isolated area, the sludge temperature increased rapidly as a result of the work done. The increased temperature reduced the available net positive suction head (NPSH). Eventually the sludge temperature increased to a temperature which caused the pump to cavitate. When the pump cavitated, the pump amperage decreased below the expected range.

It is not clear why the problem only occurred with the G and H slurry pumps. Possibly the sludge was more compacted around these pumps. The G and H slurry pumps were elevated 68 inches off the floor in 1988, the last time the pumps were operated. During the same period the V1 and the V2 pumps were at their present location, 8 inches off the floor. It is suspected that the sludge in the vicinity of the H and G slurry pumps was not suspended in 1988 and may have continued to compact since the sludge's introduction into the tank.

The other two pumps, V1 and V2, maintained proper loading throughout the test. The measured depths of unsuspended sludge at the three locations as a function of pumping hours are shown in Table II. The pumping hours for the B2 riser location are based on the H pump. As a result of the problems experienced by the H slurry pump, the sludge removed at the E1 riser is attributed to the V2 slurry pump rather than the H pump. During testing, the sludge height under the B2 riser actually increased.

The effective cleaning radius observed in the tank 42 PVT is estimated to be 5.2 meters which is approximately 80% of the predicted value. Figure 3 shows the area of the tank cross section that is within the cleaning radius of 5.2 meters for the V1 and V2 slurry pumps . Approximately 30 percent of the sludge is within the ECR of these two pumps.

CONCLUSIONS

The performance of the slurry pumps during the Process Verification Tests in ESP was below expectations based on previous testing. Significantly higher bearing water leak rates were observed with the QV pumps in tank 51. The leak rate for the B1 pump in tank 51 was so excessive that it could not be operated at any speed without overriding the pump's bearing water pressure interlock. The bearing water leak rate for the tank 42 standard slurry pumps was near the expected range.

The observed ECR for both the standard and QV slurry pumps were well below predicted values. The quad volute pump's ECR was approximately 54% of the design basis of 40 feet. All four slurry pumps will be required to operate at maximum speed to fully

suspend the sludge in tank 51. The tank 42 standard slurry pumps' ECR was approximately 80% of the predicted value. However, based on the ECR of the standard slurry pump, all four of the standard slurry pumps operating at full speed will not be able to fully suspend the sludge in tank 42.
The results of the PVT indicated the majority of the sludge mobilization occurred during the first 50 hours of operation. However, sludge continued to be mobilized at reduced rates throughout testing . A steady state condition in which all of the sludge was suspended was never reached in either of the two processing tanks. The release of hydrogen during initial startup of slurry pumps was maintained below ten percent of LFL. This indicates that with the existing monitoring and control procedures, the hydrogen release can be maintained below its flammability limit in the tank vapor space.

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35-3

DETERMINATION OF URANIUM DISTRIBUTION IN THE EVAPORATION OF SIMULATED SAVANNAH RIVER SITE WASTE (U)

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ABSTRACT

The results of an experimental program addressing the distribution of uranium in saltcake and supernate for two Savannah River Site waste compositions are presented. Successive batch evaporations were performed on simulated H-Area Modified Purex low-heat and post-aluminum dissolution wastes spiked with depleted uranium. Waste compositions and physical data were obtained for supernate and saltcake samples. For the H-Area Modified Purex low-heat waste, the product saltcake contained 42% of the total uranium from the original evaporator feed solution. However, precipitated solids only accounted for 10% of the original uranium mass; the interstitial liquid within the saltcake matrix contained the remainder of the uranium. In the case of the simulated post-aluminum dissolution waste; the product saltcake contained 68% of the total uranium from the original evaporator feed solution. Precipitated solids accounted for 52% of the original uranium mass; again, the interstitial liquid within the saltcake matrix contained the remainder of the uranium. An understanding of the distribution of uranium between supernatant liquid, saltcake, and sludge is required to develop a material balance for waste processing operations. This information is necessary to address nuclear criticality safety concerns.

INTRODUCTION

Carbon-steel tanks are used at the Savannah River Site (SRS) for the interim storage of nuclear waste. The waste consists of two phases: a sludge and its associated supernate. The solution is strongly alkaline with a high salt content (i.e., sodium nitrate, nitrite, aluminate, etc.). To minimize the volume of waste, the supernatant solution is evaporated and stored as a mixture of concentrated liquid and crystalline solids (i.e., saltcake). The evaporation process consists of successive heating and cooling of the solution. During heating, water is evaporated resulting in salt saturation of the solution. Upon cooling, salts, in excess of their solubility at the lower temperature, crystallize forming a saltcake layer. The residual supernate is then reprocessed to repeat the cycle. This process is repeated with the supernate solution until the waste tank is filled with saltcake and a minimal volume of saturated supernate. Final disposal of the saltcake will be accomplished by dissolution of the saltcake and processing through the Defense Waste Processing Facility (DWPF). Decontamination of the high-level radioactive waste solutions will be accomplished by the precipitation of cesium and potassium with sodium tetraphenylborate and adsorption of strontium by monosodium titanate in the In-Tank Precipitation (ITP) process. The slurry will be concentrated by filtration and washed. The resulting decontaminated salt solution filtrate will be stabilized in Saltstone. The concentrated slurry will be transferred for vitrification in DWPF.

Fissile isotopes of uranium are contained in the high-level nuclear waste that will be processed for permanent disposal via the ITP and sludge processing facilities (1). An understanding of the distribution of uranium between supernatant liquid, alkaline solids (saltcake), and sludge is required to develop a material balance for waste tank farm processing operations. This information is necessary to address nuclear criticality safety concerns. Two concerns are that fissile uranium, contained in the saltcake, will either not dissolve during salt dissolution or will dissolve at a much slower rate than saltcake. Either mechanism could result in uranium settling in the tank and collecting in sufficient quantity to form a critical mass. The solubility of uranium in alkaline salt solutions has been measured to better understand the behavior of the fissile material (2,3). This paper presents the results of an experimental program addressing uranium distribution behavior in an effort to provide guidance for safe dissolution and removal of saltcake. The work presented in this paper provides for an increased understanding into the distribution of uranium in saltcake and supernate for two SRS waste compositions. The two waste compositions simulated and investigated were H-Area Modified Purex (HM) low-heat and post-aluminum dissolution wastes.

EXPERIMENTAL

Simulated Waste Solutions

Simulated supernate wastes with H-Area Modified Purex (HM) low-heat and post-aluminum dissolution compositions, both spiked with depleted uranium, were used in the evaporation studies. The low-heat waste composition is based upon a 1981 waste inventory estimate (4). The post-aluminum dissolution waste composition is based upon an in-tank sludge processing demonstration in 1983 (5). The nominal compositions of the simulated test solutions are shown in Table I. For preparative purposes, aluminum nitrate, $Al(NO3)39H2O$, was dissolved separately with excess sodium hydroxide, NaOH, in distilled water to form sodium aluminate, $NaAlO2$, for the low-heat waste. Because of the lower nitrate level in the post-aluminum dissolution waste, a 65 wt % $NaAlO2$/19 wt % NaOH reagent was used instead of aluminum nitrate. For both simulants, the remaining components were mixed with distilled water and the aluminate solutions were added slowly while stirring. A 250 g U/L solution was prepared with depleted uranyl nitrate ($UO2(NO3)26H2O$) in 0.1 M nitric acid. This uranium solution was added to the waste simulants at a concentration of 100 mg U/L solution. The solutions were then allowed to stir for one week. After stirring, the solutions were filtered with a 0.45 mm filter to remove insoluble uranium solids.

Evaporation Procedure

Batch evaporation of simulated low-heat and post-aluminum dissolution wastes were conducted. The starting solution volume for the evaporation tests was approximately four liters. The starting solution densities of the low-heat waste and post-aluminum dissolution waste were 1.29 and 1.24 g/mL, respectively. Successive batch evaporations were performed on the waste simulants until the density of the supernate was approximately 1.5 g/mL. The evaporations were performed in a 6 L Teflon vessel heated by resistance-heated Chromalox elements. The solutions were heated to boiling; the overhead vapors were condensed and collected in a beaker. The volume of water removed for each evaporation step was based upon the estimated amount of evaporation required to obtain a solution density of 1.5 g/mL. Once the desired volume of water was evaporated, the boiling solution was removed from the evaporator. Small (i.e., 15-30 mL) samples of the boiling solution were collected in test tubes for analysis. the remainder was collected in a volumetric flask. These solutions were cooled to room temperature to allow crystallization of sodium salts. The supernate in the volumetric flask was decanted from the solids (saltcake) and the supernate solution density was determined. If the density was less than 1.5 g/mL, the decanted supernate was returned to the evaporator as the feed solution for the next evaporation step. This evaporation and precipitation procedure was repeated until the resulting supernate density was approximately 1.5 g/mL.

Sample Analysis

Supernate from each of the 15-30 mL supernate/solid samples collected in each evaporation step was decanted and the saltcakes were centrifuged to remove as much interstitial liquid as possible. The supernate and saltcake fractions were analyzed to determine their volume, weight, and chemical composition. Saltcake fractions were prepared for analysis by dissolution in distilled, deionized water. Extensive physical and chemical data (e.g., boiling points and densities, weight percent interstitial liquid in saltcake, and trends in chemical compositions) pertaining to

each evaporation step and the resulting supernate and saltcake were collected. Uranium analysis was performed on unfiltered and filtered (0.2 mm) solutions. Uranium was analyzed by laser-activated fluorescence and by x-ray fluorescence. Unfiltered solutions were analyzed for the remaining constituents. Concentrations of aluminum (Al), sodium (Na), silicon (Si), and phosphorous (P) were analyzed by inductively-coupled plasma (ICP) emission spectroscopy. Nitrate (NO3-), nitrite (NO2-), phosphate (PO43-), oxalate (C2O42-), and sulfate (SO42-) were measured by ion chromatography. Hydroxide (OH-), aluminate (AlO2-), and carbonate (CO32-) were determined by titration.

RESULTS AND DISCUSSION

Low-Heat Waste Evaporation and Characterization Data

Table II lists the evaporation data for the simulated low-heat waste. Four successive evaporations were performed on the low-heat waste which increased the solution density from 1.29 g/mL to 1.44 g/mL. The boiling point of the starting solution was 106.3C while the boiling point of the supernate feed in the fourth evaporation was 117.9C. The total volume reduction for the low-heat waste was approximately 70% based upon the supernate remaining after the fourth evaporation plus the total volume of solids from each evaporation. The solution volume was reduced by 87%. No solids crystallized from solution after the first evaporation, but solids did crystallize from solution in the remaining evaporations. The density of saltcake from each successive evaporation increased from 1.74 g/mL to 1.86 g/mL. The weight percent interstitial liquid in saltcake was determined from the volume of liquid removed from saltcake during centrifuging. The amount of interstitial liquid in saltcake was 14 wt % after the second and third evaporations and 18 wt % after the fourth evaporation.

Table III provides the composition of the starting solution and the supernate solutions after each evaporation step for the low-heat waste. As shown, the hydroxide concentration increased with each evaporation from 1.22 M in the starting solution to 5.22 M after the fourth evaporation. Nitrate initially increased after the first evaporation and then decreased with each evaporation. Nitrite and aluminate increased with each evaporation. Uranium increased from 8.1 mg/L to 31.4 mg/L. The uranium concentrations are significantly lower at the 1-2 M hydroxide concentrations in the low-heat waste as compared to existing data (3) thus suggesting that the low-heat waste was not saturated with uranium in the starting solution. However, the uranium concentrations in the 3-6 M hydroxide range are slightly higher than the existing data. The data suggest that uranium solubility may be significantly affected by concentrations of other components of the waste. In addition, as discussed by Karraker (3), measurements of the solubility of U in strongly basic solutions depend upon the method used for solution make-up.

Saltcake was not produced during the first evaporation step because the waste components of the low-heat waste had not reached their solubility limits. Saltcake was produced after the second, third, and fourth evaporations. Table IV shows the composition of saltcake after each evaporation step. The weight percents in Table IV are based on the total saltcake weight which includes the sodium salts plus water. Weight percent water was calculated from the difference between the weight of the sodium compounds and the total weight of saltcake. The calculated weight percent water is assumed to be fairly accurate because the total sodium moles calculated from the common sodium compounds associated with each ion were nearly equal to the sodium moles analyzed by inductively-coupled plasma emission spectroscopy. As shown in Table IV, the concentration of major components in the saltcake remain fairly constant for each evaporation while the weight of solids from each evaporation decreases with each succeeding step. The major component by weight is nitrate (~50 - 70 wt % based on total saltcake weight for each evaporation). The weight percent undissolved uranium and dissolved uranium in interstitial liquid remaining after centrifuging for each evaporation is approximately 0.0009%. The weight percent water in saltcake decreased from 27 and 24% after the second and third evaporations to 8% after the fourth evaporation.

Post-aluminum Dissolution Waste Evaporation and Characterization Data

Table V shows the evaporation data for the simulated post-aluminum dissolution waste. Three successive evaporations were performed on the post-aluminum dissolution waste which increased the solution density from 1.24 g/mL to 1.47 g/mL. The boiling point of the starting solution was 104.3C and the boiling point of the supernate feed in the third evaporation was 122.5C. The total volume reduction for the

post-aluminum dissolution waste was approximately 69% based upon the supernate remaining after the third evaporation plus the total volume of solids from each evaporation. The solution volume was reduced by 80%. Solids crystallized from solution after each evaporation step and the saltcake density increased from 1.41 g/mL to 1.76 g/mL. The concentration of interstitial liquid in saltcake determined from centrifuging was 6 wt % after the first evaporation and approximately 20 wt % after the second and third evaporations.

Table VI shows the composition of the starting solution and the supernate solutions after each evaporation step for the post-aluminum dissolution waste. As shown, the hydroxide concentration increased with each evaporation from 2.75 M in the starting solution to 8.80 M after the third evaporation. Nitrate initially increased after the first evaporation and then decreased with each evaporation. Nitrite and aluminate increased with each evaporation. The uranium concentration was initially 45.0 mg/L and then decreased and remained constant at approximately 20 mg/L. The uranium concentrations, as a function of hydroxide, are comparable to the Karraker data (3) at the low hydroxide range but slightly higher in the 5-9 M hydroxide range. This suggests that the uranium solubility is strongly dependent on hydroxide concentration but may also be affected by concentrations of other components.

Saltcake was produced during each evaporation step for the post-aluminum dissolution waste. Table VII shows the composition of saltcake after each evaporation step. The weight percents in Table VII are based on the total saltcake weight which includes the sodium salts plus water. Weight percent water was calculated from the difference between the weight of the sodium compounds and the total weight of saltcake. The calculated water content is assumed to be fairly accurate because the total sodium moles calculated from the common sodium salts associated with each ion were essentially equal to the sodium moles analyzed by inductively-coupled plasma emission spectroscopy as shown in Table VII. As shown in Table VII, as in the case of the low-heat waste saltcake, the major component is nitrate (~ 35 - 50 wt % based on total saltcake weight for post-aluminum vs. ~ 50 - 70 wt % for low-heat). The concentration of undissolved uranium and dissolved uranium in interstitial liquid decreases from 0.0056 wt % to 0.0036 wt %. The amount of water in saltcake increases from 23 wt % after the first evaporation to 37 wt % after the third evaporation.

Uranium Distribution

Low-heat waste was subjected to four successive evaporations. Saltcake was produced from each of the last three evaporations. The percentage of available uranium contained in the saltcake produced by each successive evaporation (i.e., undissolved uranium and dissolved uranium contained in interstitial liquid) decreased with each successive evaporation. For low-heat waste, the fraction of available uranium contained in each saltcake is 22, 18, and 8% for the second, third, and fourth evaporations, respectively. This data trend is likely the result of a combination of two factors. First, each successive evaporation produces less saltcake than the previous one. This results in a decrease in the amount of interstitial liquid. Second, the uranium concentration in the supernate is increasing. This would indicate that the uranium solubility limit has not been reached. Therefore, the ratio of uranium contained in the saltcake to uranium in solution is decreasing. Simply stated, the salts (i.e., NaOH, $NaNO_3$, $NaNO_2$, etc.) in this particular solution have reached their solubility limit for the second, third, and fourth evaporations while the uranium has either not reached, or has just reached, its solubility limit by the fourth evaporation. This theory suggests that the majority of uranium contained in saltcake produced by the second and third evaporations is present in interstitial liquid. The effect of the four evaporation cycles, as a whole, on the low-heat waste is that 42% of the uranium is contained in the composite saltcake.

Similar to the observations with low-heat waste, the percentage of available uranium contained in the saltcake produced by evaporation of post-aluminum dissolution waste decreases with each successive cycle. For post-aluminum dissolution waste, the fraction of available uranium contained in saltcake is 49, 22, and 19% for the first, second, and third evaporations, respectively. These percentages (specifically that for the first evaporation) are higher than those observed for the low-heat waste. This is expected in light of the fact that the post-aluminum dissolution waste starting solution had a uranium concentration of 45.0 mg/L and that 78% of the uranium was contained in solids larger than 0.2 mm. The large size would suggest that the uranium was colloidal and would easily separate from solution. Therefore,

the first evaporation likely caused a large portion of the uranium to aggregate with the saltcake. A similar pattern was repeated for the second evaporation; however, a much smaller percentage of the uranium was contained in solids larger than 0.2 mm. The effect of the three evaporation cycles, as a whole, on the post-aluminum dissolution waste is that 68% of the uranium is contained in the composite saltcake. An estimate of the amount of dissolved uranium contained in interstitial liquid is discussed in the next section.

Dissolved Uranium in Interstitial Liquid of Saltcake

The amount of dissolved and undissolved uranium contained in the interstitial liquid of the low-heat waste saltcake was estimated from the experimental data. The estimate was obtained from the weight percent water contained in a saltcake sample and the composition of the supernate above the saltcake. From this data, the volume of supernate remaining in the saltcake after centrifuging was determined. Using the uranium concentration of the supernate and the volume of supernate remaining in the saltcake, it was possible to estimate the quantity of dissolved uranium that is present in the saltcake. The difference between the measured total uranium contained in the saltcake and the dissolved uranium is undissolved uranium. Using this method, it was estimated that 82, 100, and 16% of the uranium contained in the saltcake produced by the second, third, and fourth evaporations, respectively, of low-heat waste was dissolved. Data presented in the Uranium Distribution section of this document showed the fraction of available uranium contained in the saltcake produced by the second, third, and fourth evaporations of low-heat waste was 22, 18 and 8%, respectively. Coupling this data with the above estimates would predict that the fraction of available uranium present as undissolved uranium in saltcake from the second, third, and fourth evaporations of low-heat waste is 4.0, 0.0 and 6.7%, respectively. The effect of the four evaporation cycles, as a whole, on the low-heat waste is that an estimated 10% of the total uranium originally present in the salt solution is precipitated as a solid.

The fact that only 10% of the total uranium originally present in the salt solution is precipitated as a solid is not unexpected. Three pieces of evidence support this estimate. First, tests showed that 84 to 96% of the uranium contained in the low-heat saltcake from the three evaporations was removed by rinsing of the saltcake with a highly concentrated sodium salt solution (6). Second, documented information shows that precipitated uranium does not readily dissolve in highly caustic salt solutions (2,3). Third, the percentage of uranium contained in the salt solution rinses that was less than 0.2 mm in size was relatively large (i.e., 67 to 98% for the second, third, and fourth evaporations). Coupling these three pieces of evidence suggest that a large fraction of the uranium contained in the centrifuged, low-heat waste saltcake samples was in fact uranium dissolved in the interstitial liquid.

The amount of dissolved uranium contained in the interstitial liquid of post-aluminum dissolution waste saltcake was estimated. Using the method described above, it was estimated that 14, 79, and 32% of the uranium contained in the saltcake produced by the first, second, and third evaporations, respectively, of post-aluminum dissolution waste was dissolved. Data presented in the Uranium Distribution section of this document showed the fraction of available uranium contained in the saltcake from the first, second, and third evaporations of post-aluminum dissolution waste was 49, 22 and 19%, respectively. Coupling this data with the above estimates projects that the fraction of available uranium present as undissolved uranium in saltcake from the three evaporations of post-aluminum dissolution waste is 42, 4.6 and 13%, respectively. The effect of the three evaporation cycles, as a whole, on the post-aluminum dissolution waste is that an estimated 52% of the total uranium originally present in the salt solution is precipitated as a solid. This value shows that a substantial portion of the uranium is present in a solid phase.

CONCLUSIONS

Successive evaporations were performed on simulated low-heat waste and post-aluminum dissolution waste spiked with depleted uranium. Waste compositions were determined for supernate and saltcake samples. For the H-Area Modified Purex low-heat waste, the product saltcake contained 42% of the total uranium from the original evaporator feed solution. However, precipitated solids only accounted for 10% of the original uranium mass; the interstitial liquid within the saltcake matrix contained the remainder of the uranium. In the case of the simulated post-aluminum dissolution waste; the product saltcake contained 68% of the total uranium from the original

evaporator feed solution. Precipitated solids accounted for 52% of the original uranium mass; again, the interstitial liquid within the saltcake matrix contained the remainder of the uranium.

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35-4
OPTIMIZATION OF WASTE LOADING IN HIGH-LEVEL GLASS IN THE PRESENCE OF UNCERTAINTY

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ABSTRACT
Hanford high-level liquid waste will be converted into a glass form for long-term storage. The glass must meet certain constraints on its composition and properties in order to have desired properties for processing (e.g., melt electrical conductivity, melt viscosity, and liquidus temperature) and acceptable durability for repository acceptance. The Optimal Waste Loading (OWL) models, based on rigorous mathematical optimization techniques, have been developed to minimize the number of glass logs required and determine glass-former compositions that will produce a glass meeting all relevant constraints. There is considerable uncertainty in many of the models and data relevant to the formulation of high-level glass. In this paper, we discuss how we handle uncertainty in the glass property models and in the high-level waste composition to the vitrification process.
Glass property constraints used in optimization are inequalities that relate glass property models obtained by regression analysis of experimental data to numerical limits on property values. Therefore, these constraints are subject to uncertainty. The sampling distributions of the regression models are used to describe the uncertainties associated with the constraints. The optimization then accounts for these uncertainties by requiring the constraints to be satisfied within specified confidence limits.
The uncertainty in waste composition is handled using stochastic optimization. Given means and standard deviations of component masses in the high-level waste stream, distributions of possible values for each component are generated. A series of optimization runs is performed; the distribution of each waste component is sampled for each run. The resultant distribution of solutions is then statistically summarized.
The ability of OWL models to handle these forms of uncertainty make them very useful tools in designing and evaluating high-level waste glasses formulations.

INTRODUCTION
In immobilizing high-level liquid waste for long-term storage, it is advantageous to produce the minimum volume of immobilized waste glass (for a given volume of waste),

thus minimizing vitrification and disposal costs. The Optimal Waste Loading (OWL) models, based on rigorous mathematical optimization techniques, have been developed to determine the minimum number of glass logs required to immobilize Hanford high-level liquid waste. This optimization varies the glass-formers composition to maximize the waste loading (and minimize the glass volume), such that all processing and product (durability) constraints on the glass are satisfied.

Because there is uncertainty in both the property model predictions and in the waste feed composition for vitrification, the question naturally arises as to how the uncertainty affects the calculated waste loading and resultant number of logs required. In this paper we describe our approaches for addressing this question. We discuss techniques we are developing for addressing each of these uncertainties separately. As these techniques mature, we will integrate these and other techniques to handle all appropriate uncertainties.

In this paper, we briefly describe the general constrained optimization problem, which is the basis for the OWL model. We then present an overview of the OWL model formulation, followed by a discussion of the glass property models that are used in the constraints for optimization. We then describe our techniques for handling property model and waste composition uncertainties, and present some results for each. We end with some brief concluding comments.

THE GENERAL CONSTRAINED OPTIMIZATION PROBLEM

In the general constrained optimization problem, the objective is to find the multidimensional point x that produces the maximum (or minimum) value of some function f and meets a set of criteria called constraints. The problem would be stated as:

maximize f(x)
subject to h(x) = 0
g(x) 0

where

f is the objective function which we want to maximize (or minimize). It is a single (scalar) function of the unknown variables
x is a vector (x1, x2, ... , xn) of the variables over which we optimize
h is a vector function containing the equality constraints (h1(x), h2(x),..., hp(x))
g is a vector function containing the inequality constraints (g1(x), g2(x),..., gq(x))

Each equality and inequality constraint is also a function of the unknowns. In general, the objective function and any or all constraints may be nonlinear.

GLASS FORMULATION OPTIMIZATION MODEL

In the simple glass formulation problem, frit (glass formers) is added to a single waste composition, and the mass fraction of waste in the glass (waste loading) is maximized such that all the constraints are satisfied. The frit composition is varied as part of the optimization. The problem formulation is shown in Fig. 1 and briefly described below. More detailed information on the model can be found in Hoza (1).

FIG. 1

The simple waste optimization problem can be generally stated as follows:

minimize number of glass logs required
  or
maximize waste loading in the glass

subject to mass balance constraints
  property model component bounds
  solubility constraints
  glass property constraints

The objective function and constraints will be discussed in the next two sections.

Objective Function

The goal is to minimize the number of glass logs necessary to immobilize a waste of the specified composition. This can also be achieved by maximizing the waste loading

(fraction of the glass that is waste).
Constraints
There are four classes of constraints in the model.
The first, the mass balance constraints, are equalities which define the relationships involved in the formation of glass from its components. These include an overall balance and component balances for all components.
The second, the property model component bounds, limit the range of the composition (mass fraction) values each component can have in the calculated glass composition. They reflect the composition region over which the glass properties were experimentally determined, as part of the Composition Variation Study (CVS) (2), and define a polyhedron in composition space that specifies the region over which the glass property models are considered valid. Compositions outside these limits will not necessarily produce unacceptable glasses. Rather, these compositions represent regions for which the glass property models must be extrapolated. Promising glasses outside these limits would have to be evaluated experimentally to determine their acceptability.
The third, the solubility constraints, limit the maximum value for the mass fraction of selected components (Cr_2O_3, F, P_2O_5, SO_3, and noble metals). They are intended to represent solubility limits for the specified components. These limits cover component species not included among the species used in the glass property models. Eventually these solubility limits will be replaced with thermodynamic calculations that predict insoluble species.
The fourth, the glass property constraints, utilize the glass property models developed in the CVS. These are discussed in the next section.
GLASS PROPERTY MODELS
The glass property models are equations empirically fit to data, i.e., glass compositions and property values (melt viscosity, melt electrical conductivity, and durability in this work). Liquidus temperature models have also been developed but were not used in this work. The modeling approach and the calculation of uncertainty are addressed in the balance of this section.
Modeling of Properties
The property models are empirically fit linear and nonlinear (in composition) models. The models were developed as part of the Composition Variation Study (CVS) and are described in Hrma, Piepel, et al. (2). The CVS has been performed in five phases (CVS-I and CVS-II Phases 1-4). The models used in this work were based on data obtained through CVS-II, Phase 2.
The CVS used statistical mixture experiment design and optimal experimental design methods and software to select the glass compositions tested throughout the CVS. The glass composition region included is expected to contain glasses that might be made from various waste types expected to be processed at Hanford.
The model of each property is of the form
Eq. (1)
where bi and bij are the coefficients of the first- and second-order terms, respectively; xi is the mass fraction of component i; and 10 is the number of components considered in the study. The components included in the models are SiO_2, B_2O_3, Na_2O, Li_2O, CaO, MgO, Fe_2O_3, Al_2O_3, ZrO_2, and Others, which accounts for all species other than the nine specifically included. For the linear property models, all bij are zero.
The glass properties used in this work were viscosity, electrical conductivity, and durability (actually rate of release of boron) by either the Product Consistency Test (PCT) or Materials Characterization Center Test (MCC-1). The current version of OWL includes PCT Li and Na releases, and no longer uses MCC-1 releases.
Calculation of Uncertainty
Predictions made with a fitted property model are subject to uncertainty in the fitted model coefficients. The uncertainty results from the random errors in property values introduced during testing and measurement as well as minor lack-of-fit of the empirical model relative to the true relationship.
The uncertainty in a predicted property value for a given glass composition is defined as
Uncert=M[xTSx]0.5
Eq. (2)
where
M = multiplier, which is usually the upper 95th percentile of a

t-distribution
[t.95(n-p)], where n is the number of data points used to fit the model and p
is the number of fitted parameters (coefficients) in the model
x = glass composition vector expanded in the form of the model
x T = transpose of glass composition vector expanded in the form of the model
S = covariance matrix of the estimated parameters (coefficients)

For linear (first-order) property models, x is the usual glass composition vector. For nonlinear models, the vector is augmented by second-order terms. For example, if there are two second-order terms, x12 and x2x4, the usual composition vector (x1, ... ,x10) becomes (x1, ... ,x10, x12, x2x4).

OPTIMIZATION WITH GLASS PROPERTY MODEL UNCERTAINTY

The method used to account for glass property model uncertainty in the glass optimization and results of optimization calculations with property model uncertainty are given in the next two sections.

Method

This model accounts for uncertainty in the glass property constraints by using uncertainty to narrow the feasible region determined by glass property models. This approach changes the form of the glass property constraint to

Eq. (3)

When Uncert = 0, this constraint is the same as for the model that does not account for property model uncertainty. Figure 2 shows the effect on a ternary diagram (for a waste + frit + recycle mixture. The idea is the same for a waste + frit system, but a ternary diagram better helps visualize the concept). A single linear glass property constraint with upper and lower limits is shown on the figure. The regions in the triangle with dark shading are infeasible (the constraint cannot be satisfied in those regions). The unshaded region is feasible. The lightly shaded regions represent those compositions that become infeasible when property model uncertainty is considered. Alternately, it can be viewed as the shrinkage of the feasible region due to uncertainty. The shading around the glass composition point represents the uncertainty in the glass composition resulting from uncertainty in the waste composition. Methods for dealing with this uncertainty will be discussed in the section on waste composition uncertainty.

Fig. 2.

Results

The effect of property model uncertainty on maximum waste loading was examined for four Hanford double shell tank waste types. Table I summarizes the results of calculations that explore this effect for two constraint sets.

TABLE I

When the full constraint set is used (first and second rows in the Table I), there is no difference between the waste loading with uncertainty and the waste loading without uncertainty in the glass property models. This is not surprising. The uncertainty in the glass property constraints effectively tightens the glass property constraints, but not enough to make a difference. The binding constraint for each case is still the same as for the case without uncertainty, so the glass property constraints and their uncertainties are irrelevant (for these cases; this will not always be the result).

When only the glass property constraints (viscosity, electrical conductivity, and durability) are used (third and fourth rows in Table I), the following occurs:

Waste loading is reduced. As expected, the uncertainty in the glass property constraints makes a difference. The percent reduction in waste loading as a result of considering the uncertainty is on the order of the uncertainty in the binding constraints.

The uncertainty in the glass properties is much greater than it is for the full-constraint case. Because the calculated uncertainty is a function of where the point is located in composition space, this indicates that these points are in composition regions where less experimental data are available and may even be outside the experimental region. Examination of the glass compositions for these cases (which are not in the table) confirms this. Several of the component compositions are outside the upper and lower limits on the ten components (because those limits were dropped for these cases).

OPTIMIZATION WITH WASTE COMPOSITION UNCERTAINTY

The method used to account for waste composition uncertainty in the glass optimization and results of optimization calculations with waste composition uncertainty are given in the next two sections.

Method

The basic approach taken to address the optimization in the presence of waste composition uncertainty problem relies on the stochastic modeling method (3). Using this method, the strategy is to generate a large number of possible waste compositions based on the composition error structures, and for each of these, to generate a waste loading. The distribution of waste loadings can then be analyzed. The main steps in this method, as applied to this problem, are:

Develop probability distributions for the masses in the high-level vitrification feed of each of the components followed in the OWL models, based on estimates of means and standard deviations. For this work, all mass distributions were assumed to be normal (Gaussian).

Sample the above distributions and developing N waste composition input sets (mass fraction basis). Sampling the distributions provided masses for each of the species tracked. Latin Hypercube Sampling (4) was used because it provides better coverage of the composition distributions than simple random sampling with fewer samples. Given these masses and the total mass of the waste, the mass fractions of all species were determined and normalized to 1.000.

Run the N waste composition sets through the OWL glass formulation model to calculate the optimal waste loadings for each waste composition set.

Analyze the resulting distribution of waste loadings for the N input sets.

For this work, uncertainties in waste components were assumed to be statistically independent (i.e., uncorrelated). This is likely an unrealistic assumption, but knowledge of composition uncertainty correlations was insufficient to account for them in this work. Future efforts will account for them once they are adequately quantified.

Results

The method described above takes distributions in the masses of all relevant species, performs a series of calculations, and produces a distribution of waste loadings. This section looks at how the waste loading distribution is related to the input distributions, and what one can conclude from the output distribution?

The relationship between the output distribution and the input distribution depends on the constraints--which constraint(s) is/are binding and whether the same constraint is binding for all cases or the binding constraint changes for different runs. The following situations are possible; they are listed in order of increasing complexity.

The same single-component constraint is always binding.

The same multiple-component constraint (e.g., durability) is always binding.

The binding constraint is different for different runs.

No feasible solution is possible for some runs.

The waste selected for the sample calculation represents the least complicated situation. For this case, the binding constraint was consistent over all generated waste composition sets. This binding constraint was the upper limit on a single waste component (P_2O_5). As expected for this case, the waste loading varied inversely with the mass fraction of P_2O_5. Because the upper bound on P_2O_5 was the binding constraint for optimization, higher concentration of P_2O_5 causes a lower maximum waste loading fraction (WLF).

What can one conclude for this single-component-limited case? If the generated waste component mass distributions reflect reality, and if N is set appropriately high, then the sample input sets are increasingly likely to cover the range of possible waste composition sets. Each WLF is the highest WLF that will produce glass meeting the property constraints for an input waste composition set. Therefore, the distribution of optimal WLFs represents the possible range of optimal WLFs given the uncertainty defined for the input high-level waste stream.

Figure 3 shows the distribution of the optimal WLF and the reverse cumulative distribution of the optimal WLF, with cumulative probability increasing as WLF decreases. The cumulative distribution can be interpreted as follows: for any WLF calculated by maximizing the WLF subject to constraints as per OWL optimization, the cumulative distribution represents the probability that that WLF can be achieved given the waste composition and its associated uncertainty and error structure. For

example, if the WLF is 0.038, the probability of being able to achieve that WLF is 0.85.
Fig. 3.
The above analysis was for the simplest case; the same single-component constraint is always binding. How would the results change for binding multiple-component constraints or for changing binding constraints? Subsequent work will have to examine this issue, but a cumulative distribution (as in Fig. 3) could still be developed and used as described above.

Issues

Because this was a preliminary look at the application of stochastic modeling, many assumptions were made to simplify calculations. These assumptions, which are addressed below, will be revisited in future work.

Two independent (and inconsistent) determinations of the total mass are available, the sum of the sampled masses and the measured mass (actually measured volume and density). Some technique to reconcile the two is needed.

The distributions for each component were assumed to be independent. This is unlikely to be true for several reasons (e.g., relationships of components in frit, waste, and recycle; correlations in analytical uncertainties; and imposed correlations among component mass fractions because they must sum to one).

The simple case examined had the same binding constraint for all N samplings. This will not generally be the case. Accounting for statistical dependence between components may also change binding constraints.

CONCLUSIONS

The techniques presented here address the uncertainty in property models (which are used in specifying constraints in the optimization model) and in waste feed composition. The latter technique needs further development to address the issues identified. Combining the two techniques would allow formulation of glasses in the presence of both types of uncertainty.

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35-5

HIGH VOLUME REDUCTION AND GROUP PARTITION OF SIMULATED HLLW BY SUPER HIGH TEMPERATURE METHOD USING AN INDUCTION COLD CRUCIBLE

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ABSTRACT

An advanced waste management of HLLW, in which super high temperature treatment is conducted by using an induction cold wall crucible, is proposed. In this process, calcined HLLW added with minimum amount of reducing agent is heated, melted and chemically reduced, resulting in the partitioning of platinum group metals from oxide residue in the form of high volume reduced ceramics. Validity of the process concept was confirmed by the fundamental experiment. Solidified and group partitioned materials of 150-250g were obtained both by continuous and batch type treatment at the temperature of 1600C in Ar gas atmosphere. A few blocks of metal were obtained in a dark green glass material. Alkaline earth, zirconium and lanthanide elements were in volume reduced oxide. Platinum group elements, including corrosion products were successfully partitioned into metal.

INTRODUCTION

The development of the nuclear industry has reached a stage where reprocessing is an established technology. Vitrification, the first strategy, is considered as the reliable method of high level waste treatment to ensure safe immobilization of radionuclides for the final disposal into deep geological formations (1). However, current vitrification characterized by concentration involves neither recovery of by-products nor high volume reduction.

The second strategy, which is characterized by the separation of high level actinides and long lived fission products from the high-level liquid waste (HLLW) and destroy them by nuclear transmutation, has been investigated as future technologies (2). Such a strategy would favor the recovery of potentially useful fission products, such as caesium and strontium. Platinum group metals, especially rhodium and palladium, could also be recovered. However, the second strategy, based on the wet treatment involving such as the solvent extraction, adsorption on iron exchangers, etc., inevitably leads to the increased amount of secondary wastes and high reprocessing cost (3). Moreover, the partitioning nature is different from that occurring in the Purex process, so this partitioning technique is not acceptable for the current Purex process.

An advanced strategy is to have both characteristics compatible with the current Purex process and enjoying the most of the fruitful partitioning of nuclides aimed in the second strategy accompanied with neither generation of the increased amount of secondary wastes nor high reprocessing cost. A super high temperature method (SHTM) is one of the promising such strategies (4). In this method, calcined HLLW added with minimum amount of reducing agent is heated, melted and chemically reduced, resulting in the partitioning of platinum group metals from oxide residue in the form of high volume reduced ceramics. In the process, melting in a conventional furnace with high temperature wall cannot be adopted, because melting in the furnace surrounded with refractory wall inevitable leads to the generation of a fair amount secondary wastes. An induction melting in a water cooled vessel can solve this problem. However, previous investigation using an induction cold crucible was aimed at applying it to the vitrification of HLLW or volume reduction of cladding hulls (5, 6).

Objectives of the present study are to confirm the process concept of melting, smelting reduction and group separation of simulated HLLW through fundamental experiment using an induction cold crucible.

SUPER HIGH TEMPERATURE METHOD OF HLLW

Concept of the super high temperature method (SHTM) of HLLW is shown in Fig. 1. together with the conventional one.

Fig. 1.

In the current treatment, nitric acid solution which are produced during the solvent extraction of uranium and plutonium in the reprocessing of spent fuels by means of Purex method are first calcined. Then, the solution is evaporated, concentrated and oxidized, followed by the decomposition of nitrates. Usually, it is dissolved into melted borosilicate glass materials and solidified in the canister vessel. Vitrified solid containing major fission products are finally disposed into deep geology.

On the other hand, SHTM branches off the conventional vitrification method after calcination without adding any glass materials. Two kinds of scenarios, A and B, can be considered as shown in Fig. 1. They are connected in branch, so this method completes in each scenario in accordance with the development of associated technology. SHTM consists of the following concepts, regardless of the scenario:

1. Immobilization of fission product elements in HLLW by heating at high temperature without adding any glass materials.

2. Partition of easy to sublime elements, especially the highly exothermic element caesium, by heating calcined HLLW.
3. Reduction of platinum group elements to the metals followed by their separation from the oxide residue by heating calcined HLLW in an inert gas atmosphere.
4. Solidification of the residue to high volume reduced materials.

In scenario A, calcined nitric acid solute is first sublimated at approximately 1000C. During the sublimation, metals with relatively low boiling temperature and high exothermic elements, such as caesium and rubidium are evaporated. Then the residuals are heated approximately up to 1600C after the addition of the reducing agent. It is essential that the reduction takes place in a furnace accompanied with a minimum amount of secondary waste generation. After chemical reaction, elements including the platinum group are reduced to metal. Alloys containing the platinum group elements are separated from residual oxide by certain method, such as electromagnetic separation After this, separated alloys and oxides are solidified. Finally, high active waste oxides containing alkaline earth, lanthanide, actinide, yttrium and zirconium are disposed to an absolutely isolated environment from human beings.

In Scenario B, which is connected with scenario A after "Separation A", oxides containing alkaline earth, lanthanide, actinide, yttrium and zirconium are heated after the addition of a certain reducing agent. After heating, alkaline earth, actinide, and zirconium oxides are reduced together with the evaporation of high exothermic and relatively low boiling element of strontium. After this, high active waste containing actinide and zirconium alloys are separated from lanthanide and yttrium oxides. Finally, actinide and zirconium alloys are disposed.

An object of the present study is confined to "Smelting reduction A" and "Separation A" of scenario A. Smelting reduction and successive separation is realized based on the key technologies such as, induction heating in a cold crucible, control of thermodynamic stability of oxides, effective use of the difference in electromagnetic force working toward metal and oxides.

An induction cold crucible is a furnace, stirrer, electromagnetic confinement equipment of the melt out of contact with the cold wall, and solidifying mold. Usually, the crucible comprises water cooled copper wall which is segmented and electrically insulated by slits, induction coil, withdrawal machine, atmospheric chamber, etc. An induction cold crucible has long been used for casting of chemically active metals with high melting temperature, such as titanium alloys (7). This technique would be capable of applying to the melting of radio-active materials in the industrial scale in the course of recent development of surrounding technology.

There are so many parameters concerning the cold crucible process, such as the structure of the crucible, arrangement of the coil, dome height, output of the generator, frequency of the alternating current, withdrawal velocity, species or dimension of the raw material to be melted, etc. Combination of appropriate parameters results in stable melting and excellent quality of the solidified ingot (8).

Relation between atomic number and standard free energy of oxidation of fission products and corrosion products at 2000K are shown in Fig. 2. In general, actinides have lower free energy of oxidation than platinum group metals and higher than the majority of lanthanide and yttrium. This difference in thermodynamic stability allows actinide to be separated from most of the other fission products.

Fig. 2.

A reduction agent is chosen taking the following condition into account.
1. Reducing ability is necessary and sufficient for the partitioning of platinum group metals from other residual oxides.
2. Reduction takes place by adding as small an amount of reducing agent in mass and volume as possible.
3. Melted and solidified oxide after reduction has absolutely small leach rate
4. Reducing agent is cheap and poisonless.
5. Reducing agent is not the isotope of the fission products.

EXPERIMENTAL METHOD

Oxides of the simulated liquid waste were synthesized from elements containing larger than 1g/Mg-U amounts through calculation by the ORIGEN code as shown in Table I. Corrosion products such as iron, chromium and nickel were added. Those elements

with an atomic number of 35, 36, 41, 53, 54, 61 were not added mainly because of the difficulty in obtaining. A 5 mass % boron compound was added to decrease the melting point of calcined HLLW.
Silicone nitride is one of the promising reducing agents satisfying the requirements described in the previous section, since solidified glass is used as the stable material in the current strategy.
An induction cold crucible is 45mm in inner diameter with 12 number of slits, width and length of which are 0.2 and 150mm, respectively. Graphite wall of 3mm thickness was inserted into the cold crucible to promote heating low conductive HLLW. Insertion of graphite is not essential to this process, because a high frequency generator on the order of GHz enables the induction heating of low conductive HLLW without insertion of graphite. A multi turn induction coil surrounds a water cooled crucible made of copper. After carbon steel, which was the primary and was used as the pulling rod, was inserted from the bottom into the crucible where the electromagnetic field was most effective, atmosphere in a chamber was replaced with Ar gas.
The crucible, coil, high frequency generator, etc., are provided with cooling water, then power is gradually supplied. With an increase in supplied power, the primary melts to rise in a dome shape. A melt is electromagnetically stirred and the top of the melt is out of contact with the wall. Simulated HLLW, which was agglomerated in advance to prevent from spreading over a chamber by free convection, was supplied on the melt dome by a vibratory feeder.
Major experimental condition in continuous treatment is shown in Table II. This was determined by the scientific approach using a precise numerical model (9). Batch treatment is almost the same as that of a continuous one except that the withdrawal equipment is not installed and larger crucible of 70mm diameter is used.
RESULT
In a batch treatment, simulated HLLW materials were initially melted on the crucible bottom and steadily on the viscous melt surface agitated slowly. After cutting the solidified materials longitudinally on the plane including the symmetry axis, they were observed being separated into two lumps, metal and oxide, when the temperature is 1600C. Generally, metal was observed on the bottom and oxide was above. The separation is not solely responsible for the gravity. Cohesion due to agitation of the melt or interfacial tension between metal and oxide, followed by macro separation by the electromagnetic force, would also be responsible. Up to 250g simulated HLLW was processed in batch type treatment.
When the treatment temperature was lower, reduction and melting were insufficient. Metal and oxide were mixed in the form of small particles and the macro separation was not established.
In a continuous and high temperature treatment, the material was melted first by the heat of the primary. Successive feed of materials results in the macro separation of metal on the center and oxide near the wall. With an increase in fed material, macro separation was not clearly observed from the top opening and the melt was stirred accompanied with the foaming probably due to the generation of nitrogen gas during the reaction.

$$6MxO + Si3N4 = 6Mx + 3SiO2 + 2N2 \quad \text{Eq. (1)}$$

However, after cutting the solidified material, macro separation of metal by the electromagnetic levitation provided to be established below the foaming.
In a steady state, raw materials were melted by the heat conduction from inductively heated metal and graphite wall. Reaction continued successfully and 150g simulated HLLW was treated.
Wearing of graphite was small in the experiment. However, it will become a problem when the equipment is enlarged to an industrial scale and works long run. This would be overcome by using high frequency generator on the order of GHz.
One of the longitudinal sections of the solidified material obtained from the continuous treatment is shown in Fig. 3. A few blocks of metal were obtained in a dark green glass material. Two small metals in the oxide seemed to be produced as a resultant of falling of droplet, which cannot be sustained as a large amount of metal in the oxide melt by the electromagnetic force, on the solidification front of oxide. Densities of oxide and metal are measured as 4.1x103 and 8.4x103kg/m3, respectively. From the mass conservation during the chemical reaction, volume ratio between oxide and metal are estimated as 4.5:1.0. Meanwhile, the experimentally

obtained ratio is between 3.5:1.0 and 5.0:1.0. Experimentally obtained ratio was approximately in agreement with that of calculation.
Fig. 3.
Mass fraction of simulated elements into oxide and metal phase obtained by chemical analysis using inductively coupled argon plasma spectroscopy, together with that theoretically obtained on the assumption of stoichiometric reaction, are shown in Fig. 4 as a function of atomic number. Generally, experimental results agree with that of theory. Some discrepancy would be responsible for the shortage of the reducing agent on account of free convection during feed. From thermodynamic calculation, oxide of Te is expected to be reduced to metal. However, Te was not observed neither in metal nor in oxide. Boiling temperature of metal is 990C, so the element seemed to be evaporated during reaction.
Fig. 4.
Alkaline earth, zirconium and lanthanide elements were contained in volume reduced oxide. On the other hand, platinum group elements including corrosion products were successfully partitioned into metal. From this result, it is expected that the long lived fission products of actinide having similar thermochemical properties with lanthanide could also be separated from the metal including the platinum group.
CONCLUSION
Concept of SHTM, which is an advanced waste management of HLLW, was proposed. SHTM is comparable with the current vitrification process and enjoying the partitioning of elements without adding any glass materials. In the process, an induction cold crucible was used, resulting in the minimum amount of secondary wastes.
Validity of the process concept was confirmed by the fundamental experiment. Solidified and group partitioned materials of 150-250g were obtained both by continuous and batch type treatment.
A few blocks of metal were obtained in a dark green glass material. Alkaline earth, zirconium and lanthanide elements were in volume reduced oxide. Platinum group elements including corrosion products were successfully partitioned into metal.
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STATUS OF IDAHO CHEMICAL PROCESSING PLANT (ICPP) WASTE MANAGEMENT TECHNOLOGY DEVELOPMENT PROGRAM*
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ABSTRACT
A program has been in place at the Idaho Chemical Processing Plant for the past two years to identify and define technologies for treating and disposing of sodium bearing liquid radioactive waste, high-activity radioactive calcine, and irradiated spent nuclear fuel (SNF) stored at the Idaho National Engineering Laboratory (INEL). The sodium bearing waste (SBW) and calcine have resulted from past ICPP reprocessing operations and cleanup conducted since 1953. The irradiated spent fuel consists of various fuel compositions and ranges from complete fuel elements to fuel pieces for which no reprocessing flowsheet has been identified. The development program has resulted in a very successful systems analysis approach affecting final disposal of the waste and spent fuel. Use and integration of all applicable waste treatment and disposing technologies currently available within the DOE complex, at universities, and in private industry are described.
INTRODUCTION
Between 1953 and April, 1992, when DOE announced curtailment of reprocessing, irradiated nuclear fuel had been reprocessed at the ICPP. As a result of the decision to curtail reprocessing in 1992, the ICPP put in place a Technology Development Plan to identify and implement acceptable technologies for disposing of sodium-bearing liquid radioactive waste, radioactive calcine, and irradiated spent fuel stored at the INEL.
A unique and practical systems analysis program was developed to allow immediate assessment of technical treatment and disposal options by calculating high and low level waste volumes, near-term and life cycle costs of processing options, and required time lines for a given scenario. Extensive repository performance assessment and preliminary waste acceptance criteria were developed. Hundreds of waste and disposal treatment options were evaluated; available technologies developed throughout the DOE complex, at universities, and with private industry through partnerships, cost sharing, and contracts were integrated to create the options. Separations technologies, HLW vitrification technologies, and LLW grout technologies were included in the development. A strong program for stakeholder involvement was developed. The long-term spent fuel treatment plan was largely superseded by high priority fuel transfers to new underwater storage basins and expanded evaluations of strategies for handling all DOE owned spent fuel.
PROGRAM OBJECTIVES
The nation's radioactive waste policy has been established by the Nuclear Waste Policy Act (NWPA), which requires the final disposal of SNF and radioactive waste in accordance with U.S. Environmental Protection Agency (EPA) and Nuclear Regulatory Commission (NRC) standards. In accordance with these regulations and other legal agreements between the State of Idaho and the DOE, the DOE must, among other requirements, 1) complete a final Environmental Impact Statement by April 30, 1995, 2) evaluate and test sodium-bearing waste pre-treatment technologies, 3) select the sodium-bearing and calcine waste pre-treatment technology, if necessary, by June 1, 1995, and 4) select a technology for converting calcined waste into an appropriate disposal form by June 1, 1995.
The primary objective of the Waste Management Technology Development Program (WMTDP) is to develop and demonstrate safe, cost-effective, and environmentally responsible methods for the conditioning, interim storage, qualification, and final disposition of SNF and radioactive wastes. The program will, ultimately, recommend and implement technologies and processes to facilitate the conditioning and certification of SNF and radioactive waste for permanent disposal. The program will also provide technical support to resolve current SNF and radioactive waste interim storage issues, such as SNF characterization and removal of sodium bearing waste from the existing tank farm. Success will be measured in terms of safety, life-cycle cost, regulatory compliance, and waste volume. Emphasis will be placed on process robustness to ensure that processes will apply not only to INEL SNF and radioactive waste, but will also foster transfer of technologies to address the conditioning of a broad range of SNF and radioactive wastes throughout the DOE complex.
Assumptions made in developing the WMTDP include: 1) Treatment and immobilization processes must accommodate radioactive liquid and calcine wastes, 2) Existing facilities will be utilized to the extent practical, 3) Actinides and fission products will be collected into a HLW stream for disposition approaches involving

constituent separation, 4) Low-activity waste will meet NRC Class C, or lower, limits, and 5) Funding will be available to support signed agreements, consent orders, and Federal Facility Compliance Agreement mandates.

SYSTEMS ANALYSIS

DOE Order 4700.1 outlines a logical systems engineering approach to problem evaluation and resolution. This process consists of six principle steps (3):

Define the problem and establish the program goal
Determine the functional (baseline) requirements
Identify alternatives that meet the functional requirements
Develop and evaluate the alternatives
Recommend preferred alternatives for selections and implementation

The WMTDP is taking a systems engineering approach to the development of technologies for the treatment and disposition of radioactive waste. The system engineering process considers all aspects of systems requirements from the earliest stages of design through development, testing and operation. The process supports project management by ensuring that technical control is on a level and integrated with funds, cost, schedule, and performance controls. The fundamental philosophy is the systematic narrowing of a large number of candidate process options through identification and evaluation, laboratory and component-scale testing, non-radioactive and radioactive pilot-scale testing, and implementation.

The evaluation employs systems analysis tools and techniques and considers process development technical data and test results. Ultimately, process options that best meet the stated criteria will be selected by decision makers for development into an operating facility to condition and prepare SNF and radioactive waste for placement in a final storage facility. In selecting processes for further evaluation and testing, candidate options are evaluated against the following criteria: 1) Safety of process to workers, the public, and the environment; 2) Life-cycle cost (i.e., development, construction, operations, decontamination and decommissioning , disposal, etc.); 3) High-activity and low-activity waste volumes generated; 4) Final waste forms; 5) Flexibility of the process to handle multiple input streams; and 6) Compliance with all applicable regulations.

Through mid 1993, using program goals and baseline requirements as guides, extensive investigations were conducted to identify potential technologies that could be employed at the ICPP to treat and dispose of radioactive liquid and calcine wastes. Identified technologies were then sub-divided into pretreatment , separations, and immobilization categories and combined into process systems (flowsheets) capable of handling ICPP radioactive liquid and calcine waste. Additionally, an analysis of alternatives for removing sodium waste from the existing tank storage volume was performed and recommendations were made to DOE concerning processes for treating sodium waste inventories without having to construct additional storage tanks. Processes that did not meet the baseline requirements, could not be developed in time to meet regulatory requirements, or were viewed as technically inferior to other technologies were eliminated from further consideration. Processes that still showed promise following initial development were advanced to the next stage of analysis. These technologies have resulted in 27 viable waste treatment process combinations.

Mass balance calculations resulting from laboratory testing and literature data were made with radioactive waste compositions of current and projected inventories being used as inputs. Mass balances were then used to evaluate candidate technology flowsheet performance relative to ICPP radioactive waste streams and to develop estimates of cost, schedule, throughputs, and radioactive waste volumes.

This approach to analyzing the processing , storage, and disposal of radioactive liquid and calcine waste at the ICPP was implemented to facilitate informed decision making and understanding in the resolution of stakeholder issues. This approach evaluates candidate processes and disposal forms using a consistent set of assumptions and data. Additionally, the potential impacts of existing uncertainties are being evaluated, resulting in the recommendation of programs to investigate and resolve those uncertainties.

SEPARATION TECHNOLOGIES

There are many treatment and disposal options which can be potentially applied to the ICPP radioactive sodium-bearing liquid and solid HLW calcine wastes. However, those which include separation processes to split the high activity, low volume components (actinides and fission products) from the inert, high volume, low

activity fraction are greatly desirable in order to reduce the volume and, therefore, cost of the final, high activity, repository-destined waste. The Modified Consent Order between the State of Idaho and the DOE required that the ICPP contractor LITCO select and investigate three base separation technologies possibly applicable to both SBW and calcine. This investigation must occur prior to the required 1995 EIS Record of Decision to choose preferred treatment and disposal options for both waste streams. However, due to the nature of the waste streams, these base technologies must be complemented with auxiliary upstream and downstream technologies such as mercury and cesium removal and/or calcine dissolution. These base technologies are necessary if: 1) secondary waste is to be minimized, 2) desired fission product decontamination factors are to be achieved, and 3) desired low activity waste classifications (e.g., NRC A, B, or C) are to be obtained. The following sodium-bearing waste and calcine base separation and auxiliary technologies (as tabulated in Table I) were tested with surrogate and/or actual wastes at laboratory, bench and small pilot-plant scale levels.

TABLE I

The majority of the testing was accomplished at ICPP facilities, but for cases where the technology was immature or not part of the previous ICPP experience (i.e. freeze crystallization), the development work was subcontracted to private firms. Detailed test results of those technologies listed in Table I can be found in Ref. 1, 2, 3, and 4. Brief status descriptions of key base and auxiliary technologies still under ICPP investigation are as follows.

Freeze Crystallization (FC) is applicable only to sodium-bearing waste. Freeze Crystallization is a developing technology which concentrates liquid waste by removing heat and thereby freezing one or more components of the liquid to a solid phase. In the case of sodium-bearing waste, this solid phase is water in its pure form (ice). When the ice forms, the mother liquid becomes supersaturated in sodium nitrate which is then precipitated and separated from the remaining contaminated SBW. Since the presence of sodium causes caking and agglomeration difficulties in calcination, the depleted sodium liquid becomes amenable to calcination with less cold additives. Non-radioactive testing has shown the process to be technically plausible, but equipment difficulties pose likely challenges which may be insurmountable for full-scale operation.

The freeze crystallization concept is also the principle in the evaporation/precipitation separation technology. However, the difference is in the method by which the water is removed from the waste. By heating the sodium-bearing waste water is evaporated and sodium nitrate is precipitated upon reaching saturation. Like freeze crystallization the evaporation/precipitation concept is not applicable to calcine.

The simple precipitation/neutralization process, applicable to both SBW and dissolved calcine, involves increasing the pH of the acidic aqueous wastes to levels that promote heavy metal and polycation precipitation. Testing shows that when pH levels are increased between 8 and 10, TRU components, toxic metals and a large fraction of the strontium also precipitates to the sludge, leaving a low activity supernate highly concentrated in sodium. Additional fission product (Cs, Sr) removal from the supernate would be necessary, sludge quantities would be large, and chemical conditioners, flocculants and coagulators would be required to improve liquid-solid separations.

After precipitation, the low activity, high sodium, supernate fraction could then be treated further via electrohydrolysis (EH), an auxiliary technology employing both electrohydrolysis and membrane separation concepts. EH utilizes cation and anion selective membranes and an electromotive force to split waste into hydrogen and hydroxide ions and is applicable to both the treated low activity fractions of sodium-bearing waste and/or dissolved calcine. As a result, the waste is substantially diluted, and relatively pure sodium hydroxide and nitric acid are recovered for recycle and reuse in future ICPP decontaminations. Development efforts continue to focus on feed chemistry to eliminate the presence of ions that cause frequent membrane fouling.

The base technology of radionuclide partitioning via the TRUEX process developed by Argonne National Laboratory is recommended for both sodium-bearing waste and dissolved calcine. This technology consists of continuous counter-current solvent extraction performed in centrifugal contractors and a solvent (organic phase) of 0.2 M CMPO and 1.4 M TBP in a hydrocarbon diluent. The TRUEX process has shown excellent

decontamination factors on actual SBW and dissolved calcine wastes, and is currently the recommended base treatment option for the EIS Record of Decision. A pilot plant is currently available for studying the process parameters. International collaborations are also in progress for developing a radionuclide partioning concept utilizing phosphine oxide as an extracting agent.

As with the precipitation and freeze crystallization technologies, TRUEX will also require auxiliary technologies to remove high concentration fission products such as 90SR and 137Cs. For 90Sr, the SREX process is being developed and studied extensively at both laboratory and pilot levels. Like TRUEX , SREX is a counter-current solvent extraction method utilizing a crown either extractant. Cs137 removal is accomplished by ion exchange technology. Testing to date indicates that the most promising is ammonium molybdophosphate (AMP). One engineered form of this material, suitable for radiological use, is on a modified polyacylonitrile (PAN) resin. However organic supports are known to be less thermally and radioactively stable than inorganic supports.

Despite these development challenges, the TRUEX, SREX, Cs ion exchange series of separation technologies yield the most cost effective, technically proven, and flexible train of unit operations for sodium-bearing waste and calcine treatment. These operations, pending stakeholder approval, will be the EIS Record of Decision recommendation for ICPP waste separations.

HLW VITRIFICATION TECHNOLOGIES

Throughout the United States and the rest of the world, the accepted vitrified HLW form is borosilicate glass. The French began developing this technology in the 1950's and the British and Japanese have patterned their vitrification processes after the French. West Valley, Savannah River, and Hanford are each in various stages of implementing glass technology to vitrify their waste. The INEL, historically, has pursued a different technology. Essentially all of the INEL HLW is stored at the ICPP in the form of calcine. Because of the composition of the calcine (high in cladding materials), a significant reduction in the immobilized calcine volume could be achieved by immobilizing it as a glass ceramic rather than as a glass. The volume decrease results from a combination of higher waste loading and higher density for the glass ceramic form. During the last 15 years, a small HLW Immobilization group has developed very durable glass ceramic waste compositions for all types of ICPP calcines (alumina, zirconia, fluorinel, zirconia-sodium, and fluorinel-sodium).

Although the glass-ceramic form produces a minimum volume for the ICPP calcine, the structured systems analysis described above showed that life-cycle costs for treating ICPP HLW could be minimized if the calcine were redissolved and separated into high- and low-activity fractions and these fractions immobilized into high-level glass and low-level grout forms. The final decision on technology implementation will occur as part of the INEL EIS Record of Decision to be issued in June 1995. However, since the glass form now seems to be a likely candidate for immobilization of the ICPP HLW, the development effort is now focussing on this waste form. The major waste form development objectives for this activity include:

Develop durable glass waste forms for ICPP wastes using laboratory-scale equipment and waste simulants.

Demonstrate vitrification flowsheets on pilot plant scale using waste simulants.

Identify process control requirements for waste qualification.

Demonstrate waste immobilization with radioactive wastes.

In parallel with the above items, establish programs for quality assurance, records management, and waste form qualification.

Demonstrate process equipment at near full scale.

Provide recommendations and process data for a waste treatment facility design.

The most comprehensive suite of glass waste form fabrication experiments to date has been performed to demonstrate direct vitrification of zirconia calcine. The approach used in conducting this experimental program illustrates the general approach used by the ICPP HLW Immobilization Program to develop waste forms.

The experimental program included preparation of laboratory-scale waste glass samples from simulated (non-radioactive) zirconia calcine, fabrication of glass waste forms using a pilot-scale melter, and preparation of laboratory-scale waste glass samples from actual (radioactive) zirconia calcine. The laboratory experiments were used to develop and test waste formulations. This work is performed most efficiently and economically at a small scale. However, there is technical risk

inherent in scaling a process up from bench-scale to production. Pilot-scale tests were used to gather data for process scaleup, thereby reducing the technical risk. Hot tests were used to confirm the results obtained from cold testing, thus further reducing technical risk.
Performance of the experimental waste glasses was compared using standardized static leach tests (MCC-1). The results of these tests showed the leach rates from the glass samples were consistently less than 1 gram/m2 day. The similarity of results obtained from waste glasses prepared in different ways using simulated and actual calcine validates the approach being pursued.
Significant work remains to be done at the ICPP to meet these objectives. To date the following have been achieved:
Specified waste streams for immobilization from proposed processes
Developed predictive programs for glass waste formulations
Prepared glasses from surrogate wastes for proposed processes
Measured leach resistance of surrogate waste glasses
Prepared glass from radioactive zirconia calcine (archived for durability studies)
Procured and modified equipment for additional hot tests
Prepared a performance assessment for disposal of INEL wastes in a geologic repository
Completed feasibility studies for the HLW glass plant process
Significant milestones planned for the remainder of FY-1995 are:
Issue the first draft of the Waste Compliance Plan
Prepare and test simulated waste glasses from feed streams currently being identified by separations process development.
Complete installation of equipment for hot tests
Determine equipment needs for pilot-scale demonstration

LLW GROUT TECHNOLOGIES

Following separations, the low activity fraction, like the high activity portion, will require stabilization prior to interim or final disposal. ICPP investigators have employed laboratory scale testing and system analysis techniques over the past year and, as a result, have chosen to concentrate stabilization efforts on grouting techniques (as opposed to glass). These techniques will ensure a low cost, high waste loading waste form that can be readily transported from interim to final storage. Furthermore, analyses have indicated that Portland cement, due to its availability and performance record, is the preferred grout over other encapsulating thermosetting and thermoplastic materials. However, small development efforts are ongoing with both sulfur-polymer and polyethylene cements.
Non-radio-active laboratory testing at ICPP with cement grout has concentrated on optimizing grout waste formulations in terms of performance (leach resistance and compressive strength) and high waste loadings. This optimization is being accomplished by varying parameters such as the water-to-cement ratio, the addition of admixtures for rheology and leaching control (plasticizers, slag-fly, clays, etc.), the curing time, and the waste loading. Early results indicate that the water-to-waste ratio is probably the most significant single parameter affecting chemical and strength resistance of hardened cement mix. Current test data show that a 10% waste loading (by weight under a dry bases) passes initial strength tests and waste loadings as high as 70% may be achieved depending on the waste source and pretreatment methods. Contracted studies have also been conducted at Pennsylvania State University to determine waste loadings possible for ICPP surrogate waste originating from the radionuclide separation of both calcine and sodium-bearing. Prior to grouting, preconditioning through either chemical or thermal methods may also improve waste loadings and grout performance. Because highly soluble nitrate salts in the waste will easily leach, denitration and the addition of mineral like compounds may be required upstream of the grout process. Bench scale thermal tests methods such as high-temperature calcination (via a rotary kiln dryer or evaporation on a production scale) will reduce and remove all the nitrates in the waste to an oxide form. This same testing has also indicated that the addition of clay or silica during thermal denitration may help form insoluble, leach resistant sodium-alumina silicates within the formulation, thereby greatly enhancing its performance.
Extensive development of the low-level grout process will continue through the use of a comprehensive low-level grout pilot plant which is currently in the design phase. The pilot plant will accommodate three methods of mixing waste with cement, since mixing is considered one of the key equipment challenges facing the grout

process. High quality mixing is available via batch mixing, but large volumes of rinse solution would be generated. In-line mixing would provide high throughput with high quality, but process control would be difficult. In-container mixing, on the other hand, provides a low cost simplified alternative, but the drums may not be completely filled. Completion of the initial pilot-plant test phase will determine both the optimum mixing method as well as define the thermal denitration equipment for preconditioning.

Development efforts related to the polyethylene encapsulation of low activity waste are currently under review to ensure a suitable alternative to the preferred Portland cement grout process. Formed through the polymerization of ethylene gas, low density polyethylene provides a matrix with a relatively open structure. This matrix can be processed at relatively low temperature and pressures to produce final formulations of 70 wt% waste loadings that meet all performance criteria. However, continued development is required to ensure that the matrix can handle the higher radioactivity and heavy metal concentrations existing with ICPP wastes.

Like polyethylene, sulfur polymer cement (SPC) is under investigation as an alternative to base Portland cement grouts. As sulphur polymer cement concrete has gained popularity as a construction material, it became evident that it would be capable of immobilizing low activity wastes. Produced by combining elemental sulfur with dicyclopentadiene, SPC behaves similarly to polyethylene, except SPC will react with certain chemical species in the waste. Advantages of SPC include possible high waste loadings (50%), relatively low processing temperature, recovery from out-of-specification batches via remelting, and resistant to acids and corrosives. Disadvantages include the existence of adverse chemical reactions, incompatibly with bases and oxidizers, and potentially poor thermal and radiation stability.

INTEGRATED FACILITY

The integrated facility which incorporates the above technologies has been named the Waste Immobilization Facility (WIF). The WIF will house a combination of technologies for processing liquid sodium-bearing waste, other decon and process solutions, and calcine. The technologies include calcine dissolution, transuranic extraction (TRUEX), strontium extraction (SREX), cesium ion-exchange (CslX), low and high activity waste immobilization, and interim storage for the immobilized waste forms. Amalgamation of recovered mercury will also be included. A vitrification plant will be built for immobilizing high-activity waste and a grout plant for immobilizing low-activity waste. This process is designed to minimize the volume of the high-activity waste stream and the cost of producing the low-activity waste stream. When the WIF is operational, it will have surge storage tanks to collect and characterize any liquid waste generated at the ICPP.

WIF buildings are arranged in a complex of six structures co-located on a common site.

- Separations Building
- HLW Vitrification Building
- LLW Grouting Building
- Vitrified HLW Interim Storage Building
- Grouted LLW Interim Storage Building
- Bulk Chemical Storage Building

The process and storage buildings are designed for a 40-year life. The facilities provide for remote operations and maintenance of the process with appropriate radiological and biological shielding for operator safety. Heating, ventilation, and air conditioning (HVAC) systems provide additional confinement barriers to limit spread of radioactive contaminants. The storage facilities are sized for a seven-year production period. The interim storage facilities' layout allows storage capacity to be increased to accommodate an additional fourteen-year production of vitrified HLW canisters and grouted LLW drums.

WIF process systems are designed to handle a range of waste streams including liquid SBW and HLW calcine materials currently held in ICPP inventory and similar radioactive waste materials anticipated to be generated from future ICPP activities, and in particular from D&D operations. Process flow streams within WIF vary as the feed streams change; therefore, unit operations within the facility are designed to handle the maximum flow rate of liquids and/or solids that would occur in the plant if it were processing each of the feed streams on a dedicated campaign basis.

The design throughput of the facility is based on being able to treat the calcine inventory in 30 years. This requires the facility to be sized for 185 gallons per

hour liquid throughput. The actual facility throughput will vary based on the feed material being processed. For example, processing SBW will produce approximately 1/10 canister of HLW glass and 44 drums of LLW grout per day. Processing 100% zirconium calcine feed will produce approximately 1/2 canister of glass and 14 drums of grout per day.

WIF is divided into eleven functional areas:

1. Calcine transport, receiving, dissolution, and solids separation
2. Liquid waste receiving, storage, solids separation, and dissolution
3. TRUEX, SREX, ion-exchange, and organic mixed waste thermal oxidation
4. HLW evaporation, pretreatment, condensate treatment, and solids separation
5. HLW calcining, vitrification, and offgas treatment
6. HLW canister filling, handling, decon, transfer, and storage
7. LLW denitration, pretreatment, and grout mixing
8. LLW drum filling, handling, transfer, and storage
9. Mercury collection, washing, amalgamation, packaging, and storage
10. WIF PEW collection, storage, evaporation, and acid recycle
11. Hot and cold process offgas treatment; service and hazardous waste collection

Ultimately, WIF will contain all of these functional areas; however, to minimize cost impacts to the DOE budget while still meeting the waste treatment schedules required by the State of Idaho, the facility will be constructed in two phases. Phase 1 will consist of functional areas 2, 3, 7, 8, 9, 10, and 11. This will enable the facility to process SBW into a liquid HLW and a grouted LLW. Interim storage for both fractions will be provided. Current plans are for this phase to begin operation in approximately 2009 which will result in the ICPP Tank Farm being emptied of SBW by the Consent Order date of 2015. Phase 2 will add the remaining functional areas to enable the facility to treat calcine, vitrify the HLW, and store the resulting glass waste form. This phase will begin operation in approximately 2014 and operate for approximately 30 years to treat the inventory of calcine at the ICPP.

STAKEHOLDER INVOLVEMENT

One of the six major goals of the DOE Environmental Management Program is to develop a strong partnership between the department and its stakeholders. As stated by Tom Grumbly, "Stakeholder involvement was one of the key activities that lead recently to the successful conclusion of negotiations of changes to the Hanford Federal Facility Agreement and Consent order..." The public involvement programs at other sites were benchmarked, resulting in defining some lessons learned such as providing stakeholders with clear information, involving them as early as possible, and keeping the working meetings small and informal. A public participation plan has been developed based on these principles to involve stakeholders in the evaluation of technologies for immobilizing calcine and liquid sodium-bearing waste. As part of the plan, workshops are held with various interested stakeholder groups to discuss the technology development options, potential stakeholder concerns, and establish the relative importance to each outside interest group of the concerns. The resulting determination of stakeholder values with regard to future processing options will be incorporated in the final recommendation to the State of Idaho of a preferred technology as required by the December 22, 1993 Court Order of the DOE/Navy/State of Idaho agreement.

CONCLUSIONS

1. After 2 years, the ICPP Technology Development Program has been highly successful in identifying cost effective, environmentally acceptable, and safe options for treatment and disposal of ICPP sodium-bearing liquid and calcined solids radioactive wastes. Due to deteriorating spent fuel, the emphasis on spent fuel management has largely shifted to near-term storage improvements rather than long-term treatment and disposal actions.
2. A successful systems analysis program has been developed to allow immediate evaluation of near-term and life cycle costs, waste volumes, and related information.
3. Separations has been identified as cost effective and required technologies have been demonstrated using hot laboratory and cold pilot plant techniques for both sodium-bearing and calcine radioactive wastes.
4. Immobilization methods for HLW by glass formation and low-level waste by improved grout have been identified as leading options for processing ICPP radioactive wastes.
5. Required facilities necessary to implement waste treatment and preparation for

disposal in the required time frame have been identified.
6. Maximum use has been made of available technology through DOE Complex sites, private industry, and technical universities.

RECOMMENDATIONS

1. Continue to apply the elements of the ICPP technology Development Program to the ICPP and expand applicable program elements to the INEL and other DOE sites.
2. Continue to work with other DOE sites, universities and private industry to utilize applicable, available technologies and transfer technologies as appropriate.

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SEPARATION AND IMMOBILIZATION OF RADIONUCLIDES FROM IDAHO CHEMICAL PROCESSING PLANT HIGH-ACTIVITY WASTES

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ABSTRACT

Partitioning of radionuclides from Idaho Chemical Processing Plant wastes has been demonstrated using aqueous separations processes. The two major waste streams consist of an acidic liquid waste high in sodium content (1.3 M) and a solid calcine which is readily dissolved (greater than 98%) in 5M nitric acid to produce an acidic aqueous stream. Actinides are removed from the aqueous feed streams by a solvent extraction process utilizing organophosphate compounds such as CMPO, DHDECMP or phosphine oxide derivatives. The raffinate from this process is then treated to remove strontium by another solvent extraction process utilizing selective crown ethers or treated to remove strontium and cesium using chlorinated cobalt dicarbollide. The raffinate from the crown ether strontium removal process is subjected to cesium removal by use of inorganic sorbents or solvent extraction using crown ethers. All of the above processes have been demonstrated, on the laboratory-scale, using actual high-activity waste streams. Based on the results of tests with simulated and actual wastes, the acidic waste streams have been decontaminated to levels below NRC Class A Low Level Waste. The concentrated high-activity radionuclide product streams would be immobilized as a glass waste form. The effect of inert materials from the separations processes, primarily Zr, P and Mo as well as radionuclides on glass waste loadings has been evaluated.

INTRODUCTION

The Idaho Chemical Processing Plant (ICPP), which is located on the Idaho National Engineering Laboratory (INEL), has reprocessed irradiated nuclear fuel since 1953 to recover uranium-235 and krypton-85 for the U.S. Department of Energy. The resulting acidic liquid radioactive waste was solidified to a high-level waste (HLW) calcine, and stored in stainless-steel bins enclosed in concrete vaults. About 3800 m3 of radioactive HLW calcine is currently stored at the ICPP. In addition to the HLW calcine, approximately 6.9 million liters of high-activity acidic-liquid waste are stored at the ICPP. This liquid is a result of decontamination activities, evaporator bottoms and solvent wash activities and cannot be calcined directly because of its high sodium content. In the past, this waste was blended with extraction cycle raffinates and calcined. Since October 1993, all extraction

raffinates available for blending with the sodium-bearing waste (SBW) have been calcined.

Reprocessing activities at the ICPP were halted in April 1992, and since that time an aggressive development effort has been underway to evaluate technologies for processing and immobilizing the liquid and calcine radioactive wastes. A rigorous systems analysis approach was used to develop a model for evaluating and comparing waste treatment technologies. This model was used to evaluate several technologies and combinations of technologies (1). The key criteria used to evaluate technologies included: life-cycle cost (with and without HLW disposal costs), five-year cost, cost-time profile, HLW volume, total waste volume, and capability of meeting regulatory requirements. Aqueous separation processes were selected as having the highest probability for meeting regulatory requirements for processing SBW and also resulted in the lowest life-cycle costs for all treatment options considered.

The HLW calcine is composed primarily of metal oxides from inert materials such as Zr and Al (from fuel claddings), Ca (added for fluoride corrosion control in the calciner), and B and Cd (used as soluble nuclear poisons in the dissolution process). Radioactive materials including the TRU elements and fission products (primarily Cs and Sr) comprise less than 1 wt% of the calcine. Aluminum and zirconium based calcines represent the greatest inventory of the calcine at approximately 20% and 80% respectively. A summary of calcine compositions is given in Table I.

TABLE I

The average composition of SBW is given in Table II. The liquid SBW has an average nitric acid concentration of 1.4M, a nitrate concentration of 4.5 M and a sodium concentration of 1.3M. Radionuclides in the SBW that will require treatment and/or immobilization are U, Np, Pu, Am, Cs, Sr, and possibly Tc. Other hazardous constituents in the SBW that may require treatment are mercury, lead, chromium and cadmium.

TABLE II

The SBW is readily amenable to aqueous separation processes, but the calcine must first be dissolved prior to treatment. Dissolution of most calcines can be achieved using 5 M HNO_3 at 90o C in about 30 minutes. The calcine/acid mixture must be continuously mixed to facilitate effective dissolution. The heel of undissolved calcine from a batch dissolution is left in the dissolver and mixed with fresh acid and a new calcine charge. Laboratory and bench-scale data indicate that greater than 98% of the calcine can be dissolved in 10 sequential batch dissolutions. This suggests that a calcine heel will not build in the dissolver and that undissolved calcine, which will be fed directly to the glass melter, will not be a significant factor in glass composition or waste loading.

The flowsheet for separating the actinides, strontium and cesium from the aqueous acidic feed solutions is shown in Fig. 1. This flowsheet includes solvent extraction processes for the removal of actinides and strontium and an ion-exchange process for the removal of cesium. The high-activity waste stream from the separation processes is concentrated and sent to the vitrification process. The decontaminated waste raffinates would meet NRC Class A LLW criteria and be immobilized as grout for near surface disposal. These processes will be discussed in more detail later in this report.

The advantages offered by this flowsheet include: significant reduction of HLW volume, capability to process several waste streams in one facility, and a very consistent feed composition to the HLW vitrification process. The major disadvantage of this flowsheet is the additional capital cost required for the separations process, but analyses indicate this cost may be largely offset by the reduction in the size and throughput of the HLW vitrification facility.

Fig. 1.

Figure 1. Aqueous Separations Flowsheet for ICPP Acidic Waste

Actinides (U, Np, Pu, Am) and Tc are removed from the aqueous waste streams by solvent extraction processes utilizing bifunctional neutral organophosphate compounds such as CMPO, DHDECMP or phosphine oxide derivatives. The TRUEX process, which uses a solvent comprised of 0.2M CMPO and 1.4M TBP in a paraffinic hydrocarbon diluent, has been successfully demonstrated on both simulated and actual calcine and SBW (2). In a series of 3 batch contacts with fresh TRUEX solvent in each contact, >99.995 % of the actinides were removed from actual SBW. The actinides were effectively stripped from the solvent by 0.04M HEDPA. Tests in centrifugal

contactors, with simulated SBW containing neodymium as an americium surrogate, have demonstrated >99.93% neodymium extraction in 4 stages. It has also been shown that mercury can be selectively partitioned from the acidic waste and the actinides in the TRUEX process.

Other potential actinide extractants, primarily DHDECMP and phosphine oxide derivatives have been tested with simulated and actual SBW. These solvents offer promising alternatives to the TRUEX process and will be evaluated in detail, including tests in centrifugal contactors, in the near future.

Strontium is selectively removed from the acidic raffinate from the TRUEX process by the SREX (strontium extraction) process. The SREX process utilizes a crown ether extractant of 0.2M di-(tert-butyldicyclohexo) 18-crown-6 in 1-octanol (3,4). Recent test results indicate 1.0M TBP in a hydrocarbon diluent is also an effective solvent for the crown ether. The SREX process has been demonstrated on simulated and actual SBW and dissolved calcine. The SBW poses the greatest challenge for this process as sodium and potassium, present in 4 to 5 orders of magnitude higher concentrations than strontium, will be partially extracted and reduce the amount of strontium extracted. Greater than 99.99% of the strontium was removed in six successive batch contacts with actual SBW and fresh solvent. The same degree of separation from actual dissolved calcine was accomplished with 3 successive contacts. Testing of the SREX process in centrifugal contactors is in progress, but results are not yet available.

Two more important issues relative to the SREX process have been investigated. The first is the purity of the substituted crown ether extractant. Extractant samples from two suppliers in the U.S. and one in Russia were tested. The results of one U.S. produced crown ether and the Russian crown ether were comparable. The other U.S. produced crown ether had significantly lower strontium distribution coefficients than the other two. It is postulated that differences in the isomeric composition of the extractants result in different levels of strontium extraction due to interferences from alkali and alkaline earth elements. The second issue is the availability of substituted 18-crown-6 material in commercial-scale quantities. The U.S. manufacturer of the better crown ether has developed the capability and has demonstrated production in kilogram quantities. These issues are significant for the potential future use of the SREX process on a production scale.

Cesium removal is accomplished by ion exchange using ammonium molybdophosphate (AMP). AMP is effective at removing cesium in nitric acid media, but must be placed on an engineered support for use in a fixed-bed column. AMP on a polyacrylonitrile (PAN) support developed and produced in the Czech Republic (5) has been tested with simulated and actual wastes at the ICPP. An equilibrium capacity for AMP-PAN of 150 g Cs/ kg AMP was experimentally determined. Testing of AMP-PAN in fixed-bed columns is in progress.

The disposition of the AMP-PAN has not been determined. One option is to load the AMP-PAN with cesium and then remove the bed material and send it to the vitrification process. Alternately, it may be feasible to elute the cesium from the AMP-PAN. In this case, the cesium-rich eluent would be added to the actinide and strontium-rich streams prior to concentration/vitrification.

A collaborative program between the Khlopin Radium Institute (KRI) of St. Petersburg, Russia and the INEL has resulted in testing of chlorinated cobalt dicarbollide for the removal of strontium and cesium from ICPP wastes (6). The process was effective at selectively removing the fission products Sr and Cs; however it may be difficult to implement because of safety concerns relative to the nitrobenzene-based solvent. Recent work at KRI may have produced a new solvent that would be acceptable for use in the United States. Testing of this process in centrifugal contactors is planned for 1995.

In order to meet the NRC Class A LLW criteria and to avoid operational problems with the centrifugal contactors and fixed-bed columns, the liquid feed streams must be clarified prior to the separations processes. Data from actual calcine dissolutions suggest that radionuclides are concentrated by about an order of magnitude in the undissolved solids from the dissolution process. Preliminary estimates indicate up to 99.99% solids removal must be achieved to meet the 10 nCi/g limit in the low-level grout. This level of solids removal is assumed to be more than adequate to preclude any equipment operational problems.

The amount of HLW glass produced from the concentrated radionuclide containing streams from the separations processes was estimated for zirconium-based calcines.

Zirconium-based calcines represent about 80% of the calcines at the ICPP and on a mass basis, over 70% of the total waste. Assumptions and process considerations used for this estimate are as follows:

99% Zr calcine is dissolved, undissolved solids report to HLW fraction
66 wt% of the Zr in the feed to TRUEX reports to the HLW fraction
0.04M HEDPA strip, 150 l strip/400 l aqueous feed to TRUEX, 2 moles P/mole HEDPA
60 g Cs/kg AMP-PAN, 80 wt% AMP on AMP-PAN, 1 mole P/mole AMP

Based on these considerations, the composition of the feed stream to the HLW vitrification process can be estimated. The amount of glass produced will be based on limiting concentrations of certain components to maintain solubility in the glass and not produce unacceptable process conditions such as excessive corrosion or glass viscosity. Assumptions for the limiting compositions in glass are shown below:

1 wt% loading of radionuclides in glass. If the feed stream to vitrification was pure oxides of the radionuclides, a 1 wt% loading would produce a centerline glass temperature of 3500C.
7 wt% elemental zirconium loading in glass.
1.5 wt% elemental phosphorous loading in glass.
1 wt% elemental molybdenum loading in glass.

Based on these assumptions, glass volumes were calculated for the limiting case of each element. Zirconium has the greatest impact on glass volume, resulting in about 2300 m3 of glass. Phosphorus contributes to about 1100 m3 of glass, while molybdenum contributes to 300 m3. If 1 wt% of radionuclide oxides were the only material fed to the HLW melter, only 22 m3 of glass would be produced. It is apparent from these calculations that zirconium (partially extracted in the TRUEX process) has the largest effect on glass volume and phosphorus (used as a stripping reagent in the TRUEX process) has the second largest effect.

SUMMARY

Aqueous separation processes proposed for treatment of the two high-activity waste streams at the ICPP have been demonstrated to be feasible and cost effective. The waste streams can be decontaminated to levels below the NRC Class A LLW criteria, allowing for near-surface disposal as grout. Clarification of the waste prior to processing is a very critical aspect of decontaminating the wastes. Waste loadings of key elements from the separations processes indicate that the primary driver for glass volume is zirconium and the second largest driver is phosphorus. These two elements are the focus of development efforts to 1) maximize weight percent loadings in glass and 2) minimize the amount of each element reporting to the HLW vitrification process.

ACRONYMS

AMP	ammonium molybdophosphate
CMPO	octyl (phenyl)-N,N-diisobutylcarbamoylmethylphosphine oxide
DHDECMP	dihexyl-N,N-diethylcarbamoylmethylphosphonate
HEDPA	1-hydroxyethyl-1,1-diphosphonic acid
HLW	high-level waste
ICPP	Idaho Chemical Processing Plant
INEL	Idaho National Engineering Laboratory
KRI	Khlopin Radium Institute
LLW	low-level waste
NRC	Nuclear Regulatory Commission
PAN	polyacrylonitrile
SBW	sodium-bearing waste
SREX	strontium extraction
TBP	tributylphosphate
TRUEX	transuranium extraction

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Session 36 -- Utility and Government Issues in Spent Fuel Storage/(Including Dry Casks)
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36-1

A PROGRAMMATIC OVERVIEW FOR THE DOE-OWNED SPENT NUCLEAR FUEL PROGRAM

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ABSTRACT

This paper provides a programmatic overview of the current status of DOE-owned spent nuclear fuel, resulting principally from the stoppage of weapons production and the reduction in the weapons stockpile. The paper also describes how DOE will manage the effort leading to ultimate disposal, and problems and issues associated with this responsibility.

INTRODUCTION

Since the 1940's, the Department of Energy (DOE) and its predecessor agencies, have operated a complex of defense nuclear facilities for the purpose of manufacturing, testing, stockpiling, and then dismantling nuclear weapons. In recent years, as a result of the changing political situation, the United States has halted production of new weapons, and then agreed to make dramatic reductions in the size of the nuclear weapon stockpile. Because of the stoppage of weapons production and the reduction in the weapons stockpile, DOE must now deal with the nuclear materials legacy of the Cold War.

DOE currently owns and stores about 2700 metric tons of heavy metal (MTHM), most of which is stored in facilities in or near Richland, Washington; Idaho Falls, Idaho; and Aiken, South Carolina. While disposal of such a substantial amount of spent nuclear fuel (SNF) represents a significant problem, it is only a small portion of over 100,000 MTHM processed since the start of the weapons program, and will increase by only about 100 MTHM over the next 40 years.

In addition to the weapons program SNF, DOE has provided nuclear fuel to a number of research reactors, both in the United States and abroad, requiring only that the SNF from these reactors be returned to DOE for ultimate disposition.

PROGRAM OVERVIEW

This paper provides a programmatic overview of the current status of DOE-owned SNF, how DOE will manage the effort leading to ultimate disposal, and problems and issues associated with this responsibility. Because the paper is general in approach, specific details concerning issues and site-specific problems are left to other papers in the program.

The accumulation of DOE-owned SNF is, currently, resting primarily in storage pools. Some of this SNF has been in underwater storage well beyond the limits initially planned and has degraded to the point where continued storage, as is, can lead to safety problems. This situation combined with the fact that the SNF stored in multiple locations is made up of over 90 fuel forms grouped into six categories presents DOE with formidable challenges as it seeks a path forward for permanent disposal.

DOE-OWNED SNF VULNERABILITIES

In August 1993, the Secretary of Energy requested that the condition of DOE-owned SNF storage facilities be assessed. Although no conditions were identified that required immediate attention to prevent harm to the workers or to the public, 106 environmental, safety and health vulnerabilities were identified regarding the current inventory of spent fuel. In addition to these vulnerabilities, five DOE facilities and three burial grounds were identified that warranted priority

management attention to avoid unnecessary increases in worker radiation exposure and cost during cleanup.
A critical generic vulnerability identified in the assessment of the DOE SNF inventory noted that, unlike the high-level waste from the first stage of reprocessing, there was no path forward for ultimate disposition of DOE-owned SNF. In the past, DOE-owned SNF had been managed under a plan to achieve ultimate disposition by converting the liquid high-level waste resulting from reprocessing into a vitrified waste form for disposal in a geologic repository. However, in 1992, DOE discontinued reprocessing thereby leaving an inventory of spent fuel to be managed in addition to the inventory of liquid high-level waste.
In response to the assessment of storage facilities for DOE-owned SNF, a three-phased action plan was developed to resolve the identified vulnerabilities. The Phase I Action Plan addressed 31 of 33 high priority vulnerabilities, and 48 of the lower priority issues. The Phase II Action Plan released in April 1994 covered 81 vulnerabilities. As a result, all vulnerabilities were covered by a complete or partial individual action plan. The completion of the partial action plans identified in Phase II was addressed in the Phase III Action Plan released in October 1994. With this release, all vulnerabilities have now been addressed, one-quarter of the 430 corrective actions have been completed and 15 vulnerabilities have been completely resolved. The Phase III plan constitutes the final Headquarters report, and represents the completion of the Secretary's initiative to assess DOE's SNF facilities and the first step toward preparing DOE-owned SNF for interim storage.
MANAGING DOE-OWNED SNF
Using a system engineering methodology to define the program requirements, the path forward for permanent disposal has been defined as consisting of three primary functional activities:
Stabilization of existing fuels and facilities (Assure Existing Safe Conditions);
Development of stable, long-term, cost-effective interim storage pending disposition (Achieve Interim Storage);
Preparation for permanent disposition. Currently the planning base assumes disposal in a geologic repository. (Prepare for Disposition).
In undertaking these activities, DOE seeks to make its key SNF policy decisions in conformance with a process established by the National Environmental Policy Act (NEPA). As required by NEPA, the DOE is systematically evaluating the potential environmental consequences of spent fuel management and policy alternatives prior to making its management decisions.
To support DOE's decision making regarding management of DOE-owned SNF, a number of environmental documents have been or will be prepared. These documents include:
an Environmental Assessment for the return of DOE-owned SNF from foreign research reactors that have immediate significant problems with on-site storage of the SNF;
a Programmatic Environmental Impact Statement to address the management of SNF throughout the DOE complex, and
an Environmental Impact Statement for Foreign Research Reactor SNF that does not have immediate on-site storage problems.
Additional site-specific spent fuel NEPA reviews are underway or planned.
DOE also has a number of evaluations underway to consider significant programmatic or policy issues related to possible licensing by the Nuclear Regulatory Commission (NRC), and whether some of the DOE-owned SNF should be considered as Resource Conservation and Recovery Act (RCRA) material falling under Environmental Protection Agency regulations. These issues are discussed later in this paper.
In a real sense, NEPA, through the Programmatic EIS, will drive the DOE-owned SNF program strategy for some 40 years on a national level. The draft Programmatic EIS that has been issued addresses potential environmental impacts associated with alternative management approaches, e.g. regionalization which would distribute existing and projected DOE-owned SNF among certain DOE sites based primarily on fuel type. A final programmatic EIS on DOE-owned SNF is scheduled to be issued by April 15,1995, following review and revision based on stakeholder comments on the draft EIS, and the Record of Decision by June 1,1995.
Another EIS that is being prepared involves SNF from foreign research reactors that contain U.S. origin enriched uranium. This document will evaluate the potential environmental effects of establishing and implementing a policy that is consistent with the long-standing nuclear non-proliferation policy of the United States. This

EIS, scheduled to be issued in draft later this year, will defer to the programmatic EIS for siting alternatives, but evaluates the environmental consequences for a stand-alone, site- specific management approach for SNF from foreign research reactors.

PROGRAM POLICY ISSUES

The resolution of existing and potential programmatic issues will ultimately depend on the completion of the NEPA process. Until that process is completed, near-term issues and vulnerabilities must be addressed to the extent possible. At this time, the DOE-owned SNF program faces three primary policy issues:

- Should DOE-owned SNF be placed in the first geologic repository?
- Should new interim dry storage facilities be developed and licensed by the NRC?
- Should some DOE-owned SNF be regulated under RCRA?

Should DOE-Owned SNF be Placed in the First Geologic Repository?

The Nuclear Waste Policy Act of 1982, as amended, mandated that the Federal Government pursue the development of deep geologic repositories for the disposal of civilian spent nuclear fuel, high level waste and other highly radioactive waste. In 1985, the President authorized the co-location of defense nuclear wastes and civilian nuclear waste. Although the statute was silent on the disposal of DOE-owned SNF, in 1994, the DOE General Counsel concluded that the NWPA authorizes disposal of DOE-owned SNF in the civilian repository conditioned upon an intra-agency agreement that requires payment of a fee adequate to cover the full cost of the disposal. This conclusion is consistent with Act's stipulation that the EIS accompanying the repository site recommendation to the President does not need to consider alternatives to geologic disposal for spent fuel.

It is important to note that the NWPA limits the first repository to 70,000 MTHM until a second repository is in operation. The Office of Civilian Radioactive Waste Management has allocated, as a planning basis, 10 percent of the first repository for defense materials. Within this 10 percent allocation, DOE may include some DOE-owned SNF. While there is currently no consensus on a method of assigning metric ton heavy metal equivalence to high-level waste, it is anticipated that the total inventory of high-level waste will exceed the 7,000 metric ton heavy metal planning allocation set by the Office of Radioactive Waste Management. The Department is evaluating an option of placing both DOE SNF and Defense HLW in the first repository, up to the 7,000 MTHM limit, with factors like risk, safety, cost, etc. determining how much and what type of each would go when.

Should New Interim Storage Facilities be Licensed by the NRC?

Management of DOE-owned SNF until its disposal is expected to require new interim storage capacity to span the time period until the geologic repository is available. However, since the current statutory framework does not allow licensing of new DOE interim storage facilities for DOE-owned SNF by the NRC, a decision must be made regarding whether DOE should proceed toward implementation of NRC licensing requirements and reviews for new facilities. One option available under the current statutory framework is to conduct a "licensability review", which would provide an independent technical review by the NRC of the adequacy of the proposed facility. This approach will require DOE to agree to pay a fee to the NRC for the review. Other options are also under evaluation.

Should Some DOE-Owned SNF be Regulated Under RCRA?

With passage of the Federal Facility Compliance Act of 1992, which addresses management of mixed wastes, questions have been raised concerning the applicability of RCRA waste management regulations to DOE-owned SNF. In light of decisions to discontinue reprocessing of DOE-owned SNF, the Department is evaluating potential regulatory frameworks, but is committed to safely and responsibly managing SNF regardless of the regulatory regime, i.e. under RCRA or the Atomic Energy Act. Preliminary evaluations indicate that only a small fraction of the many types of DOE-owned SNF may potentially exhibit the characteristics of a RCRA hazardous waste. Process knowledge or more specific ongoing evaluations will be required to completely resolve this issue. Discussions are ongoing with the Environmental Protection Agency on this issue.

MASTER LOGIC SCHEDULE

As noted earlier in this paper, the DOE complex is wide-spread with a large number of fuel types residing in several sites. The management of these fuel types leading to final disposition is controlled by the several NEPA documents that are being produced for the entire complex as well as individual sites. To manage this vast

array of activities, as well as to enable the decision maker to maintain an overall picture of the on-going activities stretching out over a 40 year period, DOE has developed a Master Logic Schedule (MLS). The MLS consists of several levels of detail ranging from the Summary Chart and Logic Diagrams shown in Figs. 1 and 2 to progressively more detailed schedules.

The MLS can be thought of as a snapshot in current time, recognizing that as some of the NEPA documents are completed, and Record of Decisions issued, the activities may change. For example, the MLS was developed using the planning base of regionalization by fuel type. The MLS assumes that certain spent fuel types will be moved to selected DOE sites. Under this option, chemical processing of the DOE-owned SNF could take place at the Savannah River Plant, and possibly the Idaho National Engineering Laboratory, pending NEPA evaluations. The schedule assumes that certain DOE-owned SNF may be declared a RCRA waste, and that DOE will seek a licensability review by the NRC of new dry storage facilities to be acquired. However the MLS also assumes that RCRA and the licenseability review is done during the Major Systems Acquisition process, thus not impacting the schedule. Delays due to law suits, permit acquisitions, and various reviews could impact the schedule significantly, and will require schedule modifications.

If this is not the decision reached by the Department by June 1,1995 there will be substantial changes in the MLS. Similarly, as the Technology Integration Plan matures from its current state (as reflected in the December 1994 revision 0), activities and milestones in the MLS will change. To provide for this, the MLS is accompanied with a data base that provides ready revision of the MLS to reflect decisions as well as actual schedules. The MLS will become part of the integrated SNF Program Plan, which will be issued in FY 1995. This Plan will be a living document, being updated periodically to reflect current issues and approaches to resolve them. The Program Plan will serve as a control mechanism for the SNF program, guiding budget and other planning and decision-making activities.

CONCLUSION AND SUMMARY

Safe, efficient, and effective management of DOE-owned SNF in preparation for ultimate disposal (in a geologic repository) is a complex and difficult task. DOE needs to ensure that the necessary policies and programs, technologies, hardware, and facilities are all developed, integrated and available in a timely manner. DOE is utilizing system engineering at both the program-level and site-level to achieve this.

It is expected that as the environmental documents, studies, and findings are completed, the requirements and direction of the program may undergo changes and adjustments. The management practices and tools that DOE has put in place are considered to be sufficiently responsive and far reaching so as to enable management to make appropriate responses in a timely manner to keep the DOE-owned SNF program on track.

Because the program for dealing with DOE-owned SNF is relatively new, and is not expected to complete its mission until well into the next century, it is critical that stakeholders be actively engaged. In keeping with DOE's openness initiative, the public review process for the Programmatic Spent Nuclear Fuel EIS entailed four scoping periods, 20 public hearings, and the receipt and consideration of about 5100 public comments. Also, DOE is currently working with the Citizen Advisory Boards at Hanford, Idaho, and the Savannah River Site to secure stakeholder input on the recently issued Strategic Plan, and the Foreign Research Reactor EIS.

Through the completion of these activities:

- issuance of the Plans of Action to address vulnerabilities,
- issuance of the Programmatic SNF EIS and Record of Decision, - resolution of the 3 key program issues,
- issuance of the SNF Strategic Plan,
- preparation of the MLS and Program Plan,
- issuance of the Technology Integration Plan, and
- issuance of SNF program requirements documentation,

the DOE will have set the initial framework and planning basis for a complete path forward. With all of this in place by the end of FY 1995, the program will be focusing in FY 1996 on getting on with the job of completing the resolution of urgent health and safety vulnerabilities, developing and implementing solutions for new, dry interim storage, and tackling the technical and regulatory issues associated with complying with repository acceptance criteria. DOE believes it has

the right program in place to carry out this important task and complete the implementation of the path forward.
Whether resolving technical and regulatory issues, or interacting with stakeholders in this dynamic program, NEPA reviews will play a key role in our activities, both now and in the future. But the specific path forward will be determined in detail by the programmatic EIS and supporting program documents. These planning documents will form the basis of an overall framework that will enable us to overcome the challenges of DOE-owned SNF management and to meet our program commitments.

36-2
CRITICAL POLICY ISSUES FACING DOE WITH RESPECT TO THE STORAGE AND DISPOSAL OF DOE-OWNED SPENT FUEL

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ABSTRACT

The Department of Energy (DOE) Office of Spent Fuel Management, EM-37, is proceeding with actions to implement safe and cost effective long-term interim storage of its Spent Nuclear Fuel (SNF) prior to ultimate disposal. In order to address identified shortcomings in the current storage of DOE-owned SNF, EM-37 developed the "Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities." Cognizant DOE sites will implement the actions for the individual facilities that are specified in the Plan of Action. The third and final phase of this plan was issued in October 1994. This Plan of Action identified four critical policy issues that confront DOE with respect to safe and effective interim storage of SNF and preparation of SNF for final disposal. These issues are:

What is the Path Forward for Geologic Disposal of the Department of Energy's Spent Nuclear Fuel? How much, if any, DOE-owned fuel will be placed in the "first" repository? How will the selection be made? What will happen to SNF not placed in the first repository?

Should New Interim Storage and Conditioning Facilities be Licensed by the Nuclear Regulatory Commission? What will the standards be for DOE's interim storage facilities? How will DOE address concerns with respect to "self-regulation?"

Should Hanford N-Reactor Fuel be Stored in a Dry Configuration? Dry storage of N-Reactor and other DOE SNF continues to be an actively discussed topic.

What Should the Approach Be If Some DOE-Owned Spent Fuel is Deemed Unsuitable for Extended, Interim Dry Storage or Direct Geologic Disposal? The issue here is whether DOE should retain the option and capability to process certain "at risk" SNF.

This paper summarizes progress to date in addressing these issues. In all cases, a final decision has not been reached with respect to every aspect of these issues; however, in each case DOE has identified a preferred option in the form of a recommended technical approach or path forward. Each of these issues is discussed in turn, in light of the most current information available.

INTRODUCTION

In August of 1993, Secretary O'Leary commissioned a comprehensive baseline assessment of the environmental, safety, and health vulnerabilities associated with the storage of spent nuclear fuel (SNF) in the DOE complex. During October 1993, a multi-disciplinary Spent Fuel Working Group, directed by the Assistant Secretary for Environment, Safety and Health, and comprised of DOE employees and contractors, assessed 66 facilities spread across 11 sites. This assessment was performed to determine the inventory and the condition of the Department's Reactor Irradiated Nuclear Material, which includes spent nuclear fuel and reactor irradiated target material. The assessment also evaluated the condition of the facilities that store spent fuel and identified the vulnerabilities and problems that are currently associated with these facilities.

In order to address the identified vulnerabilities and associated problems, EM-37 developed the "Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities." In developing this plan, four critical policy issues were identified that significantly affected the future direction of DOE's spent nuclear fuel program. Resolution of these issues was considered necessary in order to permit important programmatic decisions to be made so that a clear path could be established to address the identified vulnerabilities and related problems. This paper presents the current status of DOE's efforts to resolve these critical policy issues.

WHAT IS THE PATH FORWARD FOR GEOLOGIC DISPOSAL OF THE DEPARTMENT OF ENERGY'S SPENT NUCLEAR FUEL?

Background

The Nuclear Waste Policy Act of 1982 and its 1987 Amendment (the Act) mandates that the Federal Government pursue the development of deep geologic repositories for the permanent disposal of civilian spent nuclear fuel, high-level waste, and other highly radioactive waste. In 1985, the President accepted the Secretary of Energy's recommendation that defense nuclear waste should be co-located with the civilian waste. General Counsel concluded in a legal opinion dated March 23, 1994, that the Act authorizes the disposal of DOE-owned spent nuclear fuel in a geologic repository, conditioned upon payment by the DOE of a fee adequate to cover the full cost of disposal. The DOE, however, has made no final decision at this time regarding the specific strategy for the ultimate disposition of DOE-owned spent nuclear fuel.

The Act limits the capacity of the first repository to 70,000 metric tons heavy metal until a second repository is in operation. DOE's Office of Civilian Radioactive Waste Management (OCRWM), however, has allocated as a planning basis 10 percent of the repository capacity for defense wastes (7,000 metric tons heavy metal, if the capacity of the first repository is at least 70,000 metric tons heavy metal). Projections of waste inventories show that the first repository, with a legislative limit of 70,000 tons, does not have adequate capacity for either all civilian, or the total civilian and defense waste.

The path forward for ultimate disposition of DOE-owned spent nuclear fuel must be addressed in a manner consistent with the three DOE-owned spent nuclear fuel program systems engineering functions: remediation of vulnerabilities in existing storage, achievement of reliable interim storage, and preparation of DOE-owned spent nuclear fuel for disposal. In addition, three assumptions are made when developing options for ultimate disposition of DOE-owned spent fuel. These assumptions are: 1) The current OCRWM program will be successful in constructing, licensing and operating repositories for the disposal of high-level radioactive waste; 2) Development of long-term interim storage, with active institutional controls in place, will be required to store DOE-owned spent nuclear fuel, pending repository availability; and, 3) There is a distinct possibility that some DOE-owned spent nuclear fuel may not be adequately characterized to support the OCRWM schedule to submit its license application to the Nuclear Regulatory Commission in 2001.

Discussion

Three planning options have been developed for consideration. Each option assumes emplacement in a geologic repository as DOE's proposed strategy for disposition of its inventory of spent nuclear fuel. These options would integrate DOE's spent nuclear fuel into the program being conducted by OCRWM for the disposal of civilian spent nuclear fuel and establish the framework within which to evaluate DOE-owned spent nuclear fuel against the requirements for qualifying spent nuclear fuel for geologic disposal. The options considered are:

Option 1 -- DOE-owned spent nuclear fuel and vitrified high-level waste would be authorized for disposal in the first repository. The total quantity of DOE-owned spent nuclear fuel and vitrified high-level waste would not exceed 10 percent of the repository capacity. Disposition of remaining DOE-owned spent nuclear fuel and vitrified high-level waste would not be decided until the DOE recommendation on the need for a second repository. In addition, under this option all DOE-owned spent nuclear fuel would be aggressively stabilized and prepared for repository disposal.

Option 2 -- All DOE-owned spent nuclear fuel would be authorized for disposal in the first repository, along with all high-level waste, assuming the 70,000 metric tons heavy metal limit is removed by Congress.

Option 3 -- All DOE-owned spent nuclear fuel would be deferred to a second repository.

Recommended Path Forward

Option 1 is recommended because it provides a path forward for the geologic disposal of DOE-owned spent nuclear fuel, while minimizing impacts on OCRWM's schedule for the first repository. Option 1 leaves undisturbed the existing repository capacity allocations for civilian spent nuclear fuel and DOE-owned materials. Option 1 also does not assume an increase in the capacity of the first repository. This option is also recommended because it would constitute a significant step toward satisfying stakeholder concerns over de facto disposal of DOE-owned spent nuclear fuel at

storage sites.
SHOULD NEW INTERIM STORAGE AND CONDITIONING FACILITIES BE LICENSED BY THE NUCLEAR REGULATORY COMMISSION?
Background
New interim storage facilities will be required for DOE's spent nuclear fuel since additional time is needed to finalize a decision regarding the location and timetable for long-term disposal of this fuel. However, it is very likely that DOE's spent fuel will eventually come under Nuclear Regulatory Commission (NRC) jurisdiction, since it will be required to meet waste acceptance criteria for geologic disposal, and given the likelihood of new legislation being enacted to address external regulation of new DOE facilities. Further, it appears that the technical standards that would be required for NRC licensability of any facility associated with spent fuel storage or disposal are reasonable. Therefore, it is prudent that DOE pursue an evaluation of the current regulatory framework and technical requirements associated with NRC licensability of new spent fuel interim storage facilities. This evaluation will provide the data and information required for a future decision regarding licensing of DOE facilities.
Discussion
A number of options exist as to how to best pursue application of NRC requirements to DOE's spent nuclear fuel facilities. Three options were formulated to capture and highlight the identified issues and address the variety of implementation approaches. These options can be summarized as follows:
DOE Directive Option -- This option would be to design future SNF interim storage facilities to comply with DOE technical requirements and proceed to construction without NRC review. This option relies on existing DOE spent fuel requirements defined in the DOE directives system.
Licensability Review Option -- This would consist of designing future SNF interim storage facilities to comply with NRC technical requirements and a confirmation of compliance through NRC licensability reviews. This type of independent technical review of a DOE facility by the NRC has been previously conducted (e.g., Fast Flux Test Facility) and can be performed within the current statutory framework. To facilitate this type of NRC review, this option adopts NRC technical and safety requirements utilized for commercial SNF.
Full Licensing Option -- This option would require design of future SNF interim storage facilities to comply with NRC requirements and begin construction once an NRC license can be obtained. This option differs from the other options primarily in that it would require external regulation of DOE facilities and thus represents a change from the status quo. While this option would result in external regulation, it would also require congressional action for full implementation.
Recommended Path Forward
The need for flexibility early in the DOE-owned SNF program and a desire to position DOE for potential future external regulation were determined to be the predominate selection criteria. The best option, which could be implemented under the current statutory framework and would assist in transitioning to external regulation, is the Licensability Review option. The Licensability Review option would also assist in verifying the implementation of NRC criteria to DOE spent fuel and facility design. Moreover, it would provide a strong independent technical review that could improve the credibility of DOE with its stakeholders and the general public. Finally, a licensability review would facilitate a detailed evaluation of the cost, schedule, and safety benefit of NRC oversight of DOE facilities.
SHOULD HANFORD N-REACTOR FUEL BE STORED IN A DRY CONFIGURATION?
Background
Facility assessments documented in the Spent Fuel Working Group Report on spent nuclear fuel vulnerabilities issued on December 7, 1993, and Defense Nuclear Facilities Safety Board (DNFSB) reports have identified concerns over continued wet storage of Hanford N-Reactor fuel. DNFSB Recommendation 94-1 in part recommended the acceleration of efforts to place N-Reactor fuel into a stable interim storage configuration with a "program to be directed toward storage methods that will minimize further deterioration." Continuing wet storage may result in continued spent fuel degradation and negative impacts on future spent fuel handling and disposal. Dry storage of spent fuel, in general, is considered superior to wet storage; however, the knowledge base for dry storage of N-Reactor spent fuel is somewhat limited. An evaluation was performed in order to determine whether

long-term wet storage or dry storage is most beneficial in minimizing further fuel degradation and potential environmental insult.

Discussion

Placement of the N-Reactor fuel into dry storage has been evaluated by an Independent Technical Assessment (ITA) team comprised of industry experts, sponsored by the DOE Richland Operations Office. Results of this evaluation indicate that the fuel can be conditioned prior to placement into dry storage, and safely managed during the dry storage period. The Assessment team defined the steps and related schedule which could be followed for demonstration that dry storage of N-Reactor fuel is safe and that the fuel degradation can be arrested.

Recommended Technical Approach

The Hanford Spent Nuclear Fuel Project (the Project), in conjunction with the ITA, has developed engineered alternatives for expedited removal of spent nuclear fuel, including sludge, from the K Basins at Hanford. Based on these evaluations, a Westinghouse Hanford Company (WHC) recommended path forward for K Basins spent nuclear fuel has been developed.

The recommendation consists of a series of projects to construct and operate systems and facilities to safely retrieve, package, transport, process, and store K Basins fuel and sludge. The overall conditioning and storage scheme is based on the ITA team's proposed passivation and vault storage process. A dual purpose staging and vault storage facility provides an innovative feature that allows accelerated removal of fuel and sludge from the basins and minimizes programmatic risks beyond any of the originally proposed alternatives.

The two-phase strategy packages and moves K Basins fuel and sludge to a newly constructed Staging and Storage Facility where it is staged for conditioning. When an adjoining facility is constructed, the fuel is cycled through a stabilization process and returned to the Staging and Storage Facility for interim (40 year) dry storage.

This recommended path forward combines aspects of several of the originally developed alternatives, thus optimizing across the most significant objectives and constraints. In general, the concept of vault storage of dried, passivated metal fuel (the ITA proposal) was used. However, temporary staging of wet packaged fuel and sludge prior to conditioning was added to improve the schedule for K Basins fuel and sludge removal. Using the dry storage vault facility for temporary fuel staging avoided the need to construct a separate basin to perform this function.

The packaging, transfer, and wet staging of the spent fuel are to be conducted under a K Basins Environmental Impact Statement (EIS); a Notice of Intent for this NEPA review was scheduled to be issued in early 1995. The necessary testing, design, and procurement of critical path items for this project are to be conducted under the Hanford SNF EIS, which is scheduled to be issued in mid-1996. Preliminary schedule projections include the start of fuel removal in late 1997, complete fuel removal by December 1999, initial placement of SNF in dry storage in 2001, and complete placement in 2005.

WHAT SHOULD THE APPROACH BE IF SOME DOE-OWNED SPENT FUEL IS DEEMED UNSUITABLE FOR EXTENDED, INTERIM DRY STORAGE OR DIRECT GEOLOGIC DISPOSAL?

Background

During April 1992, the Department of Energy approved a secretarial action directing the "phaseout of reprocessing at both the Savannah River Site (SRS) and the Idaho National Engineering Laboratory (INEL)." For Savannah River, the secretarial action included an option for processing of existing inventories of aluminum clad fuel in parallel with stabilization operations over a five or six year period. Pursuant to this policy, further DOE spent nuclear fuel processing would be limited to DOE-owned spent nuclear fuel deemed "at risk" for interim storage or potentially unacceptable for the repository. "At risk" fuels are defined as those fuels which will not, or cannot, be expected to maintain safe and non-hazardous integrity during either underwater basin storage or extended, interim dry storage.

The processing of some "at risk" spent nuclear fuel in the F and H Canyons at Savannah River site is under evaluation. Also, in recognition of the unfavorable economics and the perceived proliferation risks attendant with the conventional aqueous reprocessing technology, DOE is developing an alternative technology that could prove to be useful in conditioning certain "at risk" spent nuclear fuel types. This technology, known as "pyroprocessing," employs pyrochemical methods to separate fissile materials from fission products in such a way that there is no production of

a stream of separated plutonium. Because it recycles all of the process reagents and thereby produces minimal secondary waste streams, the pyroprocess also promises to provide improved waste management through reduced high-level waste volumes and by production of highly stable final waste forms. The feasibility of application of this process to many of the DOE spent nuclear fuel types is being evaluated.
Spent nuclear fuel at various DOE facilities has been inventoried, with data being assembled on approximately 150 fuel types (e.g., cladding, dimensions, fissile compounds and alloys, chemical constituents, uranium enrichment and mass, etc.). Data are also being assembled on the condition of each fuel type and the potential for, or existence of, serious deterioration. Vulnerability assessments have been made of existing storage facilities to identify weaknesses due to aging and to deterioration of containment for radioactive materials. For a few fuel types, the existing storage facilities and deteriorated fuel condition may place the fuel "at risk" during the period prior to transfer to dry storage or during the 40 year period of extended, interim dry storage.

Discussion

Various options are available to DOE with respect to dealing with the fuel deemed to be "at risk:"

Option 1 -- Allow no chemical conditioning of DOE-owned spent nuclear fuel prior to transfer to extended, interim dry storage facilities; limit conditioning to repackaging. Chemical conditioning may be required before ultimate disposal of the spent nuclear fuel in the geological repository.

Option 2 -- Process fuel considered to be "at risk," i.e., those fuels that will not, or cannot, be expected to maintain safe and non-hazardous integrity either during storage in underwater basins or during extended, interim dry storage.

Option 3 -- In addition to conditioning fuels considered "at risk," process other DOE-owned fuels that are believed to be unacceptable for placement in the geological repository.

Recommended Technical Approach

The option for processing "at risk" spent nuclear fuels should be retained as a means to assure public and site personnel safety, and facility and environmental protection prior to implementation of extended, interim dry storage.

Selection of "at risk" spent nuclear fuel types for conditioning should be based on a thorough systems engineering study, comparing processes, technical risks, lifetime costs, and waste form volume and stability. The systems engineering study should address stakeholder concerns and involve the stakeholders at critical stages of the management decision process.

The use of conditioning technology should be considered for fuel other than just "at risk" fuels. The effect on spent nuclear fuel storage under the Resource Conservation and Recovery Act and the waste streams for the geological repository could be affected in a positive way by this conditioning and should be evaluated in a systems engineering study.

The specific approach(es) to be taken for these "at risk" fuels is being evaluated in an EIS on interim (10 year) management of materials at SRS. A range of possible approaches is being evaluated for different fuel types and includes processing to metal or metal oxides, vitrification, or continued storage. The EIS will determine the appropriate technical or management approach for these fuels, and a decision will need to be made in the near term to ensure the availability of SNF conditioning facilities such as the F and H Canyons at SRS.

In addition, reprocessing is being considered as an option under the Foreign Research Reactor EIS which is currently scheduled for a Record of Decision in late 1995.

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36-3

NATIONAL ENVIRONMENTAL POLICY ACT REVIEWS ON MANAGEMENT OF DEPARTMENT OF ENERGY SPENT NUCLEAR FUEL

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ABSTRACT
This paper gives a brief history of the origins of DOE's current inventory of spent nuclear fuels (SNF) and the effects of the end of the "Cold War" on the management of such materials. It also describes the proposed and future DOE SNF management activities, and focuses in detail on the National Environmental Policy Act (NEPA) reviews to evaluate the environmental effects of these activities.
HISTORICAL PERSPECTIVE
During the last four decades, the Department of Energy (DOE) and its predecessor agencies have transported, received, stored, and reprocessed approximately 100,000 metric tons of spent nuclear fuel (SNF). Currently, approximately 2,700 metric tons of heavy metal (MTHM) of SNF that are now stored at various locations in the United States--as shown in Fig. 1-- and overseas have not been reprocessed. This SNF is in a wide range of enrichments (that is, percent uranium 235), types, and conditions. By the year 2035, this quantity may increase by approximately 100 metric tons.
The bulk of this SNF that is now the responsibility of DOE was utilized in "production" reactors at the Hanford and Savannah River Plants that were used to produce plutonium and tritium for nuclear weapons. DOE SNF is also derived from the U.S. Navy's use of nuclear reactors for ship propulsion. Smaller amounts of SNF were generated by research reactors at other DOE facilities, such as the Oak Ridge Reservation, Argonne National Laboratory-East, Brookhaven National Laboratory, Los Alamos National Laboratory, and Sandia National Laboratory-Albuquerque, and at many non-DOE research reactors--both domestic and foreign--such as at universities, government agencies, and private firms. DOE-EM has also been assigned the responsibility for the SNF from a few special-case commercial nuclear power plants or former reprocessing facilities. Detailed listings of DOE SNF inventories at their various facilities are given in Table I (1).
It should be noted, however, that most SNF from commercial nuclear power plants--a far larger amount of SNF than the DOE SNF discussed here--is the responsibility of DOE's Office of Civilian Radioactive Waste Management (OCRWM), and such commercial SNF will therefore not be discussed in further detail in this paper. OCRWM has been studying, as authorized by the Nuclear Waste Policy Act of 1982, a site in Nevada for suitability for geological disposal of SNF and defense high-level radioactive wastes.
REDIRECTION FOLLOWING THE END OF THE COLD WAR
The end of the Cold War led DOE to reevaluate the scale of its weapons production, nuclear propulsion, and research missions. In April 1992, DOE began to phase out reprocessing of SNF for recovery and recycling of plutonium and highly enriched uranium. DOE realized the need to establish an integrated complex-wide program that provides safe and effective management of the SNF inventory, pending its permanent disposition.
Concurrently, DOE was preparing the "Environmental Restoration and Waste Management Programmatic EIS" on Department-wide waste management and environmental restoration programs, including a section on management of DOE SNF. As recently announced (60 Federal Register 4607, January 24, 1995), the DOE proposes to downscope the Environmental Restoration and Waste Management Programmatic EIS to waste management only. ["Programmatic NEPA document means a broad-scope EIS or EA that identifies and assesses the environmental impacts of a DOE program..." (definition from 10 CFR 1021.104)] When DOE committed itself to preparing a complex-wide SNF EIS as part of a court order--as discussed in more detail below--the coverage of SNF management was transferred to the more comprehensive coverage in the "Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS". The scoping and timing of these two EIS documents is also discussed further below.
CURRENT SNF STORAGE ISSUES AND THEIR NEPA IMPLICATIONS
The following text describes ongoing NEPA reviews related to SNF management and discusses other factors likely to be considered in making strategic decisions regarding interim storage of DOE SNF over the next 40 years, as well as DOE plans for future tiering of NEPA reviews that may be prepared to assist in implementation of programmatic decisions. ["Tiering" refers to the coverage of general matters in broader environmental impact statements (such as national program or policy statements) with subsequent narrower statements or environmental analyses (such as

regional or basinwide program statements or ultimately site-specific statements) incorporating by reference the general discussions and concentrating solely on the issues specific to the statement subsequently prepared. ..." (definition from 40 CFR 1508.28)]
While the Environmental Restoration and Waste Management Programmatic EIS was being prepared, a lawsuit was instituted by the State of Idaho against the further importation of SNF into Idaho, specifically from the Fort St. Vrain facility in Colorado to Idaho National Engineering Laboratory, on the grounds that an EIS had not been prepared for such shipments. In settling the lawsuit by a court order in June 1993 (slightly modified in September and December 1993), DOE and the Navy agreed to suspend almost all shipments of SNF to Idaho National Engineering Laboratory (except for limited shipments of Naval SNF) while such an EIS was being prepared. DOE and the Navy also agreed to prepare a two-part EIS (1), called the "Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS", and an EA on a very specific timetable as follows:

Issue EIS Implementation Plan	11-1-93
Complete Draft EIS and publish public notice of its availability in the Federal Register (FR)	6-30-94
Complete Final EIS and publish public notice of its availability in the FR	3-30-95
Issue and publish in the FR the Record of Decision based on the Final EIS	6-1-95
Prepare Environmental Assessment (EA) for storage of Navy SNF at sites other than Idaho National Engineering Laboratory	12-31-93

One part of the EIS was a complex-wide study of the management of SNF; the other part was a more detailed, site-specific study of environmental restoration and waste management programs (including SNF management) at Idaho National Engineering Laboratory. The alternatives being considered in this two-part EIS are listed below:
Volume 1--Programmatic Spent Nuclear Fuel Management Alternatives (2,1):
No Action, which is taking the minimum actions required for safe and secure management of SNF at or close to the generation site or current storage location;
Decentralization, which is storing most SNF at or close to the generation site or current storage location with limited shipments to DOE facilities;
1992/1993 Planning Basis, which is transporting and storing newly generated SNF at Idaho National Engineering Laboratory or the Savannah River Site;
Regionalization, which is distributing existing and projected SNF among DOE sites based primarily on fuel type or geographic location; and
Centralization, which is managing all existing and projected SNF inventories from DOE and the Navy at one site until ultimate disposition.
Volume 2--Idaho National Engineering Laboratory Environmental Restoration and Waste Management Alternatives (for details, see reference (1)):
No Action
Ten-Year Plan
Minimum Treatment, Storage, and Disposal
Maximum Treatment, Storage, and Disposal.
Following the court order mentioned earlier, DOE published in the Federal Register on September 3, 1993 a Notice of Opportunity to Comment on a proposal to expand the scope of the Idaho National Engineering Laboratory Environmental Restoration and Waste Management EIS to include Department-wide impacts related to: transportation, receipt, processing, and storage of DOE-owned SNF. Comments were accepted from September 3, 1993 through October 4, 1993. The EIS Implementation Plan summarized comments received and described the resulting scope of the EIS. The EIS Implementation Plan was issued on October 29, 1993 (3).
Three DOE site alternatives were identified in the October 29, 1993 EIS Implementation Plan as ones having existing, large-scale SNF management operations: the Hanford Site, the Idaho National Engineering Laboratory, and the Savannah River Site. Also, four Naval shipyards and the DOE Kesselring Site (in upstate New York) were identified as having experience in handling Naval SNF; these shipyards are: Norfolk, in Portsmouth, VA; Portsmouth, in Kittery, ME; Pearl Harbor, in Honolulu,

HI; and Pudget Sound, in Bremerton, WA.
In response to public comments generated in the scoping process, DOE decided to broaden the range of siting alternatives by considering other sites for future SNF management. DOE utilized a disciplined screening process, which resulted in the addition of the Oak Ridge Reservation and the Nevada Test Site as reasonable alternative sites for regionalized or centralized SNF management (4). The EIS Implementation Plan was amended on May 9, 1994 to include these additional sites. During this same time period, internal studies by DOE also identified environmental, safety, and health vulnerabilities in SNF-related units at Hanford, Oak Ridge Reservation, Savannah River and Idaho National Engineering Laboratory, which were described in a detailed report dated November 1993 (5). This was followed by DOE's three-phase "Plan of Action to Resolve SNF Vulnerabilities", which was finalized in October 1994 (6). These vulnerabilities are discussed in a paper by John J. Jicha, Jr. of DOE in Waste Management '95, session 52; its title is "DOE-Owned Spent Fuel Program Strategic Plan: Issues, Challenges, and Options."
The Draft "Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS" published in June 1994 drew approximately 5,100 comments from stakeholders and the public. DOE is presently responding to these comments and adjusting the EIS where appropriate. No sooner than 30 days after the final EIS is published, the Department will issue its record of decision. However, any particular SNF management alternative that is chosen may affect many sites, and new SNF-related facilities may be needed; other NEPA reviews may be necessary for these, with some of the NEPA reviews "tiered from" the Programmatic SNF Management EIS. The status of many such ongoing or planned tiering NEPA documents is shown in Table II. The interrelationships of the various NEPA documents discussed herein is shown in detail in Fig. 2 (2), with the letters shown in the lower right corner of specific boxes keyed to the listing of the same documents in Table II.
Note that each non-DOE domestic research reactor has appropriate NEPA coverage, as part of the Nuclear Regulatory Commission (NRC) license requirements, for on-site storage and shipping of the SNF from the reactor.
In the 1950's and 1960's the U.S. Government, acting through DOE and its predecessor agencies, began its "Atoms for Peace" program, which included a program whereby DOE furnished enriched uranium for fuel for research reactors in various foreign countries, and agreed to take back the SNF from those reactors. This program expired in 1988 for highly-enriched uranium (HEU) and in 1992 for low-enriched uranium (LEU). The United States is considering the resumption of the acceptance of SNF from such foreign research reactors in furtherance of its fundamental foreign policy and national security objectives to prevent the spread of nuclear weapons to additional foreign countries. A key element of United States nuclear non-proliferation policy has been to minimize and eventually eliminate worldwide the use of HEU--a material usable in nuclear weapons--in civil nuclear programs. Research reactors are the major civil use of HEU; if such reactors convert to LEU fuels, HEU will essentially be eliminated from civil commerce.
An Environmental Assessment (EA) was prepared for urgent shipments from eight foreign reactors (7), and an EIS is in preparation on an acceptance policy that would involve Foreign Research Reactor (FRR) SNF from approximately 42 reactors in 26 countries. The "Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS" that was mentioned earlier takes into account the SNF that might be received from these foreign research reactors. FRR SNF is discussed at length in a paper by Charles Head of DOE in Waste Management '95, session 52; its title is "The DOE Foreign Research Reactor (FRR) SNF Program and Status."
FUTURE SNF STORAGE AND DISPOSAL ISSUES AND THEIR NEPA IMPLICATIONS
The relationship of these studies with respect to planned activities by the DOE Office of Civilian Radioactive Waste Management (OCRWM) is described. These interrelated studies and actions provide a roadmap to establish a clear path forward for disposition of DOE SNF.
The main issues for the future of SNF management concern its ultimate disposition and the preparation required therefor, along with the interrelated implications of these issues for the interim storage of SNF. Because of the deteriorated condition of some of the existing DOE SNF inventory, stabilization may be necessary for interim storage as well as for packaging it for disposal. At the time of this paper,

it is uncertain whether DOE SNF would be disposed of by placement in the geologic repository now being investigated and planned by OCRWM mainly for the much larger quantities of commercial SNF from nuclear power plants. This repository is now anticipated to be available around 2010. Several related issues would need to be resolved, including: the degree of priority, if any, that DOE SNF will have vis a vis defense high-level radioactive waste; the repository's requirements on cask design and dimensions; and waste acceptance criteria. The potential disposal of DOE SNF in a geologic repository is discussed at length in a paper by Jon Thompson of DOE in Waste Management '95, session 52; its title is "A Path Forward for Geologic Disposal of DOE-Owned Spent Fuel." These latter requirements will determine the preparation and timing of the ultimate disposal of DOE SNF, but it is possible that a span of 40 years may be needed for all the DOE SNF to be placed in a repository. OCRWM is also preparing an EIS on development of a multipurpose cask design and plans to begin an EIS on the suitability of the Yucca Mountain Site in 1995. DOE's Office of Spent Fuels Management is cooperating with OCRWM on these studies to ensure that its management of DOE SNF is compatible with OCRWM's plans and requirements.

CONCLUSIONS

The end of the cold war has necessitated a change in the management of DOE SNF. The goal is to assure that all SNF management proposals are carefully analyzed through NEPA reviews to assure that the public and decisionmakers are well informed of potential environmental consequences. The numerous SNF-related NEPA documents discussed herein are illustrative of DOE's ongoing efforts to fulfill both in letter and in spirit the goals so cogently stated in section 2 of the National Environmental Policy Act of 1969, that is, "...To promote efforts which will prevent or eliminate damage to the environment and biosphere and stimulate the health and welfare of man..."

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36-4

CHARACTERIZATION OF SWISS PWR AND BWR NON-FUEL EXCHANGEABLE REACTOR CORE COMPONENTS FOR WASTE DISPOSAL PROJECTS

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ABSTRACT

Rod cluster control assemblies, part length control assemblies, burnable poison rod assemblies, flow restriction rods and incore instrumentation from 2 Westinghouse PWR's (350 MW(e)) and 1 Siemens Kraftwerk-Union PWR (965 MW(e)) as well as control rod assemblies, poison curtains, fuel channels and incore instrumentation from 2 General Electric BWR's (355 MW(e)) and (1030 MW(e)), expected with the neutrons sources to be discharged as waste over the 40 years of reactor operation, have been characterized for activities and materials for disposal projects. Packaging has been assumed 5 years after discharge from reactor. Comparisons with experimental results and further with operational and decommissioning waste are made and discussed.

INTRODUCTION

For the general licensing application made mid 1994 for the Swiss repository for short-lived radioactive waste (LLW + ILW), as well as for making projections for the repository for long-lived ILW and HLW, all the radioactive waste expected to arise in Switzerland up to the year 2040 have been characterized for activities, materials and properties up to the level of the conditioned waste package. This model inventory for disposal projects is based on waste sorts represented by average waste packages and their yearly production (1).

The wastes expected to arise from PWR (1.6 GW(e)) and BWR (1.3 GW(e)) nuclear power plants assumed to be in operation over 40 years were categorized into:

Operational waste (resins, concentrates, etc.)
Decommissioning waste
Reprocessing waste (vitrified residues, hulls, etc.) and
Non-fuel exchangeable reactor core components (rod cluster control assemblies, fuel channels, incore instrumentation, etc.).

The non-fuel exchangeable reactor core components discharged as waste have been kept separate from the decommissioning waste because they arise earlier (but with a significant fraction at the time of decommissioning). This paper presents this waste category which has not generally been addressed or reported by the radioactive waste community.

Other wastes included in the model inventory arise from medicine, industry and research. In particular, the decommissioning waste of a 600 MeV, 1.5 mA proton accelerator complex assumed to be shut down after 40 years of operation in 2036 have also been considered as the waste activity and volume are of the order of those of a 1GW(e) NPP (2).

DESCRIPTION OF THE NON-FUEL EXCHANGEABLE REACTOR CORE COMPONENTS

The types of components and the components themselves are specific to each reactor design. The raw wastes expected to arise, determined from an examination of the components on a reactor by reactor basis, are:

TABLE

The components are replaced after some years of operation when they become mechanically defective or when their operating life has ended. To estimate the numbers of components, discharged over 40 years in the framework of a model inventory, assumptions on the average operation time of the component types have to be made. To estimate the activities of the components further assumptions on the irradiation histories have to be made.

The raw wastes are materials which have been activated by the high neutron flux of the core, and in some cases, contaminated by crud deposits (components exposed to the reactor coolant water). The materials are stainless steels (structural materials), zircaloy (fuel channels), neutron absorbing materials (boron carbide (B4C), Ag-In-Cd alloys, hafnium, borosilicate glass, borated steels) and finally aluminum oxide (Al2O3) (filling material). The primary and secondary neutron sources contain, beside stainless steel, Al2O3 and boron glass, the material that generates the neutrons i.e. Cf-252 and Be/Sb-124.

RAW WASTE CHARACTERIZATION

The activation calculations have been made using the ORIGEN2 code including a cooling time of 5 years. The application of this code for these components is deficient with respect to the following points:

The neutron flux and cross sections used within the code do not take into account

for certain elements the large depression in the thermal neutron flux caused by the neutron absorbing materials (Note: for the fuel channels the calculation is correct for the metal but not for the uranium impurities within the metal because uranium is considered by the code as being in the fuel where the epithermal neutron flux is more depressed).

The code does not consider all the neutron reactions occurring from the isotopes of the elements constituting the n-absorbing materials and the metallic impurities. For instance the calculated tritium activity in the control rod boron carbide is much too low (B-10(n,t)Li-8 and secondary Li-7(n,nt)He-4 are not considered); the Be-10 (t = 1.6 106 y) activity calculated for the reaction B-10(n,p)Be-10 could be 2 orders of magnitude too low because the cross section for fast neutrons is not considered. For such cases involving disposal safety relevant radionuclide separate "hand calculations" were carried out to supplement the code.

It is expected that the development of applicable ORIGEN2 cross section libraries or other codes for radionuclide characterization of some core materials will be undertaken in the future on an international level as and when the necessity arises to specify these wastes for disposal. Such a development has just started for the reactor decommissioning wastes (3) and could include the exchangeable non-fuel core components.

The ORIGEN activity calculations have been carried out with the material compositions being specified down to the level of the major impurities. In addition, it was possible to correct for the remaining minor metal impurities in this inventory by using the results of work carried out elsewhere (4). In this work the impurities in the steels of several reactors were measured and activation calculations carried out for decommissioning waste studies. As the same type of steels are used in the components considered in the Swiss inventory, these results were added to the relevant component inventories following normalization to appropriate activities (Ni-59, etc.) arising from the main steel components with correction for different irradiation times.

Finally the crud activities were estimated by choosing an estimate for the value of the Co-60 activity per m2 (for instance 0.05 Ci/m2 for the BWR control blades after 5 years cooling) and using correlation factors for the other nuclides. For this estimation, confidential data made available to Nagra as well as data contained in Ref. 5 were used.

COMPARISON OF THIS WORK WITH OTHER EXPERIMENTAL RESULTS

Two recently published U.S. reports on the experimental characterization of reactor internals present activity results for the following components:

- PWR Westinghouse RCCA (6)
- PWR Westinghouse PLCA (6)
- BWR General Electric CRA (6)
- BWR classified CRA (7).

Comparison of the 12 activities (of interest for disposal in the USA) given in these references with the values calculated for the Swiss model inventory shows general agreement. To illustrate this 2 comparisons are shown in Table I.

TABLE I

Examination of the ratios of the activities for the PWR RCCA show that the components of the model inventory are 4 times more highly activated (long lived Ni-59, Ni-63) and have a shorter cooling time (short lived Mn-54 and Fe-55 have decayed less). More specifically it can be seen that the calculated Ag-108m and Ag-110m activities from the neutron absorbing alloy are only 50 % higher than the measured values when corrected for the higher activation.

Examination of the ratios of the activities for the BWR CRA show that the activation is the same (Ni-59, Ni-63) but the cooling assumed for the model inventory is longer (Mn-54, Fe-55 have decayed more). The discrepancy for C-14 is probably due to either the theoretical calculation or to conditions (classified CRA) not specified in Ref. 7. The crud activities (Sr-90, Tc-99, I-129, Pu-241 and Cm-242) are consistently, slightly higher than the measured activities.

CONDITIONING AND PACKAGING

Some particular types of components (poison curtains, fuel channels and incore instrumentation) that have been already discharged have been size reduced and conditioned with cement in 200 l drums. For the other components a conditioning and packaging study (8) was carried out by Nagra to define provisional waste packages

(i.e. number of components of a given type, cement quantity, internal shielding and container). The concept of a large container (18.9 and 20.2 m3) was used for the rod cluster control assemblies and the control rod assemblies. Some smaller thick-walled metallic containers (0.7 and 1.3m3), that were already available on the market, were assumed for the rest of the components types. A summary of the waste packages is given in Table II.

TABLE II

RESULTS

The detailed results of the characterization of the waste packages of the non-fuel exchangeable reactor core components (activities and material composition of the components, conditioning materials, waste package properties) can be found in the database of Ref. 1. In this paper, we present some overall results and some comparisons with the other better known waste categories (operational and decommissioning).

Figures 1 and 2 show for one of the PWR reactors and for one of the BWR reactors the activities of their packaged components expected to arise over 40 years. The activities 5 years after discharge from each reactor are summed without consideration of decay between the package production times and end of production. The grand total activities are compared with the activities of the operational and decommissioning wastes (defined 2 years after arising or NPP shut-down). The corresponding total waste package volumes of these 3 waste categories are also given. (Note: The total waste package volumes of the individual reactor core components haven been given in Table II).

Fig. 1.

Fig. 2.

The waste package volume of the reactor core components conditioned after 5 years cooling is 115 m3 / GW(e) for both PWR'S and about 1700 m3 / GW(e) for each BWR. They represent about 8 % of the PWR and 34 % of the BWR operational waste volumes. The representative percentages of the decommissioning waste are 0,4 % for old design and 1,5 % for recent design of PWR plants and 12 % for BWR plants.

Although the mass and volume of the exchangeable reactor core components discharged as waste are relatively small, their total activities lie between the activities of the operational and decommissioning waste as can be seen in Figs. 1 and 2. For all the reactors the total activity is found to be 20 times lower than the total decommissioning waste activity but 10 times higher the operational waste activity. Some radionuclide activity concentrations may exceed the waste acceptance criteria for subsurface disposal. The activities of the following long lived nuclides are greater than those of the decommissioning waste (all reactors considered together):

H-3 (12.3 y), Be-10 (1.6 10 6 y), Zr-93 (1.5 10 6 y), Ag-108 m (127 y), Cd-113 m (14.1 y), Sn-121 m (55 y), Sn-126 (1.0 105 y).

The calculated activities of the neutron sources from all reactors 5 years after discharge show the following features of interest:

- the long lived activities from the stainless steel (Ni-59, Ni-63, Nb-94) are 100 time lower than for all other reactor core components
- the H-3 activity (1 1015 Bq) is of the same order of magnitude than the activity of all other reactor core components (3 1014 Bq)
- the Cf-252 (2.6 y) activity is estimated to be 4 1010 Bq.

CONCLUSIONS

The non-fuel exchangeable reactor core components from the 5 Swiss nuclear power plants expected to be discharged as waste and packaged for disposal have been characterized for disposal projects. For such an objective, simplifications and assumptions have been made as it is not possible or relevant to consider the different irradiation histories of each of the individual components. However, as many as possible nuclide activities have been estimated and especially for the disposal safety relevant nuclides. For the first time the main features of this waste category are presented and then compared with other NPP waste categories. The radioactive waste community has almost not addressed these wastes because they are normally associated with the decommissioning waste (whereas a large fraction of them arise earlier), because they are advantageously stored for cooling before packaging due to their relative low mass and because of a lack of practical experience in characterization and packaging for most of them.

For effective packaging and disposal of individual components, the activity characterization could require significant effort if some activity concentrations

are around the limits of relatively cheap disposal. This explains the first recent U.S. publications on the experimental characterization of some real component activities based on the use of computer codes with dose rate, -rays or sample measurements. The applicability of these calculational codes (cross-sections and neutron fluxes) would certainly have to be improved especially for out-of-core materials as well as for neutron absorbing materials and their surrounding metals with respect to specific repository waste acceptance criteria and disposal route.

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36-5

TRIBAL EXPERIENCES WITH THE MRS PROCESS: A CASE STUDY OF THE FORT McDERMITT PAIUTE-SHOSHONE TRIBE

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ABSTRACT

In their efforts to find a voluntary host for a Monitored Retrievable Storage facility for spent nuclear fuel rods, the Department of Energy and the Office of the U.S. Nuclear Waste Negotiator entered into negotiations with a number of Indian tribes and states. Tribal governments and communities have had significantly different experiences with the MRS process than the states. One of the tribes most interested in hosting the MRS is the Fort McDermitt Pauite - Shoshone Tribe of northern Nevada. This small and isolated tribe has received little national attention, yet the Fort McDermitt story demonstrates the variety of political and social challenges a project like the MRS poses for tribal governments and communities, and illustrates as well some of the institutional barriers to, and requirements for, successful cooperation between the Department of Energy and Indian tribes. For example, once the tribal government decided to pursue the MRS, it became, to a degree, linked to the rising and falling fortunes of the Nuclear Waste Negotiator's Office. Understanding tribal experiences is important, both for the development of more effective waste management and disposal policies, and for achieving broader goals of equity in hazardous waste management. Hazardous waste policies, in particular, must come to reflect a better understanding of this type of

linkage, in order to become meaningful and acceptable to communities beyond the beltway.

BACKGROUND

In 1987, congressional amendments to the Nuclear Waste Policy Act authorized the creation of a single Monitored Retrievable Storage (MRS) facility, which would temporarily store spent nuclear fuel while a permanent repository was developed. Towards this end, Congress established the Office of the Nuclear Waste Negotiator, and empowered the Negotiator to work with the states and Indian tribes in order to find a willing host for the MRS. The Negotiator was given considerable leeway in developing a package of benefits for the potential host, including substantial financial incentives and a variety of public programs, such as health care, education, employment, and infrastructure development, as well as problem-solving assistance. There were several anticipated phases of negotiations: Phase I involved a $100,000 grant to study the potential of siting an MRS on state or tribal lands; Phase IIa involved an additional $200,000 for further study; and Phase IIb involved additional funding for site assessment and other expanded study, as well as entry into a contractual relationship between the parties. The moneys in Phase I and IIa were essentially "no strings attached" -- the tribe or state was free to use the funds however it wished, as long as the general goal of increasing knowledge about the MRS was pursued.

The financial incentives of siting an MRS were, obviously, considerable. In addition to the millions of dollars a year the federal government would pay the host for the facility, potential employment from construction through operation for the planned 30 year life-span of the MRS, as well as public assistance programs, would likely bring new economic prosperity to an entire community. In the balance against these economic incentives, however, were a host of very substantial public concerns and fears. Whether or not the public's fear of things nuclear is "rational", it is very real, and shapes public response to programs such as the MRS. As a result of public opposition, the Nuclear Waste Negotiator was able to find only a few states willing to discuss hosting an MRS, and only 15 Indian tribes. Of those 15 tribes, only 4 tribes proceeded, or wished to proceed, to Phase IIa. None of these negotiations have been successful, in the sense that none has resulted in the identification of a feasible MRS site. Although no MRS site has been found through negotiations with state governments, either, tribal governments and communities have had experiences with the MRS process that differ significantly from the states' experiences.

This paper will tell the story of one such tribe, the Fort McDermitt Paiute-Shoshone Tribe, whose reservation straddles the Oregon-Nevada border. The tribal government applied for and completed Phase I and IIa of the MRS process, and wished to proceed further, when the opposition of the Oregon Governor and the cessation of Congressional funding for the Nuclear Waste Negotiator's Office brought their participation in the MRS process to a halt. This small and isolated tribe has received little national attention, yet the story it has to tell is both common and unique, and demonstrates quite well the variety of political and social challenges a project like the MRS poses for tribal governments and communities. In addition, the Fort McDermitt story illustrates some of the institutional barriers to, and requirements for, successful cooperation between the Department of Energy and Indian tribes. The Fort McDermitt story is a cautionary tale, and demonstrates the internal political ramifications federal policies can have for tribal governments. Once the tribal government decided to pursue the MRS, it became, to a degree, linked to the rising and falling fortunes of the Negotiator's Office. Hazardous waste policies, in particular, must come to reflect a better understanding of this type of linkage, in order to become effective in the world beyond the beltway.

THE FORT McDERMITT CONTEXT

The Fort McDermitt Paiute-Shoshone Tribe is a small, isolated and extremely resource-poor community located on a rugged 34,650 acres of trust land on the Nevada-Oregon border. The tribe has approximately 800 members, about half of whom live on the reservation. Most of the tribal lands are leased, and so most of the tribe's growing population lives along a seven mile stretch of the Quinn River Valley, three miles south of the town of McDermitt in the Nevada portion of the reservation. The tribe has no on-going economic enterprises, other than its hayland leases, and most members of the community are dependent on government programs and subsidies. The median age on the reservation is 23 years, and an unemployment rate of around 70% plagues the tribe. Per capita income on the reservation is about

$3,000., and 70% of residents live below established poverty levels.
Services the tribal government is able to provide its members are limited, and include a Head Start facility in a decrepit old building with a leaky roof, a small and understaffed Indian Health Service clinic, and law enforcement provided by a Bureau of Indian Affairs officer who refuses to live on the reservation itself, and so lives in the town of McDermitt. The tribe has no mutual assistance agreements with the town, county or state law enforcement agencies, and these entities often refuse to assist the tribe, even in crisis situations. The housing and health care facilities on the reservation are sub-standard. The closest full-service Indian Health Service hospital is over 200 miles away, and the closest hospital of any kind is 70 miles away.
The lack of economic development on the reservation does not mean that the tribe does not prioritize development. There have been a series of failed enterprises, and there is continual dialogue within the community about what might be appropriate further development. Development efforts are hampered by the remote location of the tribe, its lack of financial and natural resources, and the relatively low educational and skill levels of many tribal members. Nonetheless, the tribe is currently considering a range of potential ventures, including the building of a dam and reservoir for tourist trade, a casino and truck stop along U.S. Highway 95, and the siting of a federal Monitored Retrievable Storage facility on the tribe's Oregon lands.
THE MRS PROCESS
According to tribal officials, Fort McDermitt paid little attention to the Office of the Nuclear Negotiator's first contact, inviting the tribe to apply for funds to study the possibility of hosting an MRS. Instead, when the tribe ran short of funds to pay for some major plumbing repairs, rather than go unpaid, the plumber suggested they contact a consultant friend of his about applying for a federal study grant he'd heard of. Although the tribe had been previously unsuccessful at obtaining federal grants, primarily because no audits of its existing programs had been conducted for years, the tribal council decided to apply for a Phase I study grant. In this round-about manner, Fort McDermitt itself initiated communication with the Negotiator's Office -- perhaps a harbinger of things to come.
The tribe's Phase I application was approved in July of 1992, but initial funding did not begin until the following October, after the tribe had hired an MRS project officer and spent weeks following a paper trail through the Department of Energy. The tribe spent the Phase I grant on hiring the tribal MRS project officer, setting up an MRS office, taking members of the tribal council on tours of related facilities, and holding one large public meeting. Phase IIa funding was spent on taking 36 more tribal members on three more tours, holding several more public meetings, continuing support for the MRS project office and officer, and meetings with county and state officials. Unlike some other tribes, who essentially used all their funds to hire outside consultants, Fort McDermitt attempted to manage its own way through the MRS process, with only part-time guidance from its Washington, D.C. attorney.
Difficulties began almost immediately, on several fronts at once. As one tribal official said, the Negotiator's Office gave the tribe its money, and very little else -- no direction, few suggestions, little concrete advice. By becoming involved with the MRS issue, Fort McDermitt opened a Pandora's box it was ill-prepared to cope with, from the sincere questions of its members to the vocal opposition of neighboring county and state entities, from the arrival of anti-nuclear activists, media and other interested parties on the quiet reservation to a degree of public scrutiny unlike anything the tribe had ever experienced.
On the domestic front, tribal members were uncertain, confused and even frightened about the notion of storing hazardous waste on the reservation. The money and the jobs sounded enticing, particularly in the economic context of Fort McDermitt. But nuclear waste is a unique issue, one which has a special capacity to raise specters and fears of the unknown, and at Fort McDermitt, tribal officials were hardly reassuring figures. Politics at Fort McDermitt are characterized by a high level of mistrust and conflict, and members were as likely to see the MRS as another opportunity for government corruption as they were to view it as a beneficial public project. Questions of corruption aside, however, tribal officials were limited in their ability to provide clear and complete information to tribal members. Their knowledge of the issue was too clearly based on the same Department of Energy

publications that were available to tribal members, and members frequently had to wait for the MRS project officer to relay their question to the Negotiator's Office -- a process that was not confidence-inspiring.

Compounding the difficulties of internal dialogue was the degree of external attention focused on the tribe. Greenpeace and other environmental groups were present at the first community meeting, and worked hard over the following months to organize opposition to the MRS, from within the tribe and from neighboring communities. Opponents of the MRS raised environmental, health and safety issues, questioned the degree of public trust and confidence in the tribal and federal governments, and posed ethical questions regarding the appropriateness of hosting an MRS, given the traditional connection of Indians and the earth. Officials of the surrounding counties questioned the capacity of the tribal government to manage a project as complex as the MRS, (despite the fact that the MRS would be managed by DOE, rather than the tribe), and local townspeople made disparaging remarks about the level of alcoholism on the reservation, and their resulting fears about allowing the tribe to become involved in anything so potentially dangerous -- and lucrative -- as the MRS. The press and other media capitalized on the more sensational aspects of the debate, and rarely managed to produce complete or objective articles and stories. When the Governor of Oregon announced her opposition to the MRS, even her carefully-worded letter to the tribal council betrayed a less than complete understanding of the MRS project, process and safeguards.

FLAWS IN THE PROCESS

Clearly, the decision to become involved in the MRS process was an independent decision made by tribal officials, and they alone are responsible for choosing this course. However, they did have reasonable expectations of assistance from the Negotiator's Office and the Department of Energy. Throughout this trying period, however, tribal officials found themselves with an unexpected and damaging lack of support. The MRS project officer recalls that her first job was to figure out what to do: "They handed out the money, and then sat back and just left us be. I couldn't believe it -- here's $100,000, with no direction, nothing." After the first community meeting, it was over a year before any federal officials involved in the MRS project returned to the reservation -- they sent lots of written materials, but in a place like Fort McDermitt, government pamphlets have a limited usefulness. Tribal officials were on their own to face the storm of external opposition, to educate and inform their members and other concerned parties, to discover and develop effective ways of conducting their public dialogue and assessing the feasibility of hosting the MRS. Despite the Negotiator's wide latitude in offering assistance to the tribe should it decide to host the MRS, during the crucial public discussion phase, the MRS project officer remembers "begging" the Negotiator's Office for some sort of media training, and receiving no assistance whatsoever. Compounding the lack of non-financial assistance, the tribe found itself caught in the institutional tug-of-war between the Negotiator's Office, the Department of Energy, and Congress. Although the Negotiator's Office was empowered to promise funding to MRS applicants, the actual moneys were distributed by DOE. This bifurcated process put the tribe in the position of receiving promises from the Negotiator, but having to establish an entirely different set of working relations with DOE in order to have those promises fulfilled. Fort McDermitt is not a tribe with extensive political experience or sophistication, and overcoming the institutional barriers commonplace to the federal government was a severe drain on its resources.

To further complicate the picture, Congress itself began to take a closer look at the MRS issue. Important members of Congress were reaching the conclusion that the MRS was politically unfeasible, and possibly an inefficient, unnecessary and/or dangerous obstacle in the path of the development of a permanent repository. This new attitude in Congress forced the Negotiator's Office to begin fighting for its very survival -- leaving it with less and less time and financial resources to devote to its previously primary task of negotiating with potential hosts.

More and more time elapsed between contacts with Fort McDermitt, more and more uncertainty developed within the relation between the tribe and the Negotiator's Office, and more and more tribal members and opponents of the MRS interpreted this uncertainty and delay as a sign that things were not going well. Tribal officials' inability to provide concrete answers was seen by some as further evidence of inability or untrustworthiness and opponents capitalized on this attitude. When

funding did not arrive as planned, some tribal members believed, (or were encouraged by MRS opponents to believe), that tribal officials had embezzled the money, rather than accept explanations that Washington was delaying. When tribal officials expressed uncertainty in their discussions with county and state officials, this was interpreted as their own lack of knowledge or inability to make decisions, rather than as a direct reflection of the unstable position of the Negotiator's Office. Some tribal members, those who had voiced their misgivings of the federal government at the start, now blamed the tribal government for having gotten involved with the feds at all.

In short, the public perception of the legitimacy of tribal government actions became linked with the rising and falling fortunes and attentiveness of the Negotiator's Office. Worse yet, this linkage was seen by many as the fault of tribal officials, rather than a result of inattentiveness, or error on the part of the Negotiator's Office.

CONCLUSIONS

To be sure, the Nuclear Waste Negotiator's mandate was not to hold the hands of interested tribes, and, indeed, too close a connection between the tribal government and the Negotiator's Office might have appeared to compromise the political autonomy of the tribe's decision. The decision to host an MRS was, however, supposed to be an informed decision, a public decision, a political decision based on concrete information as well as extensive debate and dialogue. All parties concerned knew that any discussion of hazardous waste disposal was likely to be complicated by public fears and concerns, sensational press coverage, and difficult decisions about acceptable risks and benefits. The process of simply handing tribes $100,000. to study an MRS led to a number of tribes in turn handing the money to various consultants -- leaving the tribe no richer in knowledge than it began, and certainly no more likely to host an MRS.

At Fort McDermitt, tribal officials made a concerted effort to engage in a serious investigation of the feasibility of an MRS on tribal lands. In this effort, they received unconscionably little assistance from the Negotiator's Office or the DOE. As a result, the tribal government was left on its own to determine how to conduct its investigation, an intrinsically inefficient process of reinventing the wheel over and over again, as well as a potentially politically devastating experience. When the Negotiator's Office began its dealings with Fort McDermitt, it could have made some effort to determine the resources and capacities of the tribal government, and figured out suggestions for proceeding that were contextually appropriate. It could have offered media training, or recommended some other entity for the purpose. It could have prioritized consistent communication with tribal officials, and made itself more available as a resource to tribal members and officials. It could have brokered the relationship between DOE and the tribe, rather than allowing a bifurcated funding process to confuse tribal officials and drain tribal resources. It could have used its own resources to help ensure that the debate which buzzed around Fort McDermitt included all the pertinent facts and information, rather than allow it to be permeated with basic misunderstandings of the MRS and MRS process. Instead, the Negotiator's Office engaged Fort McDermitt in the MRS issue, promising funding and assistance, and then left tribal officials largely on their own to face a situation more complex than any of them could have anticipated. Despite the fact that Congress has halted further funding for the search for a willing MRS host, and despite the fact that the opposition of the Oregon Governor to an MRS on Fort McDermitt's Oregon lands is the equivalent of a political death-knell to the project, controversy around the MRS issue continues to swirl at Fort McDermitt. Months after the cessation of funding, a consortium of environmental groups held an anti-MRS conference in the town of Fort McDermitt, bringing in MRS opponents from other tribes who managed to resurrect tribal members' concerns about mismanagement of their MRS funds and the trustworthiness of their tribal officials. Independent media types are still arriving on the reservation, pursuing leads about corrupt tribal officials and naive tribal members.

Whether or not Fort McDermitt ever hosts an MRS, the conflict and mistrust that have followed from the tribal council's decision to simply study the issue will linger in the tribe's political arena for years to come. Certainly, tribal officials bear the responsibility for the outcome of their own choices. Just as certainly, however, the Office of the Nuclear Waste Negotiator, the Department of Energy, and Congress bear the responsibility for the creation and pursuit of a public policy which failed

utterly to take into account the unique aspects, resources and vulnerabilities of tribal contexts. The MRS policy, as pursued by these entities, also failed to take into account the dilemmas of hazardous waste disposal as a public issue, an issue surrounded by complex public fears and concerns, and one which requires very careful planning for public participation and responsiveness. Because of these failings, there is no feasible MRS site today. Just as importantly, the government of a small, isolated and impoverished tribe has lost a significant portion of its most valuable resource, public trust and confidence -- as has the federal government.

Session 36-A--Panel Discussion: Alternative Technologies for the Stabilization of DOE Spent Nuclear Fuels and Other Reactor Irradiated Materials
Co-chairs: Chris Whipple, ICF Kaiser;
Dieter A. Knecht, Lockheed Idaho Technologies Co.
36-A-Panel
A panel discussion was held to present a wide range of perspectives on alternative technologies for stabilizing DOE spent nuclear fuel and other irradiated materials. Panelists included Jim Werner, Director of Strategic Planning and Analysis; Jim Gallagher, Executive Vice President of Westinghouse; Marilyn Meigs of BNFL; Brian Costner, Director of Energy Research Foundation; and Alan Hoskins, manager of INEL Spent Nuclear Fuels Program of Lockheed Idaho Technologies Co. and co-chairs included Dieter Knecht of Lockheed Idaho and Chris Whipple of ICF Kaiser and chair of the National Academy of Sciences Board on Radioactive Waste Management. Each panelist gave a 5-minute presentation, followed by audience questions and answers for the rest of the session.
In the panelist presentations, Jim Werner introduced the topic concept and provided a brief overview of technical options, including processing the fuel, without separation of fissile materials, as a means of stabilizing some of the fuels. Jim Gallagher presented the progress at Hanford and path forward, including providing a more rational regulatory structure to modify overlapping regulations, using the NEPA process with input from all stakeholders, and proceeding with the engineering and management talent to achieve the desired results. Marilyn Meigs offered an option based on BNFL's experience with processing of metallic Magnox fuel, which has similar characteristics to N-reactor fuel. Brian Costner gave reasons why reprocessing should not be the "default" option and offered a path forward in which decisions should be based on the following criteria: to resolve the health and safety risks, resolve the technical issues with long-term storage and final waste form, and re-evaluate the costs of the options rather than using creation of jobs and the old self-regulation as a basis. Alan Hoskins provided the wide range of characteristics of DOE spent nuclear fuels and technical options currently under consideration. Questions and comments were lively and covered a wide range of perspectives, ranging from technical questions concerning acceptability of minimal fuel treatment to policy of overall repository viability and timing. There were strong views both against and for reprocessing, but there seemed to be some agreement that the presumption in favor of reprocessing that existed in the "old" days was no longer considered to be valid. As an indication of the success of the panel discussion, the chair of the State of New Mexico Environmental Evaluation Group expressed his thanks that an open, free discussion on this topic was held.
Session 37 -- Poster - Mixed Waste I
Co-chairs: Earl W. McDaniel, ORNL;
Terry Sams, MMES;
Jeanette Berry, ORNL
37-1
MIXED WASTE INTEGRATED PROGRAM WASTE DESTRUCTION/STABILIZATION TECHNICAL AREA PROGRAM HIGHLIGHTS*
John McFee
IT Corporation
Jeanette Berry
Martin Marietta Energy Systems
ABSTRACT
The Mixed Waste Integrated Program was established to plan, manage, and integrate the U.S. Department of Energy's mixed waste programs for research, development, demonstration, testing, and evaluation of innovative treatment technologies. The current status of the program is described including activities in process systems

integration, regulatory interfacing, and development activities in the waste destruction and stabilization technical area.

INTRODUCTION TO MIXED WASTE INTEGRATED PROGRAM TECHNOLOGY DEMONSTRATIONS

The Mixed Waste Integrated Program (MWIP) has coordinated technology development and demonstrations throughout the U.S. Department of Energy (DOE) and in private industry as needed to generate design data to improve mixed waste treatment processes. The MWIP has endeavored to meet the following objectives:

Conduct pilot-scale demonstrations of mixed waste treatment systems, treating actual waste within three years. Cumulatively, these pilot-scale demonstrations will be capable of treating a minimum of 90% of the current mixed low-level waste inventory.

Transfer technologies to industry for commercial use and application to DOE sites and identify opportunities to retrofit existing projects with more effective technologies.

Demonstrate and validate commercialized technologies to contribute to world leadership of U.S. industry in environmental science and technology.

Typically, experiments conducted to develop technology are based on a single-unit operation and use surrogate wastes, although in certain cases an actual waste stream may be used. Confidence in the performance of emerging technologies is gained through demonstration of complete systems on actual wastes. The confidence of waste management customers in system performance increases the likelihood that an emerging technology will be implemented to treat DOE mixed waste. MWIP demonstrations are geared toward combining several unit operations using actual waste streams and are typically conducted at a DOE site (see Fig. 1). Demonstrations in progress are the fixed hearth plasma arc furnace system, being demonstrated at Argonne National Laboratory-West with technical oversight from the Idaho National Engineering Laboratory, and a vitrification system, being demonstrated at Oak Ridge National Laboratory, with technical oversight from the Savannah River Site.

The MWIP recognizes that technology development must address each unit operation as part of a treatment system to ensure that all treatment system elements are compatible and that the resulting system is an improvement over the alternatives. The fixed hearth plasma demonstration consists of several principal investigators whose combined work ensures that development efforts result in a technically successful integrated process system. The Program coordinates oversight of each project and interacts with representatives of the site that is hosting the project demonstration. The Program is critical in moving unit operations from the development stage of an integrated system to the demonstration stage, while ensuring that results are directed toward resolving key problems.

Both the plasma and the vitrification demonstrations have significant involvement from the private sector, including universities. In FY94, 48% of funding was directed to private industry, and 16% of funding was directed to universities. Therefore, the goal of obtaining industrial and academic involvement in project success has been met.

This paper summarizes activities associated with the development of the plasma hearth process (PHP), as well as other MWIP activities undertaken by the Waste Destruction and Stabilization (WD/S) technical area. Detailed reports on the technical progress of many MWIP-supported development activities are presented separately in this conference.

PROCESS SYSTEMS INTEGRATION THROUGH THE TECHNOLOGY RESOURCE TEAM

A technology management process, depicted in Fig. 2, was published in "A New Approach to Environmental Research and Technology Development at the U.S. Department of EnergyAction Plan," January 25, 1994. It provides the framework for identifying technology needs and developing effective, acceptable technology solutions. Integral to the design of this technology management process is teaming: the successful integration of the needs of potential technology users, suppliers of technology-based solutions, and stakeholders.

The Technology Resource Team (TRT) was established as the technology resource arm of the Mixed Waste Focus Area, and the MWIP has contributed to its success. The team fosters improved communications and coordination among customers and the research and development community to overcome obstacles to progress in solving mixed waste problems and to achieve the goals of the department's new approach to technology development. Members take a proactive approach in providing technical information (e.g., life-cycle cost and performance data), technical assistance, and tools (e.g.,

decision-making tools and design guidance) for the evaluation and selection of technologies and treatment systems. Members work closely with customers to address and resolve issues associated with the evaluation and selection of mixed waste technologies, treatment systems, and facilities. The TRT has supplied technical experts to help accomplish the following:

- Communicate technology development activities, capabilities, and benefits to customers
- Define DOE complex and site needs data from the Mixed Waste Inventory Report, Conceptual Site Treatment Plans, Draft Site Treatment Plans, Preliminary Site Treatment Plans and through site interactions
- Evaluate the results from the development and demonstration activities
- Establish consensus with regard to engineering systems and analysis tools
- Assist in the selecting, recommending, and implementing emerging technologies to comply with the Federal Facility Compliance Act (FFCA) at sites requiring them.

REGULATORY INTERFACES

A strategic objective stated in the "Mixed Waste Characterization, Treatment, and Disposal Focus Area Strategic Plan for Technology Development," April 4, 1994, is as follows: "Gain regulatory acceptance of new technology . . . through early involvement by the regulators . . . as indicated through achieving permitting of developed technologies."

Interagency contacts by which this objective can be achieved have been developed. The key in gaining acceptance of new technology is to involve stakeholder, regulators, and end-users in the technology development process. To this end, the MWIP and the National Technical Working Group (NTWG) have developed a proposal to exercise a DOE/U.S. Environmental Protection Agency (EPA) Interagency Agreement. The proposal includes the following tasks:

- Develop a permit roadmap for emerging mixed waste technologies
- Prepare waste analysis plans for nonincineration processes (characterize waste to treat)
- Develop a trial burn test plan
- Implement a technology review and exchanges.

Furthermore, the Office of Technology Development has identified a federal employee to facilitate regulatory and stakeholder interaction, including the National and Western Governors' Associations. A Draft Site Implementation Plan outlines actions necessary to obtain permits for the fixed hearth plasma demonstration to be conducted in Idaho. These activities provide consistency for stakeholder involvement.

PLASMA HEARTH PROCESS DEVELOPMENT ACTIVITIES

One of the technical areas in which technology development is required is the WD/S area. In past years this technical area sought, evaluated, and supported selected waste treatment technologies for application across the DOE complex. Recently the support has been focused on demonstration and development of the PHP and the associated technical and programmatic concerns.

The PHP is being developed as an waste treatment technology that can accept nearly all of DOE's solid mixed waste and convert the waste to melted slag and metal. This medium- or large-capacity system is capable of addressing a major waste treatment issue for the DOE. Recent estimates of the amount of DOE waste that can be accepted by a PHP system exceed 150,000 cubic meters. Figure 3 provides a "block diagram" of the PHP process. Included with the block diagram is an indication of the related design issues associated with each processing step represented by the block and the currently funded Technical Task Plans (TTP). The following discussion summarizes those issues and the related TTPs.

The DOE waste from current operations, waste retrieved from past activities, anticipated environmental restoration, and anticipated decontamination and decommissioning wastes all require characterization prior to treatment to ensure compatibility with the process and compliance with applicable permits. The PHP is capable of accepting drummed waste without shredding and with few process-imposed limits on the waste constituents. Because the process accepts a wide variety of wastes, MWIP is supporting an enhanced waste-package characterization technique in the PNFA process. The goal of the characterization process is reduction in the hazards of manual waste characterization. The status of this characterization program is being reported separately by P. Lange (1) in the conference.

Additionally, the MWIP is participating in the joint EPA/DOE NTWG to address

regulatory issues related to waste characterization requirements and their impact on DOE thermal treatment systems. Typically, Resource Conservation and Recovery Act permits for thermal treatment systems require complete waste characterization to ensure compliance with the permit conditions. The NTWG is discussing the need for this complete characterization of mixed waste for PHP processing. Relaxation of the characterization requirements could reduce operating costs and reduce risks associated with handling and manual sampling for characterization.

The next block in Fig. 1 represents the PHP primary chamber. The PHP can accept whole drums of solid and sludge wastes. This has been demonstrated by the processing of three MWIP specified wastes in FY94. Organic sludges, inorganic sludges, and heterogeneous debris were successfully processed, resulting in a slag phases that pass the EPA's toxicity characteristic leaching procedure. These tests convincingly demonstrated that the process can accept combustibles and noncombustibles with nothing more than automatic control system correction of the air flows. These tests and other significant test program results are reported by R. Geimer (2) in this conference.

The PHP testing continues to refine the designs and confirm operability using a nonradioactive pilot plant, a nonradioactive bench-scale system, and a planned radioactive bench-scale system. Three design issues being studied are melt/slag removal, radionuclide partitioning, and refractory selection for improved crucible lifetime. These design studies are reported by R. Geimer (2), R.L. Gillins and S.D. Poling (3), C. Dwight et al. (4), and A.L. Keilpinski et al. (5) at this conference. A separate program is evaluating plasma torch design improvements, using thin-plate cooling as a means of extending plasma torch lifetime.

The PHP uses a secondary combustion chamber (SCC), or afterburner, for complete oxidation of organics in the gas phase. This is represented by the third block in Fig. 1. Designs of secondary combustion chambers are well-developed and not expected to be a problem. An alternative SCC design is being designed for the bench-scale radioactive pilot plant to demonstrate an nonincineration alterative. The use of a plasma-heated (nonfossil fuel) SCC is reported by R. Geimer (2).

The DOE expects that PHP gaseous emissions will meet all regulatory requirements, because commercially available off-gas treatment systems have been demonstrated for numerous thermal treatment systems. Alternatively, the DOE is considering a minimum emission system demonstration using the PHP as a testing platform. The controlled emissions program was begun in FY94 with an identification of the options available:

Off-gases monitored on-line for process shutdown in excursions

Off-gases monitored on-line with automatic diversion to either holdup systems or additional treatment systems

Off-gases captured, monitored, and released after acceptable test results.

The preferred processes to meet the objectives of these options have been identified, and design studies are underway at DOE's Western Environmental Technology Office in Butte, Montana.

To meet the monitoring requirements of the controlled emission demonstrations, as well as to address a common stakeholder concern, the MWIP is supporting development of two-stack monitoring systems for on-line detection of off-gas pollutants. R. Gritzo (6) reports on an alpha detector that is being developed for real-time monitoring of airborne alpha emissions, and D. Otteson et al. (7) report on the use of tuneable diode lasers for continuous monitoring of volatile organic compounds. Both activities are reported elsewhere in this conference.

The current PHP design minimizes secondary wet wastes using an off-gas cleanup system that first removes all particulates in a baghouse, followed by processing through high-efficiency particulate air filters and removal of the acid gases in a wet scrub system. Spent scrubber solutions are expected to be a problem in the DOE's treatment systems, and MWIP has supported an innovative immobilization technology: chemically bonded phosphate ceramic waste forms. This activity is reported by D. Singh and A. Wagh (8) and A. Wagh et al. (9) in this conference. Polymer solidification of waste is being investigated by J. Heiser et al. (10), and polyethylene encapsulation demonstration is described by P. Kalb et al. (11). E. Franz et al. (12) have developed proposed performance criteria to be met by these waste forms.

STEAM REFORMING

The MWIP steam-reforming program has been restrained due to funding limitations in for this fiscal year. Testing has recently been resumed for the demonstration of

selected MWIP wastes. These wastes have been selected from a specified list of defined surrogates developed under MWIP funding. In FY94, only program planning activities were accomplished. The status is summarized by L. Bustard et al. (13) in another paper in this conference.

TECHNICAL SUPPORT ACTIVITIES

The WD/S technical area participated in several activities supporting DOE Headquarters (DOE-HQ) and field offices this year. Most significant is the support of DOE-HQ in the review of the site treatment plans being developed by field offices in compliance with the FFCA. The WD/S technical area reviewed several conceptual-site treatment plans, two-draft site treatment plans, and two proposed-site treatment plans. These reviews provided a technical review for DOE-HQ and identified potential applications for innovative technologies. The WD/S technical area also supported three field office activities, with senior technical review of proposed designs for waste treatment systems.

A recent development in the MWIP is the interest in alternatives to thermal treatment. In response, MWIP is developing a Technical Area Status Report to summarize low-temperature waste destruction processes available or under development. This report identifies numerous alternatives available for gaseous or aqueous wastes, but limited alternatives were identified on solid waste oxidation using low temperature processes. K. Dickerson and C. Brown (14) report on an incineration alternative in this conference.

CONCLUSION

The MWIP has been supporting numerous research, development, and demonstration activities focused on treatment of the DOE's mixed low-level waste problems. The PHP has shown good promise of being capable of treating a wide spectrum of the DOE's waste. The MWIP has developed a complex, interrelated program of TTPs addressing waste characterization, waste treatment, secondary waste treatment, waste minimization, and process monitoring designed/developed to bring the PHP process to a successful and rapid demonstration. Successful demonstration, scheduled for early FY96, will substantially address stakeholder concerns over the DOE's current inability to treat mixed waste.

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RECENT PROGRESS IN THE DEVELOPMENT AND DEMONSTRATION OF THE PLASMA HEARTH PROCESS FOR MIXED WASTE VITRIFICATION

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ABSTRACT
The Plasma Hearth Process (PHP) demonstration project is one of the key technology projects in the Department of Energy (DOE) Office of Technology Development (OTD) Mixed Waste Integrated Program (MWIP), with potential application in the treatment of a wide variety of DOE mixed wastes. The ultimate goal of the project is a comprehensive field-scale demonstration of the PHP system for treatment of actual DOE waste. The field-scale demonstration is currently targeting alpha-contaminated low-level waste now in storage at the INEL.
Initial project activities during testing and development of the PHP have been and continue to be conducted on a nonradioactive basis. A relatively simple near full-scale batch operated PHP system was designed and constructed for Phase I proof-of-principle testing. Testing has been conducted on a wide range of simulated DOE waste types representative of a large fraction of the mixed waste now in inventory. Most recent testing for the MWIP has involved the testing of waste matrices having been spiked with hazardous constituents (organics and heavy metals) and surrogates for radioactive materials. Extensive sampling of the products and the offgas from the process was conducted. Destruction of principal organic hazardous constituents (1,2-dichlorobenzene and naphthalene) was found to range from 99.996% to >99.9999%. Heavy metals spiked into the feeds were consistently far below Toxicity Characteristic Leaching Procedure (TCLP) limits in the vitrified product. Data from the analysis of stack emissions were compared to EPA requirements, and particulate and metal emissions were well below emission limits.
Additional testing has been conducted on pilot-scale hardware to evaluate the performance of materials in the processing zone and the mechanics of material handling. Several types of high temperature refractories have been evaluated, and some found to perform much more reliably in the aggressive processing zone. Several methods for handling materials have been explored, particularly in the separation of molten products of steel and slag, and their subsequent removal from the processing chamber. Based on these tests, design of a fully integrated pilot-scale PHP and air pollution control system capable of continuous operation is currently in progress.

INTRODUCTION

The Plasma Hearth Process (PHP) is recognized as one of the more promising solutions to DOE's mixed waste treatment needs, with potential application in the treatment of a wide variety of DOE mixed wastes. The PHP development project is one of the key technology projects in the Department of Energy (DOE) Office of Technology Development (OTD) Mixed Waste Integrated Program (MWIP). This ultimate goal of this project is a successful demonstration with a field-scale PHP system for treatment of actual DOE mixed wastes. The field-scale demonstration is currently targeted for treatment of 55-gallon drums of alpha contaminated low-level waste now in storage at the Idaho National Engineering Laboratory (INEL).
The PHP is a high temperature vitrification process using a plasma arc torch in a

refractory lined chamber to destroy organics and stabilize the residuals in a nonleaching, vitrified waste form. Plasma arc technology is an innovative technology that has exhibited commercial success, primarily in its use for production of high purity alloys and other specialty metals. This project involves the adaptation of existing, well understood, and commercially available technology. The key characteristics of the PHP include: 1) extremely high temperature operation that completely destroys organics; 2) creation of a very stable, leach resistant final product; 3) the ability to accept a very wide range of waste types without pretreatment; 4) the ability to treat waste without removing it from the container; 5) generation of separate slag and metallic phases, allowing segregation and possible reuse of the metal; and 6) the preference of many radionuclides (especially actinides) to migrate to the stable slag phase. A more complete description of the PHP concept, its inherent operating characteristics, and its benefits is available in previous references (1-4).

Table I outlines the various PHP test systems that will be used for the technology development project. The terms "pilot-scale," "bench-scale," and "field-scale" are relative to this PHP project. Phase I involved batch tests using a proof-of-principle (POP) system. Phase I is now completed, and the results of these tests are the principal subject of this paper. Successes during this initial battery of tests led to the establishment of the multi-phase PHP development project. Phase II, recently initiated, is divided between nonradioactive pilot-scale (NPS) testing and radioactive bench-scale (RBS) testing. The radioactive field-scale (RFS) PHP system represents Phase III of the PHP development project, the final field-demonstration of the treatment of INEL alpha-contaminated low-level waste.

TABLE I

In addition to the above PHP systems, a nonradioactive bench-scale (NBS) PHP system is available for testing and research in the demonstration project. This system, commercially funded and operated, is located in the Science Applications International Corporation (SAIC) Science and Technology Applications Research (STAR) Center. The STAR Center system supports all phases of the demonstration project. It is used for short-term, small-scale tests where a design or operating issue can be resolved quickly and cost-effectively. Figure 1 presents a photo of the NBS hardware at the STAR Center.

Fig. 1.

The NPS, while designated as a pilot-scale, is essentially a full-scale system. The NPS will demonstrate continuous processing of 55-gal drums containing nonradioactive surrogate waste materials. Hardware and system operability and conformance to environmental criteria will be the focus of the NPS testing. RBS testing, using a PHP approximately 1/10th of the scale of the NPS, will include radioactive materials to assess the fate and partitioning of radionuclides within the process. The RFS will be essentially the same size as the NPS. Upon a successful field demonstration, the RFS will be considered for conversion to a fully operational treatment system for DOE mixed wastes.

Initial project activities during testing and development of the PHP have been and continue to be conducted on a nonradioactive basis. The nonradioactive testing segment of the project has two major goals which will aid the implementation of the field demonstration: to provide development and confirmation of the functionality of the PHP hardware; and to ensure that the process will produce results that are acceptable from a regulatory and safety standpoint. The culmination of the nonradioactive development effort is the NPS system. With the NPS, PHP development and implementation will proceed rapidly, and with reduced implementation risk because it will operate at near-field demonstration rates.

The remainder of the paper consists of three sections. The first section summarizes the results of nonradioactive testing using the POP system. The second section focuses on advanced concept tests using both the POP and NBS systems. The third section incorporates these results into the ongoing design effort for the NPS system.

PROOF-OF-PRINCIPLE RESULTS

A relatively simple, batch-mode PHP system was constructed in 1991 to test the PHP concept of waste treatment. Figure 2 presents a schematic diagram of this proof-of-principle system. Figure 3 is a photo of the POP primary melter hardware.

Fig. 2.

Fig. 3.

Three test campaigns (1-4) have been conducted with the POP system. Listed in Table II, the simulated wastes types tested represent a large fraction of the current DOE mixed waste inventory. Of the three campaigns, the MWIP tests included the most diverse waste types and the most extensive sampling and analysis. The results from the previous two campaigns were consistent with corresponding data from the MWIP tests; thus, the following discussion highlights the MWIP test results.

TABLE II

Table III summarizes key results from the MWIP tests. Each test involved processing two 30-gal drums of material. The test approach included processing the drums at an approximate rate of one per hour. Actual processing times varied from 44 to 81 minutes per drum, which translate into the feed rates listed in Table III. No attempt was made to find the a maximum feed rate for the different waste types; the feed rate could have been accelerated in most cases.

TABLE III

The only particulate emission control device employed with the POP system was a conventional baghouse. Tests with all feeds listed in Table II included stack sampling for particulate matter. Emissions were consistently below the proposed regulatory limit of 0.015 gr/dscf; MWIP data are shown in Table III.

The two organic sludge tests included determination of destruction and removal efficiency (DRE) for two POHCs. The concentration of each POHC in the organic sludge was 3.5% by weight. Table III includes the DRE results. Test 6 involved a process upset (temporary shutdown of the plasma torch) caused by an electrical short. While this upset apparently decreased the DRE for naphthalene compared to the previous test, the DRE still exceeded regulatory requirement of 99.99%.

Throughout the PHP development project the vitrified product (slag) consistently has been highly leach resistant, even though no effort has been made to optimize the product composition. The feed for the MWIP tests included spiking each of the RCRA metals cadmium, chromium, lead, and nickel to 0.1% in the waste. The resulting slags passed TCLP even though their RCRA metal concentrations were sufficiently high to flunk the TCLP had they not been leach resistant. In several cases, the resulting slag was more crystalline than glassy (due to a lack of glass forming constituents in the feed material), but still demonstrated a high degree of leach resistance. Table III includes TCLP results for the spiked metals (nickel is not a criteria metal for TCLP).

The integrity of the slag matrix from four of the six MWIP tests was evaluated using the Product Consistency Test (PCT). DOE scientists developed the PCT for assessing vitrified high level waste (5). Table IV presents the PCT results in comparison to published data (5,6). All slags with the exception of heterogeneous debris slag from Test 3 showed similar or superior leach resistance for the common analytes (Si and Na) with respect to the benchmark EA Glass. Even the more crystalline PHP slags compared favorably with engineered glasses for which the PCT is normally applied. Given that optimization of waste form performance was not an objective of the MWIP tests, the leach resistance performance of the slags is encouraging.

TABLE IV

Of particular note is the extremely low leachability of the cerium from the slags. Cerium (a plutonium surrogate) was spiked into the wastes at a concentration of 0.3%. The cerium consistently partitioned (94% to >98%) to the slag product. This result is desirable, as the radionuclide can be expected to be bound within a demonstrably stable matrix. Another 1% to 2% partitioned to the metal product that is recovered separately from the slag. The less volatile RCRA metals also favored the slag phase, although nickel has a significant propensity for the metal product. The volatile metals were captured to a high degree with the baghouse dust. Stack emissions were compared to Clean Air Act Tier II requirements and found to be well within emission limitations (4).

DESIGN SUPPORT STUDIES

The next step in the (nonradioactive) PHP development is to transition from a batch-operated process to a continuous process. Therefore, studies to support the design of a pilot-scale system with continuous capability have been a recent focus of the program. These tests allow various design concepts to be tested in either the POP or NBS systems to examine their efficacy. Based on actual test results, these design concepts are incorporated into the design, modified and re-evaluated, or discarded. The majority of the tests have focused on two areas: slag/metal separation and materials of construction.

Slag/Metal Separation and Removal

In the crucible, the molten slag and metal phases are naturally immiscible and the metal phase collects at the bottom of the crucible because it is much denser than the slag. The current PHP design approach is to recover the slag and metal phases separately in order to present DOE with an opportunity to recycle the metal while further reducing the volume of material requiring disposal.

Product removal was evaluated systematically in a series of focused tests; SAIC utilized both the POP system and the NBS system for this work. First, isolation of the molten metal phase from the slag phase by means of an underflow weir was evaluated. Once isolated in a collection area adjacent to the crucible, the metal would be relatively easy to remove by pouring. The challenge was to keep the metal molten as it passed through the underflow weir. The successful approach included high temperature, thermally conductive materials to keep the vicinity of the weir hotter than the freezing point of the metal. The hydrostatic head of the slag resting on the metal caused the metal layer on the other side of the weir to rise above the height of the metal layer in the crucible. Removal of the slag phase by tilt pouring was the next operation demonstrated. Here, the challenge was to remove the slag by tilting the crucible to pour off the slag while still maintaining isolation of the metal separated via the underflow weir. The POP system was used for these tests.

The NBS system was used to evaluate simultaneous separation and removal of the slag and metal from the crucible. Again, an underflow weir was used to separate metal from slag. Both molten streams were poured into collection vessels by continuous overflow instead of by tilt pouring. A small amount of metal "short-circuited" to the slag overflow spout and was found in the slag collection vessel. This confirmed the advantages of tilt pouring over continuous overflow for slag and metal removal.

Materials of Construction

Areas in the PHP where materials of construction are an important issue are: the crucible which contains the molten slag and metal, and the ground path that carries the plasma current from the molten bath out to electrical ground. Several types of high temperature refractories have been evaluated for the aggressive processing zone in the crucible. Alumina-chromia, magnesia-chromia, high alumina, magnesia, magnesia/carbon, and silicon carbide refractories were tested. A variety of grades and forms (brick, castable and rammable) of these materials were included in the evaluation. Alumina-chromia and magnesia-chromia (in brick form) are the best performers in the corrosive environment created by contact with the slag. All the materials tested held up well in contact with the molten steel phase under nominal waste feeds. However, certain waste feeds high in water content or other oxidizers caused failure of silicon carbide refractories whether in contact with the slag or the metal. Materials of construction tests have been conducted on both the POP and NBS systems.

Although water-cooled steel typically served as the ground path, tests in the NBS have shown that magnesia/carbon refractories are conductive enough to serve as a ground path. This enables water-cooling to be minimized, if not eliminated.

NONRADIOACTIVE PILOT-SCALE SYSTEM DESIGN ACTIVITIES

Successful product removal concepts and promising refractory materials identified from the design support studies are being incorporated into the design of the NPS. The NPS will demonstrate continuous PHP operation while treating whole 55-gal drums of nonradioactive waste materials. Thus, the NPS will include waste feed and product removal systems that will allow continuous operation. The design basis for the NPS is 1,000 lb/hr of a defined nominal heterogeneous feed. As with the MWIP tests, NPS test parameters will include destruction of RCRA organics and partitioning of RCRA metals and surrogate radionuclides. Unlike the POP system used for the MWIP tests, the NPS system will include a more comprehensive air pollution control system (APCS). Design of the NPS is nearing completion, and fabrication and installation will be undertaken in the spring of 1995. A summary of the current design is provided below.

PHP Hardware

The waste feed system will be a horizontal, water-cooled cylinder with an air-lock for loading the drums to be treated, and variable speed hydraulics for ram-feeding the drums into the plasma chamber. As with the current feeding approach, the drums will be fed axially. A constant flow of purge gas will protect the feed system from convective heat transfer from the plasma chamber. The NPS will utilize the same 1.2

MW plasma torch that was used for the POP system; however, because the NPS design will optimize the energy density of the plasma chamber and will include greater thermal insulation, the maximum NPS processing rate will be significantly higher than that of the POP system. The plasma chamber will be a vertical cylinder with a domed cap. The inner surface will be refractory lined, while the shell will be water-cooled. The plasma torch penetrates the domed cap to reach the crucible, which will be mounted to the bottom of the chamber. The torch will have a computer controlled three-axis-of-motion positioner with a manual override. Removal of the molten phases from the crucible will be accomplished by tilting the entire crucible one direction to pour the slag and the other direction to pour the metal. Two separate vessels will be positioned in a melt collection chamber located below the crucible. These vessels will be a 55-gal (or smaller) drum for the slag and a reusable cast iron pig mold for the metal. Figure 4 presents a sketch of the NPS primary melter hardware.

Air Pollution Control System

The offgas from the plasma chamber will pass through a gas-fired secondary combustion chamber before entering the APCS. The first APCS unit will be an evaporative cooler, which will use a water spray to rapidly quench the gas exiting the secondary combustion chamber to 400F. The quenched offgas will pass through a baghouse and then a HEPA bank to remove essentially all of the particulate matter. The filtered offgas will then be quenched to saturation to remove acid gases using an integrated quench/packed bed scrubber. At the scrubber exit will be a mist eliminator followed by an electric reheater. The reheater will heat the saturated gas to above its dew point so that the gas will be dry when it passes through the induced draft fan and stack.

CONCLUSIONS

The PHP has proven to be very promising as a robust, effective technology for solving the bulk of the DOE mixed waste problem. The proof-of-principle tests demonstrated the broad applicability of the PHP to safely treat diverse waste matrices. The design support studies have shown that the PHP can be made to operate continuously. Based on these successful tests, the PHP development project is proceeding with the design of new systems that will address questions regarding continuous service with wastes that include radioactive hazards. Continuous service is primarily being addressed with the NPS system. Radioactive service is primarily being addressed with the RBS. The experience gleaned from testing with these two systems will be incorporated into the design of the RFS, for which permitting efforts have begun.

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RADIOACTIVE DEMONSTRATION OF THE PLASMA HEARTH PROCESS ON SPIKED AND ACTUAL LOW-LEVEL MIXED WASTE*

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ABSTRACT
The Plasma Hearth Process (PHP) is an innovative thermal treatment technology being developed to demonstrate its effectiveness as a treatment process for the large quantities of radioactive mixed waste currently stored at a number of U. S. Department of Energy (DOE) sites. PHP technology development is sponsored by the Department of Energy's Office of Technology Development (OTD) and will be demonstrated for the OTD's Mixed Waste Focus Area, which is evaluating waste treatment technologies for a wide range of mixed waste types and forms. The ultimate goal of this project is to demonstrate the PHP technology by treating actual DOE radioactive mixed wastes using a field-scale unit (i.e., production size) to be constructed at Argonne National Laboratory-West (ANL-W) on the Idaho National Engineering Laboratory (INEL). The PHP technology will progress to a field-scale status through a systematic development of PHP test units (i.e., bench-scale, pilot-scale) and well defined test objectives for the units. The hardware development and testing for the PHP technology have been separated into the categories of "non-radioactive" and "radioactive." This paper describes the radioactive demonstration portion of the PHP project.
PLASMA HEARTH PROCESS DESCRIPTION
The term "plasma" refers to a highly-ionized electrically conductive gas. Plasmas can be generated by a variety of techniques, over a wide range of pressures and energy levels. The type of plasma produced in the PHP application is a dc arc-generated thermal plasma and is created by a device known as a "plasma torch." The plasma torch used in the PHP technology development project operates in the transferred arc mode. Figure 1 contains a conceptual diagram of the PHP system. Basically, the transferred arc torch uses a flow of gas to stabilize an electrical discharge (arc) between a high voltage electrode (inside the torch) and a molten pool of waste (maintained at ground potential). Because of the very high resistance to electrical current flow through a gas, electrical energy is converted to heat. Plasma gas temperatures are estimated to reach as high as 10,000oC. Even more energy is converted to heat as the electrical current passes through the melt, creating a Joule-heating effect in the molten pool.
Fig. 1.
The resulting molten products can form two phases. Operating the plasma hearth in an oxidizing (oxygen rich) mode produces a high integrity, leach-resistant slag. Operating the plasma hearth in a reducing (oxygen deficient) mode produces the slag and an additional reduced metal phase. These two phases segregate naturally due to density differences and could be recovered separately.
The PHP technology applies chiefly to solid or sludge wastes where a stabilized byproduct is required for disposal. The technology is ideally suited for heterogeneous solid wastes of nearly any category that are difficult to treat by conventional thermal technologies. One of the key advantages of the PHP technology is its ability to process a wide variety of waste materials with only minimal regard to physical or chemical nature. This minimizes the number of different treatment processes that would otherwise be needed to treat a variety of wastes. Whole drums of waste, including heterogeneous matrices, may be fed into the process chamber, minimizing pretreatment characterization, sorting, sizing, and other handling and potential exposure to toxic and radioactive contaminants. The application for which it is currently being developed is DOE solid mixed wastes, both low-level and transuranic. These wastes are found at most DOE sites and many have no identified treatment option.
The primary functional units of the PHP system are the feed system, plasma chamber, slag/metal removal system, secondary combustion chamber, and air pollution control system. Processing begins as complete drums of waste are fed to the fixed hearth plasma chamber, where heat from the plasma torch initiates a variety of chemical and physical changes. Complex organic compounds break down into noncomplex gases that are drawn from the chamber, while the remaining inorganic material melts and separates into two phases: slag and metal. The drum and inorganic materials are "drip melted" slowly into a crucible, and the molten materials are poured into waste

containers by a simple hearth tilting mechanism to avoid tap hole plugging problems. Actinides and oxidized heavy metals migrate to the slag phase which, after being removed, cools and solidifies into a glass-like, or vitrified, material. This high-integrity final waste form, similar to that selected for high-level radioactive wastes, has repeatedly shown the ability to meet or exceed disposal requirements instituted by the Resource Conservation and Recovery Act (RCRA).

Induced draft fans pull gases from the plasma chamber to a secondary combustion chamber. Once there, excess oxygen and the product gases are mixed in a very hot and turbulent environment for a sufficient residence time to complete the conversion of the gaseous organic compounds to CO2 and H2O. This secondary chamber guarantees the hazardous organics are treated completely to achieve the high destruction efficiency required by RCRA. The offgas is then scrubbed by state-of-the-art air pollution control technologies capable of removing a high degree of pollutants, producing very clean process emissions.

TECHNOLOGY DEVELOPMENT STRATEGY

The DOE OTD Mixed Waste Focus Area is sponsoring the evaluation and development of this technology by Science Applications International Corporation (SAIC) for application to DOE mixed wastes. The development process includes several PHP systems designed to address all aspects of the technology in a systematic fashion, progressing from a simple proof-of-principle unit to a field-scale prototype demonstrating full-scale operations on actual mixed waste prior to production implementation in the field. The elements of the development plan are illustrated in Fig. 2.

Fig. 2.

The first development phase, a proof-of-principle demonstration, has been completed using a simple first-generation PHP system at Retech, Inc., a manufacturer of plasma furnace systems. The tests showed that the PHP could process a variety of containerized wastes, including heterogeneous mixes, without any pretreatment. The tests also demonstrated the effectiveness of the process in destroying organics and its ability to produce a stable, leach-resistant vitrified final waste form. However, this test series did not address the behavior and fate of the actual radionuclides that will be in DOE mixed waste streams.

The second development phase features an effort to design, build, and evaluate a second-generation nonradioactive PHP pilot-scale system and to concurrently design and build a bench-scale PHP system for radioactive studies. The pilot-scale unit will be used for studying engineering/operations and evaluating advanced design concepts on surrogate waste materials. This unit will validate the operability/durability of PHP system hardware for continuous modes of operation over extended periods of time. The bench-scale system is intended to accurately simulate the behavior of radionuclides in the pilot-scale system and allow investigations of radioactive processing issues in a smaller, simpler, and less expensive unit. The relative sizes of the pilot- and bench-scale systems are compared in Fig. 3.

Fig. 3.

The bench-scale test program is structured to ensure that the plasma technology can be employed in radioactive service, and that surrogate (nonradioactive) studies on the pilot-scale unit properly model the behavior of radionuclides during treatment. These tests will demonstrate whether the PHP can successfully process radioactive wastes, especially alpha-emitters, while meeting DOE safety requirements and applicable environmental and public health regulations.

Knowledge gleaned from parallel testing of both units will be used to launch the final demonstration phase--the construction, permitting, and demonstration of a prototype field-scale PHP system for processing actual mixed waste. The field-scale system will be installed at ANL-W and will be targeted to demonstrate the treatment of 55-gallon drums of alpha-contaminated low-level mixed waste stored at the INEL. Additionally, the field-scale unit will provide full-scale verification of complete radioactive waste operations, including front-end handling through final waste form disposition. The time frame for the field-scale demonstration supports DOE commitments to various Federal Facility Compliance Agreements.

RADIOACTIVE BENCH-SCALE SYSTEM DESCRIPTION

To be located at the Transient Reactor Test (TREAT) facility of ANL-W at the INEL, the radioactive bench-scale system is configured to model the conditions of the near-full-scale nonradioactive pilot-scale PHP system currently being developed and tested at the Retech facility in Ukiah, California. The bench-scale PHP system (200

kW plasma torch compared to 1.2 MW in a full-size system) will be fabricated, assembled, and checked-out by Retech at their Ukiah facility prior to delivery to ANL-W.

The bench-scale system is a batch process, capable of feeding up to eight 1-gallon cans of waste material per test. The plasma chamber is designed to closely model the pilot-scale plasma chamber so that comparisons of radionuclide/surrogate behavior can be obtained. The Title II design of the complete bench-scale system is in progress, to be completed early in FY95. The bench-scale melter system consists of the following functional components:

- waste feed system,
- plasma chamber and crucible,
- plasma torch,
- secondary combustor, and
- air pollution control system.

Waste Feed System

A batch feeder will introduce the small waste containers into the plasma chamber. The feeder is capable of remote feeding and is oversized to allow containers of various sizes and shapes. The maximum throughput of test materials (operational feed limit) into the system is 30 lb/h.

Plasma Chamber and Crucible

The plasma chamber, commonly referred to as the melter, is a refractory-lined, double-walled, carbon steel vessel, configured as a vertical cylinder, and equipped with a closed-loop water-cooling system for chamber cooling. The top end of the cylinder has a flanged lid on which the torch is mounted and the bottom end of the cylinder has a flange-mounted hearth and crucible to collect molten material. The crucible is detachable from the removable hearth to allow batch removal of the product metal and slag.

Plasma Torch

The plasma torch is a dc transferred-arc plasma torch, model RP 75T, supplied by Retech, Inc. The torch has a hollow copper electrode and is water-cooled by a closed-loop heat exchanger. It is mounted on the lid of the primary chamber, and a three-axis-of-motion position controller controls the location of the arc inside the plasma chamber. The torch uses nitrogen as the primary plasma gas for melting, but for start-up and torch optimization other gases such as helium and argon will be used in combination with nitrogen.

Secondary Chamber

The secondary chamber is water-cooled and constructed of carbon steel lined with a refractory material. The chamber is equipped with electric heaters to maintain proper combustion and destruction temperatures. The chamber is designed to provide a minimum of two seconds residence time at a minimum of 1800oF and will be operated with a minimum oxygen content of 3% (by volume). These conditions will ensure the destruction of any gaseous organics that are released from the primary chamber. The offgas will consist primarily of CO2, N2, and H2O, with trace quantities of CO, SO2, and NOx.

Air Pollution Control System

The bench-scale system offgas will be processed and conditioned for release by an air pollution control system (APCS), which consists of the following functional components:

- evaporative cooler,
- HEPA filters (redundant),
- quencher,
- packed-bed scrubber/de-mister,
- reheater, and
- induced draft fans.

RADIOACTIVE BENCH-SCALE DEVELOPMENT STATUS

The bench-scale system is currently in the final stages of design, with a final design review scheduled for March of 1995. Upon completion of the review, fabrication of the melter system will be initiated. The procurement of major air pollution control system components, typically long-lead procurement items, is in progress so that delivery of the equipment coincides with completion of the melter system fabrication. The system will then be assembled and tested in Ukiah, California, at the Retech facility prior to delivery to ANL-W. The schedule for final installation of the system in the TREAT facility at ANL-W is aggressive, but

the system should be operational by the fall of 1995.

RADIOACTIVE BENCH-SCALE DEMONSTRATION PROJECT

The radioactive bench-scale PHP system is being designed and constructed to evaluate the treatment of actual radioactive mixed waste and to investigate the behavior of radionuclides (particularly plutonium) in the process. The bench-scale tests will investigate, in a cost-effective manner, the issues related to treating radioactive materials in the PHP before activities proceed to a more expensive full-scale demonstration. The bench- and pilot-scale units will be tested in parallel, allowing engineering development and radioactive operations to be interactively investigated.

The primary objectives of the radioactive bench-scale PHP project are:

1. to design and construct a PHP system that ensures public and facility personnel will be protected from hazards associated with the treatment of radioactive and other hazardous materials as a result of normal operations, anticipated operational occurrences, and design basis accident conditions;
2. to operate a bench-scale PHP system that accurately simulates the performance of the near-full-size pilot-scale system for surrogate/radionuclide comparisons;
3. to evaluate the behavior of radionuclides in the PHP system by means of partitioning studies, including radionuclide volatilization and deposition; and
4. to conduct treatability demonstrations on actual mixed waste to demonstrate organic destruction and to produce stable, leach-resistant final waste forms.

It is critical to the development of the PHP technology to determine the behavior and partitioning of specific radioisotopes, particularly those that are alpha-emitting. The technology can only be considered successful if it treats mixed waste without significant radiological problems, such as excessive releases of volatile radioactive species or concentration of actinide species that could result in criticality concerns. The bench-scale tests will focus on determining the relative quantities of radionuclides of interest that are retained in the slag phase or partitioned to the metal phase, and the quantity that is volatilized or otherwise transported to the offgas. The bench-scale tests will also establish the validity of the simulated waste tests conducted on the pilot-scale unit, particularly in the area of partitioning, and demonstrate that the PHP can effectively treat actual radioactive mixed waste.

The first period of bench-scale tests at ANL-W will be conducted on three categories of simulated or actual wastes:

1. simulated waste materials including surrogates for radionuclides of interest;
2. the same simulated waste materials as in (1) above, spiked with known amounts of the radionuclides of interest; and
3. actual radioactive mixed wastes.

Phase 1 Simulated Waste Tests

Simulated waste tests will be performed to checkout the system operationally and to verify the ability of the bench-scale system to model the performance of the pilot-scale system. these tests will be closely coordinated with the pilot-scale testing and will establish whether the bench-scale test results can be scaled up to full-scale. Successful demonstration of scale-up will allow correlation of the testing results between the bench-scale and pilot-scale systems. The waste materials to be tested will be a variety of simulated waste mixtures with known quantities of hazardous materials and surrogates for various radionuclides, duplicating the simulated waste mixtures to be processed through the pilot-scale system. Recipes for these simulated waste materials are based on simulation of actual mixed wastes to be processed in the bench-scale system in Phase 3.

These tests will also evaluate the partitioning of a variety of radionuclide surrogates and toxic metals in the PHP system, including stack emission rates. Partitioning refers to both the distribution of radionuclides in the three product phases (slag, metal, and offgas) and the uptake of radionuclides by the components of the system itself, such as the refractories. The methodology for evaluating radionuclide partitioning in the PHP system is as follows:

- Aspects that control the partitioning of radionuclides in the PHP system have been examined and the design of the bench-scale system conforms to the pilot-scale system in these aspects, where possible; and
- Tests with simulated waste materials, including the radionuclide surrogates cerium and cesium, run on the pilot-scale system will be reproduced on the bench-scale system.

It will be determined to what degree the radionuclide surrogates in the pilot-scale

nonradioactive system mimic the partitioning of the same surrogates in the radioactive bench-scale system. If reasonable agreement is obtained, the program will proceed to the next step of comparing radionuclide surrogates to actual radionuclides; otherwise, the bench-scale system will be reconfigured and/or operational modes adjusted until reasonable agreement between the two systems is achieved.

Although most system parameters (such as melt pool surface-to-depth ratio, refractory chemistry, melt pool temperature and chemistry, chamber size, offgas space velocity, turbulence, and power density) that affect the partitioning of radionuclides can be either matched or scaled appropriately, the complex interrelationships between design and operational parameters that affect partitioning make exact similitude impossible. An iterative approach, alternating testing and system modifications, may be required to achieve the desired level of similitude.

Phase 2 Spiked Radionuclide Waste Tests

The spiked radionuclide tests will evaluate the partitioning of actual radionuclides within the system and confirm whether surrogates for radionuclides realistically model their radioactive counterparts. The wastes materials to be tested in this phase will be the simulated waste materials from the Phase 1 tests spiked with known quantities of radionuclides. Each Phase 2 test container will be spiked with one or more of the radionuclides Pu, depleted uranium, or Cs137.

The partitioning of the spiked radionuclides will be compared to the partitioning of their respective surrogates. If the partitioning of radionuclides closely follows that of corresponding surrogates in the bench-scale system, then it can be assumed that the surrogates are valid in this application. Having previously demonstrated corresponding behavior of surrogates in the two systems during Phase 1, it can be concluded that the surrogates used in the pilot-scale system accurately simulate the behavior of radionuclides in a full-scale system. If the surrogate/radionuclide behaviors are significantly different but their behaviors can be reliably correlated, these correlations will be applied to full-scale results. Additional tests will examine the effects of varying parameters, such as radionuclide concentrations and the presence of chlorides, on radionuclide partitioning.

Phase 3 Actual Mixed Waste Tests

Testing on actual waste is intended to demonstrate that the PHP technology can easily treat a variety of mixed waste types into final waste forms that meet leachability criteria for disposal. The materials to be tested will be a variety of actual wastes from the INEL inventory of stored mixed waste. These wastes will be removed from their drums and repackaged into the smaller containers to be fed through the bench-scale system. The alpha low-level mixed waste (-LLMW) to be used in the PHP test project will be composed of Pu52-contaminated (weapons-grade plutonium) waste from the Rocky Flats Plant. It is currently stored at the INEL Radioactive Waste Management Complex. These a-LLMW waste categories include combustibles, uncemented inorganic sludges, and uncemented organic sludges. Heterogeneous solid wastes (mixed paper, metal, glass, etc.) will be obtained from low-level mixed wastes currently stored at ANL-W. These streams will be treated individually and in combinations to demonstrate the technology's relative insensitivity to variations in waste feed compositions. The effects of varying system stoichiometry on final form performance will be investigated. Finally, the effects of adding waste-form-enhancing additives will be investigated.

Test Regulatory Objectives

Emissions data will be collected during selected portions of the above-described tests to evaluate the performance of the technology with respect to regulatory requirements. Bench-scale system data will be used to support permitting of the field-scale system. The types of data to be collected include metals emission levels, radionuclide emission levels, organic destruction efficiencies, dioxin and furan generation, and NOx generation rates.

RADIOACTIVE FIELD-SCALE DEMONSTRATION PROJECT

The final activity in the PHP demonstration project will be the construction and operation of a production-sized PHP system for treating actual INEL mixed waste. This major program element will be the first step in transferring the PHP technology to the end-user. Specific mixed waste streams stored at the INEL will be selected and processed through the PHP field-scale system, which will be a complete waste treatment system from front-end waste handling through the production of a

disposable final waste form. Installed at the ANL-W TREAT facility, the unit will have a 1.2 MW torch and a throughput of up to two 55-gallon drums per hour, depending on waste characteristics.

The objectives of the field-scale demonstration are to:

verify that radioactive mixed waste can be handled in a way that is safe and complies with environmental regulations,

demonstrate the PHP's versatility in treating a wide variety of actual mixed wastes,

demonstrate the permitting and operation of a versatile DOE mixed-waste thermal treatment facility,

establish accurate life-cycle costs for the PHP, and

demonstrate the technology's readiness to be implemented complex-wide.

Due to both the aggressive project schedule and funding limitations, PHP development focuses on treating 55-gal drums of waste. Drummed waste represents over half the volume of DOE solid mixed wastes in storage. Current plans are to process up to 600 drums of mixed waste in the initial demonstration phase, in three test runs of at least 100 hours. Modifying the waste feed system or even the melter itself for receiving other waste containers, e.g. 4 x 4 x 8 ft wood boxes, will not be precluded in the field-scale design. Production treatment facilities may employ multiple feed systems connected to a single PHP, or may involve PHP units that are dedicated to nominal container types, e.g. drums, boxes, bulk (uncontainerized) solids, liquids, etc. The main thrust of the field-scale demonstration is to establish a treatment facility to begin operation on a significant fraction of the mixed waste inventory; subsequent work can address modifications to treat the remainder of the wastes.

FIELD-SCALE DEVELOPMENT STATUS

A preconceptual design of the field-scale system was completed in the first quarter of FY95. Much of the information contained in the preconceptual report was gleaned from the design activities on the full-size pilot-scale system currently underway. A conceptual design report is in work and will be issued in mid-FY95. Title design activities will follow the NEPA process, which is expected to require an environmental assessment.

NONRADIOACTIVE STAR CENTER SYSTEM DEVELOPMENT SUPPORT

An existing fixed-hearth plasma system at the SAIC Science and Technology Applications (STAR) Center in Idaho Falls has been used to investigate a variety of PHP design issues. These issues include slag/metal separation, hearth/crucible configuration development, and testing of refractory materials for wear and durability. The STAR Center system is roughly the size of the radioactive bench-scale unit and includes an early model of the same plasma torch. Hearth/crucible design, primarily of the electrical ground path, was the objective of the Radioactive Bench-Scale Design Support Studies conducted in the STAR Center. Out of this work a promising design based on a metal ground path and rammable refractory has been selected for long-duration testing. Two other viable (but less desirable) hearth/crucible configurations have also been developed. They can be used if the first choice does not meet the challenges of long duration testing. These results and the future testing planned at the STAR Center will be incorporated into the radioactive bench- and pilot-scale designs and operational philosophies.

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ARGONNE-WEST FACILITY REQUIREMENTS FOR A RADIOACTIVE WASTE DEMONSTRATION*

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ABSTRACT

At Argonne National Laboratory - West (ANL-W), near Idaho Falls, Idaho, facilities that were originally constructed to support the development of liquid-metal reactor technology are being used and/or modified to meet the environmental and waste management research needs of DOE. One example is the use of an Argonne-West facility to conduct a radioactive waste treatment demonstration through a cooperative project with Science Applications International Corporation (SAIC) and Lockheed Idaho Technologies Company. The Plasma Hearth Process (PHP) project will utilize commercially-adapted plasma arc technology to demonstrate treatment of actual mixed waste. The demonstration on radioactive waste will be conducted at Argonne's Transient Reactor Test Facility (TREAT). Utilization of an existing facility for a new and different application presents a unique set of issues in meeting applicable federal, state, and local requirements as well as the additional constraints imposed by DOE Orders and ANL-W site requirements. This paper briefly describes the PHP radioactive demonstrations relevant to the interfaces with the TREAT facility. Safety, environmental, design, and operational considerations pertinent to the PHP radioactive demonstration are specifically addressed herein. The personnel, equipment, and facility interfaces associated with a radioactive waste treatment demonstration are an important aspect of the demonstration effort. Areas requiring significant effort in preparation for the PHP Project being conducted at the TREAT facility include confinement design, waste handling features, and sampling and analysis considerations. Information about the facility in which a radioactive demonstration will be conducted, specifically Argonne's TREAT facility in the case of PHP, may be of interest to other organizations involved in developing and demonstrating technologies for mixed waste treatment.

BACKGROUND

Introduction

Argonne National Laboratory (ANL), one of the nation's largest energy research and development organizations, is operated by the University of Chicago for the U.S. Department of Energy (DOE) with sites in Illinois and Idaho. At Argonne National Laboratory - West (ANL-W), near Idaho Falls, Idaho, facilities that were originally constructed to support the development of liquid-metal reactor technology are being used and/or modified to meet the environmental and waste management research needs of DOE. One example is the use of an Argonne-West facility to conduct a radioactive waste treatment demonstration through a cooperative project with Science Applications International Corporation (SAIC) and Lockheed Idaho Technologies Company. The Plasma Hearth Process (PHP) project will utilize commercially-adapted plasma arc technology to demonstrate treatment of actual mixed waste. The demonstration with radioactive waste will be conducted at Argonne's Transient Reactor Test Facility (TREAT).

Argonne Experience

Argonne's experience began at the dawn of the nuclear age and throughout its history nuclear activities have played the central role. Argonne is a recognized leader in alpha confinement and remote operations technologies--key elements in a realistic PHP development program. At ANL-W, metric ton quantities of plutonium and uranium are a part of daily operations. Some 25,000 plutonium items in 2600 containers and 73,000 uranium items in 2140 containers exist on the site. For the past 10 years, plutonium and actinide fuel has been cast in the Experimental Fuels Laboratory. In gearing up for applying the ANL electrometallurgical process to the DOE spent fuel problems, new facilities and analysis techniques have been developed. As a result, new capabilities have been developed for process control, actinide and fission product waste streams, materials control and accountancy, criticality control, employee and public safety, and regulatory permit processing. From decades of transuranic (TRU) facility/equipment design and operations, a few simple lessons learned are now applied to all ANL-W TRU projects: 1) simplify the process, equipment, and facility to the maximum extent; 2) minimize confinement penetrations;

3) minimize moving parts, and 4) limit the energy potential. These principals are being implemented in the PHP Support Systems in TREAT, described in a subsequent section of this paper.

Plasma Hearth Process Program Description

Plasma-arc technology is an existing, commercially available processing method used primarily in the high-purity metals industry. Adaptation of this technology to treatment of mixed waste, and demonstration of this application on actual mixed waste is the focus of the PHP Project. The PHP is a high temperature vitrification technology which utilizes a plasma arc torch to melt waste material into a fixed hearth. The melted waste solidifies into separable metal and glassy-slag phases. Results of testing to date show promising results with regards to volume reduction of waste, destruction of hazardous organic constituents, stabilization of hazardous inorganic constituents in the vitrified product, and resistance to leaching from the final product. One of the greatest benefits of the PHP is the apparent flexibility to process a wide variety of waste forms without the need for pretreatment steps such as shredding or the need for additives to achieve the final product. Also, the PHP discharges much less offgas than incinerators, which are currently being scrutinized by regulators and the public. The PHP Project is funded by DOE's Office of Technology Development (EM-50), specifically the Mixed Waste Integrated Program. Programmatic aspects and results of non-radioactive PHP testing are widely published. (1,2,3)

The PHP program ultimate goal is to conduct a full-scale demonstration using actual mixed waste. To ensure success in this goal, two PHP systems will be built and tested prior to the full-scale system. A bench-scale PHP system will be constructed at the ANL-W TREAT facility to test the treatment of radioactive material containing waste forms for assessment of radiological considerations. The bench-scale system is approximately one-tenth scale of a full scale system in terms of torch power and processing rate. In parallel, a pilot-scale PHP system will be built and operated on non-rad waste to ensure system operability and reliability at a large scale as well as conformance to all environmental criteria.

Three phases of testing will be conducted in the bench-scale PHP experiments. First, simulated waste, which will include radionuclide surrogate materials like cerium, will be tested to check out the PHP systems operationally and to verify the ability of the bench-scale system to model performance of the pilot-scale system. Then, simulated waste matrices will be spiked with radionuclides, specifically plutonium, cesium, and uranium. Radionuclide concentrations in various locations throughout the PHP primary system will be measured. Finally, tests will be performed on actual waste. The actual waste streams will include inorganic sludge, organic sludge, combustibles, and a heterogeneous mixture of waste matrices. The actual waste tests are expected to demonstrate the PHP technology's relative insensitivity to variations in waste feed compositions. Also, the system will be operated to vary stoichiometry for the purpose of evaluating final waste form performance. The effects of including waste-form-enhancing additives to the feed material will also be investigated in the actual waste tests.

All of the test material to be treated in the PHP system will be packaged in small (3.8 liter) metal containers. Approximately eight containers will be processed per test. Approximately 39 tests will be processed (312 containers) in the PHP experiments, expected to be conducted over a 6- to 9-month time period. For the spiked waste tests, each container may include up to 500mg plutonium, 20g depleted uranium, and 3700 Bq of cesium-137. The radionuclides for the actual waste tests will consist of plutonium in quantities much less than the spiked tests.

TREAT History

The TREAT facility at Argonne-West was originally constructed in 1958 as an experimental, air-cooled, thermal reactor used for fuel and material testing under transient conditions. Reactor operations started in 1959, and continued for 35 years; over 2800 reactor transients have been conducted. The facility was upgraded to modern standards in 1989. The TREAT facilities existing infrastructure is being utilized to conduct the non-reactor PHP experiment. The features which make TREAT suitable for the PHP experiments include the large high bay area (35m x 21m x 23m high), double HEPA-filtered building exhaust, existing electrical and water supplies, overhead crane coverage, and truck access. TREAT support systems specific to the PHP Project are described in the following section.

PHP SUPPORT SYSTEMS IN TREAT

New support systems are being provided to the TREAT facility that will enable the PHP process equipment to be operated safely and effectively to perform the experimenters' testing and evaluation mission. Existing TREAT systems will be supplemented by additional systems for non-reactor nuclear experiments. These new systems will provide: a secondary radiological confinement to the PHP process equipment which is independent of the TREAT building; inert gases; exhaust and ventilation; communications, monitoring, alarming; process residue sampling and handling capabilities; and extended utilities and fire protection. The addition of these support systems considerably broadens the usefulness of the TREAT facility and will result in a versatile installation capable of fulfilling the facility needs to support developmental testing and evaluation of other waste processing technologies. The new support systems will be designed and installed in accordance with DOE, Environmental Protection Agency (EPA), and ANL requirements as outlined in the following sections.

General Design Requirements

Experiments in TREAT are currently constrained by the following principal requirements established by TREAT: a) Plutonium is limited to a maximum of 150 g and 1705 g biological equivalent of Pu-239 per experiment; b) Double containment is required for all test specimens containing plutonium; and c) the primary containment shall be designed to retain its integrity during all planned testing conditions, and at least one level of containment must be ensured during the maximum credible accident conditions.

Compliance to DOE Orders requires a determination of the natural-phenomena-hazard classification. DOE Order 5480.28 uses a graded approach: structures, systems, and components are assigned to one of five performance categories (PCs) in accordance with performance criteria given in DOE-STD-1021-93. Due to the low radioactive inventory involved, the PHP experiment will be classified as PC-1. Design and evaluation criteria for structures, systems, and components for earthquake, wind, and flood are based on DOE-STD-1020-94.

Specific standards to which the PHP support systems are being designed include: U. S. DOE Order, DOE-6430.1A, "General Design Criteria"; National Fire Protection Association Codes and Standards; American National Standards Institute (ANSI) N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities"; ANSI-N317, "Performance Criteria for Instrumentation Used for In-Plant Plutonium Monitoring"; ANSI-N323, "Radiation Protection Instrumentation Test and Calibration;" U. S. DOE Order 5400.5, "Radiation Protection of the Public and the Environment"; U. S. DOE Order, 5480.5, "Safety of Nuclear Facilities"; U. S. DOE Order, 5480.11, "Radiation Protection for Occupational Workers"; International Conference of Building Officials, "Uniform Building Code"; and ANL-W "Environment Safety and Health Manual."

The PHP bench-scale experiments also require consideration of environmental requirements. A permit-to-construct will be obtained from the State of Idaho in accordance with the Clean Air Act. The EPA granted a permit for the bench-scale experiments under the National Emission Standards for Hazardous Air Pollutants. A categorical exclusion in accordance with the National Environmental Policy Act, paragraph B3.10 of Appendix B to Subpart D of 10 CFR 1021, was approved by DOE. The experiments will be conducted under a Resource Conservation and Recovery Act treatability study in accordance 40 CFR 261.4.

System Descriptions

An 11m x 14m x 7m high secondary confinement structure will be provided for backup confinement of radiological and hazardous materials during normal and abnormal operations of the PHP without reliance on the TREAT building. This structure will be located within a high-bay section of TREAT having a floor area of 735 m2, a ceiling height of 23m, and a 55,000kg bridge crane. The secondary confinement will have removable roof panels that allow equipment handling, and is designed for ease of modification, if required.

A process off-gas and ventilation system that contains the process off-gas and secondary confinement ventilation air and discharges it from the TREAT facility stack is being provided. This system includes redundant exhaust fans and HEPA filtering of the off-gas and ventilation air to reduce or limit radioactive contamination levels in the effluent gases to acceptable limits.

The ventilation system maintains a negative pressure within the secondary confinement structure relative to the TREAT building to prevent radioactive

contamination spread to other parts of the building. The system is configured such that air flow sweeps from the cleanest regions anticipated to areas within the secondary confinement having successively higher potential for contamination.
A fire protection system will provide detection and suppression of fire (by water sprinkler) within the secondary confinement and shall provide for alarms to personnel and the ANL-W fire department.
An extensive communications, monitoring, and alarm system will incorporate telephone and area-wide audio communications, monitor and record PHP secondary-confinement and process operational conditions (normal and abnormal), and provide both operator and facility alarms at preset conditions and process-parameter thresholds. This system includes extensive video equipment for monitoring and recording the process operation.
Provisions will be made for materials handling that services solid-material-entry and -discharge points. This includes logistics and equipment, as required, for the transport of waste containers to the experiment and the packaging and disposition of process wastes. A major feature is a 6m x 1.5m x 2.7m high glovebox with built-in equipment for lifting, translating, sample drilling, and weighing the PHP hearths. The glovebox will confine alpha contamination. Hearths approximately 1.22m in diameter and weighing up to 816kg will be handled within this structure. The enclosure will also be used for the extraction of representative samples from the hearth, HEPA filters, and acid-gas scrubber liquor.
Extensive radiation monitoring instrumentation will be provided to measure radiation fields and the levels of radioactive materials that are airborne or deposited on surfaces to quantify the level and type of radiation and/or airborne radioactive contamination, provide data to support operations to assess the radiological-safety conditions, and to warn personnel of abnormally high radiation fields. Radioactive emissions from the TREAT stack will be monitored by an emissions-monitoring system to quantify the level and type of radioactive effluent.
Several utility systems are being provided that enhance the usefulness of the facility. An electrical-power system connects to the existing TREAT electrical system to supply the normal- and redundant-diesel/generator power systems and distribution necessary to support the test equipment. Feeders having capacities of 600A and 400A at 480V are being provided, along with an 18kVA uninterruptible power supply.
Cooling-water systems for equipment cooling including both normal service water (12 liter/s at 290K) and a chemistry-conditioned supply of 0.3 liter/s are being provided.
An inert-gases-supply system will furnish nitrogen, helium, and argon to the PHP on process demand for torch gases, for purging the camera and view ports, and for purging the waste-feed system. Nitrogen flow may be up to 25 liter/s from a liquid nitrogen tank and vaporizer. Helium and argon flow rates up to 0.025 and 0.13 liter/s, respectively can be furnished. These flow rates could be easily increased, if needed.
A breathing-air system is being provided that furnishes breathing air to support up to 4 workers for use during suited entries into potentially contaminated areas and equipment.
All of these PHP support systems will be adequately instrumented, where practical, to monitor parameters having significant impact on the systems' operation or capabilities to provide their respective functions. Significant data will be recorded for subsequent system-performance analysis. Alarms are being provided to warn operations personnel whenever significant parameters are outside of established limits.

CONCLUSION

The personnel, equipment, and facility interfaces associated with a radioactive waste treatment demonstration are an important aspect of the demonstration effort. Areas requiring significant effort in preparation for the PHP Project being conducted at the TREAT facility include confinement design, waste handling features, and sampling and analysis considerations. Information about the facility in which a radioactive demonstration will be conducted, specifically Argonne's TREAT facility in the case of PHP, may be of interest to other organizations involved in developing and demonstrating technologies for mixed waste treatment.

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DEVELOPMENT OF PLASMA VITRIFICATION TECHNOLOGY FOR CONTAMINATED SOIL AT THE SAVANNAH RIVER SITE

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ABSTRACT

The Mixed Waste Integrated Program (MWIP) of the United States Department of Energy's Office of Technology Development is developing treatment technologies for a wide variety of materials containing mixed low-level waste, i.e., having low levels of radioactivity along with hazardous constituents. Vitrification is a promising treatment technology for many of these wastes, including contaminated soil such as that found at the Savannah River Site.

Proof-of-principle tests were performed to demonstrate the feasibility of both ex-situ and in-situ vitrification of contaminated soil by means of a plasma torch. A mixture of 89% as-excavated Savannah River Site sandy clay loam with 11% lime addition was tested. Vitrification of a mixture of this feed, in a 10" diameter crucible with a non-transferred arc plasma torch at a nominal 160 kW, was successful. The process produced homogeneous glass (albeit with local compositional variations), surrounded by a skull of incompletely reacted feed. Characterization of the resultant product durability using the Product Consistency Test (1) showed elemental leaching well below the Environmental Assessment glass (which is often used as a minimum standard of glass acceptability in high-level waste glass assessment) for both the glass and the skull regions. Future tests should include doping the soil with hazardous constituents to enable further verification of the wasteform integrity via the Toxic Characteristic Leaching Procedure (2).

In-situ operation was mimicked in the test crucible by segregating the lime additive from the soil within the crucible. Making full use of the available torch maneuvering capabilities (which would likely exceed those of a torch used in-situ) failed to produce a homogeneous melt. Therefore, intimate mechanical mixture of the additive with the soil appears crucial to the success of SRS soil vitrification, and must be included in design considerations for in-situ operation.

INTRODUCTION

The Mixed Waste Integrated Program (MWIP) of the United States Department of Energy's Office of Technology Development is funding efforts to develop treatment technologies for a wide variety of materials containing mixed low-level waste (MLLW), i.e., having low levels of radioactivity along with hazardous constituents. Vitrification is a promising treatment technology for many of these wastes, including contaminated soil. Soils with MLLW constituents are present at many sites across the Department of Energy (DOE) complex, as well as in industrial settings such as the naturally occurring radioactive material (NORM) associated with drilling operations in the oil and gas industries. High temperatures are required for soil vitrification, making plasma technology attractive for this application.

The processing requirements of one soil type may not be directly transferrable to another, due to differing morphology and chemical composition. Rather than testing a "generic" soil which might ultimately prove to have limited applicability, a representative contaminated soil material was desired for proof-of-principle testing of the plasma vitrification method. Savannah River Site (SRS) soil was used for

these tests because SRS is a large (300 square miles) DOE site where nuclear materials production and processing operations have been carried out over several decades, resulting in significant quantities of soil requiring remediation.
This paper describes the plasma vitrification tests of SRS soil which were performed at the Diagnostic Instrumentation and Analysis Laboratory (DIAL) of Mississippi State University. Both ex-situ and in-situ treatment methods are of interest; therefore, the tests were designed to demonstrate features of each.

TEST MATERIAL

At SRS, the soil most likely to be contaminated with low levels of radioactivity is that found near processing facilities. For building stability, these facilities were generally sited on a particular group of soil series. Soil excavation for construction of facilities and burial of waste resulted in mixing of diagnostic horizons to the extent that individual soil series cannot be identified. The resulting mix of material is classified, using United States Department of Agriculture (USDA) soil taxonomy, in the Great Group Udorthents. The particle size distribution for this group typically is sandy clay loam (USDA nomenclature), with an estimated breakdown of 55% quartz sand, 30% kaolinite clay particles, and 15% silt. Thus, despite the inevitable local variability in soil composition throughout the site, there is a high probability that most of the contaminated soil at SRS is quite similar in particle size distribution and mineralogy. Although a sandier soil is generally found as topsoil throughout the site, this sandy top layer would have been removed during facility construction, exposing the sandy clay loam. For the tests described herein, clean (nonradioactive) sandy clay loam was excavated from the soil pile associated with the construction of a non-radioactive site facility.
Proof-of-principle tests of sandy clay soil vitrification have been performed previously (3). From those tests, it had been determined that use of a fluxing additive was required for homogeneous vitrification, and that commercial, agricultural limestone was suitable for this purpose. Two tests had been run: one with no additive and one with 30% by volume of lime additive. Although lack of a flux additive precluded complete vitrification of the soil, too much flux was used in the second test, which resulted in precipitation of multiple crystalline phases upon cooling, again impairing product homogeneity.
Based on these findings, the feed composition for the present tests was approximately 11 weight percent agricultural dolomitic lime, (a mixture of $CaCO_3$ and $MgCO_3$, with 20% elemental Ca and 10% elemental Mg), and 89% as-received soil (i.e., with a moisture content of approximately 15% and containing small amounts of plant material, gravel, etc.).

TEST DESCRIPTION

The objective of the tests was to examine the potential for both ex-situ and in-situ treatment of contaminated soil. Processing issues for ex-situ vitrification, in which feed material is introduced into a processing vessel, include feed composition (ratio of soil to additive), feed material handling, and off-gas emissions. Ex-situ vitrification allows the use of either the transferred-arc or nontransferred-arc mode of torch operation. Transferred-arc operation may be more efficient; since heat is conducted through the melt directly, it is more likely to promote axial mixing and avoid temperature stratification.
In-situ treatment involves inserting the plasma torch into a series of boreholes at the contaminated site, withdrawing the torch as the surrounding material at a given axial location is vitrified. Any required additives would be introduced down the borehole. In-situ treatment is attractive, in that handling and transportation of contaminated material is minimized; off-gas treatment requirements may also be reduced if the soil acts as a percolation filter for off-gas. However, the feasibility of in-situ vitrification depends on the ability to achieve product quality under in-situ processing conditions. Since a fluxing agent appears to be necessary for this type of soil to vitrify, adequate mixing of the flux with the soil must be ensured. In-situ vitrification precludes the use of transferred-arc operation, since there would be no bottom attachment point for the arc when moving the melt upwards in the borehole. The increased gas flow through the torch in non-transferred arc mode might, however, increase mixing through forced convection by the gas impinging on the melt.
With these considerations in mind, two different types of tests were performed on the SRS soil:
1. The soil and lime were intimately mixed in the crucible. The torch was fixed

both radially and azimuthally.
2. The soil and lime were segregated in the crucible. The torch was manually manipulated laterally as well as axially.
The torch was run in non-transferred arc mode for all tests to permit a direct comparison of the two test types. The first test type, essentially an ex-situ demonstration, will be rerun in transferred arc mode at a later date. The second type of test was meant to mimic the in-situ processing conditions, where the flux would be introduced into the soil without intimate mixing. The second test type was repeated twice, once with only limited lateral movement, and a second time with lateral movement specifically aimed at promoting mixing between the flux and the soil.
In all tests, the torch was first run several inches above the melt for 10-15 minutes to allow the test material to soften, thus reducing blow-off, and then lowered to a distance of approximately two inches above the surface of the test material. The duration of each test was 45-60 minutes of torch operation.
The tests were performed using a nominal 250 kW plasma torch manufactured by Plasma Energy Corporation. The torch was run in non-transferred mode, using air as the working gas. In this mode, the power output fluctuated between about 130-160 kW. The airflow through the torch varied from 12-20 scfm for these tests. The torch is installed with both axial and lateral movement capabilities. A viewport equipped with video camera enabled visual observation during the test. As described below, this capability was used in one of the tests to move the torch in order to best achieve a mixed melt.
For each test, a total of 13.44 lb of feed was used (12 lb soil and 1.44 lb lime). The test material was processed in batch mode in a 10" diameter crucible which was placed inside a larger test vessel lined with refractory brick. The test vessel was sealed and under vacuum for the tests. Off-gas was monitored by a variety of techniques, including infrared, FTIR, two-color imaging, and spectroscopic pyrometry. These measurements will be reported separately.
RESULTS
Homogeneous Feed Test
The first test, with soil and lime homogeneously mixed in the crucible, produced a homogeneous, glassy material in the center of the crucible, with a 1-2 inch "skull" of incompletely reacted feed adjacent to the crucible walls and bottom. Presumably, a more powerful plasma torch, or use of the torch's lateral maneuvering capabilities, would have completely vitrified the material. To assess product quality as a function of radial distance from the torch, samples of the product were taken at three locations: 1) from the center of the melt; 2) from the transition region between glass and skull; and 3) from the skull region itself. These were examined by X-ray diffraction (XRD) and scanning electron microscopy coupled with energy dispersive X-ray spectroscopy (SEM-EDS) (Fig.1).
XRD of samples 1 and 2 (from center and transition regions, respectively) confirmed that homogenous glass was produced, with no crystalline phases observed. XRD of sample 3 (skull region) showed the presence of both quartz (the sand in the original soil) and cristobalite, with quartz predominating. Note that the transition temperature between these two phases is 1470C, which implies that temperatures in this range were experienced even in the incompletely reacted region.
SEM results of the three samples are shown in Fig. 1. All figures are at the same scale, as shown in the figures. SEM-EDS examination of sample 1 showed the presence of a number of species: Si, Fe, Ni, Cu, and Ca. The same species were identified in sample 3, with slightly higher peaks for Ca, Fe, and Ni showing some compositional variation between the glassy center and the incompletely reacted soil/lime mix in the skull region. However, SEM-EDS examination of an area in sample 2 showed the presence only of Si--none of the other species were identified except for a very minor peak for Cu. This SEM-EDS result was obtained at two locations within sample 2 which were about 1 mm apart. This implies that this area of the transition region between glass and unreacted batch is essentially pure silica glass.
The Product Consistency Test (PCT) (1), now ASTM C-1285 procedure for determining waste glass durability, was performed for the three samples, yielding leachate concentrations as shown in Table I. Note that non-zero concentrations are measured for various cations in sample 2. This implies that constituents other than silica were indeed present in this sample, contrary to the SEM-EDS indications. Therefore, we assume that sample 2's average chemical composition is similar to that of the

other samples; the SEM-EDS result can be attributed to the small scale of the SEM-EDS sample. This shows, however, that although a uniform glass product was formed in the transition region, local (microscale) variations in chemical composition resulted.

Table I shows that all three areas had releases of Si, Na, Li, and B lower than the Environmental Assessment (EA) glass which has been used as a minimum standard for durability for high level waste glass. The skull region had a significantly higher level of Ca release, however, indicating insufficient reaction between the soil and lime. The skull region also showed a slightly lower Al release than the glassy regions.

It is notable that all three samples had roughly similar release levels. That is, even the crystalline skull region material has a durability better than EA glass, as measured by the PCT. (Note that these PCT results are on a non-normalized basis, so this conclusion must be confirmed by chemically analyzing the samples and normalizing the PCT results.) The feed was not doped with any hazardous constituents, so its ability to retain such constituents could not be assessed by a direct measure such as the Toxic Characteristic Leaching Procedure (2). This should be done to verify that both the glass and the incompletely reacted material represent a viable waste form.

Segregated Feed Tests

The main focus of these tests was to explore the ability to mix the lime and soil in-situ. From visual observation during the test and of the resultant product, there was clearly a lack of mixing between the flux and soil. In the first of these tests, with relatively little lateral torch movement, the lime clearly boiled, spattering the crucible sides and vessel viewports, while the soil remained stiff. Visual examination of the crucible contents showed extreme heterogeneity in the product, with "ledges" of unreacted soil lining the sides of that portion of the crucible which contained soil (i.e., about 270 of the circumference). Even in the center of the crucible, directly below the torch, the melt extended only a short depth below the surface, and at the sides of the crucible, the glassy portion had a thickness of only about 0.25 inches.

The test was then repeated, making full use of the visual capabilites and torch movement capabilities to try to drive the lime melt into the soil and soften the soil "ledges" in order to mix them into the melt. Although some mixing was achieved by this method, visual observation during the test still showed differences between the lime and soil regions of the crucible. Post-test examination of the material confirmed that soil "ledges" were still present around the sides of the crucible, although the product appeared more homogeneous than in the previous test.

Quantitative measurements such as those performed on the homogeneous feed will be performed on the products of these tests, but are not expected to change the following conclusions based on visual observation. The first of these two tests represents the simplest configuration for in-situ operation, that is, with additive introduced in a mass down the borehole. While some torch maneuverability may be available, it will likely be limited and the torch movement will not be "tailored" to optimize the mixing operation. The second test represents an "upper bound" to the optimization of torch operation to promote mixing. Even under these conditions, a homogeneous melt could not be produced. These tests indicate that in-situ vitrification of tyis soil type would require apparatus to mix in the flux prior to treatment with th plasma torch, which will require additional design effort and feasibility testing.

CONCLUSIONS

Proof-of-principle tests were performed to demonstrate the feasibility of both ex-situ and in-situ vitrification of contaminated soil by means of a plasma torch. To represent the SRS contaminated soils, a sandy clay loam was used; a mixture of 89% as-excavated soil with 11% lime addition was tested. Crucible vitrification of this mixture was successful, producing glass (albeit with local compositional variations) without crystallization.

In-situ operation was mimicked in the test crucible by segregating the lime additive from the soil within the crucible. Making full use of the available torch maneuvering capabilities (which would likely exceed those of a torch used in-situ) failed to produce a homogeneous melt. Therefore, intimate mechanical mixture of the additive with the soil appears crucial to the success of soil vitrification, and must be included in design considerations for in-situ operation.

ACKNOWLEDGEMENTS
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37-8
PFNA-BASED TECHNIQUES FOR CORRECTING MATRIX EFFECTS IN NON-INTRUSIVE ASSAYS OF HETEROGENEOUS WASTES
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ABSTRACT
Pulsed Fast Neutron Analysis (PFNA) is a non-intrusive technique for inspecting bulk samples whose size may range from that of a suitcase to that of a commercial-cargo truck trailer. An earlier paper (1) described techniques for performing PFNA-based TRU assays on waste drums, and reasons why such measurements would yield improved accuracy, sensitivity, and throughput over existing instruments. Here we describe PFNA-based techniques for further improving the accuracy of assay results. In principle, these measurements, based mainly upon analysis of inelastic-neutron-scattering and neutron-capture gamma rays, can be performed concurrently with TRU-assay measurements. Using such techniques, a single PFNA-based inspection system would allow substantially complete non-intrusive characterization of waste containers.
INTRODUCTION
Numerous non-destructive techniques have been developed in which physical characteristics of samples are revealed upon irradiation by neutrons. These techniques may involve analysis of neutron transmission or scattering patterns, but are often based mainly on emission of secondary radiations (e.g., fission neutrons or gamma rays from capture or inelastic scattering). Many such techniques have been reliably used for decades to characterize physically small samples, but have shown inconsistent success on large items. This paper outlines some of the most serious challenges to inspecting large samples, and briefly describes PFNA-based techniques for mitigating or correcting such effects. Indeed, one should note that while the focus is upon neutron-based active inspection techniques, similar effects arise at some level for virtually all active and passive techniques based on interactions of penetrating radiation with samples.
A sample may generally be considered small if it is much smaller than the source and detector, or if its dimensions are much smaller than the distances that separate it from the source and detector. In either case, the unattenuated neutron flux and the detector's geometric efficiency may each be assumed constant over the sample volume. Often it is also possible to neglect perturbations a small sample causes to the incident radiation field and to the number of secondary radiations escaping the sample to reach a detector. Frequently, small samples can be assumed homogeneous, allowing simple corrections to be applied for such perturbations. Such assumptions may be valid even if the sample is not strictly homogeneous, as long as the scale of sample inhomogeneities is small compared to pertinent radiation mean-free-path lengths. To the extent such assumptions are valid, they constrain the possible solutions to the inverse problem and yield accurate assay results. Difficulties arise though as sample size increases and these assumptions begin to fail.
The first set of assumptions is purely geometric. Given a sufficiently well

characterized source and detector response functions, and a sufficient number of independent observations, effects related to the spatial extent of a sample can generally be treated reliably.

Arguably the most severe difficulties arise in understanding or correcting for interactions of probe neutrons with the sample itself. These interactions perturb not only the distribution of incident neutrons in the sample, but also the energy spectrum and angular correlations that may otherwise have existed between the source, sample, and detected radiations. As differential cross sections often vary strongly with neutron energy, such perturbations greatly complicate assay results. Despite these complications, it is not uncommon to assume sample homogeneity, even when such assumptions clearly be invalid. Predictably, the accuracy of results in these cases often leaves much to be desired.

OBTAINING MATRIX DISTRIBUTIONS FROM CONVENTIONAL PFNA MEASUREMENTS

As described previously (1), PFNA measurements involve interactions of essentially mono-energetic neutron pulses with a sample. With nanosecond pulse widths and ~8-MeV neutrons, time-of-flight (TOF) measurements determine interaction depths to within several centimeters. Events are usually localized in the remaining two dimensions by collimating the neutron beam. Conventional PFNA inspections are based mainly upon gamma rays from inelastic scattering of uncollided neutrons (though for very large or dense samples it is important to also consider those scattered neutrons whose energy is above the inelastic-scattering threshold).

Despite the highly penetrating nature of neutrons used for PFNA, corrections must be made for neutron removal from the uncollided beam at different depths. Fortunately, PFNA measurements yield enough information to reliably correct for such matrix effects by iteratively analyzing the observed gamma-ray signals. By constraining the corrections to match total measured neutron removal from the beam, corrections for moderate-size samples usually converge satisfactorily after a single iteration.

Conventional PFNA measurements performed in this manner allow distributions to be mapped for many elements, including C, N, O, Al, Si, and Fe, for example, and have yielded spatial resolution of ~125 cm3. These maps are uniquely detailed in their combination of spatial resolution and elemental specificity, and can be used either independently, or in conjunction with complementary techniques (e.g. computed x-ray tomography) to develop detailed models of neutron and photon transport within a sample (2). Such models can, in principle, give very good matrix corrections for both neutron and photon interactions in a sample.

Indeed, while elemental maps such as those described above certainly provide information important for developing matrix corrections, it is important to note that the list of elements mapped directly by PFNA does not include H, perhaps the single commonly occurring element that strongly affects neutron transport over the entire range of energies pertinent to existing non-intrusive techniques. Although space does not permit more than passing reference, it is simply noted here that in practice, a variation on the iterative algorithm described above yields very good maps of H distribution. Combined with maps provided directly by conventional PFNA measurements, inferred H distributions provide a very complete description of the physical system and the basis for detailed understanding of radiation transport in a sample. This information promises to be of value for both PFNA-based inspections and for correcting matrix effects in other active and passive assay systems as well.

PULSED THERMAL NEUTRON ANALYSIS (PTNA)

The prompt radiations used for PFNA inspections are observed over several tens to hundreds of nanoseconds after a neutron pulse, depending on sample depth. The pulsing frequency is chosen so no more than one pulse exists in the sample at any time. Cargo containers therefore require a pulsing frequency not more than ~5 Mhz, while pulsing rates for 208-liter waste drums can be much higher.

Although signals for PFNA measurements are induced by the uncollided pulses of essentially mono-energetic and mono-directional neutrons, various elastic and inelastic scattering processes populate lower-energy neutron groups . Neutrons in these groups have no useful angular correlation with the uncollided beam, and have reaction cross sections much different in general from those of the mono-energetic source. Because signals induced by scattered neutrons do not have the same intrinsic spatial correlation as those from the uncollided beam, PFNA inspection stations are usually designed to minimize such signals. The signals induced by scattered neutrons, can nevertheless reveal important complementary information for assaying various elements including Cl, Hg, 235U, and 239Pu, for example.

Evolving typically over millisecond time scales, the time structure of signals induced by neutrons in low-energy groups cannot be observed at the high frequencies ordinarily used for PFNA. Such measurements are possible however if the normal uniform pattern of PFNA pulsing is modulated into "macropulses" consisting of a chain of several nanosecond-wide pulses repeated at the usual interval, followed by a quiescent period up to ~1 ms. Conventional PFNA measurements can be performed during each macropulse, while concurrent Pulsed Thermal Neutron Analysis (PTNA) measurements of signals induced by lower-energy-group neutrons can be performed by analyzing signals between macropulses. Signals occurring after each macropulse are induced by neutrons whose energy spectra have progressively lower mean energy. This time-energy correlation is important, because the known variations of cross section with neutron energy (e.g., for fission and radiative capture) are the basis for matrix corrections not available otherwise.

SELF-SHIELDING CORRECTIONS IN PTNA MEASUREMENTS

Figure 1 shows MCNP simulation results of the radiative-capture gamma-ray production for different neutron-energy groups and time windows after a discrete pulse of mono-energetic neutrons. In the simulation, a 208-liter drum in an inspection cavity was irradiated by an isotropic 4.0-MeV neutron source. The drum was filled with a 150-kg homogeneous mixture of H, C, O, Al, Si, and Fe in ratios typical for mixed wastes (3). At the center of the drum was placed a 2-cm sphere containing pure CCl4, whose density was varied from 10% to 100% of normal density (1.59 g/cm3). The gamma-ray production rate has been plotted as a function of CCl4 density for each energy group and time window plotted. To more clearly illustrate the effect self-shielding has on gamma-ray production from different neutron groups, gamma-ray production rates have been normalized to those observed at the lowest density of CCl4 considered.

Fig. 1.

With a thermal-neutron cross section for Cl of ~33 b, the CCl4 suffers significant self shielding from thermal neutrons, even with CCl4 at only 25% of full density. Yet due to their smaller absorption cross section, higher-energy neutrons existing at <1 s are nearly unaffected by changes in CCl4 density. Cross sections for many elements vary inversely as neutron velocity at low energies, with strong resonances often appearing in the range up to ~1 keV. By comparing production rates of secondary radiations in different time windows with known cross section variations as a function of energy, the effective degree of self-shielding can be determined. Thermal-neutron-induced responses provide good sensitivity to small amounts of material, while systematic errors associated with self-shielding can be corrected for larger quantities of material by considering responses from higher-energy neutron groups.

An important complication to the shielding correction just described is presented by the presence of H. With mass virtually the same as that of the neutron, and a thermal-neutron elastic-scattering cross section of ~80 b, the H distribution in the sample strongly affects both the neutron energy spectrum and spatial distribution. Fortunately, the 0.33-b thermal-neutron capture cross section provides an intrinsic gauge of these effects. Because capture gamma-ray production from H occurs in exactly the same places that the probe neutrons are being moderated, these gamma-ray signals are a sensitive monitor of the local degree of moderation. In general, increasing H in the sample may either increase or decrease the total production rate for secondary radiations, but by monitoring thermal neutron capture from H itself, the degree to which this occurs can be determined. In particular, 2.2-MeV H gamma rays observed at early (e.g., <1 s) and late (e.g., > 200 ms) times reveal the net local effects of moderation and absorption. The effective degree of moderation determined from these responses can be applied as a correction to other capture gamma-ray signals measured concurrently. This is especially true for elements whose cross sections vary in the same 1/v manner as H. After correcting for the local H distribution, residual corrections may be applied for matrix absorption and self shielding effects, as described above.

CONCLUSIONS

Pulsed Fast Neutron Analysis is capable of mapping many elemental distributions. In principle, such maps provide a very complete basis for detailed predictions of neutron and photon transport. While such models can be valuable for accurately correcting matrix effects, other measurements may be performed on a PFNA platform to yield assay results relatively insensitive to moderation and self-shielding. PTNA

measurements are based on observations of secondary radiations in different time windows following a neutron pulse. By observing the time-dependent evolution of capture gamma rays from H as well as the secondary radiations from other elements of interest, differences in neutron energy spectrum and reaction cross sections make possible corrections for each of these effects in a heterogeneous matrix. By modulating the typically uniform pattern of PFNA pulses into "macropulses," PFNA measurements can be performed concurrently with such PTNA measurements, despite the much different time scales associated with the processes being measured in each case.

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37-9

FULL-SCALE INTEGRATED TECHNOLOGY DEMONSTRATION OF THE POLYETHYLENE ENCAPSULATION PROCESS FOR IMPROVED FINAL WASTE FORMS

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ABSTRACT

A full-scale integrated technology demonstration of a polyethylene encapsulation process, sponsored by the U.S. Department of Energy (DOE) Office of Technology Development, was conducted at the Environmental and Waste Technology Center at Brookhaven National Laboratory (BNL) in September 1994. Polyethylene encapsulation has been developed and tested at BNL as an alternative final waste form technology for improved treatment of low-level radioactive (LLW), hazardous and mixed wastes. A fully equipped production-scale system, capable of processing over 900 kg/hr (2,000 lb/hr), has been installed at BNL. The demonstration covered all facets of the integrated processing system including pre-treatment of aqueous wastes, precise feed metering, extrusion processing, on-line quality control monitoring, and process control.

INTRODUCTION

Following over 10 years of bench-scale research, development, and testing at BNL, the polyethylene encapsulation process was successfully demonstrated at production-scale during a Technology Demonstration sponsored by the U.S. Department of Energy's Office of Technology Development on September 13 and 15, 1994. Over 80 attendees representing DOE and the private sector participated. Figure 1 is a photograph taken during the demonstration. Simulated aqueous nitrate salt waste representing actual wastes at Rocky Flats Plant and a generic DOE mixed waste ash surrogate were processed. All aspects of the fully integrated system required for successful processing were included in the demonstration. Pre-treatment of the aqueous waste surrogate was accomplished using a vacuum dryer system marketed by VECTRA Technologies, Columbia, SC. Quality assurance monitoring of the waste and binder ratios was conducted using a transient infrared spectrometer (TIRS) developed at Ames Laboratory.

Fig. 1.

BACKGROUND

Polyethylene is a thermoplastic material with a relatively low melting point that can be processed at temperatures of 130 - 150C. Low temperature processing minimizes volatilization of hazardous constituents in waste streams and reduces the need for complicated off-gas and secondary waste treatment. Micro-encapsulation is accomplished when waste material is mixed with molten polyethylene and allowed to cool. Process development studies at BNL with polyethylene encapsulation have shown

successful process applicability to a wide range of waste types including nitrate salts and other evaporator concentrates, sludges, blowdown solutions, incinerator ash and ion exchange resins. Waste form performance has been thoroughly investigated (1,2,3) and, in general, is well above minimum requirements specified by the Nuclear Regulatory Commission (NRC) (4,5) and the Environmental Protection Agency (EPA) (6). Waste can be encapsulated with greater efficiency (i.e. more waste encapsulated per drum) and with better waste form performance using polyethylene than is possible with conventional solidification agents like cementitous materials. For example, polyethylene can incorporate up to 70 dry wt.% evaporator salt concentrates compared with a maximum of 12 dry wt.% for the best hydraulic cement formulation (7). Accounting for differences in waste form densities, polyethylene achieves at least a four-fold improvement (by weight) in the amount of waste incorporated per disposal drum.

PROCESS DESCRIPTION

The BNL polyethylene encapsulation process utilizes a versatile, industry tested, modified single-screw plastics extruder. Dry waste and polyethylene binder are continuously fed to the extruder by individual dynamic feeders. Distributive mixing within the extruder produces a homogeneous molten mixture that is extruded directly from the die into a waste container, such as a 208 liter (55 gal.) drum. The waste container is allowed to cool forming a solid monolithic waste form.

Based on bench-scale successes of the polyethylene encapsulation process with a range of waste types, scale-up to a full-scale encapsulation system that would match production capacity at a typical DOE treatment facility was planned. The BNL Demonstration and Full-Scale Test Facility encompasses a fully integrated encapsulation process including waste pre-treatment, material conveying, precise metering of feed materials, extrusion processing, and process control. Figure 2 is a photograph of the full-scale facility.

Fig. 2.

A nitrate salt solution (evaporator concentrate containing 35 wt.% dissolved solids) and a mixed waste ash were prepared and used as surrogate DOE waste streams for the demonstration. The composition of the nitrate salt waste surrogate is given in Table I. Wastes containing residual moisture require pre-treatment to bring them to dryness prior to encapsulation. Pre-treatment was accomplished using an indirect heated, stirred vacuum dryer. This technique was proven during bench-scale research and development and scale-up feasibility testing to produce free-flowing dry powders with less than 1% moisture that were amenable to extrusion.

A production-scale Liquid Volume Reduction System (RVR-200) was supplied by VECTRA Technologies, Inc., Columbia, SC. The blender/dryer portion of the RVR-200 can be seen on the right hand side of the photograph in Fig. 1. Dryer capacity is 757 liters/day (200 gal./day) equating to 408 kg (900 lbs.) dry salt/day for the nitrate salt surrogate containing 35 wt.% solids. The dryer was specially fabricated for the BNL Full-Scale Test Facility and incorporates design features to allow flexibility in testing various wastes. The liquid is charged to the vessel through a top port and discharged through a bottom 152 mm (6 in.) diameter, pneumatically operated ball valve, into a variable speed grinder to reduce particle size. The final product is emptied to 208 liter drums maintained under negative pressure to prevent powder dispersion.

Material feed is accomplished with two dry material feeders (AccuRate Model 610, Whitewater, WI) converted to a loss-in-weight (LIW) system (Merrick Industries Model 510, Lynn Haven, Fl). This system delivers accurate predetermined mix ratios of waste and binder to the extruder. The LIW control system consists of three computer controllers arranged in a feedback loop: a master controller that specifies the total feed rate and two slave controllers, one for each feeder, which calculate their respective feed rate as a fraction of the total feed rate depending on the waste/binder ratio. "Pacing" software allows the master controller to decrease the master reference feed rate if an underfed condition is sensed by either individual feeder. The master reference signal will decrease until both feeders can satisfy the lowered demand. The accuracy of a LIW system is better than 1% deviation from setpoint.

Waste and binder are dynamically fed to a 114 mm (4.5 in.) production-scale extruder (Davis-Standard, Pawcatuck, CT) with an output capacity of 900 kg/hr (2,000 lb/hr). The extruder is equipped with five electrically heated and water cooled zones and two die zones. Solid state differential thermocouple controllers maintain

temperatures to within 1C. Zone temperatures and pressures, melt temperature and pressure, current draw, and screw speed are constantly monitored.
An on-line monitor to provide real-time waste composition data for the polyethylene/waste melt as it is extruded has been developed at Ames Laboratory (8). The monitor is based on a technique known as transient infrared spectroscopy (TIRS). The monitor is installed at the end of the extruder and operates by inducing a small temperature differential on the surface of the extruded melt, reading the infrared spectra, then providing computerized analysis to convert the spectra to a waste composition. A photograph of the Ames TIRS system is shown in Fig. 3. By calibrating the monitor with spectra for known waste loadings, the instrument provides real-time data on the actual waste loading of encapsulated waste exiting the extruder. The TIRS system can be used to continuously check for any variations in the waste/binder ratio and provide on-line quality control data. If a significant variation is detected, immediate corrections can be made to the waste/binder feed ratio.
Fig. 3.
An open-loop, integrated process control and data acquisition system coordinates material feeding and extrusion processing. Process control is accomplished with computer software (LabVIEW for Windows, National Instruments, Austin TX) installed on a standard IBM compatible PC. The process control computer monitors the extruder output rate by weighing the quantity of material exiting the extruder. A low-profile scale located under the drum being filled sends a signal to the process computer via an RS-232 serial port. This signal is converted to an output rate by the process control software which, in turn, sends a 0-10 V signal to the Loss-in-Weight Master Controller as the updated total feed rate. This method allows the feed rate to match the output rate. Other process parameters, such as extruder zone temperatures and pressures, can be acquired and stored for future quality assurance.
Confirmation testing was conducted on the final waste form products from the full-scale demonstration for QA/QC monitoring and waste form performance. Core samples were taken from the final waste forms to test compressive yield strength as a measure of mechanical integrity, and for density measurements to check homogeneity of the waste and polyethylene binder. Compressive yield strength data for waste forms containing 30, 40, 50, and 60 wt% nitrate salt waste are represented graphically in Fig. 4 and range from 2150 psi to 2420 psi. Waste form density as a function of waste loading is given in Table II. Compressive strength and density results closely resemble data previously measured for laboratory-scale specimens (7).
Fig. 4.
CONCLUSION
The successfully completed production-scale technology demonstration at BNL has shown that polyethylene encapsulation is a viable new process for treatment of many problem mixed waste streams within DOE and the commercial sector. Over the past 10 years, it has progressed from proof-of-principle through bench-scale testing to production scale technology demonstration. The full-scale system at BNL's Environmental & Waste Technology Center complements a fully equipped process development and testing facility for investigating encapsulation technologies for various waste streams. Successful integration of waste pre-treatment, feed metering, extrusion processing, on-line monitoring, and process control during the production-scale demonstration provides necessary confirmation of process viability prior to "hot" testing with actual wastes. A full-scale field demonstration of a mobile polyethylene encapsulation system using actual mixed waste is currently planned. The polyethylene encapsulation process will then be ready for implementation within the DOE complex and the commercial sector.
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37-10

ENCAPSULATION/SOLIDIFICATION OF TRITIATED WASTE WATER USING POLYMERS AND POLYMER IMPREGNATED CEMENTS

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ABSTRACT

Tritium, especially in the form of tritiated water, is extremely mobile and no suitable treatment currently exists. Brookhaven National Laboratory has developed methods of stabilizing tritiated water in polymer impregnated cements (PIC) in which a porous (cement) waste form is first prepared, solidifying the waste material. The cement or concrete waste form is then impregnated with a thermosetting resin, and the resin is cured in place. The open porosity of the cement becomes filled with polymer, and, as a result, water infiltration and leach pathways are reduced. Other thermosetting polymer systems were also investigated including water extendible polyester styrene (WEP) for the stabilization of low concentration tritiated liquid waste streams. The water extendible polyesters were also used with a water absorbing starch polymer additive.

Tritium leachability from polymer impregnated or encapsulated forms relative to ordinary Portland cement (OPC) waste forms was used to determine performance. The effective diffusion coefficients (De) were calculated from the release rates. For the baseline OPC the De was 3.6x10-8 cm2/sec. De for the polymers were 3.5x10-9, 1.4x10-9, 1.2x10-9 and 9.7x10-10 cm2/sec for the PIC with a high alumina cement, the PIC with OPC, the WEP with starch polymer addition, and the plain WEP respectively. With the low activity waste, the baseline PC had reasonable performance by itself. Performance gains were 10 times for the high alumina cement-PIC, 26 times for the OPC-PIC, 30 times for the WEP/starch and 37 times for the WEP.

INTRODUCTION

During FY94 Brookhaven National Laboratory (BNL) was contracted by the Department of Energy's Office of Technology Development to perform mixed waste treatability studies using alternative polymer waste forms. Included in these studies was the encapsulation/solidification of liquid waste contaminated with low levels of tritium. Tritium, especially in the form of tritiated water, is extremely mobile and no suitable treatment currently exists.

In the mid-1960's to mid-1970's BNL developed and patented methods for tritium fixation using polymer impregnated waste forms (1). Polymer impregnated cement (PIC) waste forms are typically prepared in a two-step process. A porous waste form was first prepared, solidifying the waste material. Tritiated water was fixated as either a cement hydrate or adsorbed on a material such as silica gel incorporated in a cement matrix. The second step involved impregnating the cement or concrete waste form with a thermosetting resin, then curing the resin in place. The open porosity of the cement becomes filled with polymer, and, as a result, water infiltration and leach pathways are reduced. Tritium fixation was demonstrated by measuring laboratory leach rates in water. Full scale lysimeter testing was conducted at the Savannah River Site. Resin impregnated concretes were investigated in Japan for

waste containers and the composites thoroughly characterized (2). The method proved reasonably successful and resulted in waste forms of greatly reduced leachability and enhanced durability. Current concerns about the safe disposal to tritiated waste have lead to the re-investigation of this method as well as examining some of the newer polymers such as water extendible polyester styrene.

The tritium waste water treated in the earlier studies contained high levels of tritium (curies/L). The cost of tritium has since lead to improved recovery methods and the levels of tritium in liquid waste streams has been greatly reduced. The performance advantage of polymer impregnated cement may not be as great if the tritium source term is low. BNL investigated the use of polymer impregnated cement (PIC) for the stabilization of low concentration tritiated liquid waste streams. The waste stream used was tritiated cooling water, containing 50,000 picocuries/mL tritium, obtained from the High Flux Beam Reactor at Brookhaven National Laboratory. In addition to PIC a water extendible polyester styrene (WEP) was also investigated. Polyester resins are a commonly used family of thermosetting resins. By adding an emulsifying agent the resins can be made compatible with water. The water ends up encapsulated in tiny microcells evenly dispersed through out the polyester-styrene polymer.

FORMULATION

A PIC is made by impregnating a cement or concrete form with a liquid resin and then curing the resin in place. This fills the void space of the cement with polymer and reduces leaching pathways. Typical resin systems that can be used for the impregnation include; polyester-styrenes, methacrylates, furfuryl alcohol, etc. Several methods of curing the resins can be employed. Curing is a chemical reaction that results in the polymerization of the monomer. Polymerization can occur through gamma irradiation, at elevated temperatures using a thermally activated initiator, and at room temperature using a catalyst promoter combination. The latter two are preferred for polymer impregnation.

The polymerization of unsaturated monomers such as methacrylates and polyester-styrene is typically a chain reaction. Polymerization can be initiated by the action of a free radical on a monomer molecule, which leads to polymer chains consisting of thousands of monomer molecules. Free radicals can be formed by the decomposition of a relatively unstable material called an initiator or a catalyst. Peroxides are commonly used as initiators. The peroxide molecule splits at the O-O bond and when subjected to heat or in the presence of a promoter, forms two free radicals that have unpaired electrons and, thus, are very reactive.

Promoters can be used instead of temperature for ambient temperature curing of catalyzed monomer systems. Promoters (also called accelerators) are chemical compounds that induce the decomposition of a peroxide catalyst by breaking the O-O bond. This reaction can take place at a wide temperature range, depending on the promoter-catalyst system used.

Cure time is dependent upon temperature, promoter-catalyst combination and concentration, and admixtures (or contaminants) that may retard or enhance the set. Gel times (the time after which the resin viscosity increases rapidly and can no longer be poured or worked) were set at 3 hours for this study. Gel times can be manipulated easily by the resin manufacturer or the catalyst-promoter supplier. The reaction is exothermic and results in an autoaccelerating reaction that must be properly controlled.

Using a thermally activated initiator allows the greatest degree of freedom during the impregnation step. The waste form can be immersed in the liquid for as long as necessary to achieve full penetration of the form by the resins. Once the form is fully saturated the resin can be cured by heating the waste form. Thermally activated initiators exist that can be activated at low temperatures (~50C). With room temperature cures that require a promoter-catalyst combination the reaction begins as soon as the combination are brought together. This results in the resins having a limited "pot-life" or time that they remain at low viscosity and can still be used for impregnation. The waste form impregnation must be completed prior to the gel-time of the resin.

A methacrylate resin system was selected to fabricate the laboratory samples. Methacrylate monomers (acrylic) are a low viscosity, commonly used, family of polymers. A methacrylate resin manufactured by the 3M Company (3M 4R Concrete Restorer) was used. It is a modified high-molecular weight methacrylate (viscosity = 5-10 cps). The system consists of dicyclopentadienyl methacrylate and isooctyl

acrylate. Room temperature polymerization is accomplished using a cobalt octoate promoter (reducing agent) and cumene hydroperoxide initiator (oxidative catalyst). The resin was polymerized using a promoter-catalyst combination that yielded a 180 minute pot-life. Since the laboratory samples were small (3 to 20 cm) the impregnation time was low and samples were cured at room temperature without concern about impregnation and gel times.

The initial step in the PIC process is the preparation of the cement-based waste form. In previous BNL studies, a water to cement ratio of 0.2 was used which is below the theoretical value required to fully hydrate the cement. By using less water than required for hydration, all free tritiated water will be bound to the cement. The low water to cement ratio results in a dry paste that has to be vibrated (or tamped) into a mold. This also results in a more porous waste form that is more easily impregnated with the resin. BNL investigated ways of reducing the porosity of the initial cement form and pumpability of the cement paste. Lower porosity would result in slightly higher waste loadings and increased performance of the cement matrix in terms of leaching but would make impregnation more difficult. Increasing the pumpability or workability of the paste would result in reduced processing by eliminating the vibration compaction or tamping step. A super plasticizer was added to the paste to increase slump (more fluid) at the same water to cement ratio. The paste was molded into 10 cm diameter by 20 cm length cylinders and allowed to cure for 24 hours. The resultant waste form was apparently very low porosity, judged from microscopic inspection. A simple resin soak method of impregnation was not deemed feasible for such a low porosity matrix. To increase the depth of penetration into the sample a vacuum impregnation technique was attempted. The specimens were placed in a vacuum chamber for 6 hours to remove the air from the pore structure. Resins were then introduced, while maintaining the vacuum and the sample allowed to soak. After one hour the vacuum chamber was brought to ambient pressure, the samples were allowed to soak for an additional hour and then placed into sealed plastic containers and allowed to cure. After curing, the samples were cross-sectioned and the depth of penetration determined. The resins could only penetrate 2-3 mm into the cement matrix (see Fig. 1). The addition of pressure above ambient during the second soak phase did not increase the depth of penetration. Specimens of the higher porosity, low-slump cement which had high porosity were impregnated by placing specimen in a vat of resins for 2 hours after which they were transferred to plastic containers, sealed and allowed to cure. Cross-sectioning of these samples showed complete penetration of the resin into the sample. From these results it was apparent that the higher porosity cements had a great advantage over the lower porosity pastes during the impregnation process. All laboratory test specimens were prepared using a 0.2 water to cement ratio and no slump increasing additives.

Fig. 1.

SAMPLE PREPARATION

BNL fabricated bench scale samples of tritium encapsulated in ordinary Portland cement, PIC or WEP. Five lots of samples were prepared consisting of a baseline of portland cement, two types of PIC, and two formulations of WEP. The baseline was made by mixing tritiated water with type II Portland cement. The water-to-cement ratio was 0.35, a typical value used to make a workable cement paste. The cement paste was mixed in a conventional mixer for 2 minutes and the mix poured into a cylindrical mold and vibrated for 30 seconds. The cement was allowed to cure for thirty days before testing. The samples had average densities of 1.95 g/cm3. The final waste form contained 19,500 picocuries/cm3 (10,000 picocuries/g).

PIC samples were made by mixing tritiated water with either Portland type II cement (PC) or a high alumina cement (HAC) [Secar 80, LaFarge Corp., high calcium aluminate]. The water-to-cement ratio was 0.2 for all PIC samples. The paste was allowed to cure for 48 hours and removed from the mold. Waste forms were fabricated using the room temperature curing techniques described earlier. The resin and catalyst were premixed together prior to introducing the waste form. The sample was then immersed in the resin to allow the resin to infiltrate and fill the pore structure of the waste form. The gel time was set to be approximately three hours. The cement form was immersed for 2.5 hours, removed from the resin, replaced into the sample mold which was then capped and the final waste form was allowed to cure. All waste forms were fully impregnated, as verified by cross-section analysis of sample waste forms prepared with non-tritiated water. Waste form bulk density, before and after resin impregnation, and weight of resin absorbed during

impregnation were measured as a matter of process control. The PC-PIC samples had an average density of 1.78 g/cm3 and used 20 % (by weight) resin. The HAC-PIC samples had an average density of 1.88 g/cm3 and required 15.3 % resin for impregnation. The final waste forms contained 23,500 picocuries/cm3 (13,250 picocuries/g) for the PC based PIC and 26,300 picocuries/cm3 (14,000 picocuries/g) for the HAC based PIC.
The resin was a high molecular weight acrylic (3M4R 5742) manufactured by the 3M Company. The resin was purchased unpromoted and required the addition of a promotor (cobalt octoate) and an initiator (cumene hydroperoxide) for polymerization to occur. Once the initiator is added the polymerization reaction begins. The useful working time of the resin is about 90% of the final gel time. After this time the viscosity increases to a point that further impregnation is not possible. The gel time was set at three hours by varying the promotor/catalyst ratio. [It must be stated that the initiator and promotor are not to be added together or a violent reaction will occur. The promotor is added to the resin, thoroughly mixed and then the initiator may be added. The promotor can be added in advance, and the prepromoted resin has a shelf life of up to six months.]
There are several methods which can be employed to facilitate the impregnation of the cement paste. The simplest is to immerse the waste form in the resin at ambient pressure. This is suitable for high porosity materials or thin sections were depth of penetration is not limiting. Another way of impregnating introduces the resin to the form under vacuum. The vacuum is used to remove the air in the pore structure of the waste form which impedes resin flow into the pores. Much greater depth of penetration can be obtained this way. A final method of impregnation follows the vacuum technique but after the form is fully immersed in the resin, the system is pressurized to force the resin further into the sample. For the bench scale samples which are of limited diameter, the least complicated technique of immersion at ambient pressures was chosen.
The WEP samples were made by mixing tritiated water with an emulsifiable polyester styrene resin using a high shear mixer. The resins (Aropol WEP 662P) were obtained from Ashland Chemicals, Inc., prepromoted with cobalt and were initiated with methylethylketone peroxide. Two lots were prepared; the first lot of WEP had no additives, the second lot contained 0.2 % (by weight of WEP resin) of an absorptive starch polymer. The starch polymer was capable of absorbing 400 times its weight in water and was added as a dry powder to the WEP resin. It was hoped that the starch polymer would help hold the water more tightly in the waste form. The resin manufacturer recommends a maximum water content of 50%. For both lots the water content was 32% of the resin by weight (24% of the weight of the final waste form). The water was slowly added to the resins while the shear mixer was running. After complete addition of the water, mixing was continued for one minute, the MEKP catalyst was added and mixing continued for an additional thirty seconds. The mix was then poured into cylindrical molds and allowed to cure. Gel time of the resins was approximately five minutes. The polyester based samples had average densities of 0.99 g/cm3 for the plain WEP and 0.98 g/cm3 for the WEP containing the starch polymer. The final waste forms contained 24,500 picocuries/cm3 (24,800 picocuries/g) for the plain WEP and 24,500 picocuries/cm3 (25,000 picocuries/g) for the WEP and starch polymer.

WASTE FORM PERFORMANCE EVALUATION

Tritium leachability from polymer impregnated or encapsulated forms compared with ordinary Portland cement (OPC) waste forms was used to determine performance. Leaching of these samples was performed following the Accelerated Leach Test. This standard method, developed at BNL (3), predicts a sample leach rate provided data fits a diffusion controlled model. Samples tested were nominally 3.0 cm in diameter and 3.0 cm in length. The test involved 13 leachant changes over an 11 day period. The volume of leachate was modified from 3.0 liters to 0.5 liters to achieve reasonable count rates. Since the contaminant is tritiated water and the leachate is water there will be no concentration effects and the change in leachate volume has no effect on the test results. Specimens were suspended using monofilament fishing line approximately into the center of each solution. Duplicate samples were leached at 20C. Tritium concentrations were determined by liquid scintillation counting.
The effective diffusion coefficients (De) were calculated from the release rates (Table I). Figure 1 shows the cumulative fraction leached as a function of time for each of the solidification methods. For the baseline OPC the average De was 3.6x10-8 cm2/sec. Average De for the polymers were 3.5x10-9, 1.4x10-9, 1.2x10-9 and 9.7x10-10

cm2/sec for the PIC with HAC, the PIC with PC, the WEP with starch polymer addition, and the plain WEP respectively. With the low activity waste the baseline PC had reasonable performance by itself. Performance gains were 10 times for the HAC-PIC, 26 times for the PC-PIC, 30 times for the WEP/starch and 37 times for the WEP. The difference in leach rates of the PC-PIC and HAC-PIC is probably due to the lower polymer loading of the HAC forms. Leach rates appear to be in line with polymer content with pure polymer being lowest. The diffusion of tritium through the polymer is likely the rate determining step (diffusion occurs at a slower rate than the exchange of tritium between bound and unbound water molecules). Figure 2 graphically depicts the relationship of polymer loading to effective diffusion coefficient. As the amount of polymer in the waste form increases, and hence the wall thickness of the diffusion barrier increases, the leach rate appears to decrease.

TABLE I

Fig. 2.

Fig. 3.

CONCLUSIONS

In terms of leachability of tritium pure polymer (WEP) performed slightly better than waste forms containing 20% polymer (PIC) and much better than plain portland cement (OPC). The final choice of stabilization/solidification method will depend on the waste form performance requirements, cost and processing requirements. The processing and cost are lowest for the OPC and for low concentration waste streams the performance may be adequate. As the waste stream concentration (of tritium) increases the performance of OPC is expected to be greatly reduced. Where high performance is required the polymer systems may be more suitable. The performance differences for the WEP and PIC were not that great and as such a compromise must be made between material cost of the polymers and processing costs. Thermosetting polymer resins cost vary from $0.60/lb for common polyester-styrene resins to $6.00/lb for specialty resins. For polymer impregnation commonly available, low cost resins can be used. The cost of these should range from $0.60/lb to $1.00/lb. For the pure polymer system with aqueous waste a water extendible resin must be used and these are generally more expensive ranging from $1.50/lb to $3.00/lb. Choosing a 55 gallon drum and using the densities and formulations of the laboratory samples results in the following requirements: OPC waste forms use 662 lbs of cement; WEP waste forms use 345 lbs of polymer resin; and PC-PIC waste forms use 544 lbs of cement and 163 lbs of polymer resins. Assuming $0.05/lb for cement, $0.60/lb for resins for PIC and $1.50/lb for WEP resins results in a materials cost of $33 for OPC, $125 for PC-PIC and $517 for WEP. Processing for OPC and WEP are simple mixers; standard cement or in drum mixers for OPC and a high shear mixer for WEP. Both systems are commercially available. For the PIC a multi-step process is required; mixing and curing of the cement followed by impregnation which may require an evacuation step for 55 gallon drum scale. The additional cost (capital, maintenance, and operations) of these added steps must be considered.

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PROPOSED WASTE FORM PERFORMANCE CRITERIA AND TESTING METHODS FOR LOW-LEVEL MIXED WASTE*

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ABSTRACT

Proposed waste form performance criteria and testing methods were developed as guidance in judging the suitability of solidified waste as a physico-chemical

barrier to releases of radionuclides and RCRA regulated hazardous components. The criteria follow from the assumption that release of contaminants by leaching is the single most important property for judging the effectiveness of a waste form. A two-tier regimen is proposed. The first tier consists of a leach test designed to determine the net forward leach rate of the solidified waste and a leach test required by the Environmental Protection Agency (EPA). The second tier of tests is to determine if a set of stresses (i.e., radiation, freeze-thaw, wet-dry cycling) on the waste form adversely impacts its ability to retain contaminants and remain physically intact. In the absence of site-specific performance assessments (PA), two generic modeling exercises are described which were used to calculate proposed acceptable leach rates.

INTRODUCTION

The U.S. Department of Energy (DOE) is developing processes and systems for the treatment and disposal of radioactive and hazardous mixed wastes accumulated during decades of weapons production and other operations. The treatment and disposal of these wastes has to be performed in a way that minimizes exposure to operational personnel, assures the health and safety of the public, and protects the environment from the risks associated with the release of hazardous chemical and radioactive components from the waste. These wastes must be managed also according to applicable State and Federal regulations, as well as DOE orders.

There is no single set of regulations which governs the treatment and disposal of all of DOE's low-level mixed waste (LLMW), although in 40 CFR Part 268.42 the EPA has identified technology-based treatment standards for a limited number of mixed wastes. Management of the chemically hazardous component of mixed waste must meet EPA regulations in accordance with the requirements of the Solid Waste Disposal Act, as amended by RCRA in 1976, and the Hazardous and Solid Waste Amendments Act of 1984. In cases where the state's regulations are more stringent than EPA's, the state's regulations prevail, according to the Federal Facility Compliance Act of 1992 (FFCA). The management of the radioactive components of the waste is governed by the guidance in DOE Order 5820.2A, Chapter III, Management of Low-Level Waste. If the waste is to be transported, it must meet the Department of Transportation and State transportation requirements.

The first barrier to releases of contaminants after disposal in a multiple barrier system is the solidified waste form. In the past, waste form characteristics were considered to be of secondary importance to the hydrogeochemical characteristics of the disposal site. Experience gained in operating low-level waste (LLW) disposal sites has made it apparent that waste forms play an important role in isolating the radioactive and other toxic components of the waste. Degradation of waste forms in disposal has resulted in releases of the contaminants from the disposal units into the environment. Also, poorly solidified waste forms, due to chemical and mechanical instabilities, have resulted in incompletely solidified waste unsuitable for disposal, thus requiring reprocessing at additional cost.

In order to judge whether or not the waste form will retain the hazardous components, waste form performance criteria are necessary. Such criteria, related specifications and test methods will allow comparison of technologies for treatment systems, demonstrate regulatory compliance, introduce uniformity into waste form comparisons across the DOE complex, improve quality control, help predict long-term waste form performance, and generate data for input to models for site performance assessment. The criteria and testing methods proposed here provide guidance for those developing and testing waste forms.

WASTE FORM PERFORMANCE CRITERIA AND PERFORMANCE ASSESSMENT MODELS

Waste form performance criteria are meant to provide some level of assurance that the treated waste will perform as expected over a defined period of time. To establish the necessary level of performance the waste form must be viewed as part of the entire waste disposal system; including components such as the engineered structure, the container, the geochemistry of the site and its climate. After estimates have been made quantifying the various components of the disposal facility and its environment, a performance assessment (PA) model can be used to estimate the dose that would be received as a result of disposal under the assumed conditions. By varying the quantities used to describe individual functions or performance, the PA model can be used as a tool that provides, within rather broad ranges, an assessment of how the various components of the disposal system interact to alter the final dose estimate. Waste form performance, in the sense of release rates, is a major

factor in the performance of the system. While it is desirable to use PA models developed for site-specific applications to develop criteria for each site, those PAs are several years away. Meanwhile, there is an urgent need within DOE to develop generic criteria so that waste treatment processes and site designs can proceed and be tested against the criteria.

To establish leach rate guidelines for use in the waste form performance criteria (Tables I and II), two sets of PA modeling exercises were used. Each set included two generic sites, humid and arid. One set was conducted for radionuclides by IT Corp (1). for the DOE Mixed Waste Treatment Project and the other for RCRA metals by Brookhaven National Laboratory (2). Similar methods and assumptions were used. However, the regulatory criteria for the two are different. For the radiological modeling an effective dose equivalent of 25 mRem/year (DOE Order 5820.2A) is used as the pass/fail limit. For RCRA metals the limit that was chosen is a metal concentration that is 100 times the EPA drinking water standard in ground water at the bottom of the disposal trench (3).

Results from the radiological assessment indicate that the major contributors to the dose at a generic site are 238U and its daughters, 228Ra, and 99Tc. The generic PA model for a humid site uses a release rate for the waste form of 8.12 x 10-6 per year (8 parts per million per year), which yields an effective dose equivalent greater than the DOE limit. For an arid site, the same waste form release rate results in an annual effective dose equivalent that is below the DOE limit. It is the inventory, the geochemistry and the hydrology of the site that control doses over long times (>1000 years). Considering the large uncertainties of the estimated doses, and the fact that release rates measured in the laboratory test procedures are conservative since the tests expose the waste form to far more water than a disposal environment, we have chosen a maximum leach rate of 10-5 per year as a reasonable generic value for wastes containing long-lived or high toxicity radionuclides. This value is subject to change if site-specific PAs show that site characteristics will accommodate higher (or require lower) leach rates.

These results, as well as those from work at Savannah River Site (SRS) (4), indicate that the radionuclides that have the greatest impact are those with long half-lives and some of the daughters that are generated by their decay. It is evident that large quantities of U/Th wastes will result in significant long-term doses at humid sites, even for a waste form with a very low release rate. This is the result of the extremely long half-lives and the limited credit that can be claimed for the long-term performance of waste forms and engineered structures. Thus, for some radionuclides limits on inventory in the disposal facility are necessary.

The presence of long-lived radionuclides makes it clear that there needs to be an approach by which doses are attenuated, if not reduced in total, for the long-lived radionuclides. This can be achieved by requiring very low releases from the waste form. Radionuclides with shorter half-lives can be sequestered sufficiently long by the disposal facility to allow them to decay to levels at which there is little concern, assuming other features of the disposal facility perform as expected. As a result, very low leach rates only need to be required for long-lived (or high hazard) radionuclides, while short-lived radionuclides can be released at higher rates without increasing risk. This concept places emphasis on waste form quality (regarding release) where it is needed, while accepting a lower quality and, therefore, usually lower cost waste form when appropriate. This is reflected in the three levels of waste form performance in Table I. These three categories were determined by a method which includes calculation of a figure of merit (FOM) based on allowable limits of intake (ALI) (5), the half-life and a generic inventory of DOE LLW. This method is described in Ref. 2.

A similar argument can be made for non-radioactive hazardous constituents of the waste. Materials regulated under RCRA are found in an estimated 70,000 cubic meters of stored low-level mixed waste and in currently generated waste streams that have an annual generation rate of about 7700 cubic meters per year (6). From a preliminary review of the gross composition of mixed wastes (personal communication with Wayne Ross, Pacific Northwest Laboratory) approximately 2 weight % of the waste consists of lead, 1.9 weight % is chromium, 1.5 weight % is nickel, 0.02 weight % is mercury (all in one waste stream), 0.02 weight % cadmium (two waste streams) and 0.02 weight % beryllium (two waste streams). Releases of these materials must also be considered with respect to requirements in the DOE order regarding protection of public health and groundwater resources.

Unlike assessment of doses resulting from radionuclides, a PA is not required for RCRA constituents. However, to calculate what the guidelines should be for the limit on the release rate of RCRA constituents from the waste form, a PA method similar to the one described above for radionuclides was employed (2).
Assessments of releases from an underground disposal unit using the Disposal Unit Source Term (DUST) Code (7) were made for lead (Pb), chromium (Cr), and mercury (Hg). This was one of the codes which was used by IT Corp in the PA modeling discussed above. The DUST Code is a one-dimensional code that models the transport of contaminants from waste containers to the disposal unit boundary. The parameters were chosen to match as closely as possible the conditions used in the IT Corp. PA. The effects of several parameters were examined. Of greatest interest was the effect of a waste form's fractional release rate on contaminant concentration at the base of the disposal unit and how these concentrations differed at humid and arid disposal sites. Although it is certain that the water infiltration rate affects the fractional release rate, the intent was to examine the effects of varying conditions on contaminant transport by varying one parameter at a time. A second reason for the DUST modeling exercise was to see how concentrations of Pb, Cr, and Hg (which correspond roughly to potential dose exposures for radionuclides) changed as a function of time. A value one hundred times the EPA drinking water standard for the specific RCRA metals was used as a target (at the bottom of the disposal unit) concentration.
Waste form fractional release rates from 10-8 to 10-2 were used in the calculations. From this exercise release rate targets for waste form performance were developed that are not based on regulations, but on performance needed under a set of disposal conditions. While actual allowable release rates should be determined on a site-specific basis, this process allowed the setting of an annual release rate of 10-5/yr as a numerical guideline. The modeling indicates that at this release rate the groundwater concentrations below the trench slightly exceed the target concentration. However, the modeling is highly conservative, since it did not consider factors such as sorption by the soil and solubility limits of contaminants.
PROPOSED CRITERIA AND TESTS
The modeling results show that the fractional release rate is a prime consideration in assessing long-term performance of and subsequently developing criteria for mixed waste forms. The approach to developing the criteria presented in this paper is based on the following:
Release of contaminants by leaching is the single most important parameter by which the effectiveness of a waste form is judged.
The waste form should retain the original desirable qualities of the solidification agent.
Acceptable release rates of contaminants from waste forms are defined by analysis of releases from disposal facilities through Performance Assessment modeling.
Solidified LLMW contains <100 nCi/gm of TRU
Only inorganic RCRA hazardous components are addressed here. It is assumed that organics have been removed or destroyed in a pretreatment step or are below regulatory limits.
The waste form is a monolith.
Since we believe that the rate of release of contaminants from a waste form is the single most important parameter exhibited by the solidified waste, then leachability is the primary property against which a waste form is judged and, as such, is also used to assess the impact that several types of stress can have on the solidified product. Consequently, the first tier of tests includes a leach test required by EPA and a leach test designed to determine the net forward leach rate of the solidified waste. The first tier tests should be performed first to determine acceptability. Only on passing the given specifications for the leach tests should other tests be performed. The second tier of tests is designed to determine if a set of "assaults" on the solidified waste adversely impact its ability to retain contaminants and to remain physically intact. They are meant to be representative of potentially detrimental factors (either external to the waste form or as a result of internal processes such as irradiation). Some second tier tests are not appropriate for all materials and can be omitted for certain solidified wastes. This is left to the judgment of those doing the testing. However, it is important that all materials should be tested at least once with all of the second tier tests because the history of solidification processes is replete with surprise failures. In the case where

standard tests are not available, the following is suggested. For radiation stability, if a waste form is expected to be exposed to greater than 106 R over a period of 300 years, then it should be tested by irradiation with a source of high energy gamma rays. The total exposure should be equivalent to the total exposure that the waste form will accumulate over 300 years. For testing the stability in water, the samples should be immersed in distilled water for 90 days. In the case of testing for freeze/thaw stability, the apparatus and cycling schedule of ASTM B-553 (8) should be used at temperature ranges of 60C (3C) to -20C (3C). It is recommended that once nondestructive testing (NDT) has been developed for examining waste form integrity, it should be used to supplement or replace compressive strength methods in order to reduce the time and cost involved. A compilation of the proposed criteria and tests is presented in Table II and a more detailed discussion is presented in Ref. 2.

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WASHING TECHNIQUES AS AN ALTERNATIVE TO INCINERATION

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ABSTRACT
The Environmental Protection Agency (EPA) best demonstrated available technology under the Resource Conservation and Recovery Act (RCRA) for many hazardous wastes is incineration (1). However, public pressure has caused state and federal regulators to consider tightening emission standards on incineration units. Accordingly, the U. S. Department of Energy (DOE) Environmental Management (EM) and the Office of Technology Development are exploring alternatives to incineration to address the large amounts of existing hazardous and mixed waste. Thus, the purpose of this paper is to review washing technologies as an alternative to incineration.
Soil washing is the extraction of contaminants from excavated soil by mixing the soil with water, solvents, surfactants, or chelating agents (2). Soil washing is a combination of physical and chemical treatments performed on soil in an aqueous solution. It is often used as a size segregation technique with washing of fines from coarse soil. The process removes contaminants that reside in specific grain-size domains, separates the waste stream into "cuts", and focuses on treatment appropriate to the contaminant/grain-size relationship. Usually, heavily contaminated soils are treated several times in a multistage counter-current system. The contaminated water or solution is then treated for removal of contaminants.
Early development and implementation of soil washing technology was based primarily on the use of water as the extracting solvent (3). This process is most applicable in the remediation of soils containing highly water-soluble constituents or soils containing low concentrations of silt and/or clay particles. Currently soil washing processes include extraction media ranging from alkaline mixtures of ionic and nonionic surfactants and bioremediating agents to acid extractions. Early soil washing applications were for topsoil rather than for buried and otherwise stored waste, although some buried waste and contaminants would be amenable to this approach. Additionally, washing processes may provide added benefit for treatment of debris.
INTRODUCTION
The EPA best demonstrated available technology under RCRA for many hazardous wastes is incineration (1). It has been shown to be effective in meeting the destruction and removal efficiency requirements of RCRA and the Toxic Substances Control Act. However, public pressure has caused state and federal regulators to consider tightening emission standards on incineration units. In May 1993, EPA announced a de facto moratorium on the permitting of new incinerators until new emission standards are established. Accordingly, DOE-EM and the Office of Technology Development are exploring alternatives to incineration to address the large amounts of existing hazardous and mixed waste.
Approximately 250,000 m3 of mixed low-level waste (MLLW) is currently retrievably stored within the DOE complex (4). An additional 900,000 m3 of MLLW is expected to be generated through waste operations, environmental restoration, and decontamination and decommissioning activities over the next five years (1). Treatment capacity for the majority of these wastes can be described as: 1) no proven treatment technology exists, or 2) currently available technologies need extensive modification to be capable of treating radioactive waste. The existing MLLW matrices as a percentage of the total stored mixed waste are listed in Table I (5). Although the concentrated organic material requiring destruction is not large, organic compounds may be spread throughout these matrices in low concentrations.
Soil washing is the extraction of contaminants from excavated soil by mixing the soil with water, solvents, surfactants, or chelating agents (2). Soil washing is a combination of physical and chemical treatments performed on soil in aqueous solution. It is often used as a size segregation technique with washing of fines from coarse soil. The process removes contaminants that reside in specific grain-size domains, separates the waste stream into "cuts", and focuses on treatment appropriate to the contaminant/grain-size relationship. Usually, heavily contaminated soils are treated several times in a multistage counter-current system. The contaminated water or solution is then treated for removal of contaminants.
TABLE I
Early development and implementation of soil washing technology was based primarily on the use of water as the extracting solvent (3). This process is most applicable in the remediation of soils containing highly water-soluble constituents or soils

containing low concentrations of silt and/or clay particles (usually <15% smaller than 50 m). The EPA Superfund Innovative Technology Evaluation (SITE) program now includes soil washing processes using extraction media ranging from alkaline mixtures of ionic and nonionic surfactants and bioremediating agents that act as biosurfactants (6) to acid extractions for removing heavy metals from contaminated soils and solids (7). These processes represent a significant extension of existing soil washing techniques presently being used on a wide scale in Germany and the Netherlands.

Early soil washing applications were for topsoil rather than for buried and otherwise stored waste, although some buried waste and contaminants would be amenable to this approach. For example, if the contaminant is soluble and the contaminated surfaces are accessible (removed from bags and containers), it may be possible to decontaminate the bulk of such buried waste by washing, leaching and, extraction. Additionally, washing processes may provide added benefit for treatment of debris by envoking the debris rule (see Debris Washing).

EPA (8) has the following insights on soil washing:

"Soil washing is basically a volume reduction technology that uses wash water to separate contaminated soil into two fractions: a large fraction of relatively clean, coarse soil and a smaller fraction of fine soil/sludge containing the concentrated contaminants.''

"Hydrophobic contaminants can be difficult to separate from soil particles into the aqueous washing fluid.''

Complex mixtures of contaminants in the soil make it difficult to formulate a single suitable washing fluid that will remove all the different types of contaminants from the soil.

"A high percentage of clay and silt (e.g., >>30-50%) in the soil usually indicates that soil washing will be unfavorable due to the amount of time and money required to treat this volume of contaminated soil.''

"Chelating agents, surfactants, solvents, and other additives are often difficult and expensive to recover from the spent washing fluid and then recycle in the soils washing process.''

Identification of the contaminants is important. Volatile organics may require removal in a pretreatment step or treatment of air emissions. Solubilities of contaminants is important in deciding whether wash solution additives are needed. Metal speciation is important in metal solubility.

Soil particle-size distribution and chemical analysis of the contaminants are important characterization data for soil washing.

TECHNOLOGY DESCRIPTION

Soil washing technology is commercially available through numerous vendors. A search of EPA's Vendor Information Systems for Innovative Treatment Technologies (VISITT) data base, version 3 (9), identified 19 soil washing vendors with processes in various stages, from laboratory bench-scale to full commercial scale. Each soil washing vendor uses a slightly varied process train and/or reagents for removal of contaminants (10, 11, 12, 13). As reported by the vendors, these processes are in the following stages of development: three bench-scale, two pilot-scale, and 14 full-scale. Additionally, numerous soil washing demonstrations (including solvent extraction and debris washing) have been conducted as part of the EPA SITE program (14).

Soil Washing

Soil washing is an ex-situ process that incorporates size classification and vigorous scrubbing of soil particles with water to remove heavy metals or organic contaminants. Soil washing can be and has been used as a single-stage, stand-alone technology where applicable or coupled with other on-site remediation technologies to achieve desired final contaminant levels. In some cases, water-soluble surfactants, chelating agents, acids, or bases may be used to facilitate contaminant removal. Soil washing is based on the observations that: (1) contaminants tend to concentrate in the fine and organic fractions of the soil (e.g., silt, clay, humic matter) due to physical and/or chemical adsorption; and 2) contaminants associated with the coarse soil fraction (e.g., sand, gravel) are often present as a thin surface coating that can be removed by vigorous scrubbing and attrition of the soil particle surfaces (9) (Fig. 1) (15). Many soil contaminants, both organic and inorganic, tend to chemically or physically attach to the silt and clay fractions of the soil. The silt and clay, in turn, tend to attach to coarse sand and gravel

particles. The various processes used in soil washing break the silt and clay away from the coarser fractions and scrub the coarser fractions, resulting in clean sand. Fig. 1.
Thus, the objectives of the technology are: 1) to reduce the volume of soil that require further treatment or disposal by concentrating the contaminants, 2) to transfer the contaminants to a more easily treated aqueous phase, and 3) to produce a washed soil product that meets clean-up criteria.
A "basic" soil washing treatment system (Fig. 2) includes four major sub-systems: 1) screening (soil preparation), 2) washing, 3) soil and water separation, and 4) waste water and sludge treatment and management (9,16,17). Screening is employed to remove the oversized soil fraction. The initial process is accomplished with vibrating screens and then with attrition scrubbers. Water is then added, and the resulting slurry is separated into coarse-grained sands and fine-grained materials, typically by use of hydrocyclones. Underflow from the hydrocycLones contains the coarse-grained sands that require treatment, typically by air flotation. Surfactants may be added to the sand to aid in reducing the surface tension binding the contaminant to the particles (18). This allows the contaminants to "float." They are then removed by air flotation and forwarded onto sludge management. Overflow from the hydrocyclones contains the fine-grained, contaminated material. This fraction can then be either processed by an alternative technology or dewatered and disposed of off-site. The small volume of contaminated residuals are subsequently treated by appropriate destructive or immobilizing processes such as thermal desorption, chemical extraction, biodegradation, solidification, or vitrification. Sludge management typically requires additional treatment such as polymer addition, thickening, and dewatering. Cleaned soil may be either redeposited on-site or otherwise beneficially used as backfill or industrial sand. For MLLW, if soil is treated in such a way that the waste can be considered low-level waste, then the treated soil can be disposed of at an appropriate facility (e.g., Nevada Test Site). Process water is cleaned of contaminants and recycled for further use in the system. Fig. 2.
Other variations to soil washing technology include debris washing, solvent extraction, solvent washing, and in situ soil flushing. General descriptions of debris washing and solvent extraction are presented below. Solvent degreasing was used successfully at the Oak Ridge K-25 site, but its use was stopped in order to avoid exposing the workers and the environment to the hazardous solvents. In situ soil flushing has been demonstrated and consists of water or an aqueous solution, injected into, or sprayed onto, the area of contamination and the contaminated elutriate collected and pumped to the surface for removal, recirculation, or on-site treatment and reinjection (2). During elutriation, the flushing solution mobilizes the sorbed contaminants by dissolution or emulsification. Handling and treatment of the elutriate are required, and delivery and recovery systems are needed. Medium-solubility organic compounds and soluble salts are the most likely contaminants to be treated.
Debris Washing
Sludge and debris washing technologies are similar to soil washing, but typically require additionally stages for screening, mixing, and scrubbing, as well as variations within the stages (e.g., sorting, high-pressure washing, solvent treatment). Debris is defined by EPA as a solid material exceeding a 60-mm particle size that is intended for disposal and that is: 1) a manufactured object, or 2) plant or animal matter, or 3) natural geologic material (19). Further, any mixture where the debris portion comprises the largest amount of material present by volume as determined by visual inspection is classified as debris. The Hazardous and Solid Waste Amendments to RCRA allow hazardous wastes to be land-disposed only if they are treated or can be land-disposed in units satisfying the so-called no-migration standards in RCRA section 3004. Land disposal includes any placement of hazardous waste in a landfill or other facility specified in RCRA 3004 (k). Alternative treatments are permitted for debris in lieu of waste-specific treatments. Application of these regulations to qualifying MLLW can result in the following (19):
- simplifying processing and associated characterization requirements;
- shortening the research, development, demonstration, testing, and evaluation time frame;
- reducing the cost of processing and disposal; and

reducing the potential health risks to operations personnel and the public by reducing waste treatment and handling processes.

In addition to washing technologies, alternative treatment technologies that are applicable to treatment of MLLW include: biodegradation, physical extraction (e.g., abrasive blasting, scarification, grinding, spalling, high-pressure steam and water), thermal extraction (e.g., high-temperature metals recovery, thermal desorption), chemical destruction (e.g., oxidation and reduction), thermal destruction (e.g., incineration, plasma arc), macroencapsulation, and microencapsulation (19). Debris washing technologies include: water washing, solvent washing, chemical foams and gels, acid washing caustic washing, liquid-phase solvent extraction, and vapor-phase solvent extraction. The EPAs Risk Reduction Engineering Laboratory and IT Corporation have developed an on-site metallic and masonry debris decontamination process (14, 20,21). The entire system is mounted on two 48-ft flatbed semi-trailers and can be readily transported to the site. This system can be applied to various types of debris (scrap metal, masonry, stones) contaminated with pesticides, PCBs, or metals. The net costs for on-site debris decontamination ranged from $50 to $75 per ton and the system is estimated to be able to handle from 50 to 120 tons of typical debris per day.

Solvent Extraction

Solvent extraction is potentially effective in treating contaminants by separating solids (sludges or soils) into three fractions: oil, water, and solids. As the fractions separate, contaminants are partitioned into specific phases. Typically solvent extraction employs three treatment steps: soil washing using a solvent, soil drying, and solvent regeneration (12).

First, contaminated solids, slurries, or waste waters are fed into the extractor. The extraction fluid (solvent) is circulated through the contaminated matrix to wash the soil. Liquified gas solvent may be used to extract organic compounds, oil, and grease from contaminated sludges and soils. Controlled temperature and pressure optimize the washing procedure. Typically, more than 99% of the organic compounds are separated from the feed. Following phase separation of the solvent and organic constituents, treated water is removed from the extractor, while the mixture of solvent and organic constituents passes to the separator through a valve where pressure is partially reduced. In the separator, the solvent is vaporized and recycled as fresh solvent. The organic compounds are drawn off from the separator and either reused or disposed. Hot inert gas or solvent-free gas is heated and injected into the soil to complete drying of the soil.

Extractor design is different for contaminated waste waters and solids. For waste water, a tray tower contractor is used, whereas for solids a series of extractor/decanters operating counter-currently are employed. Extraction efficiencies of 90 to 98% have been achieved on sediments containing PCBs, while laboratory experiments have yielded 99.9% removal efficiencies for volatile and semivolatile organic compounds. Estimated costs for PCB cleanup range from approximately $150 to $450 per ton of waste.

Several different solvent extraction technologies are available. For example, the Resources Conservation Company's Basic Extractive Sludge Treatment (B.E.S.T) technology utilizes triethylamine as the extraction agent (14). Triethylamine has an inverse miscibility in which it is hydrophobic above 20C and hydrophilic below 20C. This property allows the process to extract both aqueous and nonaqueous compounds by changing the temperature of the solvent, and solids can be dewatered while simultaneously extracting organic contaminants. The Low-Energy Solvent Extraction Process (LEEP) uses common organic solvents to extract and concentrate organic contaminants (14). Contaminants are removed from the soils by hydrophilic leaching solvents then concentrated by use of a hydrophobic stripping solvent. Bradtec has developed and tested a dilute, aqueous based, extraction process that combined dissolution with dilute selective solvents, contaminant recovery and solvent regeneration for treatment of radionuclide and metal contaminated soils (22). Finally, the Center for Hazardous Materials Research uses an acid extraction treatment system in the soil washing process that uses hydrochloric acid to extract contaminants from soils (14).

INPUT AND OUTPUT STREAM CHARACTERISTICS

Ex situ soil washing is applicable to soil, sediments and sludges contaminated with hazardous organic compounds, radionuclides, and toxic heavy metals (Table I). Specifically the process can be applicable for treatment of soils contaminated with

wood-preserving chemicals (e.g., PCP, creosote), organic solvents, electroplating residues (e.g., cyanides, heavy metals), paint sludges, pesticides, and petroleum and oil residues (8). Particle-size distribution is the key parameter for determining the feasibility of soil washing. Typically for soil washing to be economical, the contaminated material size distribution should not consist of more than 40% passing 45 microns or 325 mesh (9). Materials finer than this begin to reduce the amount of recovered clean soil for redeposition. In addition, the contaminated soil should contain less than 20% by volume of solid organic material such as leaves, roots, and twigs. For nonmobile treatment units, potential sites should have a minimum of 5,000 tons of material in order to justify the fixed costs of mobilization and demobilization. In situ soil washing (flushing) is most applicable to highly permeable soils. All input streams must be sorted, separated and prepared prior to soil washing (23).

TABLE II

There are four waste streams generated during soil washing: 1) contaminated fines and humic compounds from the soil washing unit, 2) waste water, 3) sludge from the soil washers, and 4) air emissions (16). Soil washing is typically considered a waste minimization, volume reduction process which reduces the original amount of material that needs to be remediated by an ultimate destruction or immobilization technology. Thus, solid output soil washing streams can be further separated into two broad categories: 1) coarse-grained cleaned material for on-site backfilling and redeposition and 2) fine-grained material requiring further treatment for regulated disposal. Discharge water may need treatment to meet discharge standards; however, this water should be recovered and reused to the maximum extent possible in the washing process. The residual solids from waste water treatment process, such as spent carbon and sludges, must be treated before disposal. Any air emissions must be collected and treated to meet regulatory standards.

TECHNOLOGY STATUS

Soil washing is an available technology that has been used as a remedial action for contaminated soils. Several soil washing demonstrations have been conducted by the EPA at Superfund sites. For buried waste that is conceptually the same as soil but more heterogeneous, soil washing is in the development stage. Doubtless, modifications will have to be made to handle buried waste and the secondary waste streams. This concept may prove effective for a given scenario but not necessarily in general. Once the concept has been accepted as effective, then details such as emptying bags and containers need to be developed. The most difficult part to develop may be proving that the treated waste is clean. The whole point of the treatment is to cleanse the bulk of the waste of contamination and to take advantage of the debris rule. A technique must be developed to ensure that the treated waste is truly free of contamination. Without such assurance, the treated waste will be disposed as if it were still contaminated, even if it is not. Finally, the wash water/leachate/extract is a secondary waste. Cleaning and recycling this secondary waste stream must also be developed if the concept proves to have merit.

Several full-scale soil washing plants with capacities of 6.5 to 30 tons/h are in operation in the Netherlands and Germany (24). Soils (165 tons) from the United States have been shipped to the full-scale facility at Moerdjik, the Netherlands, for testing (25). This plant has an annual treatment capacity of 80,000 tons. However, emphasis in Europe is on removal of semivolatile organic compounds, polyaromatic hydrocarbons, PCBs, and pesticides; none of the plants are presently removing radionuclides or heavy metals from soils. Also, most of the soils and sediments being remediated are sandy-textured (>85% being >50 m diameter).

For the specific contamination found at each individual site considering this alternative technology, further development is required. Field screening and material handling as needed in conjunction with sediment removal, and further treatment will be required following removal. Additionally, a systems tradeoff may be required to determine if the process has merit for the proposed application.

REMOVAL EFFICIENCY

Based on the various individual site characteristics, removal efficiencies are reported anywhere from 85 to 99% for organic compounds, 80 to 99% for semivolatile organic compounds, 50 to 99% for oil and grease (and other petroleum hydrocarbons), 84 to 88% for PCBs, 93 to 99% for cyanides, and 50 to 90% for heavy metals (see Fig. 3) (8,9). As a very general statement, applicable contaminants can usually be removed from coarse soil fractions (greater than 200 mesh) with relatively high

removal efficiencies (95 to 99.9%) at low to moderate cost, while fine soils, silts, and clays (<200 mesh) yield only moderate contaminant removal efficiencies (50 to 90%) at a moderate to high cost (9). Volatile organic contaminants often are easily removed from soil with a 90 to 99% efficiency. Semivolatile organic contaminants may be removed to a lesser extent with use of a surfactant (40 to 90%), while metals and pesticides, which are less soluble in water, often require acids or chelating agents for removal (8).
Fig. 3.
Bench-scale studies have been conducted evaluating the effectiveness of various extraction media on the removal of plutonium (Pu) and americium (Am) (26,27,28,29). Extraction media included (1) pH 12.5 NaOH solution, (2) a 2% HNO3, 0.2% HF, 2% pine oil, and 5% Calgon solution, and (3) 2N HCl. Soils from five DOE sites were extracted at a 1:1 soil-to-solution ratio in a rotary extractor for 6 min at 70 rpm. The extractions appeared to reduce concentrations of Pu and Am in the fractions >105 m and concentrated the radionuclides in the <105 m size fractions. Based on wide variations in the results, the authors concluded that the selection of extraction media will have to be tailored for each soil. Research funded under the Uranium in Soils Integrated Demonstration at Fernald, Ohio, indicated that soils containing 400 to 500 mg/kg uranium (soils sampled from within the Fernald production area and around a low-level waste incinerator) can be lowered to concentrations of <50 mg/kg using extraction media that are selective for uranium leaching. The best results have been obtained using an extraction medium that includes a strong reducing agent and strong complexing agents for uranium (30, 31). Carbonate leaching in combination with attrition scrubbing also appears to be a possible method for removing uranium to levels <50 mg/kg for the soil sampled within the production area (32).
COST OF TECHNOLOGY OR EQUIPMENT
A wide range of soil washing costs have been reported in the literature. For example, EPA (16) estimated the cost range for soil washing to be between $60 and $230 per ton. Trnovsky et al. (33) estimated the present worth from the estimated capital and operating and management costs for several remedial action alternatives at the Petroleum Products Corporation Superfund Site in Pembroke Park, Broward County, Florida and estimated the unit cost of soil washing to be $50 to $65/ton. Soil washing when used as a pretreatment to solidification/stabilization resulted in significant cost savings by decreasing the volume of soil that needed to be stabilized.
The following are EPA comments on soil washing costs (8): (1) treatment and disposal of contaminated residuals can be a major expense; (2) design field tests can be expected to range from as low as $100,000 to more than $500,000 (1989 costs); (3) costs for a German (Harbauer GmbH) full-scale soil washing facility are reported as, capital cost $6,000,000 (1986 dollars) for a 15 to 20 ton/h facility, O&M costs of $150/ton soil including water treatment, and sludge disposal costs of about $50/ton soil, and (4) processing costs for other less complex European soil washing operations are estimated at $73 to $110/ ton soil.
SUMMARY AND REQUIRED DEVELOPMENT
The primary advantage of soil washing is volume reduction of wastes requiring ultimate disposition. Soil washing can be cost-effective as a pre-processing step in reducing the quantity of material to be processed by another destruction and/or immobilization technology. It may also be used to transform the feed into a more homogeneous condition to augment subsequent treatment systems operations. Additionally, soil washing provides a closed system that remains unaffected by external conditions, and hazardous wastes can be excavated and treated on-site (34). Contaminants in soils containing a high fraction of silt and clay-sized particles are, typically, strongly adsorbed and are difficult to remove by soil washing. Hydrophobic contaminants generally require surfactants or organic solvents for removal from the soil, while complex contaminant mixtures in soils, as well as frequent changes in the contaminant compositions, make it difficult to design a single washing fluid that will consistently and reliably remove all of the contaminants (8).
A major need in development of soil washing technology is a clear understanding of the acceptable cleanup concentrations of contaminants. The nature of the extraction medium depends on characteristics of the organic compound, radionuclide, and/or heavy metal in the soil and the physicochemical characteristics of the soil; thus, each contaminant and soil needs to be investigated to properly assess the

probability of decontamination. Simple soil washing, using water as the extraction medium, will not suffice in cleanup of heavy-textured soils containing a variety of radionuclides, heavy metals, and organic contaminants.

A policy needs to be established about disposition of the cleaned waste (e.g., use of the debris rule where the treated waste is no longer handled and disposed of as RCRA hazardous waste), and a positive characterization technique needs to be developed to assess the cleanliness of the treated waste (i.e., quality assurance/quality control for the treatment).

Additionally, the need exists to prove in the laboratory that buried waste can be cleaned by such an approach. Although a technique may have proven effective in soil washing, the chemical and physical characteristics of buried waste may differ sufficiently from contaminated soil for the approach to also work effectively. Of course, the technique should be acceptable, as developed, for treating contaminated backfill and soil surrounding buried waste. After proving that the concept is useful, details such as removing waste from bags and containers need to be developed before field testing. To claim any benefits from the treatment, a positive test for the cleaned waste must be developed. In addition, a treatment cycle must be developed for the secondary waste stream to recycle the wash water/leachant/extractant and concentrate and to dispose the removed contaminants.

In summary, until the moratorium on permitting incinerators is resended and/or the emission standards are established washing technologies are viable alternatives to incineration. Furthermore, washing technologies may provide added benefits to the treatment of debris for mixed low-level waste through use of the debris rule.

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STEAM REFORMING OF DOE COMPLEX WASTE SIMULANTS

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ABSTRACT

Sandia National Laboratories has worked with Synthetica Technologies and Manufacturing and Technology Conversion International (MTCI) to demonstrate the applicability of their commercial steam reforming technologies for treating DOE low-level mixed wastes. Previously, Synthetica successfully demonstrated destruction of a Sandia formulated lab trash simulant. During November, 1994 Synthetica did not adequately process the aqueous halogenated organic liquid mixed waste simulant (MWTP-2110) formulated by the DOE Mixed Waste Integrated Program (MWIP). Testing at MTCI is ongoing and initial results appear to be favorable. Approximately 200 lbs each of the MWIP aqueous halogenated organic liquids (MWTP-2110), and absorbed aqueous and organic liquids (MWTP-3113/3114) simulants have been processed. At 1650 F, destruction efficiencies of greater than 99% were obtained for tetrachloroethylene, toluene, and 1,2 dichlorobenzene. Product gases consisted primarily of H2, CO2, CO, and CH4 and had higher heating values of up to 355 BTU/SCF. Conclusions concerning the suitability of the MTCI process for treating DOE mixed wastes will be drawn upon the completion of testing.

INTRODUCTION

Steam reforming is a process of reacting organic materials with high temperature steam to produce CO, CO2, and H2. Halocarbons also produce mineral acids (e.g., HCl). Traditionally steam reforming has been utilized as a process for producing molecular hydrogen from hydrocarbons. (1) More recently steam reforming has been developed as a non-oxidative alternative for waste destruction. (2) In the steam reforming environment, products of incomplete combustion (PICs) commonly seen in oxidizing environments such as polychlorinated dioxins and furans are not produced. Furthermore, by operating with an excess of steam, the formation of thermodynamically reformed products such as polynuclear aromatics may be greatly suppressed.

Sandia National Laboratories has worked with Synthetica Technologies and Manufacturing and Technology Conversion International (MTCI), two suppliers of steam reforming technology, to demonstrate and extend the applicability of steam reforming to DOE complex wastes. Synthetica markets a steam reforming apparatus that employs a two step process for waste destruction, gasification in a feed device at temperatures up to 1100F followed by destruction in an electrically heated high temperature reactor (2100F) known as a detoxifier. The MTCI single step process, is an intermediate temperature (1600F) fluidized bed that is heated by pulse combustion. The pulse combustor results in highly turbulent combustion mixing that significantly enhances heat transfer. The combustion gases do not mix with the feedstock or product gases. Therefore, both technologies destroy wastes in an oxygen free environment, and result in a hydrogen rich product gas.

Arrangements were made for each company to demonstrate their technology on mixed waste simulants formulated by the DOE Mixed Waste Integrated Program (MWIP). (3) The simulants that were tested include aqueous halogenated organic liquids (MWTP-2110), and absorbed aqueous and organic liquids (MWTP-3113/3114). The simulant compositions as employed in these tests are outlined in Table I. The formulations in Table I may differ slightly from other formulations as they were taken from a draft document. Testing was to be carried out in duplicate with 200 lbs of simulant being treated in each test. Metrics included characterization of the effluent gas and residual matter including the working fluid (condensed water).

TABLE I

SYNTHETICA TESTS

In previous tests conducted using Synthetica's drum feed evaporator and detoxifier on a Sandia devised simulant, mass reductions of about 20:1 and volume reductions of

greater than 150:1 were demonstrated. (4) Also, a simple mass balance verified the steam reforming chemistry (consumption of water and waste). The simulant chosen for these tests represented a week's worth of lab-trash generated in the SNL hot cell and consisted primarily of shredded solid carbonaceous material such as lab coats, paper towels, and plastic with about 5% by weight organic liquids. Despite a noticeable in-leakage of air, the exhaust gas was found to be well enough characterized and to contain low enough levels of pollutants that permits could be sought from local regulatory agencies. Therefore the conclusion of the tests was that steam reforming could be a practical alternative for SNL to pursue for treating its own mixed and low-level radioactive waste. Similar results were anticipated for other wastes.

The MWIP aqueous halogenated organic liquids simulant (MWTP-2110) proved to be more challenging to the Synthetica system. The amount of tetrachloroethylene and dichlorobenzene in the waste simulant was reduced as indicated in Table I in order to accommodate limitations for chlorocarbons in the detoxifier. Also, the amount of the inert vermiculite was scaled down by about 40% so that the entire volume of waste simulant could fit into a single drum. The resulting formulation with its high levels of volatile organics could not be successfully processed with the equipment currently available at Synthetica. During the attempted processing of this waste simulant, several problems were encountered.

A significant leakage of air into the system occurred as the reactor temperatures increased during startup. This problem was initially diagnosed as a ruptured burst disk. Replacement of the disk and subsequent startup of the system and commencement of the test ensued. Approximately two hours and thirty minutes into the test run, an abnormally high pressure drop across the absorber bed was observed. At the same time symptoms of a plugged condenser were observed by test personnel. As a result of these problems, the test director terminated the run to protect the equipment. Investigations conducted after the system had cooled revealed the absorber bed was plugged with soot, while the condenser was completely plugged by large crystals of what was believed to be naphthalene, and a second black tar-like substance. Investigation by Synthetica personnel into possible causes for the plugged absorber bed revealed a cracked reactor bed which allowed leakage of air into the system. Other possible explanations for the plugged bed include inadequate steam and design limitations of the current system for processing waste containing large amounts of volatile organics. Toluene and naphthalene comprised 19.5 and 5 percent of the MWTP-2110 waste simulant respectively.

Two weeks after the attempted test run, Synthetica and Sandia mutually agreed to terminate further test efforts. Sandia's decision was based on several factors including system repair time, contractual period of performance, and optimism on successful completion of other mixed-waste tests given the simulant recipes and current equipment.

Synthetica has designed new system components that when fabricated, should improve their ability to process wastes such as those proposed in these tests.

MTCI TESTS

Testing of mixed waste simulants is currently underway at MTCI. Currently, one test has been performed with each of the two feedstocks shown in Table I. Duplicate tests have yet to be performed. During the tests, the composition of the effluent gas (CO, CO2, H2, O2, N2, CH4, C2H4, C2H6, C3H6, and H2S) was measured by gas chromotography on a five minute interval. Several samples were also collected by an independent laboratory for GC/MS analysis to provide confirmation of the results. The flue gas from the pulse combustors was also periodically analyzed by a combustion analyzer. At least four solids samples were withdrawn from the reactor during each test to help evaluate mass and volume reduction and the fate of the RCRA metals and radionuclide surrogates in the simulants. At the conclusion of each test, the entire volume of bed material was removed to assist in this analysis. Samples of condensate were also taken. Temperatures were recorded at 1 minute intervals, flow rates were recorded at 20 minute intervals, and the condensate collected was measured at 10 minute intervals. During the duplicate tests, gas sampling will be performed to allow for the measurement of dioxins and furans via EPA procedure 23.

In the first test that was performed at MTCI, 195 lbs of the 3113/3114 simulant was processed at 1500-1550F. During the second test, 198 lbs of the 2110 simulant was processed at 1600-1650F. During each of these tests, silica sand was used as the bed material. Due to the large number of samples collected for analysis many of the

results from the two tests are currently unavailable. However the results that are available appear to be favorable.
Table II summarizes the destruction efficiencies obtained for many of the key organic components present in the tests. At 1550 F, the destruction efficiency of 1,2 dichlorobenzene exceeded 98% while that of naphthalene exceeded 99%. These compounds are recognized as being very thermally stable and therefore represent something of a worst case for the system. Elevating the temperature to 1650 F in the second test resulted in raising the destruction efficiency of 1,2 dichlorobenzene to greater than 99%. MTCI is currently considering the addition of a second stage, higher temperature reactor to their system in order to achieve even greater destruction efficiencies.
TABLE II
The composition of the reformer effluent gas for each test is summarized in Table III. A complete mass balance has not yet been performed. However, the primary gaseous products were H2, CO2, CO, and CH4. The large quantity of CO2 is indicative of water/gas shift chemistry rather than oxidative combustion. The H2S shown in Table III probably originates from the ion exchange resin in the simulants. The HCl formed from the chlorocarbons was collected in the scrubber downstream of the reformer. Also included in Table III are heating values for the product gas. The heating value of the gas can be utilized in the MTCI system by feeding the gas to the pulse combustion system.
TABLE III
CONCLUSIONS
The equipment available at Synthetica at the time of these tests was inadequate to process the aqueous halogenated organic liquid mixed waste simulant (MWTP-2110). Based on this outcome and other considerations further testing was not attempted. Synthetica is in the process of designing new system components that when fabricated, should improve their ability to process wastes such as those proposed in these tests.
Testing at MTCI has not been completed. However, initial results for the MWIP aqueous halogenated organic liquids (MWTP-2110) and absorbed aqueous and organic liquids (MWTP-3113/3114) simulants appear to be favorable. At 1650 F, destruction efficiencies of greater than 99% were obtained for tetrachloroethylene, toluene, and 1,2 dichlorobenzene. Product gases consisted primarily of H2, CO2, CO, and CH4 and had higher heating values of up to 355 BTU/SCF. A more complete evaluation of the suitability of the MTCI process for treating DOE mixed wastes will be possible when the remaining tests are complete and the data is analyzed. Complete mass balances including the fate of the metals and radionuclide surrogates will be available.
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NEEDS DRIVEN VITRIFICATION OF MIXED RADIOACTIVE WASTES
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ABSTRACT
It is important to make the best use of the available technology to aid in the safe

and efficient disposal of wastes at the U.S. Department of Energy (DOE) sites. A relatively mature technology, waste vitrification has been found to be cost effective, environmentally sound, and well suited for a variety of radioactive waste types. Scientists began studying the technique of incorporating radioactive waste into glass during the 1950's. Experimental melters have been used since 1975 to perfect the vitrification process. Over the past 20 years, scientists and engineers at the U.S. Department of Energy's Savannah River, and Hanford sites, and at the West Valley Demonstration Project have been developing expertise to characterize waste, tailored formulations to produce durable waste glasses, and developed technologies to effectively process and vitrify these wastes. Recently the glass melter in the Defense Waste Processing Facility (DWPF) was started up at the Savannah River Site (SRS). This facility is designed to remotely process and vitrify the high-level radioactive waste currently stored at the SRS. Startup of the melter was a vital part of the facility qualification, and involved treatment of simulated waste. The qualification phase will continue until late 1995 when radioactive wastes are scheduled to be processed in the facility.

The Environmental Protection Agency has declared vitrification the "best demonstrated available technology" (BDAT) for stabilization of heavy metals and high level radioactive waste. Nine nations have either chosen, or are considering, glass as a solid waste form for their highly radioactive wastes. By choosing glass, they minimize the lengthy, costly testing required of other waste forms.

Vitrification is cost effective and can produce an extremely durable glass form, which can isolate contaminants from the environment for thousands of years. Benefits of the application of glass technology to the disposal of wastes include:

well characterized waste form stability,
extremely low release rates in environmental conditions,
excellent mechanical and thermal stability,
no combustible or pyrophoric properties,
retention of essential release properties even if disturbed or mechanically damaged during storage or disposal,
low generation of potentially respirable particles,
ability to accept high loadings of heavy metals,
ability to accommodate fluctuating waste types with negligible effect on release properties,
ability to combust limited amounts of organics (including carcinogens),
volume reduction, reducing disposal costs
total destruction of asbestos,
and scale of facility adaptable to accommodate a wide range of disposal needs.

Compliance with environmental waste disposal regulations can be very expensive. While vitrification capital costs may be somewhat higher than conventional waste solidification techniques, such as cementation, the expense is more than offset by the savings from volume reduction and reduced monitoring inherent in the more durable waste form. The high temperature in the melter also destroys organic materials, thus further reducing the regulatory burden of disposal.

In the last decade there has been increased commercial interest in waste vitrification, and a growing convergence of the traditional commercial glassmaking and metallurgical approaches to waste glass production. The metallurgical approach is characterized by extremely high operating temperatures with molten metals present, strongly reducing conditions, and products that are not dependent on variations in appearance or small volume defects. In the metallurgical tradition, the glassy phase is a waste byproduct which should be minimized, and which results from winning metals from ores, or in secondary smelting operations. In the metallurgical tradition many of the refractories used are basic and are incompatible with production of high durability glasses. For example, a basic oxygen furnace for iron production may have a slag that is saturated with lime to maximize the scavenging of sulfur and phosphorous. A refractory of CaO may function for two years in this service, but would be dissolved in one week's time if subjected to a soda-lime-silicate bottle glass at the same temperature. By comparison, commercial glasses are usually acidic melts, and efforts to minimize furnace rebuilding result in the use of durable, acidic refractories: silica, high alumina, high zirconia, or high chromium oxide refractories, which are generally not compatible with a molten metal phase. Resolution of these conflicting requirements is one of the issues being addressed.

Parallel to the metallurgical tradition is the commercial glass making tradition. Initial waste glass processes, equipment and controls were based in the commercial glass making tradition, adapted to the requirements for working with radioactive materials, and slurries and sludges. In the glass making tradition, the glass product and its forming or final characteristics are the reason for the glass making operation. Thus, glass product formulation is rigidly controlled, with raw materials selected for their uniformity in composition and physical or handling characteristics. The incorporation of regulated waste products into such operations are difficult because of the inherent variations in source material. This additional process variation, and the increased regulatory requirements have minimized use of waste materials that are not produced internally. However, recycling, environmental and litter control regulations are increasing the use of internally-generated and post-consumer recycle in the glass bottling and television / computer monitor production plants.

RADIOACTIVE WASTE TREATMENT

The Mixed Waste Integrated Program seeks to develop and demonstrate technologies which can make significant improvement on existing commercially mature treatment systems. Vitrification has the potential to treat approximately 75% of the mixed (RCRA hazardous and radioactive) wastes being stored at DOE sites. The waste glass systems that are being considered are soda-lime-silica, borosilicate, aluminosilicate, nepheline, basalt, iron enriched basalt and phosphate glass systems. Glass ceramic systems have also been considered, many of which can be potentially produced as glass and subsequently heat treated to develop crystalline phases.

The ability to vitrify waste depends upon how well the waste is characterized, and on the flexibility of the melter and of the glass formulation. Considerations are physical form of the waste, size, water and organic contents, handling characteristics, ash composition, volatile materials, and extent of remote handling required. Relatively minor mass fractions of radioactive materials can cause orders of magnitude differences in the cost of personnel protection, offgas treatment required, and the capital, operational and maintenance costs. Waste is characterized using the most advanced technologies available, and the best way to vitrify the waste products is determined. Hazardous species in almost any material can be characterized using tools that include inductively coupled plasma-atomic emission spectroscopy; inductively coupled plasma-mass spectroscopy; atomic absorption spectroscopy, ion chromatography; x-ray diffraction; and x-ray fluorescence. For radioactive elements, gamma ray spectroscopy and alpha energy spectroscopy may be combined with more traditional counting methods.

The behavior of nearly every element in the periodic table has been studied in waste glass development. SRS has used a systems approach and process models have been developed to ensure that the melt's viscosity, durability, liquidus, and other important properties can be estimated from the melter feed ingredients. For more complex problems, the product composition control system, built upon proprietary statistical process control algorithms, allows for random variations, and suggests corrective actions. As a result, a durable glass product is reliably produced, while minimizing process upsets.

Melters are being applied to the immobilization, volume reduction and stabilization of High Level Radioactive Waste (HLW) Low Level Mixed Wastes (LLMW), Transuranic (TRU), Low Level, and chemical (RCRA) wastes. LLMW and TRU types generally do not require heavy shielding to control radiation doses to operating and maintenance personnel. The low penetrating radiation levels of these wastes allow them to be treated in facilities where direct contact maintenance can obviate the expense of remote maintenance facilities. Direct contact maintenance allows the adaptation of commercially available equipment, which has been precluded by the restricted maintenance conditions in High Level Waste facilities. Thus, rather than using expensive custom melter designs, the LLMW and TRU processes can use adaptations of commercial glass making equipment. However, these wastes share many contamination and environmental release concerns with High Level Waste. Thus, they require waste-form quality, contamination control features, and production control practices similar to the High-Level Wastes. When treating LLMW the requirements for protection of the operating personnel from contact with the wastes vary dramatically, dependent upon the source and type of the waste. For example, the hazards associated with natural and low enrichment uranium are primarily those of potential heavy metal

poisoning. In contrast, the 10 - 100 nanoCurie per gram low level transuranic wastes are potential sources of ingestion of maximum permissible body burden dosages of plutonium, since they may contain small "hot spots" or particles which could become airborne and result in personnel contamination through ingestion or respiration. DOE has needed a facility where commercial pilot scale equipment could be operated on surrogate (non-radioactive) simulations of typical DOE waste streams. The DOE/Industry Center for Vitrification Research (Center) was established in 1992 at the Clemson University Department of Environmental Systems Engineering, Clemson, SC, to address that need. The Center aids in the application of vitrification technology to the immobilization of hazardous, radioactive, and mixed wastes, by combining technology and lessons learned in commercial glass making with DOE contractor experience and DOE needs. During surrogate operations the melters are checked for:
1. general operability (i.e. does the unit perform as advertised)
2. general limits of processability (e.g. operating temperature range),
3. normal and specialized maintenance requirements,
4. compatibility with waste glass formulations (i.e. materials of construction, since high temperature equipment can be rapidly destroyed by chemical attack if materials of construction are not suitable)
5. interfaces with the rest of the treatment system (pretreatment and offgas treatment requirements),
6. product conformance to regulatory requirements, and
7. as far as possible the operating economics (life cycle costs, manpower requirements, utilities, raw material costs).

The non-radioactive pilot scale testing at Clemson Univ. has been a necessary step in adapting commercial technology to DOE's needs. Efforts are now being made to make the next transition. to demonstration on actual waste at the waste generation site, using equipment suitable for production treatment of small sites. A "Transportable Vitrification System" (TVS) is being completed in FY95 by SRS for field demonstrations of vitrification at various DOE sites on contact handled wastes. The Modular system will have generic melter feed, offgas treatment and laboratory capacity. Alternative melter systems can be substituted for the original melter after initial operations, to handle variations in physical or chemical properties of the wastes. The entire system can be transported via tractor trailers to allow vitrification of a variety of hazardous waste streams at different sites. The "TVS" has become a main focal point of SRTC vitrification activities, with waste glass formulations and alternative melter configurations to deal with specific waste issues.

RADIOACTIVE WASTE TYPES SUITABLE FOR VITRIFICATION

Vitrification has been demonstrated to be capable of incinerating organic materials and incorporating the resulting ashes in a number of inorganic glass systems. Initial operations have focussed on the vitrification and volume reduction of inorganic sludges resulting from waste water treatment. The high temperature melting process destroys the chemical compounds associated with such wastes, and ties poisonous elements up into a durable matrix. It has been demonstrated that soda-lime-silicate, aluminosilicate, and borosilicate glasses bind hazardous elements.

Current wastes of the beta-gamma type that are most suitable for vitrification are SRS M Area sludges and filter papers resulting from uranium forming and cladding, and similar waste water treatment sludges from Oak Ridge, Rocky Flats Plant and Los Alamos Scientific Laboratory. The M Area sludges have been contracted for on-site vitrification treatment by Duratek Co. There is enough inventory to keep a large scale facility operating for about two years, after which the equipment will be available for treatment of other wastes. A model "Upfront" delisting petition is being prepared for SRS M-Area sludge so that these wastes can be delisted from RCRA concern, allowing them to be disposed as Low Level Waste. Again, economics plays a major role in the effective disposal of these wastes, the major justification being that delisting of the hazardous portion of these wastes will allow more efficient disposal. Vitrification is clearly justified for these wastes because it will reduce the waste volume relative to the cement waste form, and will provide an increase in waste form durability: In fact, the vitrification process won over a competitive cementation process for the M Area sludge treatment because it was able to demonstrate lower disposal volumes and costs.

Alpha emitting transuranic wastes (TRU) are comparable to High Level Waste in

biologic risk, but require smaller scale operations. In these waste types, major goals are to convert small volumes of liquid or solid flammable wastes into small waste forms that are easy to handle, but provide a high degree of contamination control (isolation). Waste glass is essentially a sealed source, which reduces the dispersibility of a wastes during accidents, and the transferability of activity by contact. The high temperature melting process combusts organics and reduces the volume of this waste category, which primarily consists of solutions, contaminated plastic, cloth and paper products. Current investigations at SRS include formulation of glasses for vitrifying surplus transuranic solutions. If the TRU solutions were declared waste, they would be considered mixed waste because of their low pH and high nitrate concentrations. Vitrification will allow the storage and subsequent shipment and use or disposal of Pu, Np, Am and Cm.

SELECTION OF WASTE STREAMS FOR VITRIFICATION DEMONSTRATIONS

Fiscal year 94 operations focussed on waste water treatment sludges from SRS, ORR, Rocky Flats and Los Alamos. Treatability studies have been conducted on SRS and ORR sludges, and surrogate pilot scale operations conducted for SRS and ORR sludges. Surrogate pilot testing for RF and LANL sludges are being completed in FY 95. This category, including pond wastes constitutes about 1/3 of the entire DOE inventory of Low Level Mixed Wastes. Initial operations of the Transportable Vitrification System will be on and Oak Ridge Reservation sludge of this type. Surrogate (non-radioactive) pilot-scale demonstrations are being completed on the Rocky Flats Plant Precipitate sludge and LANL TA-50 sludge. Correlations for X-ray Fluorescence (XRF) to Ion Coupled Plasma Emission Spectroscopy (ICP-ES) analyses will be completed, which will help to define the durable glass processing region of the CaO-Fe2O3-SiO2 ternary system, and provide a rapid composition measuring device for use in field demonstrations.

In fiscal year 95, the Savannah River Site has been tasked through MWIP to investigate vitrification technology of high mercury and high organic waste streams. The waste streams that will be investigated are as follows:

1. Mercury Contaminated Storm Sewer Sediments from ORNL
2. Ion exchange resin used at SRS and ORNL.
3. Other Waste Streams

Due to the chemical composition and physical characteristics of these waste streams, they are very strong candidates for vitrification treatment. These waste streams also represent wastes that are found throughout the DOE complexes, and thus finding a viable solution to their disposal is imperative. In this task, crucible-scale studies will be performed on surrogates of the materials to determine the optimum vitrification processing limits. Once these are determined, vitrification studies will be performed with the actual radioactive waste streams and pilot-scale studies will be performed with the surrogates.

ORNL Storm Sewer Sediments - This waste stream was selected in conjunction with ORNL. ORNL is currently investigating Hg removal processes. This waste stream appears to be amenable to both ORNL processing and our vitrification treatment. The goal is for this waste stream to be treated in the Transportable Vitrification System after the WETF sludge is treated. The analyses of this waste are currently being reviewed for accuracy. It appears that the waste is very high in silica content and contains about 20,000 ppm Hg. ORNL expects that with the Hg removal process that the Hg levels could be reduced to 30-100 ppm. As soon as a final analyses of this waste is received, crucible-scale studies will be completed. Pilot-scale studies will be completed in the melter which proves to be the most applicable from crucible-scale studies (temperature and processing limits).

Ion Exchange Resins used at SRS and ORNL - Since many sites are currently or are planning on using ion exchange resins to treat wastes, it was decided to determine if vitrification was a viable treatment method for encapsulating the resins. Two types of resins have been identified. One is a resorcinol-formaldehyde resin for Cs removal in HLW. It is being considered for use at the SRS, ORNL, and Hanford. The other resin is from reactor facility operations, where it is used to treat the storage basin water. One resin's approximate composition after removing Cs from waste is 35.5 wt% C, 7 wt% Na, 0.95 wt% Cs, 45.5 wt% O, 5.8 wt% H, and 0.07 wt% K. Data will be compiled from pilot-scale demonstrations with resins and simulated wastes. Resins from reactor operations are composed of divinylbenzene / styrene copolymer beads with the cation resin having a sulfonic acid group attached and the anion resin having a quantenary ammonium group attached. The cation resins used in

the reactor areas have hydrogen ions attached for exchange and the anions have hydroxide ions attached for exchange. The major species which are trapped on the resins are Na, Ca, K, Si, NO3-, SO4-2, and Cl-, and the resins contain Cs-137, Sr-90, and Tritium.
The optimum glass composition for ion exchange wastes will be determined, as well as the maximum waste loading. Vitrification studies with actual wastes will be performed at the SRS, with the best estimate of the typical waste loading of the resin. Pilot-scale studies will be completed in the Stir-Melter since it seems to be the most amenable to vitrification of this type of waste. It is considered a mixed waste because of the Cs that remains on the resin and because the resin is organic. Crucible-scale studies will be performed to optimize the glass composition and waste loading. A mercury waste stream containing radioactive cesium was selected in conjunction with the Environmental Protection Dept. of SRS. The waste is on the Mixed Waste Inventory Report and was also in the Draft Site Treatment Plan as being scheduled for treatment. An acceptable treatment method had not been previously determined, so the applicability of vitrification will be tested after ion exchange treatment to separate the cesium from the soluble mercury. Crucible studies will be performed with the clean resins and the spent resins to determine the optimum glass composition and processing conditions. Once this is determined, the surrogate resin will be treated on a pilot scale.
The Environmental Protection Division of SRS also asked for help with treating soils from spill remediation and toxic characteristic contaminated debris. Contamination can be the result of hazardous chemical spills, or low levels of TRU or mixed fission products. Currently, these wastes are disposed of without further processing by shallow land burial. Vitrification of these materials will bind the hazardous and radioactive species, reducing the risk of ground water contamination. This waste stream consists of soils, sand, and associated debris (rocks, wood, etc.) and associated water/liquid resulting from cleanup activities of spills and remedial actions contaminated with Toxicity Characteristic constituents and radioactive materials. This material was generated from site wide general SRS operations (Operations Waste and ER). This contact Handled waste has Am-241, Cs-137, Pu-238, and Pu-239. The contamination is dispersed through the matrix. Presence and levels of hazardous constituents will vary dramatically based on the clean up activities. Usually the containers are only hazardous for one or two of the TC constituents.
Toxic Characteristic Metal Contaminated Debris - Comprised of non-combustible debris material (metal, floor tiles, light bulbs, broken thermometers, recorders, equipment, etc.) which has one or more toxic characteristic metals and/or organics as a hazardous constituent(s). Generation is from site wide operations (transition and decommissioning and decontamination activities). Generation is expected to be 1063 m3 by 12/97, 1077 m3 by 12/98 and unknown quantities after that. The inventory is 5.6 m3. Radionuclides vary depending upon the location where the waste was generated.Cadmium, chromium, lead, mercury, and asbestos are the contaminants.
Spent Filter Cartridges and Carbon Filter Media-Filter cartridges were used in SRS fuels facilities to remove particulates, consisting of mercury salts and depleted uranium from process streams. They contain U-235 and U-238. It is also suspected that the filters contain Cr and Pb. Cylindrical filter cartridges are constructed of a PVC exterior frame surrounding a fibrous cartridge media. Ronningen-Petter filters (Dover Corp) Extended Area Bags are constructed of pleated spun-bonded polyester. The pleated filter is bonded to PVC which is molded on top and bottom. No metal is used in their construction. Mercury salt and particles of depleted uranium are the expected impurities on the cartridges. The projected generation is at a rate of about 0.6 m3 until 12/99. The current inventory is 0.8 m3. Future generation will be subject to LDR storage prohibitions. Contamination is dispersed through matrix.
SRS MELTER SYSTEMS
The range of melter technologies demonstrated, or being considered includes Joule-heated melters, induction melters, high temperature plasma torch systems, resistively heated refractory metal bushings, and microwave systems. Each system has advantages for application to specific waste streams. Tailoring technologies to particular waste steams helps to maximize effectiveness and efficiency of waste vitrification options.
Slurry Fed Melters (SFM), also known as HLW melters, Joule Heated Melters, and Ceramic Melters, have been developed in the U.S., Europe and Japan for the conversion of HLW to borosilicate glass for permanent disposal. Laboratory and pilot

scale operations have been conducted to develop equipment, glass compositions and control methods. The majority of these systems have been developed for remote operation, with essentially no maintenance over a 2 to 4 year melter life. The relatively high cost of these melters is the result of the specialized requirements of remote installation, operation, maintenance and removal. The systems were developed for vitrification of waste water treatment sludges, and are very suitable for that operation. The slurry fed melters are the result of several generations of melter development based originally on commercial, electrically heated melters, adapted to the requirements of remote, non-contact maintenance, where the only available tools for changing out melter components or entire melters are a crane with hooks and an impact wrench. Major differences between HLW Melters and Commercial Melters have been: the use of metal shells to contain glass contact refractories and thermal insulation, the development of specialized slurry feeding and glass pouring systems, and the use of nickel based alloys for electrodes carefully matched with glass composition control. The HLW style melters will be used for initial SRS LLMW treatment in the M-Area Sludge treatment. High temperature melters and stirred melters are also being demonstrated since they appear to have cost and operating advantages.

High Temperature Melters are based on commercial melter materials which permit operation at temperatures higher than those of the HLW systems. The commercial systems are also based on direct contact maintenance, allowing possible cost reductions for LLMW treatment. As in all high temperature systems, the life of the system, and a large fraction of the cost is limited by the materials of construction. Glass systems include borosilicate, aluminosilicate, calcia-aluminosilicate, basalt, and soda-lime-silicate systems. As a result of the broad range of glass compositions and refractory and electrode materials available, a major need for commercial waste glass systems is selection of composition limits, operating temperatures and materials of construction: For optimum operations these can vary from one waste to the next. A second major need in commercial glass systems is sealing of the systems to control airborne contamination during the melter feeding and glass pouring operations and demonstration of slurry feeding of wet sludges. A vendor supplied system is being investigated at Clemson Univ., and as the initial melter for the Transportable Vitrification System.

Stirred Melter The current stirred-melter was developed to combine the high production rates and high glass quality features of the Joule-heated melters with the low-cost, compact, easily maintained features of the pot melters. Melt rates in the stirred-melter are believed to be proportional to melter volume, rather than melt surface area as in conventional melters. A small scale system has been installed at Clemson University.

Similar to the continuous pot melters are the commercially available bushing melters which are used to melt high quality fiber glass. Pots for these systems are fabricated from platinum group metals, and are heated by passing current through the pot. The commercial systems are economically viable because of very low consumption rates of the platinum, and the ability to recover the platinum for reuse. SRS is considering small systems based on commercial bushings for specialized transuranic waste melting.

Induction melters have been used for about 40 years at the SRS for production melting of uranium, lithium, plutonium and aluminum alloys, for vacuum hot pressing of plutonium dioxide, and to initiate transuranic fluoride reductions. They are reliable, low cost, high throughput devices for the melting of metals which directly couple to the radio frequency fields established by the induction coils, or the material to be melted can be indirectly heated by RF coupling to the graphite or silicon carbide crucible. The metal melting systems have used graphite and silicon carbide crucibles, with magnetic stirring to assure alloy uniformity.

Graphite Electrode Arc Melters are the basis for stainless steel and ferro alloy production, as well as such diverse operations as acetylene, phosphate, and fuse-cast refractory production (zirconia, alumina and chromia). Commercial units can be as large as 50 feet in diameter with electrodes in excess of 2 meters in diameter. The electrodes for such systems require constant renewal, since the graphite reacts with the melt and with oxygen in the melter atmosphere. Methods are available for electrode feeding, including forming and graphitization of the electrodes as they are consumed. Many melters are run "cold top" in open crucibles, where the feed batch provides the furnace cover. Refractories are selected based on

the chemical system being processed, and refractory life varies dramatically dependent upon product and the amount of thermal cycling. Most of these systems operate in foundry environments, and have not required sealing of the electrodes to the furnace, or fume control during pouring. As a result, most commercial melter systems have uncontrolled air leakage around electrodes, and do not provide sufficient fume control or prevent contamination of the outside of the vessel. Combustible gasses can be formed by carbothermic reaction, pyrolysis of the treated material or by water-gas-shift reactions between carbon and water. The most extensive research of graphite electrode arc systems for DOE wastes have been conducted by INEL, and by PNL at the Massachusetts Institute of Technology (MIT). A bench scale systems is being supplied to SRTC by INEL, which will be installed in a SRS glovebox for LLMW and TRU testing. A larger pilot scale unit is being vendor supplied by the developer of the MIT unit to determine if such a unit should be considered as a replacement for the TVS when processing high metal content wastes. Plasma Arc Melting is being investigated at Mississippi State University and the Georgia Institute of Technology. Furnaces may be refractory lined, or of the cold crucible type. Refractory lined systems must have control limits to minimize glass and metal attack, and overheating by the plasma. Plasma arc torch design is a mature technology, and is available from a number of vendors, but torches require rebuilding on a monthly basis. Torch failure can result in uncontrolled coolant addition to the treatment furnace. Treatment furnaces have been offered commercially, but except for primary metals production they have been used primarily for recycle of metals or laboratory testing. SRS is following the work at other sites on equipment development, and supporting brief tests at universities to determine if SRS wastes are compatible with the emerging designs. A Russian designed hybrid plasma/induction melter is being obtained for testing at Georgia Tech., which combines cold crucible melting with plasma melting.

Cold Crucible Induction Melters Melter manufacturers in France, Russia and the United States, have developed cold crucible induction melters, where the water cooled induction coil is in direct contact with the material being melted. The steep temperature gradients produced adjacent to the induction coil cause a "skull" melting effect, where the material being produced forms its own crucible. In waste melter operations this means that the need for replacement of crucible materials is avoided, at the cost of reduced melting rate. STRC is considering such technology for future generations of High Level Waste melters, and for high salt content wastes.

Demonstrations of refractory life, control of melter pouring, control of wasteglass quality, control of combustible offgasses, control of high vapor pressure metals, control of volatilization and entrainment of radioactive materials, integration to offgas systems, and maintenance of the torch under radioactive conditions are major portions of all SRS testing.

SUMMARY

Vitrification offers many attractive waste stabilization options. Versatility of waste compositions, as well as the inherent durability of a glass waste form, have made vitrification the treatment of choice for high-level radioactive wastes. Adapting the technology to other hazardous and radioactive waste streams will provide an environmentally acceptable solution to many of the waste challenges that face DOE today.

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VITRIFICATION STUDIES WITH DOE LOW LEVEL MIXED WASTE WASTEWATER TREATMENT SLUDGES

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ABSTRACT

Vitrification studies with simulated Low Level Mixed Waste (LLMW) sludges were performed at the Savannah River Technology Center (SRTC). These studies focused on finding the optimum glass compositions for four simulated LLMW wastewater treatment sludges and were based on both crucible-scale and pilot-scale studies. Optimum compositions were determined based on the maximum waste loading achievable without sacrificing glass integrity.

Crucible-scale study results indicated that 45 wt% waste loading was obtainable with a Oak Ridge Reservation (ORR) West End Treatment Facility (WETF) simulated sludge when various additives were used. For a simulated Rocky Flats Plant (RFP) Precipitate sludge, durable glasses were produced with 75 wt% waste loading at higher temperatures and with 65 wt% waste loading at lower temperatures. Charcoal additions had to be used in manufacturing these glasses to deter the formation of a sulfate salt layer. Simulated Los Alamos National Laboratory (LANL) TA-50 sludge was also tested. A durable glass was produced at higher melting temperatures with 65 wt% waste loading.

Pilot-scale study results indicated that durable simulated Savannah River Site (SRS) M-Area sludge glasses could be produced with up to 90 wt% waste loading when melted in higher temperature melter systems and up to 85 wt% waste loading when melted in lower temperature melters. Pilot-scale studies were also performed using simulated ORR WETF sludge based on the crucible-scale findings. The most durable glass produced was melted in a high temperature melter system and contained 45 wt% waste loading.

INTRODUCTION

The Department of Energy (DOE) has chartered the Mixed Waste Integrated Program (MWIP) to investigate waste forms for LLMW. Vitrification of the wastes is a main focus of the MWIP investigations. To help develop the alternative waste forms, MWIP has funded the SRTC to perform vitrification studies. SRTC's vitrification effort is in conjunction with the DOE/Industrial Center for Vitrification Research (Center) located at Clemson University.

Vitrification studies in fiscal year 1994 involved both crucible-scale and pilot-scale studies with simulated wastewater treatment sludges. The simulated sludges tested included SRS M-area wastewater treatment sludge, ORR WETF sludge, RFP Precipitate sludge, and LANL TA-50 sludge.

SRS M-Area sludge has evolved from wastewater treatment of Ni plating line sludge. The sludge contains a large amount of Si from the M-Area filtration process which uses perlite as the filter aid. The Resource Conservation and Recovery Act (RCRA) metal of concern is Ni while the radioactive element of concern is U. The inventory of this sludge is about 1,100,000 gallons, and it is currently being held in storage tanks after treatment. It has been characterized by C.M. Jantzen of SRTC and a surrogate composition was developed (1). Crucible studies with both simulated and actual sludge were performed by C.M. Jantzen as part of an M-Area treatment study; however, results will not be reported in this paper since the studies were performed under a separate project scope.

ORR WETF sludge has resulted from treatment of nitrate-containing wastes by biodenitrification. The sludge contains a large amount of Ca and a small amount of organics. The RCRA metals of concern are Ba, Cd, Cr, Pb, and Ni, while U is the radioactive element of concern. The current inventory of this sludge is approximately 7,100 m3, and it is currently being stored in 500,000 gallon tanks. It was characterized by W.D. Bostick of Oak Ridge National Laboratory (ORNL), who developed a surrogate formula and recipe.(2) Ce was substituted on a molecular weight basis for uranium, so the behavior of the radioactive element could be monitored.

RFP precipitate sludge has evolved from chemical precipitation of aqueous waste from plutonium recovery operations, and thus is a transuranic (TRU) waste. The sludge is generated during chemical precipitation of radioactive elements from liquid wastes. The major constituent of this waste is Fe, but it also contains a small amount of nitrates. The RCRA metals of concern are Cr, Ni, Pb, Cd, and Ag, while the radioactive element of concern is Pu. The current inventory of this material is >3 m3, which is currently stored in 55 gallon drums. The sludge was characterized by RFP personnel and a surrogate composition was developed (3). However, W.D. Bostick of ORNL derived a different surrogate formula from this characterization and composition, so this composition was used for these crucible studies to be consistent with other MWIP sponsored programs (2).

LANL TA-50 sludge has been derived from a liquid waste processing plant that uses influent water containing about 100 mg/L total dissolved solids for decontamination, rinse down, and other processes. This wastewater is treated with ferric sulfate and precipitated lime. The wastewater is later filtered and a sludge remains. The major constituents of this sludge are Ca and Si. The RCRA metals of concern are Cr, Ni, Pb, Cd, and Ba, while the radioactive element of concern is U. The total inventory of this sludge is estimated at 270 m2, with about 139 m3 subject to Environmental Protection Agency (EPA) Land Disposal Restriction (LDR) prohibition, and it is stored in 55 gallon drums (2). Based on available analytical data, W.D. Bostick of ORNL derived a surrogate formula (2), which was used in these studies, but information was not available on the level of RCRA metals in the actual sludge, so a standard spike of 500 ppm was incorporated. For the crucible studies, Ce was used as the substitute for uranium.

The compositions of the simulated sludges used in these studies on a normalized weight percent oxide basis are given in Table I. Anions, such as nitrates and sulfates, and waters of hydration from the batch materials were not included in the tables, since they are not of concern during crucible-scale vitrification.

Crucible-scale studies were performed using these surrogates (with the exception of the SRS M-Area sludge, since they were already performed) in order to determine the types and quantities of glass forming additives to be added to each waste. The information gained is necessary to perform the pilot scale demonstrations at the Center. Eventually, simulants of all of the wastewater sludges will be vitrified in demonstrations at the Center, but, as of now, only the SRS M-Area and ORR WETF demonstrations have been completed. Only results from the glass characterizations performed for these demonstrations will be discussed here.

EXPERIMENTAL

As mentioned earlier, Bostick's surrogate recipes were used in the crucible-scale studies. The recipes were used to make approximately 500 grams of each type of waste. The wastes were blended with various glass formers in order to determine the optimum glass compositions for pilot scale testing. The amounts and types of glass formers to be used were determined from previous crucible studies performed at other DOE sites, where applicable, or were determined from expected glass making regions of ternary diagrams. The compositions to be tested on the pilot-scale were determined based on waste loadings, PCT results, and TCLP responses. The batch compositions tested for each waste type are given in Table II, along with the associated batch number and melt temperature. For the SRS M-Area studies and one of the pilot-scale ORR WETF studies (OR12P), the compositions tested at the Center are given, along with the associated melter instead of melt temperature. In Table II, hydrous borax is abbreviated as HB and diatomaceous earth is abbreviated as DE. In the studies, additives were added as listed in Table II with the exception of Na_2O which was added in the form of Na_2CO_3. Batches OR1 and OR7 were used in the pilot-scale studies at the Center in the Stir-Melter and EnVitCo melter, respectively. Charcoal had to be added to Batches RF6 and RF10 to deter the

formation of a sulfate salt layer. Batches LA4, LA5, and LA6 were tested to try to utilize the Minimum Additive Waste Stabilization (MAWS) concept. Using RFP simulated sludge as the glass additive, up to 100% waste loadings were tested.

For the crucible-scale studies, approximately 70 gram batches were made of the glass compositions shown in Table II. The batches were placed in covered high purity (99.8%) alumina crucibles and placed in a furnace at the specified melt temperatures for 4 hours. After 4 hours, the crucibles were removed from the furnace and the glasses were air quenched to room temperature. In the pilot-scale studies, the batches were continuously fed to the melters. Glass samples were taken once steady-state conditions were met. This usually occurred after three melter volumes of glass had been produced.

For all studies performed, the glasses were analyzed for chemical constituents and phase assemblage. The chemical constituent analysis was performed on the glass product using Inductively Coupled Plasma - Emission Spectroscopy (ICP-ES) and Atomic Absorption (AA) Spectrometry. Phase assemblage was characterized using X-ray Diffraction (XRD) and Scanning Electron Microscopy (SEM).

To assess the integrity of the glass, the Product Consistency Test (PCT) (4) and the Toxicity Characteristic Leaching Procedure (TCLP) (5) were also performed. The PCT is a crushed glass leach test that measures the releases of B, Si, Na, and other elements in 90C ASTM Type I water over a period of seven days and is the standard test used for determining the durability of High Level Waste (HLW) glasses (4). Each glass sample was tested in triplicate and submitted for leachate analysis, with the results averaged and normalized. The PCT results were compared against the HLW Environmental Assessment (EA) glass PCT values (6) to determine the stability of the glass in water.

While the PCT is the accepted durability test for HLW, the TCLP is recognized as the standard test method for determining the hazardous nature of a waste. For the scoping crucible-scale tests, the TCLP was performed on +100 mesh (>0.150 mm) crushed glass. In general, EPA tests are usually performed on larger size glass specimens, as was the case with the pilot-scale glasses. Thus, the crucible-scale results reported here provide a conservative estimate of the leach resistance since approximately 200 times more surface area was exposed to the leaching solution. TCLP extractions were performed on the glass, and the resulting leachates were analyzed by ICPES. The TCLP results were compared against the more restrictive of either the TCLP or RCRA Land Disposal limits.

CHEMICAL CONSTITUENTS

Table IIIA , B lists the oxide composition of all of the glasses produced in the vitrification studies (7,8,9,10,11). The batches that did not produce homogeneous glass were not analyzed, and thus are not included. As stated above, batches OR1 and OR7 were used in pilot-scale demonstrations. The pilot-scale SRS M-Area and ORR WETF glasses are designated by the "P" after the Batch ID.

PHASE ASSEMBLAGE

The phase assemblage of the glasses as determined by XRD are given in Table IV (8,10,11) XRD results are not included for the pilot-scale glasses produced at the Center. For the most part, volume percent analyses were not available, since standards for these compounds are not available at SRTC for determinations. Quantitative XRD analysis was performed on glasses OR1-OR3 because a standard was available. The largest volume percentage of crystals detected was found in the glass OR3, which was around 9%, while glasses OR1 and OR2 had less than 0.5%. The unusually high amount of crystals in glass OR3 did not seem to greatly affect the durability. SEM analysis confirmed the XRD findings.

Although quantitative results were not available for the RFP sludge glasses, a comparative analysis of the peak intensities on the XRD spectra indicated that larger quantities of crystals were present in glasses RF8 and RF9. The relative intensities of the other spectra were only slightly above background. The crystalline phases detected did not seem to significantly affect the durability results. SEM analyses could only verify the presence of the hematite (Fe_2O_3) in the glasses.

For the LANL simulated sludge glasses, the results show that at least one glass from each ternary system contained crystals. In all cases except for LA18, the glasses that contained crystals were the glasses with the highest waste loading. SEM analysis of the glasses verified the presence of crystals in LA3, LA4, LA7, LA17, and LA18 and verified that the remaining glasses were free from crystalline phases.

PCT RESULTS

The PCT data in g/L for all of the fabricated wastewater treatment sludge glasses is given in Table V (7,8,9,10,11). The normalized releases of B, Si, and Na were well below the EA accepted values (6) for all glasses with the exception of glass LA17. The RCRA metals of concern and Ce, which was used as the radioactive surrogate for some of the glasses, were only released in very small amounts. In most cases, they were near the detection limits. Some of the releases were not applicable, since they contained only trace amounts of B. Glass LA17 was the only glass tested to actually exceed the EA accepted PCT values (6), so it was considered unacceptable for further pilot-scale testing.

TCLP RESULTS

The TCLP data in mg/L for all of the wastewater treatment sludge glasses are contained in Table VI (7,8,9,10,11). As mentioned earlier, the TCLP was performed using a modified procedure for the crucible study glasses, while the TCLP for the pilot-scale glasses was performed by outside vendors on standard sample sizes. Results for the remaining RCRA metals are not included in the table since they were not included in the glass compositions.

All of the SRS M-Area pilot-scale, ORR WETF (with the exception of OR1P, OR10 and OR12P), RFP Precipitate (with the exception of RF8 and RF9), and LANL TA-50 (with the exception of LA1, LA3, LA13, and LA18) simulated sludge glasses performed better than the EPA limits.

OR1P exceeded the listed EPA limit for Ni, which is based on the RCRA disposal limits. However, new regulations which were effective December 19, 1994 raised the limit for Ni to 5.0 mg/L (12), so this glass would be considered acceptable. OR10 exceeded the EPA limit listed for Pb and Ni, which is the RCRA land disposal limit. However, the Pb release did not exceed the TCLP limit of 5.0, and Ni does not exceed the new Ni limit of 5.0 mg/L. Once again, the tests for the crucible-scale glasses were performed on very conservative particle size glasses so this consideration must be taken into account when looking at the overall quality of these glasses. OR12P exceeded the listed Ni limit, but does not exceed the new regulation Ni limit.

Glass RF8 exceeded the Cd, Ni, and Ag EPA limits and glass RF9 exceeded the Cd limit. However, the RFP Precipitate simulated sludge used to fabricate these glasses was spiked with 500 ppm of the RCRA metals, which was much higher than what was actually present in the sludge according to chemical analyses. By taking the elevated spike levels into consideration and adjusting the TCLP releases accordingly, the releases of Ag and Cd are reduced to 0.036 and 0.057, respectively, for glass RF8, and the release of Cd for RF9 is reduced to 0.012.

No information was available on the amount of Ni present in the sludge from the chemical analyses, so this value was not reduced for glass RF8. However, if the new limit is used for Ni, the glass would have passed the Ni criteria without the scaling factor being considered. These scaled results show that glasses RF8 and RF9 could possibly produce acceptable glass when the actual sludge is vitrified because of the lower amounts of RCRA metals that will be present. Also, it must be remembered that the TCLP was performed on a very conservative sample size, so the results are also very conservative.

Glasses LA1, LA3, LA13, and LA 18 exceeded the Cd limit. However, the new regulatory limits effective December 19, 1994 have raised the Cd limit to 0.19 mg/L (12), which would mean that only glass LA3 exceeded the Cd limit. These glasses will not be considered for further pilot-scale studies, since more acceptable glass compositions were found.

CONCLUSIONS

In order for glasses to be considered acceptable, they had to meet the PCT and TCLP criteria. Crystallinity in the glasses was also considered when determining a good glass, since the formation of crystals tends to decrease durability. A brief summary of the findings of all of the studies mentioned in this paper follows:

For SRS M-Area simulated sludge, up to 90% waste can be vitrified in borosilicate glass when melter systems such as the EnVitCo melter are used. Waste loadings of 85% are possible when melter systems such as the Stir-Melter are used. In both cases, hydrous borax was used as the glass forming additive.

For ORR WETF simulated sludge, crucible-scale studies showed that durable glasses consisting of 45% sludge can be vitrified at either low or high temperatures depending on the glass additives used (a combination of diatomaceous earth and hydrous borax or perlite only). The pilot-scale studies indicated that glasses

capable of passing the PCT and the new disposal limits for the TCLP can be produced at 45% or 40% waste loadings depending on the melt temperature and glass additives used.

For RFP Precipitate simulated sludge, high melting temperature glasses with waste loadings of up to 75% are possible. When only lower melting temperatures are available, waste loadings of up to 65% are possible. In both cases, a mixture of diatomaceous earth and hydrous borax were used as the glass forming additives. Crucible-scale study results indicate that 3 wt% charcoal should be added to prevent the formation of a sulfate salt layer.

For LANL TA-50 simulated sludge, a durable glass can be produced at high temperatures with 65% or 35% waste loading depending on the glass additives used (either a mixture of Al_2O_3 and SiO_2 or Na_2CO_3 and diatomaceous earth). Crucible-scale results also indicated that it was possible to produce a durable, leach-resistant glass by combining LANL and RFP simulated sludges with waste loadings of up to 100% possible. However, the reality of mixing these two waste streams is very slim since they are not located on the same site, so the composition was not recommended for further pilot-scale studies.

The fabricated glasses were plotted on associated ternary diagrams. Glasses M1P, M2P, M4P, M5P, OR1-OR3, OR1P, RF2, RF6-RF10, LA1, LA3, and LA13 fell within the known glass forming region of the borosilicate ternary system, while glasses M3P, RF4, and RF5 fell within the expected glass phase separation region. Some of these glasses (OR1-OR3 and OR1P) also fell within the suspected glass forming region of the CaO-Al_2O_3-SiO_2 ternary system, as did glasses OR7, OR7P, OR10, OR11, LA7, LA8, and LA9. Additional glasses fell within the expected glass making region of the CaO-Fe_2O_3-SiO_2 ternary system. These glasses included RF6, RF8, RF9, and LA6. Finally, glasses were also made in the known glass forming region of the Na_2O-CaO-SiO_2 ternary system. These glasses were OR12P, LA17, and LA19.

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VITRIFICATION DEVELOPMENT FOR MIXED WASTES

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ABSTRACT

Vitrification is a promising approach to waste-form immobilization. It destroys hazardous organic compounds and produces a durable and highly stable glass. Vitrification tests were performed on three surrogate wastes during fiscal year 1994: 183-H Solar Evaporation Basin waste from Hanford, bottom ash from the Oak Ridge TSCA incinerator, and saltcrete from Rocky Flats. Preliminary glass development involved melting trials followed by visual homogeneity examination, short-duration leach tests on glass specimens, and long-term leach tests on selected glasses. Viscosity and electrical conductivity measurements were taken for the most durable glass formulations. Results for the saltcrete are presented in this paper and demonstrate the applicability of vitrification technology to this mixed waste.

INTRODUCTION

The U.S. Department of Energy (DOE) is considering high-integrity waste forms to dispose of mixed low-level radioactive and hazardous waste. Vitrification is a promising waste-form option; it destroys hazardous organic compounds and produces a durable and highly stable glass. Vitrification research has been conducted through the Mixed Waste Integrated Program (MWIP) of the U.S. Department of Energy (DOE) Office of Technology Development. During fiscal year 1994, MWIP requested that Pacific Northwest Laboratory (PNL) evaluate processing limits for vitrification treatment of DOE mixed wastes. Of particular importance was the investigation of glass concentration limits for such components as halides and sulfates. PNL was the original developer of DOE vitrification technology selected for treatment of high-level waste (HLW) in the United States and other countries. The expertise and facility capabilities developed under the HLW programs are directly applicable to mixed waste vitrification development. These laboratory studies precede design of vitrification equipment and process flowsheets to permit environmentally sound treatment and disposal of mixed waste.

The waste stream considered in this study was the saltcrete waste from the Rocky Flats Plant (RFP) in Colorado. The use of large quantities of nitric acid in the RFP's production process generated a high volume waste stream primarily composed of neutralized nitrate salts. The salt solution was treated with a flocculant to help precipitate solids and radioactive material and was then processed in a clarifier. The overflow from the clarifier was sent to an evaporator and concentrated to a 35 wt% nitrate or brine salt solution. The salt solution was dried in a spray dryer and combined with cement to produce a grouted material called saltcrete. Many of the saltcrete blocks swelled and crumbled during extended outdoor storage. This work investigated vitrification of the failed saltcrete blocks.

Glass development entails directly analyzing the composition of the waste or using the best available data, and integrating that information with knowledge of glass chemistry, experience at PNL, and literature surveys to formulate canditate glass compositions. The canditate compositions are then prepared in the lab with non-radioactive chemicals and trial melts are made. The melts are inspected and characterized for leachability, homogeneity, viscosity, and electrical conductivity.

METHODOLOGY

Crucible melts containing varying amounts of simulated waste and additives (e.g., silica, boric acid, alumina, and phosphate) were prepared to define a glass composition suitable for vitrification. Initial glass compositions were based on compositions reported in the literature and previous melting experience at PNL. Additional melts were performed to refine the compositions based upon the homogeneity, viscosity, and chemical durability observed in the preceding melts.

The saltcrete waste consists primarily of portland cement, spray dried salts, and water in proportions (wt%) ranging from 55% salt, 18% cement, and 27% water to 51.3% salt, 22.2% cement, and 26.5% water. An approximate composition of the dried salts and the average portland cement composition (1) before shutdown of RFP operations are shown in Table I.

A surrogate waste was formulated from batch chemicals to match the reported composition of the saltcrete waste. Crucible melts were made with measured amounts

of surrogate waste and additives. The surrogate waste and additives were manually mixed and transferred to a fused silica, alumina, or porcelain crucible. The crucibles were placed in a furnace and held at the melt temperature (1000C to 1400C) for one to two hours. In some melts, the crucible was removed and the glass poured to permit estimation of the melt viscosity. In others, the furnace was turned off and allowed to cool slowly before the samples were removed. The cooled crucibles or poured glass buttons were then visually inspected for overall appearance, unmelted material, multiple phases, and crystalline inclusions.

A simple screening leach test provided a rapid estimate of the chemical durability of the glass formulations that were visually homogenous and had a suitable melt viscosity. The screening test resembled the EPA's Toxicity Characteristic Leaching Procedure (TCLP) in that a crushed glass sample was tumbled with the leachate. In this test, a sample of glass was crushed and sieved to a -4mm/+1mm size fraction and leached with deionized (DI) water at room temperature for 24 hours, during which time it was tumbled at 30 rpm. The mass ratio of water to sample was 10:1. The leachate concentrations were measured by inductively coupled plasma atomic emission spectroscopy (ICP-AES). Comparison of the normalized leach rate for the sample and that of standard glasses leached under the same conditions provided a comparative assessment of the samples durability.

Chloride concentrations in the final melts were analyzed by fusing the sample in Na_2CO_3 and submitting the diluted fusion for ion chromatography analysis. Matrix spike recovery of chloride by this method exceeded 90%.

RESULTS AND DISCUSSION

Glass Formulation

The primary issue associated with vitrification of the saltcrete waste is the relatively high content of salts in the waste, specifically chlorides and sulfates. Many metal chlorides have a low melting point and high vapor pressure (2) and therefore volatilize readily during glass melting. Volatilization from the melt can result in the loss from the glass of the elements and radionuclides that were to be immobilized in the vitrified waste form (3,4). Sulfates and chlorides have low solubility in typical borosilicate glass systems (<1 wt%), and when they coexist in glass, the solubility of each is even lower (5). When present in excess of their solubility limits, sulfate and chloride can form a molten salt phase on the surface of the glass with resulting processing problems (electric short-circuiting in a joule-heated melter, increased corrosion, and volatilization) and impairment of product quality (concentration of hazardous/radioactive elements into a highly leachable phase). The effort to define process limits for various problem constituents in glass included attempts to develop formulations with better retention of chloride and sulfate in the glass.

Previous work supported by the MWIP program had demonstrated the usefulness of phosphate in vitrification of wastes high in sulfate (6). The use of phosphate glass formulations promotes the decomposition and volatilization of sulfate at a low temperature, thereby avoiding the problem of secondary sulfate-phase formation. Therefore, the sulfate content of the saltcrete waste is not expected to pose a problem for phosphate glass formulations. The current work further investigated phosphate glasses as a means of increasing chloride retention in the glass melt. Phosphate glasses can be melted at low temperatures (as low as 1000C in this work), potentially reducing chloride volatility, and can accommodate large anions in the glass structure. The ability of certain phosphate glasses to retain chloride in the melt was therefore investigated in this work. Some silicate glasses were also investigated to achieve higher waste loadings and to provide a comparison to a more traditional glass for the phosphate formulations tested.

Forty-nine waste glasses were formulated and prepared for the simulated saltcrete waste. Target melt compositions and other data for several of these melts are presented in Table II. All of the formulations listed produced homogeneous, glassy melts except for SC30, which had streaks of opacity indicating phase separation or crystallization in the melt. The phosphate melts were transparent yellow, and the silicate melts were transparent green. The phosphate formulations had lower melting points and generally lower waste loading than the silicate melts.

Leach rates of the saltcrete glasses by the simple leach test previously described was comparable to that of high-level waste reference glasses leached by this same method. (ARM-1 - 0.18 g/m2; DWPF-EA - 0.32 g/m2). Leach results are expressed as the total release (grams of glass leached per square meter of surface area). Both

phosphate and silicate formulations for the saltcrete produced glasses with a durability comparable to typical high-level waste glasses. A comparison of the results from all the melts allows some general statements to be made about the glass formulations tested. Durability of the phosphate glasses appeared to be improved with increased alumina content (up to 25 wt%). Boron was observed to have a favorable effect on glass formation up to at least 6 wt% in some of the phosphate melts; however, several silicate melts with boron added to lower the melting temperature formed a separated salt phase on the surface of the melt. Analysis of this phase by X-ray florescence showed the primary components to be Na, K, Cl, and S.

Chloride retention was investigated in several melts. A low melting phosphate formulation (1000C) was spiked with increasing amounts of chloride by substituting NaCl for Na2O on an equimolar Na basis. The target chloride content of the melts was 1.6 wt% to 9.3 wt%. The products ranged from homogeneous glasses to two-phase melts. Table III presents results from this series of melts. Retention of chloride in the vitrified product ranged from about 50% to 70% after melting for one hour in an open crucible. After 24 hr uncovered in the furnace, retention of chloride in a melt with the composition of SC16 was only 15%. Chloride retention in several silicate melts (1350C) and phosphate melts at higher temperatures (1150C to 1200C) ranged from about 20% to 40% after one hour.

The above melts show that at least 2.6 wt% chloride can be retained in a homogeneous glass without visible phase separation. At 3.6 wt% chloride, opaque white streaks were present in an otherwise homogeneous, transparent yellow glass. Higher chloride concentrations led to phase separation of a molten chloride salt. The low melting temperatures of certain phosphate glass formulations appeared to improve the retention of chloride in the melt; however, significant losses still occurred and most of the chloride was lost from the melts after 24 hours uncovered at the melt temperature. Increasing chloride content of the glass generally had a negative impact on durability, largely from enhanced leaching of sodium from the glass.

CONCLUSIONS

The saltcrete waste is readily vitrified into both phosphate and silicate glasses. The resulting product is stable and substantially reduced in volume. Nevertheless, the high chloride content of these wastes poses a problem for vitrification technology due to its limited solubility in the glass matrix and high volatility from the melt. Certain glasses were shown to retain a substantial, although limited, amount of chloride. However, increased chloride concentration has a detrimental effect on glass durability and potentially on the vitrification process due to chloride volatilization. Although the low-melting temperature phosphate formulation provided significant retention of chlorides, at best only a few weight percent chloride can be contained in the glass and, even at the lower melting temperatures, chloride volatilization is a significant concern.

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ABSTRACT
Treatability studies have been conducted at the laboratory-scale to evaluate vitrification of waste water sludges at the Oak Ridge Reservation (ORR). These studies are being conducted jointly by Westinghouse Savannah River Technology Center (SRTC) and Oak Ridge National Laboratory (ORNL). These studies include testing with surrogate waste formulations at both the laboratory-scale and pilot-scale, and testing with actual waste at the laboratory-scale, pilot-scale, and field-scale. ORR was chosen as the host site for the field-scale demonstration. The Y12 West End Treatment Facility (WETF) waste water treatment sludges, which are RCRA F-listed wastes, were chosen as the candidate waste stream for the first field-scale demonstration. The laboratory-scale "proof-of-principle" demonstrations reported in this study and the pilot-scale studies planned for FY95 on the WETF sludge will provide needed operating parameters for the planned field-scale demonstration. These laboratory-scale "proof-of-principle" and pilot-scale studies also provide needed data for the evaluation of the feasibility of vitrification as a stabilization option for a variety of wastes which do not currently meet RCRA/LDR (Resource Conservation and Recovery Act/Land Disposal Restrictions) requirements for storage/disposal and/or those for which treatment capacity does not presently exist.
INTRODUCTION
Technologies are being developed by the US Department of Energy' s (DOE) Nuclear Facility sites to convert low-level and mixed wastes to a solid stabilized waste form for permanent disposal. Vitrification is one of the most important and environmentally safest technologies being developed. The Environmental Protection Agency (EPA) has declared vitrification the Best Demonstrated Available Technology (BDAT) for high-level radioactive waste (1) and produced a Handbook of Vitrification Technologies for Treatment of Hazardous and Radioactive Waste (2). The DOE Office of Technology Development (OTD) has taken the position that mixed waste needs to be stabilized to the highest level reasonably possible to ensure that the resulting waste forms will meet both current and future regulatory specifications.
Vitrification produces durable waste forms at volume reductions up to 97% (3). Large reductions in volume minimize long-term storage costs making vitrification cost effective on a life cycle basis (4). The US DOE Savannah River Site (SRS), which is operated by Westinghouse Savannah River Company (WSRC), is currently investigating vitrification for disposal of various low-level and mixed wastes (3,5).
Vitrification has been demonstrated for in laboratory studies at SRS for 1) incinerator and 2) nickel plating line (RCRA F006) waste water sludge wastes. Stabilization of the heavy metals in the glass was achieved by use of reactive additives* such as diatomaceous earth, perlite (perflo), rice husk ash, and/or precipitated silica. The process/product models developed for high level radioactive waste glass were utilized to develop glass formulations which optimize glass processability, e.g viscosity, and product durability (6). This approach is currently being applied to the ORR Y12 West End Treatment Facility (WETF) RCRA listed waste water treatment sludges discussed in this study.
Reactive Additive Stabilization Process (RASP)
Reactive high surface area silica, as a waste form additive, was shown to greatly enhance the solubility and retention of hazardous, mixed and heavy metal species in glass (3). Highly reactive silica was found to increase the solubility and tolerance of Soda(Na_2O)Lime(CaO)-Silica(SiO_2) glass (SLS) and borosilicate (B_2O_3-SiO_2) glass formulations to atomistically bond waste species (8). Highly reactive silica lowers glassification temperatures, increases waste loadings which provides for large waste volume reductions, minimizes melt line corrosion, and produces EPA acceptable glasses. The Reactive Additive Stabilization Process (RASP) can be used to vitrify 1) spent filter aids from waste water treatment, 2) waste sludges, 3) combinations of spent filter aids from waste water treatment and waste sludges, 4) combinations

of supernate and waste sludges, 5) incinerator ash, 6) incinerator off-gas blowdown, 7) combinations of incinerator ash and off-gas blowdown, 8) cement formulations in need of remediation into glass, 9) ion exchange zeolites, 10) inorganic filter media, 11) asbestos or glass fiber filters, and 12) radioactive materials including TRU wastes.

West End Treatment Facility (WETF) Waste Water Treatment Sludges

The West End Treatment Facility at the Y-12 Plant on the Oak Ridge Reservation (ORR) treats nitrate-containing wastes by biodenitrification. Wastes are neutralized with lime and mixed with nutrients (acetate, triethylphosphate, etc.) and then transferred to a bioreactor for anoxic denitrification (7). The denitrified waste slurry is then subjected to bio-oxidation to complete the removal of residual organic compounds. The sludge is removed by gravitational sedimentation. The aqueous portion is decanted and sent to an effluent polishing unit which dissociates any residual uranium from its solubilizing complex with carbonate. The solution is then treated to coprecipitate any metals (including uranium) for removal. The sludge bottoms from the WETF clarifier are stored in four 500,000 gallon tanks awaiting further stabilization. Tanks 7, 8 and 9 are full and Tank 13 is partially full. The inventory as of 1992 was 250,000 ft3 or 1,900,000 gallons.

The WETF sludge is a RCRA listed waste (the EPA waste codes are F001, F002, F005 (from treatment of solvent residues), and F006 (from treatment of plating waste). The WETF sludge is primarily calcium carbonate, biomass, and iron oxyhydroxide (8). Depleted uranium (at an average of about 0.42% U-235 isotope, vs 0.71% in natural U) is the primary radioisotope of concern in this waste stream, with very low activity contributions from Tc-99 and transuranic isotopes (Np, etc) (8). Phenolic compounds represent the principal organic hazardous constituents in the sludge. Data from Tanks 7, 8, and 9 were compiled by Bostick (7) based on data supplied by SAIC and Y.F. Weaver of the ORR Y-12 plant (7). Bostick (7) derived a surrogate recipe for WETF Tank 7 wastes (Table I) during 1993.

VITRIFICATION TECHNOLOGY DEMONSTRATION

The vitrification demonstration of the ORR WETF wastes includes the following:

- Analyze wastes (FY93-94/ORR)
- Develop surrogate (FY93/ORR)
- Surrogate "proof-of-principle" laboratory scale studies (FY93-94/SRTC)
- Actual waste "proof-of-principle" laboratory scale studies (FY94-5/ORR+SRTC)
- Surrogate pilot-scale demonstration (FY94-95/SRTC and Clemson University)
- Actual waste pilot-scale demonstration (FY95/SRTC, ORR, and RUST Federal)
- Actual waste field-scale demonstration (FY96/SRTC+ORR-Ref. 9)

The initial testing of the surrogate and actual waste "proof-of-principle" testing being carried out jointly between SRTC and ORR under a Memorandum of Understanding (MOU) are discussed in this study. The pilot-scale demonstration on ORR WETF wastes is planned for mid FY95 at RUST Federal in Clemson, SC under a Cooperative Research and Development Agreement (CRADA) between RUST and SRTC. The field-scale demonstration will take place at ORR in a Transportable Vitrification System (TVS) being provided by SRTC (9) under a separate Memorandum of Understanding.

EXPERIMENTAL

The Tank 7 surrogate waste developed by Bostick (7) was used to optimize glass formulations at the Savannah River Technology Center (SRTC) for laboratory scale "proof-of-principle" testing. Reagent grade chemicals and varying sources of SiO2, e.g. silica sand, diatomaceous earth, Perlite, or precipitated silica. Cerium was substituted on a molar basis for uranium. Waste loadings were varied from 20 to 70 wt% waste on a dry oxide basis. All glasses were melted at temperatures between 1150 and 1350C in high purity A1203 crucibles for 4 hours. Small amounts of glass, ~100 gram batches, were made to optimize the best compositions for further testing. Each vitrified sample was submitted for x-ray diffraction (XRD) analysis so that the homogeneity of the resulting glass could be determined. For glasses which were not homogeneous, the crystalline phases were identified by XRD. If analysis by XRD was inconclusive, elemental scans by x-ray fluorescence (XRF) were substituted.

Real wastes were sampled at ORR from Tanks 8 and 13 and dried at 110C. Five samples of dried sludge from each tank were dissolved and analyzed. Glasses were made with the dried sludge and reagent grade chemicals and precipitated silica. All glasses were melted at temperatures between 1150 and 1350C in high purity A1203 crucibles for 4 hours. Small amounts of glass, ~150 gram batches, were made and submitted for waste form performance assessment via the EPA Toxic Characteristic Leaching

Procedure (TCLP). Each glass was doped with 20, 50 and 100 times the RCRA hazardous metals. These doped glasses will also be submitted for waste form performance assessment via the TCLP.

RESULTS AND DISCUSSION

WETF Waste Analyses and Accuracy Mass Balance

The compositions of the WETF Tank 8 and Tank 13 sludges as analyzed at ORR are given as wt% component species in Table I. Since no standards exist to assist in determining the accuracy of a sludge analysis, an accuracy mass balance was performed (10-11). The elemental cation weight percentages measured in the sludge are converted to oxides. An oxide mass balance is performed because the major anion remaining in a waste after vitrification is oxygen. In addition, waste analysis given on a dry calcine (oxide only) basis forms the basis for determining the "glass forming potential" of a waste.

Carbon dioxide will evolve from carbonate species in the waste, e.g. the $CaCO_3$, $NiCO_3$, and Na_2CO_3, and will be vaporized during vitrification. The CO_2 that will be volatilized from the glass melt during vitrification can be calculated from the molar concentrations of Ca, Ni, and Na in the waste (Table I). Likewise, the hydroxides present in the waste will vaporize as steam during the vitrification process. The OH that will be volatilized from the hydroxide species present in the sludge, e.g. $Al(OH)_3$ and $Fe(OH)_3$, during the vitrification process can be calculated from the molar concentrations of A1 and Fe (Table I). Although organics content was not measured in this study, the organic (oil and grease) and biomass reported in Ref. 8 for the WETF sludge was added to the analyses given in Table I. The organic and biomass will also be volatilized during vitrification. The mass balance of a sludge dried only at 110C, therefore, includes the volatiles (which will be lost at the 1150C vitrification temperature) and the glass oxide components (which will combine to form the glass during vitrification). A mass balance calculation of 100±5 wt% indicates (11) that the WETF analyses determined in this study were sufficiently accurate (Table I).

The WETF Tank 8 and Tank 13 analyses determined in this study are compared to earlier analyses reported in Refs. 7 and 12. These comparisons are shown in Table I. The earlier Tank 8 and Tank 13 analyses were comparable to those determined in this study but several minor constituents of the sludge had been omitted from the earlier analyses. Moreover, the analysis reported in Ref. 8 for Tank 8 was reported to be on a dry sludge basis. Application of the accuracy mass balance technique used in this study indicated that the composition given in Reference 8 for Tank 8 was on a wet sludge basis. The conversion from wet sludge basis to dry sludge basis has been accounted for in the data presented in Table I.

The WETF Tank 7 surrogate used in the simulated "proof-of-principle" studies and in the simulated "pilot scale" studies is also given in Table I for comparison. Since the simulant was developed from earlier composition analysis of Tank 7, 8, and 9, several of the minor constituents had been omitted from the surrogate. The accuracy mass balance methodology was applied to all of the analyses of real WETF sludge and the WETF Tank 7 surrogate. All of the analyses have a high degree of accuracy but the analyses presented in this study are within the mass balance accuracy criterion of 100±5 wt% while the previous analyses are not. All of the analyses indicate that there is very little chemical variation in the WETF sludges sampled from different tanks indicating that chemically the WETF sludge is very homogeneous.

WETF Glass Formulations

The high CaO content of the WETF wastes on a dry oxide basis provided the "lime" glass forming constituent for Soda-Lime-Silica (SLS) glass formulations. The SLS glass forming system is a well known system used to make window glass and it has been extensively studied by glass chemists since 1925 (13-16). Soda-lime-silica glass is known to readily incorporate heavy metals (17). SLS glass formulations have been used for In-Situ Vitrification (ISV) to solidify contaminated soils (18-19) and a barium analog (soda-baria-silica) glass has been used at Fernald to solidify residues from uranium ore processing that contain large amounts of uranium, radium, and lead (20). SLS glasses have also been successfully fabricated from reactive sodium metal contaminated during efforts to develop sodium-cooled fast breeder nuclear reactors (21). The SLS system has also been used in laboratory "proof-of-principle" testing to vitrify M-Area wastes and SRS incinerator ashes. The use of the SLS glass forming system and the use of the Reactive Additive Stabilization Process (RASP) was shown to extends the known glass forming region in

the SLS system (3) as shown in Fig. 2.
The WETF glass formulations are being optimized in the region of the SLS system where glasses are known to form stable homogeneous glasses. The process models developed for high level waste glasses were utilized to calculate the glass viscosity and durability as a function of temperature (6).
TABLE I
To date over 140 surrogate glass formulations were tested for the ORR WETF sludges using the Tank 7 surrogate developed by Bostick(7). The surrogate glass formulations examined the following effects:

- waste loading
- melt temperature
- varying alkali additives
- varying reactive silica additives (perlite, sand, and precipitated silica)
- melt line refractory corrosion
- general refractory corrosion
- predictability of process/product models

To date more than ten different WETF laboratory-scale "proof-of-principle" glass formulations have been made with real WETF Tank 8 waste. This adjusted Tank 8 composition from Ref. 7 was used to formulate the glasses for the crucible testing "proof-of-principle" with the real WETF Tank 8 sludge (the analysis shown for Tank 8 in Table I that was determined as part of this study was not available at the time at which the Tank 8 "proof-of-principle" studies were conducted). Approximately 150 grams of each formulation were melted and found to form homogeneous glass. Waste loadings varying from 20 to 70 wt% waste (on a calcine oxide basis, e.g. the compositions given in Table I normalized for the volatiles lost on ignition) were achieved at temperatures ranging from 1150C to 1350C (Fig. 1). One of the 35 wt% loaded waste glass compositions tested is of historical significance dating from the 1608 Jamestown, Virginia settlement.
All of the initial WETF Tank 8 glasses were subjected to the EPA Toxic Characteristic Leaching Procedure (TCLP) and met the EPA Universal Treatment Standards (UTS) promulgated September 19, 1994 in the Federal Register. Severe melt line corrosion was experienced with the higher waste loaded glass formulations and the compositions were further optimized using the Tank 7 surrogate. These optimized glass formulations which contain 35 to 50 wt% actual WETF Tank 8 waste, were doped with 20, 50, and 100 times the RCRA hazardous metals to test the flexibility of the glass formulations to retain hazardous species. These glasses are currently undergoing TCLP analysis.
The optimized glass formulations used for the actual WETF Tank 8 wastes were reformulated with actual WETF Tank 13 waste. The average Tank 13 waste analysis determined in this study was the analysis used to develop glass formulations for the crucible testing "proof-of-principle" with the real WETF Tank 13 sludge. The Tank 13 optimized glass formulations will also be doped with 20, 50, and 100 times the RCRA hazardous species. All of the Tank 13 glasses will be subjected to the EPA TCLP test. All of the actual Tank 8 and Tank 13 tests will also be subjected to the Product Consistency Test (ASTM C1285).
WASTE MINIMIZATION
The 35 to 70% waste loaded glasses correspond to volume reductions of 73 to 87% for the ORR WETF waste water sludges, assuming 40 wt% solids in the sludge and a density of glass of 2.7 kg/L. Therefore, based on the 1992 tank inventory of 1,900,000 gallons of sludge, this waste will produce between 513,000 to 247,000 gallons of glass. By comparison stabilization of the ORR WETF waste water sludges in cement will cause a 2X volume increase and the final waste form volume will be 3,800,000 gallons. Comparisons of the overall waste volume reduction compared to alternative stabilization strategies in cement (Table II) shows that vitrification of ORR WETF sludges reduces the stabilized volume by 87% (Option 2 at 35% waste loading) to 94% (Option 5 at 70% waste loading) relative to the base case cement stabilization.
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Fig. 1.
TABLE II
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VITRIFICATION TREATABILITY STUDIES OF ACTUAL WASTE WATER TREATMENT SLUDGES (U)

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INTRODUCTION

Technologies are being developed by the US Department of Energy's (DOE) Nuclear Facility sites to convert low-level and mixed wastes to a solid stabilized waste form for permanent disposal. One of the alternative waste forms is vitrification. The Environmental Protection Agency (EPA) has declared vitrification the Best Demonstrated Available Technology (BDAT) for high-level radioactive waste (1) and produced a Handbook of Vitrification Technologies for Treatment of Hazardous and Radioactive Waste.(2) The DOE Office of Technology Development (OTD) has taken the position that mixed waste needs to be stabilized to the highest level reasonably possible to ensure that the resulting waste forms will meet both current and future regulatory specifications. Vitrification produces durable waste forms at volume reductions up to 97%.(3) Large reductions in volume minimize long-term storage costs making vitrification cost effective on a life cycle basis.(4) The US DOE Savannah River Site (SRS), which is operated by Westinghouse Savannah River Company (WSRC), is currently investigating vitrification for disposal of various low-level and mixed wastes (3,5). Vitrification has been demonstrated for in laboratory studies at SRS for (1) incinerator and (2) nickel plating line (RCRA F006) waste water sludge wastes. Stabilization of the heavy metals in the glass was achieved by use of reactive additives* such as diatomaceous earth, perlite (perflo), rice husk ash, and/or precipitated silica. The process/product models developed for high level radioactive waste glass were utilized to develop glass formulations which optimize glass processability, e.g viscosity, and product durability (6). This approach is currently being applied to the ORR Y12 West End Treatment Facility (WETF) RCRA listed waste water treatment sludges discussed in this study.

Reactive high surface area silica, as a waste form additive, was shown to greatly enhance the solubility and retention of hazardous, mixed and heavy metal species in glass (7). Highly reactive silica was found to increase the solubility and tolerance of Soda(Na_2O)- Lime(CaO)-Silica(SiO_2) glass (SLS) and borosilicate (B_2O_3-SiO_2) glass formulations to atomistically bond waste species (9). Highly reactive silica lowers glassification temperatures, increases waste loadings which provides for large waste volume reductions, minimizes melt line corrosion, and produces EPA acceptable glasses. The Reactive Additive Stabilization Process (RASP) can be used to vitrify (1) spent filter aids from waste water treatment, (2) waste sludges, (3) combinations of spent filter aids from waste water treatment and waste sludges, (4)

combinations of supernate and waste sludges, (5) incinerator ash, (6) incinerator off-gas blowdown, (7) combinations of incinerator ash and off-gas blowdown, (8) cement formulations in need of remediation into glass, (9) ion exchange zeolites, (10) inorganic filter media, (11) asbestos or glass fiber filters, and (12) radioactive materials including TRU wastes.

West End Treatment Facility (WETF) Waste Water Treatment Sludges

The West End Treatment Facility at the Y-12 Plant on the Oak Ridge Reservation (ORR) treats nitrate-containing wastes by biodenitrification. Wastes are neutralized with lime and mixed with nutrients (acetate, triethylphosphate, etc.) and then transferred to a bioreactor for anoxic denitrification (8). The denitrified waste slurry is then subjected to bio-oxidation to complete the removal of residual organic compounds. The sludge is removed by gravitational sedimentation. The aqueous portion is decanted and sent to an effluent polishing unit which dissociates any residual uranium from its solubilizing complex with carbonate. The solution is then treated to coprecipitate any metals (including uranium) for removal. The sludge bottoms from the WETF clarifier are stored in four 500,000 gallon tanks awaiting further stabilization. Tanks 7, 8 and 9 are full and Tank 13 is partially full. The inventory as of 1992 was 250,000 ft3 or 1,900,000 gallons.

The WETF sludge is a RCRA listed waste (the EPA waste codes are F001, F002, F005 (from treatment of solvent residues), and F006 (from treatment of plating waste). The WETF sludge is primarily calcium carbonate, biomass, and iron oxyhydroxide (9). Depleted uranium (at an average of about 0.42% U235 isotope, vs 0.71% in natural U) is the primary radioisotope of concern in this waste stream, with very low activity contributions from Tc99 and transuranic isotopes (Np, etc) (9). Phenolic compounds represent the principal organic hazardous constituents in the sludge. Data from Tanks 7, 8, and 9 were compiled by Bostick (8) based on data supplied by SAIC and Y.F. Weaver of the ORR Y-12 plant (8). Bostick (8) derived a surrogate recipe for WETF Tank 7 wastes (Table I) during 1993.

TABLE I

VITRIFICATION TECHNOLOGY DEMONSTRATION

The vitrification demonstration of the ORR WETF wastes includes the following:

- Analyze wastes (FY93-94/ORR)
- Develop surrogate (FY93/ORR)
- Surrogate "proof-of-principle" laboratory scale studies (FY93-94/SRTC)
- Actual waste "proof-of-principle" laboratory scale studies (FY94-5/ORR+SRTC)
- Surrogate pilot-scale demonstration (FY94-95/SRTC and Clemson University)
- Actual waste pilot-scale demonstration (FY95/SRTC, ORR, and RUST Federal)
- Actual waste field-scale demonstration (FY96/SRTC+ORR-Reference 10)

The initial testing of the surrogate and actual waste "proof-of-principle" testing being carried out jointly between SRTC and ORR under a Memorandum of Understanding (MOU) are discussed in this study. The pilot-scale demonstration on ORR WETF wastes is planned for mid FY95 at RUST Federal in Clemson, SC under a Cooperative Research and Development Agreement (CRADA) between RUST and SRTC. The field-scale demonstration will take place at ORR in a Transportable Vitrification System (TVS) being provided by SRTC (12) under a separate Memorandum of Understanding.

EXPERIMENTAL

The Tank 7 surrogate waste developed by Bostick (8) was used to optimize glass formulations at the Savannah River Technology Center (SRTC) for laboratory scale "proof-of-principle" testing. Reagent grade chemicals and varying sources of SiO_2, e.g. silica sand, diatomaceous earth, Perlite, or precipitated silica. Cerium was substituted on a molar basis for uranium. Waste loadings were varied from 20 to 70 wt% waste on a dry oxide basis. All glasses were melted at temperatures between 1150o and 1350oC in high purity Al_2O_3 crucibles for 4 hours. Small amounts of glass, ~100 gram batches, were made to optimize the best compositions for further testing. Each vitrified sample was submitted for x-ray diffraction (XRD) analysis so that the homogeneity of the resulting glass could be determined. For glasses which were not homogeneous, the crystalline phases were identified by XRD. If analysis by XRD was inconclusive, elemental scans by x-ray fluorescence (XRF) were substituted.

Real wastes were sampled at ORR from Tanks 8 and 13 and dried at 110oC. Five samples of dried sludge from each tank were dissolved and analyzed. Glasses were made with the dried sludge and reagent grade chemicals and precipitated silica. All glasses were melted at temperatures between 1150o and 1350oC in high purity Al_2O_3 crucibles for 4 hours. Small amounts of glass, ~150 gram batches, were made and submitted for

waste form performance assessment via the EPA Toxic Characteristic Leaching Procedure (TCLP). Each glass was doped with 20, 50 and 100 times the RCRA hazardous metals. These doped glasses will also be submitted for waste form performance assessment via the TCLP.

RESULTS AND DISCUSSION

WETF Waste Analyses and Accuracy Mass Balance

The WETF Tank 8 and Tank 13 sludges dried at 110oC and analyzed at ORR are given as wt% component species in Table I. Since no standards exist to assist in determining the accuracy of a sludge analysis, an accuracy mass balance was performed (11,12). The elemental cation weight percentages measured in the sludge are converted to oxides. An oxide mass balance is performed because the major anion remaining in a waste after vitrification is oxygen. In addition, waste analysis given on a dry calcine (oxide only) basis forms the basis for determining the "glass forming potential" of a waste.

Carbon dioxide will evolve from carbonate species in the waste, e.g. the $CaCO_3$, $NiCO_3$, and Na_2CO_3, and will be vaporized during vitrification. The CO_2 that will be volatilized from the glass melt during vitrification can be calculated from the molar concentrations of Ca, Ni, and Na in the waste (Table I). Likewise, the hydroxides present in the waste will vaporize as steam during the vitrification process. The OH that will be volatilized from the hydroxide species present in the sludge, e.g. $Al(OH)_3$ and $Fe(OH)_3$, during the vitrification process can be calculated from the molar concentrations of Al and Fe (Table I).

Although organics content was not measured in this study, the organic (oil and grease) and biomass reported in Ref. 8 for the WETF sludge was added to the analyses given in Table I. The organic and biomass will also be volatilized during vitrification. The mass balance of a sludge dried only at 110oC, therefore, includes the volatiles (which will be lost at the 1150oC vitrification temperature) and the glass oxides components (which will combine to form the glass during vitrification). A mass balance calculation of 1005 wt% indicates (12) that the WETF analyses determined in this study were highly accurate (Table I).

The WETF Tank 8 and Tank 13 analyses determined in this study are compared to earlier analyses reported in Refs. 8 and 13. These comparisons are shown in Table I. The earlier Tank 8 and Tank 13 analyses were comparable to those determined in this study but several minor constituents of the sludge had been omitted from the earlier analyses. Moreover, the analysis reported in Ref. 8 for Tank 8 was reported to be on a dry sludge basis. Application of the accuracy mass balance technique used in this study indicated that the composition given in Ref. 8 for Tank 8 was on a wet sludge basis. The conversion from wet sludge basis to dry sludge basis has been accounted for in the data presented in Table I.

The WETF Tank 7 surrogate used in the simulated "proof-of-principle" studies and in the simulated "pilot scale" studies is also given in Table I for comparison. Since the simulant was developed from earlier composition analysis of Tank 7, 8, and 9, several of the minor constituents had been omitted from the surrogate. The accuracy mass balance methodology was applied to all of the analyses of real WETF sludge and the WETF Tank 7 surrogate. All of the analyses have a high degree of accuracy but the analyses presented in this study are within the mass balance accuracy criterion of 100 5 wt% while the previous analyses are not. All of the analyses indicate that there is very little chemical variation in the WETF sludges sampled from different tanks indicating that chemically the WETF sludge is very homogeneous.

WETF Glass Formulations

The high CaO content of the WETF wastes on a dry oxide basis provided the "lime" glass forming constituent for Soda-Lime-Silica (SLS) glass formulations. The SLS glass forming system is a well known system used to make window glass and it has been extensively studied by glass chemists since 1925 (14-17). Soda-silica glass is known to readily incorporate heavy metals (18). SLS glass formulations have been used for In-Situ Vitrification (ISV) to solidify contaminated soils (19-20) and a barium analog (soda-baria-silica) glass has been used at Fernald to solidify residues from uranium ore processing that contain large amounts of uranium, radium, and lead (21). SLS glasses have also been successfully fabricated from reactive sodium metal contaminated during efforts to develop sodium-cooled fast breeder nuclear reactors (22). The SLS system has also been used in laboratory "proof-of-priciple" testing to vitrify M-Area wastes and SRS incinerator ashes. The use of the SLS glass forming system and the use of the Reactive Additive

Stabilization Process (RASP) was shown to extend the known glass forming region in the SLS system (7) as shown in Fig. 1.
The WETF glass formulations are being optimized in the region of the SLS system where glasses are known to form stable homogeneous glasses. The process models developed for high level waste glasses were utilized to calculate the glass viscosity and durability as a function of temperature (6).
To date over 140 surrogate glass formulations were tested for the ORR WETF sludges using the Tank 7 surrogate developed by Bostick (8). The surrogate glass formulations examined the following effects:

- waste loading
- melt temperature
- varying alkali additives
- varying reactive silica additives (perlite, sand, and precipitated silica)
- melt line refractory corrosion
- general refractory corrosion
- predictability of process/product models

To date more than ten different WETF laboratory-scale "proof-of-principle" glass formulations have been made with real WETF Tank 8 waste. This adjusted Tank 8 composition from Ref. 8 was used to formulate the glass formulations for the crucible testing "proof-of-principle" with the real WETF Tank 8 sludge (the analysis shown for Tank 8 in Table I that were determined as part of this study were not available at the time at which the Tank 8 "proof-of-principle" studies were conducted.) Approximately 150 grams of each formulation were melted and found to form homogeneous glass. Waste loadings varying from 20 to 70 wt% waste (on a calcine oxide basis, e.g. the compositions given in Table I normalized for the volatiles lost on ignition) were achieved at temperatures ranging from 1150oC to 1350oC (Fig. 1). One of the 35 wt% loaded waste glass compositions tested is of historical significance dating from the 1608 Jamestown, Virginia settlement.
Table II
All of the initial WETF Tank 8 glasses were subjected to the EPA Toxic Characteristic Leaching Procedure (TCLP) and met the EPA Universal Treatment Standards (UTS) promulgated September 19, 1994 in the Federal Register. Severe melt line corrosion was experienced with the higher waste loaded glass formulations and the compositions were further optimized using the Tank 7 surrogate. These optimized glass formulations which contain 35 to 50 wt% actual WETF Tank 8 waste, were doped with 20, 50, and 100 times the RCRA hazardous metals to test the flexibility of the glass formulations to retain hazardous species. These glasses are currently undergoing TCLP analysis.
The optimized glass formulations used for the actual WETF Tank 8 wastes were reformulated with actual WETF Tank 13 waste. The average Tank 13 waste analysis determined in this study was the analysis used to develop glass formulations for the crucible testing "proof-of-principle" with the real WETF Tank 13 sludge. The Tank 13 optimized glass formulations will also be doped with 20, 50, and 100 times the RCRA hazardous species. All of the Tank 13 glasses will be subjected to the EPA TCLP test. All of the actual Tank 8 and Tank 13 tests will also be subjected to the Product Consistency Test (ASTM C1285).
WASTE MINIMIZATION
The 35 to 70% waste loaded glasses correspond to volume reductions of 73 to 87% for the ORR WETF waste water sludges, assuming 40 wt% solids in the sludge and a density of glass of 2.7 kg/L. Therefore, based on the 1992 tank inventory of 1,900,000 gallons of sludge, will make between 513,000 to 247,000 gallons of glass. By comparison stabilization of the ORR WETF waste water sludges in cement will cause a 2X volume increase and the final waste form volume will be 3,800,000 gallons. Comparisons of the overall waste volume reduction compared to alternative stabilization strategies in cement (Table II) shows that vitrification of ORR WETF sludges reduces the stabilized volume by 87% (Option 2 at 35% waste loading) to 94% (Option 5 at 70% waste loading) relative to the base case cement stabilization.
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DEVELOPMENT OF A TRANSPORTABLE VITRIFICATION SYSTEM FOR MIXED WASTE

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ABSTRACT

The Department of Energy (DOE), through the Mixed Waste Integrated Program, has identified a need to move mixed-waste vitrification technology from the laboratory to the field as rapidly as possible. A great deal of work over the last few years has shown the feasibility of immobilizing selected hazardous waste streams in a vitrified product. Lab-scale work has been extended to pilot-scale tests, usually with surrogates of the actual waste. DOE felt that the technology was mature enough to allow demonstration in the field on actual wastes, with units that would be prototypic of full-sized waste treatment equipment. To this end, DOE's Office of Technology Development sponsored the Westinghouse Savannah River Company (WSRC) to specify, procure, test, and operate a field scale demonstration using mobile equipment. Oak Ridge Reservation was chosen as the initial location for the field demonstration and Martin Marietta Energy Systems (MMES) was tasked with all permitting, site preparation, and field support activities.

During September 1993, WSRC used a "Vendor Forum" to solicit preliminary proposals for the Transportable Vitrification System (TVS). A number of quality proposals were received and evaluated. A vendor was selected and detailed negotiations were completed in August 1994, at which time a contract was signed for the TVS. In parallel, WSRC opened a dialogue with MMES to explore candidate waste streams at the Oak Ridge Reservation for the first TVS vitrification campaign. After some preliminary work, a group of waste water sludges were selected. The first of these to be demonstrated with the TVS will be the West End Treatment Facility sludge. This paper describes the development of specifications for the TVS, the design and construction activities to date, and ongoing efforts for permitting, site support, and the schedule for field application.

INTRODUCTION

The Transportable Vitrification System (TVS) is an outgrowth of a much larger US Department of Energy (DOE) program that deals with the large quantity and variability of mixed low-level and hazardous waste that exist in the DOE complex. The technology development aspect of this program is administered through the Mixed Waste Integrated Program (MWIP) of DOE's Office of Technology Development (DOE-OTD). The management of this program realized that the time had come to take selected mixed waste treatment processes from the laboratory to actual field application. The TVS and related programs are Westinghouse Savannah River Company's (WSRC) response to this challenge.

The concept of treating mixed waste using vitrification evolved from the high-level waste (HLW) vitrification program. Many of the personnel involved in the present program have extensive experience with the Defense Waste Processing Facility (DWPF), a high-level waste vitrification facility now undergoing non-radioactive shakedown testing at the Savannah River Site (SRS). Vitrification offers many of the same advantages for the treatment of low-level and hazardous waste as seen in HLW vitrification_namely, the ability to "lock up" dangerous chemical species in a durable glass form while achieving sizable volume reduction. The fact that low-level and mixed wastes (LLMW) does not require extensive shielding and remote operation as in the case of high-level waste allows the construction of LLMW vitrification equipment, of comparable throughput rates to DWPF, at a cost almost three orders of magnitude less.

As in the case of HLW, work in this program began with "crucible tests" to determine the feasibility of the process and the range of successful glass formulations in the selected glass systems (1,2). This work was initially carried out with surrogates of the candidate waste streams and is presently being extended to tests of actual waste, as described elsewhere (3,4).

Based on the success of early crucible studies in the summer of 1993, it was proposed to build a mobile vitrification facility capable of treating significant quantities (i.e., tons per day) of waste. This concept has evolved into the Transportable Vitrification System (TVS). In parallel with the development of the TVS concept, WSRC surveyed various waste streams as candidates for the first demonstration of the TVS. Negotiations during the fall of 1993 led to the selection of DOE's Oak Ridge Reservation as the first host site for the TVS. This was based in part upon the large number of suitable waste streams available and upon Martin Marietta Energy Systems (MMES) enthusiastic cooperation. Selection of the candidate waste stream will be discussed in this paper.

Early in the TVS program DOE and WSRC realized that the jump from crucible studies to field application is a considerable scale up of the technology. This point of view was strengthened by our experience with similar, smaller scale melters, at the DOE/Industry Center for Vitrification Research at Clemson University's Environmental Systems Engineering Department. Problems of feed handling, emissions control, melter operability, and a host of other issues can not be investigated adequately in crucible tests; therefore, a third element was added to the TVS program, testing of surrogate and actual material in "laboratory-scale" melters. Surrogate testing will take place at the vitrification facility at Clemson, while the testing with actual waste will take place at RUST Federal's facility next door to the Clemson vitrification laboratory. Both test programs will use smaller melters than the TVS, although similar in design. This work will be reported after its conclusion later this year.

As described above, DOE and WSRC have taken an integrated approach to the application of vitrification to mixed waste. This approach is carefully structured to provide testing at three different scales proceeding from crucible to laboratory melter to field application. The first two elements of the program are described elsewhere (3). The remainder of this paper will concentrate on the development of the TVS and preparations for field application.

TRANSPORTABLE VITRIFICATION SYSTEM

Procurement of the TVS

The original specification for the TVS was contained in the "Special Consolidated Solicitation, No. E10600-E1" which appeared in August 1993. The relevant section of this solicitation, entitled "Portable demonstration facilities with offgas treatment, melters, and analytical instrumentation for melter feed and product analyses", attracted nine proposals. A technical evaluation team (D. F. Bickford, A. L. Kielpinski, and J. C. Whitehouse) reviewed and ranked the proposals. After additional review by WSRC Procurement and DOE-SR, EnVitCo Inc., of Toledo, Ohio, was selected and asked to provide a detailed proposal. The detailed proposal was received in March 1994, and awarded to EnVitCo in August 1994.

The contract calls for EnVitCo to provide a complete, integrated, transposable vitrification system consisting of four modules: Waste and Additive Materials Preparation, Melter, Emissions Control, and Control and Services. The purpose of the system is to demonstrate treatment of small to medium quantities of low-level and mixed wastes to produce a durable glass product suitable for disposal in an approved repository.

Writing specifications for the TVS was difficult due to its broad mission to treat a wide variety of (mostly unidentified) low-level and mixed wastes. Therefore, the specifications were much less detailed than those for a fixed facility designed for one or two well characterized waste streams. Instead, the TVS must deal with waste streams which range from contaminated soils to waste water treatment sludges of widely varying chemical composition.

Some of the more important specifications include the following:

- Joule heated, cold top melter capable of producing 136 kilograms (300 pounds) of glass per hour with dry feed
- melter temperature range of 1100_1400-C
- slurried or dry waste feed
- provision for the simultaneous introduction of three additive materials with mixing tanks
- offgas system to maintain the melter under vacuum conditions
- 99.99% minimum particulate removal, at one micron
- ability to remove dilute acid gases from the offgas
- design to include features to make radiological decontamination easier

modules to be capable of shipment on standard trailers

TVS Design and Fabrication

EnVitCo began design activities as soon as the contract was finalized. It soon became apparent that the TVS would not be a mobile facility. The size of the melter was too large to fit completely on a single trailer; therefore, a modular approach was adopted. Where feasible, entire subsystems (i.e., the waste and additives equipment) are contained in a single sealed unit; similarly, the control and service equipment is contained in a single module. However, the melter unit is broken down into several separate modules for shipping. Once on site, it will be assembled by bolting the modules together. The emissions control equipment is broken down into three skids for shipment. EnVitCo expects that shipment of the TVS will require five tractor trailer loads. Figure 1 is a schematic layout of the TVS.

Fig. 1.

Referring to Fig. 1, waste to be processed is delivered to the Waste and Additives Module as either dry or slurried waste. While slurried waste is pumped directly to the Blend Tank, dry waste is dumped into the Waste Hopper where a screw feeder transports it to the Blend Tank. Glass-forming additives are supplied in bulk bag containers and transported to the Blend Tank by a metering screw feeder. Water is added as necessary. The Blend Tank is placed on load cells to aid the operator in achieving the proper mix of waste and additives. An agitator in the Blend Tank homogenizes the feed. Once the batch is ready, it is pumped to the Surge Tank, which is also provided with an agitator. A recirculation loop transports the slurried feed to the melter where a side stream is drawn off and metered for introduction into the melter.

The melter contains three main chambers. The largest is the central, refractory-lined processing chamber into which the feed is introduced. This chamber contains the primary electrodes. Slurried feed forms a "cold-cap" on the surface of the molten glass that helps to reduce emissions of volatile metals. Convective currents in the glass, set up by the electrodes, draw fresh material from the cold-cap into the glass pool where the vitrification process takes place. Glass is drawn through a refractory-lined "throat" into the glass drain chamber. A nuclear-level gauge controls the glass level in the melter by moving the spindle of a submerged drain valve. When this valve is open, glass drains by gravity into glass receptacles placed on a conveyor. After filling, the containers are allowed to cool and are removed by a forklift truck for storage or disposal. The third chamber, which has a separate drain mechanism, is designed to remove sulfates that may collect on the glass pool surface. The processing of some waste streams may result in the accumulation of metals in the main melter chamber; therefore, a third "metals drain" is provided for this eventuality.

Offgas from the melter is drawn off through a refractory-lined duct to the emission control module. This module will be supplied by Anderson 2000, of Peachtree City Georgia, under EnVitCo's contract. The unit consists of a quencher, packed bed cooler, variable throat venturi scrubber, mist eliminator, reheater, HEPA filters, and fans. Treated offgas is released through a 15 m (50 foot) stack, with provisions for Environmental Protection Agency (EPA) approved particulate and gas sampling.

The Control and Services Module contains the control room for the TVS, as well as the power supply and conditioning equipment for the melter. A standard programmable logic control system will be used to control most of the operations of the TVS from the control room.

Schedule and Status

As of early January 1995, 55% of the engineering work for the TVS has been completed. Long lead items, such as refractory and power conditioning equipment, have been ordered. EnVitCo expects to deliver the TVS within the one year contract duration (i.e., before August 16, 1995).

Prior to application of the TVS on actual waste, it will be fully tested on surrogate waste at Clemson University. This shakedown test is scheduled to last six to eight weeks, including assembly and disassembly. Both equipment and procedures will be tested with a non-radioactive surrogate of the actual waste to be run at Oak Ridge. Any problems encountered will be investigated and rectified before the TVS is released for field operation. Training of operators will also take place at this time. Processing of actual waste will begin at Oak Ridge in the fall of 1995.

PREPARATION FOR FIELD APPLICATION OF THE TVS

During the design and fabrication activities described above, Oak Ridge personnel

pursued activities in three areas to support TVS field operation: selection of waste stream(s) to be demonstrated; preparation of required permit applications; and site preparation activities.

Waste Stream Selection

Oak Ridge personnel reviewed candidate waste streams from the Y-12 and K-25 plants and selected a primary and backup waste. The primary waste stream will be the West End Treatment Facility (WETF) sludge, while the backup waste stream is the Central Neutralization Facility (CNF) sludge. Approximately 40,000 kg (88,000 pounds) of waste will be required for each waste stream demonstrated.

The West End Treatment Facility was constructed to treat nitrate-bearing aqueous waste generated at the Y-12 weapons plant located in Oak Ridge, Tennessee. The facility is a batch processing unit that neutralizes the nitrate-bearing wastes prior to biological denitrifying the waste. Hydrated lime is used to neutralize the acidic wastes. The nitrate waste is converted anaerobically to nitrogen and carbon dioxide. The hydrated lime reacts with the carbon dioxide to form calcium carbonate, which precipitates as the major part of the solids generated from the process. The remaining solids are generated from hydroxide precipitation of heavy metals.

The sludge is classified as low-level mixed waste, the average uranium content is approximately 1400 ppm. Average heavy metal concentrations are barium 540 ppm, lead 240 ppm, cadmium 45 ppm, chromium 410 ppm, and silver 25 ppm. The sludge has an average calcium content of 257,000 ug/g with a pH of 8.9.

The Central Neutralization Facility, located at the Oak Ridge K-25 Site, treats liquid effluent from the mixed waste incinerator, the K-25 Site steam plant, a metal cleaning facility, and various small quantity or infrequent streams. The process uses a hydrated lime slurry to neutralize the effluent and precipitate metals. The sludge, classified as a low-level mixed waste, averages a total radionuclide activity of 1600 pCi/g, with uranium, thorium and technetium-99 as the primary radionuclides. Average concentrations of the RCRA metals in the sludge are barium 90 ug/g, silver 4 ug/g, mercury 4 ug/g, cadmium 3 ug/g, chromium 95 ug/g, and lead 110 ug/g. The sludge has an average calcium content of 58,000 ug/g with a pH of 7.8.

Permitting Aspects

A Research, Development and Demonstration permit will be necessary for the demonstration because the volume of material to be treated exceeds the limits allowed by treatability study exemptions. In addition, the Tennessee Department of Environment and Conservation has a DOE Oversight office located in Oak Ridge that will allow for close coordination with the oversight office thereby expediting the review process for permit approval.

Because low-level radioactive waste will be treated, it is necessary to invoke the radioactive processing requirements of the National Emission Standards for Hazardous Air Pollutants (NESHAP) regulations. The preliminary Rad-NESHAP evaluation for the vitrification demonstration indicates that the process will not require Rad-NESHAP approval for construction. However, due to the large volume of material and temperatures greater than 100-C, an EPA approved continuous monitoring device will be required on the process exhaust (stack). The criteria for not needing an approval to construct from EPA is based on the source having an actual effective dose equivalent (EDE) of less than 0.1 mrem/yr at the site boundary. The criteria for needing to install an EPA approved continuous monitoring device is based on having a potential EDE of greater than 0.1 mrem/yr. The potential dose is calculated with no credit for air pollution control equipment. The TVS will employ continuous sampling of stack emissions. The samples gathered will be counted with radiation detection equipment to confirm that the TVS is operating below the required release levels.

Site Preparation

The major factors considered during site selection at the K-25 Site were:

- utility requirements
- site access
- site grading (size and slope of the area, amount of preparation work required for set up)
- proximity to other activities in the plant

Several utility requirements were identified for the demonstration. The single largest utility requirement was electricity with a requirement of 1600 amps of three-phase, 60 Hz current at 480 volts. The power requirement quickly narrowed the possible sites because significant power line upgrades would be required to place the demonstration in the majority of the areas available in the plant.

A significant amount of waste will be transported to and from the demonstration site; therefore, site access needs to be conducive to truck and tanker movement into and out of the site. Sites with small staging areas and located near busy roads in the plant were eliminated.
A 1/2 acre site was required to facilitate the set up of the vitrification equipment, staging pre- and post- treated wastes, and storage of secondly wastes. A level site with an impervious surface was desired to allow erection of the equipment and to contain any spills that could potentially occur during the demonstration. In addition, a site located away from other plant activities was preferred to minimize potential exposure of plant personnel to wastes being treated by the demonstration. The selected site for the demonstration is a former UF6 cylinder storage yard located near the main power distribution station for the K-25 plant. The yard is partially covered with a concrete pad, with the remainder of the yard having a compacted gravel base. Site preparation will be limited to paving the gravel portion of the yard. A road to the site, located at the west end of the plant (away from other activities in the plant), currently exists since the area was previously used for cylinder storage.

CONCLUSION

The TVS is an ambitious response to DOE's challenge to demonstrate mixed and low-level waste vitrification on a field scale. The TVS is designed to treat significant quantities of waste while maximizing the flexibility of the system to deal with a wide range of waste streams. Engineering is well advanced on this project, and we expect to perform the first demonstration with actual waste by the end of 1995.

ACKNOWLEDGMENTS

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APPLICATION OF ADVANCED DIAGNOSTIC TECHNIQUES TO THE STUDY OF MIXED WASTE VITRIFICATION

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ABSTRACT
Results from the application of a number of different diagnostic instrumentation systems to the processing of a surrogate mixed-waste stream are reported. Measurements have been conducted by personnel of the Diagnostic Instrumentation and Analysis Laboratory of Mississippi State University on a melter located at the DOE/Industry Center for Vitrification Research (administered by Westinghouse Savannah River Company) at Clemson University. Studies included measurement of the temperature of the molten glass within the melter and analysis of the thermal and compositional characteristics of the glass as it exited the melter. Velocities and off-gas compositions were also determined. The data obtained indicate that specific facility and waste product characteristics were successfully determined. The results will also allow for future comparison to different surrogate streams and remediation technologies and may be of interest for process scaling.
INTRODUCTION
The U.S. Department of Energy, through the Mixed Wasted Integrated Program, is sponsoring advanced vitrification research and development at the DOE/Industry Center for Vitrification Research (DOE/ICVR), located at Clemson University's Environmental Systems Engineering Department (CU/ESE). The focus of this work to date has been the vitrification of surrogates of mixed waste streams found at several DOE sites (1,2). Westinghouse Savannah River Company (WSRC) has administered this work for DOE.
During the fall of 1993, DOE identified an opportunity to employ advanced techniques, under development at the Diagnostic Instrumentation and Analysis Laboratory (DIAL) at Mississippi State University, to mixed-waste vitrification processes. Specific applications include measurements at high temperatures; evaluation of organics, particulates, and toxic heavy metals in the off-gas; and the implementation of suitable control schemes based on these systems. To this end a number of instruments are at various stages of completion. WSRC recognized the potential value of the information content from these methods and a collaboration was initiated.
Two field campaigns (July 25-29 and September 12-16, 1994) were conducted by DIAL at the DOE/ICVR during the summer of 1994. The primary objectives of the work were to gather data from operating, pilot-scale melters in order to a) characterize the melt and gas stream at selected measurement locations; b) obtain useful facility data; c) demonstrate the capability of the measurement techniques; and d) enable planning for future tests. Results were obtained on two ICVR melters: the Glass Tech Stir-Melter and the EnVitCo cold-top melter. The results given here correspond to surrogate waste stream (Savannah River Site M-Area) processing using the Stir-Melter.
The measurement strategy consisted of the determination of the temperatures of the molten pool within the melter and the associated glass pour (pyrometry methods); thermal profiles and shapes of the pour (multi-color imaging system); evaluation of the heavy metals in the glass stream (laser induced breakdown spectroscopy); determination of the flow characteristics exiting the melter (laser Doppler velocimetry); and analysis of the off-gas composition (Fourier transform infrared spectroscopy). To our knowledge such a concerted effort has not been reported previously. The results obtained describe some of the important characteristics of the glass and emissions produced from this particular waste stream surrogate and will allow for future comparisons with different surrogate streams and with different vitrification technologies.
The instrumentation systems were transported to the CU/ICVR facility using one of DIAL's Mobile Instrument Laboratories (MIL's). The MIL employed is a 18 wheel-type trailer which has been configured to allow instrument operation from a remote platform. In many cases, the instruments are entirely located within the MIL and fiber optics and signal cables are routed to the particular instrument/facility interface. Figure 1 shows the various instrumentation systems with respect to the Stir-Melter. Details of this melter have been reported previously (3).
Fig. 1.

In what follows, the results of the initial campaign on the Stir-Melter are presented. Each section begins with a brief description of the instrumentation employed followed by typical results. Full details of the field measurements at Clemson as well as those recently completed at the Western Energy Technology Office (WETO) at Mountain States Energy, Butte, Montana will be reported elsewhere.

PYROMETRY

Three types of pyrometer systems with different methods of data acquisition and data reduction were employed to measure temperatures of the molten glass within the melter (Fig. 1) and the glass pour. Measurement systems employed include a two-color pyrometer (TCP), a multi-wavelength pyrometer (MWP) and a ratio pyrometer (RP). Details of these instruments can be found elsewhere (4-6). Temperature ranges from the measurements at the two facility locations are collected in Table I.

TABLE I

The Two-Color Pyrometer was used to measure the molten glass temperatures inside the Stir-Melter through the facility camera port. Small temperature fluctuations were observed and are related to periodic current fluctuations in the Stir-Melter and to feed material floating through the instrument field-of-view; the current fluctuations occur because the Stir-Melter is operated in a constant power mode.

The Multi-Wavelength Pyrometer was used to measure molten glass temperature inside the Stir-Melter through the melter's camera port and the small diameter glass stream from the melter's pour exit. Two sets of data were recorded through the melter's camera port corresponding to different positions on the melt surface. At the first location the temperature was typically about 1250 K while at the second location temperatures were about 1185 K. Temperature fluctuations were significantly larger at the second location. The decreased temperature and fluctuations may indicate that the feed, which can stay on the melt surface before incorporation into the melt is affecting the temperatures. Temperatures obtained through the camera port are comparable to those measured by the TCP. Temperature fluctuations of the Stir-Melter glass pour observed by the RP and by the MCP are also evident in the MWP data.

The Ratio Pyrometer was used to monitor the molten glass stream temperature of the pour. As a result of the slow-moving slag stream (high viscosity glass, see below) and of the uneven temperature distribution across the slag stream surface, temperatures fluctuated between 1200 and 1380 K.

MULTICOLOR IMAGING

The DIAL/MSU multicolor imaging system (MCI) was used to take images of the glass stream emerging from the Stir-Melter. The imaging system consisted of a monochrome CCD camera with a 900nm interference filter (10 nm bandpass) that was used to provide an image of the thermal radiation from the melt stream. The television signals from the camera were digitized in real time and occasionally stored in a PC by the use of a frame grabbing board. The digital images allowed measurements to be taken of the size and shape changes of the stream and the corresponding intensities provided an indication of local temperature. Images of a temperature-controlled blackbody source were used to convert the intensities to temperatures. The spatial dimensions of the image have been calibrated by capturing the image of a millimeter scale placed at the same distance from the camera as the melt stream.

Figure 2 shows a series of images demonstrating the sub-millimeter spatial resolution obtained as well as the various shapes of the melt stream. The streams can be over four centimeters long, and range in width from 10 mm at the melter exit to less than 1 mm at distances approaching 4 cm from the melter. The viscosity of the molten glass has influence on the flow shapes, but extracting quantitative viscosity information will be difficult due to the temperature gradients that give rise to the variety of shapes seen. The range of intensities and hence temperature gradients of the glass streams can also be seen in these images. To highlight the temperature gradients, the actual intensities have been reduced to 12 false colors that represent temperatures from 760 to 1000C in 20 degree intervals as shown by the gray scale at the bottom of the figure. The two images on the left show the contrast between a high temperature gradient, high viscosity flow and the more desirable, steady, low-viscosity flow. Low viscosity flows can be routinely achieved by adjusting the glass forming chemistry. Pilot-scale experiments at the IVCR permits this optimization prior to processing the actual waste stream.

LASER INDUCED BREAKDOWN SPECTROSCOPY (LIBS)

LIBS is a laser-based diagnostic technique which is being used to measure the concentration of different elements in various environments. The experimental

arrangement of the LIBS system has been described elsewhere (7,8). In brief, a frequency-doubled Nd:YAG laser beam is focussed to produce a spark at the sample. The emission from the spark is coupled to an optical fiber, the other end of which is attached to a spectrograph with an intensified diode array detector. A controller is used for operation of the detector and to transfer the data to an acquisition/analysis notebook computer.

LIBS spectra of the molten glass pour of the Stir-Melter were recorded in the spectral range between 320 nm and 770 nm. By comparing the measured wavelengths and relative intensities with standard tabulations of atomic emission spectra, most of the observed emission lines have been identified (4). An attempt to calculate the concentration ratios of various species has also been made. The relative concentrations of the identified elements have been obtained by fitting the observed LIBS spectra with theoretical spectra and also by comparing the intensity of the spectral lines. Plasma temperature is an important parameter which affects the observed LIBS spectra. By assuming local thermodynamic eqilibrium and Boltzmann statistics shortly after the laser pulse, the plasma temperature was found to be about 7500 K. The concentration ratios have been computed by using the spectra recorded during the test run with the inferred plasma temperature. Twenty LIBS spectra in each observed wavelength region (about 40 nm) were analyzed to obtain the element concentration ratios. Relative mole fractions of various elements along with associated standard deviations are given in Table II. The inferred relative concentrations from the data on July 25 and 28 are very close except for the Fe/Mg ratios is not clear at this time since the feed is expected to be similar during both testing periods. The difference in the Fe/Mg ratio might be due to a slight change in the feed composition. The LIBS measurements on the Stir-Melter have illustrated that on-line concentrations of various elements in the molten glass can be determined.

TABLE II

LASER DOPPLER VELOCIMETRY

Laser Doppler Velocimetry (LDV) is a nonintrusive optical method for measuring velocity. Two laser beams are focused at a common crossing point in the flow, and light is scattered into the detector by particles that pass through the focal volume (or crossing point). The reenforcement and cancellation of the light from the two beams modulates the light intensity received by the detector, and the frequency of this ripple signal is directly related to the particle velocity.

The LDV system was installed in a vertical riser which carried the off-gas from the melter to the downstream treatment system. Optical access was obtained through a specially made test section of 2 inch pipe. There were two openings in the sides of this test section for optical access. The windows were kept clear of condensation and other fouling by introducing a small flow of nitrogen purge gas into the window cavity. Typically, we used a flow of 3 cfh for each window. This is small compared to the gas flow 100 cfh through the test section and should have a negligible effect on the flow outside the window cavity.

Since the flow is apparently unsteady, a number of data sets with the measurement volume fixed at the center line were collected to study the change in velocity with time. Figure 3 shows the variation in mean velocity over a period of one hour. An average velocity of about 3.8 m/s with a turbulence (axial direction) of 13% of this value. However, Figure 3 shows that the velocity fluctuates significantly. It is believed that the spikes in the mean velocity are due to an accumulation of the slurry which suddenly falls into the glass melt. The slurry then volatilizes creating an increase in the outflow velocity. The velocity profile across the pipe and further analysis of the data (4) will be reported elsewhere.

Fig. 3.

FOURIER TRANSFORM INFRARED (FTIR) SPECTROSCOPY

FTIR measurements were conducted by routing a portion of the gas from the melter to a multiple-pass (White) cell which was set to a path length of 75 m. The gas stream was initially conditioned by trapping particles on a glass wool filter pack and condensing the moisture from the nearly saturated effluent using an impinger cooled in an ice bath. The gas flowed continuously through the cell and typical pressures of 100-150 torr were used with the pressure controlled by valves on the vacuum line and the pumping system for the cell. All spectra were collected using an interferometer operating at a nominal spectral resolution of 0.5 cm-1. A dual-element HgCdTe/InSb detector and a CaF2 beamsplitter provided an effective

spectral range of 1000-5000 cm-1. Background spectra were taken after evacuating the cell to a pressure of between 2 and 8 torr.
Figure 4 contains two absorbance spectra with different arbitrary scaling factors. The top trace is for a sample of the melter effluent at 150 torr while the lower trace is a difference spectrum obtained by subtracting off transitions due to CO2 and H2O present in air (an air purge was employed for the video camera port on the melter). There is considerable H2O and CO2 in the original spectrum and subtraction of the water in producing the difference spectrum is not complete. Infrared signatures correspond to CO2, H2O, CO, NO, NO2, HCN, and a trace amount of methane. The feed for these experiments consisted primarily of inorganic nitrates, carbonates, acetates, hydroxides, and sodium phosphate along with phenol, dodecane, and yeast in a water slurry. The nitrogen molecules arise from the nitrates, and the CO and CH4 are thought to be incomplete pyrolysis products from the organics and/or acetates.
Fig. 4.
Some typical products that might be expected from the pyrolysis of phenol include benzene, toluene, and other small (< C 6) hydrocarbons. Examination of the CH stretching region around 3000 cm-1 (Fig. 4) does not reveal any broad absorption band features typical of these compounds. There are some unidentified transitions, particularly in the spectral region between about 1300 and 1700 cm-1 and some of these lines are thought to result from the incomplete subtraction of water. A further analysis of this region is in progress; however, it is safe to state that all of the remaining lines are consistent with small (diatomic or triatomic) molecules. The concentrations of the molecules in this spectral region can be estimated from an assumed absorption coefficient a of le-3 (10) using the Beer-Lambert law, a=(anRc), where a is the absorbance, l is the path length of the measurement and c is the concentration. Using an absorbance of 0.08 results in a concentration of 1 ppm. This value, when scaled by the pressures for the extractive experiment, would yield an effluent concentration of 5 ppm. These concentrations should only be considered as estimates. Further quantification of the results from this study will be reported elsewhere.
CONCLUSIONS
The results indicate that all of the initial test plan objectives were satisfied. Useful facility-dependent data such as velocities and melt temperatures were obtained. Specific characteristics of the glass and pyrolysis products from the surrogate feed were also determined. Some of the capabilities of the instrumentation have been demonstrated in the Stir-Melter measurements; however, there are a number of other potential applications to mixed-waste remediation processes that are currently under evaluatiion at DIAL. For example, the LIBS system is finding use in the on-line determination of entrained heavy metals in the gas stream, and high temperature infrared signatures are available from FTIR emission experiments. Valuable information on plasma characteristics is expected from multi-color imaging. The instrumentation described here and the other systems currently under development at DIAL are presently undergoing further evaluation/demonstration- Additional field measurements are in progress.
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THE ELEMENTAL ANALYSIS OF WASTE GLASS BY X-RAY FLUORESCENCE SPECTROMETRY*

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ABSTRACT

An X-ray fluorescence (XRF) technique is reported which shows promise for the elemental analysis of waste glasses. This technique can be used for both quantitative laboratory analysis and in process control. The glass-forming melt is cast into a graphite mold and the resulting disk is annealed and polished. The disk is then analyzed with a wavelength dispersive X-ray fluorescence spectrometer and the elemental intensities are converted into concentration with a fundamental parameters routine without the necessity of matrix-matched standards. The precision of the elemental determinations are all better than one percent relative standard deviation. The XRF analysis has been compared with a reference method utilizing conventional wet chemical dissolution techniques followed by atomic emission spectroscopy. There was no significant difference in the quantitative results between these two techniques.

INTRODUCTION

X-Ray Fluorescence (XRF) Spectrometry is a widely used technique for the routine quantitative elemental determination of ceramics (1) and ordinary glass (2), but it has received little application in the field of waste glass analysis. This is unfortunate since XRF is one of the most precise and potentially accurate analytical techniques available. There are several reasons why analysts in the vitrification arena have stayed away from XRF. High-level waste glasses produce large background count rates due to their inherent radioactivity. This, however, may not be such a problem with low-level, mixed waste glasses.

Another problem facing waste glass analysis by XRF has been the difficulty in quantifying the elemental composition from the spectral line intensities. The line intensities are not only a function of the analyte content but also depend on the presence of other species in the sample due to their matrix effects. These effects include absorption of the incoming and outgoing X-rays. Another matrix effect, enhancement, results when elements, with a higher atomic number than the analyte, are excited and their resulting secondary X-rays in turn cause additional fluorescence from the analyte. One approach, empirical calibration curves must be developed by first determining empirical coefficients relating one element's matrix effects on another. This is only accomplished with the use of many standards whose compositions closely match that of the samples. This becomes impractical with waste glass samples which have complex and widely variable compositions often containing more than 20 elements. Several studies on waste glass analysis have been reported using this tedious method of empirical coefficients with numerous standards. The XRF analysis of non-radioactive, surrogate high-level waste glass was reported in 1978

by Slates (3). This method was laborious and was limited to only a few of the many elements present. Sodium, iron, aluminum, and calcium could not be quantified and the procedure required the use of numerous matrix-matched standards. A recent XRF procedure for determining iron and cerium in waste slags has been reported by Carney (4) but the results were reported to be only semi-quantitative. This procedure also required the use of forty matrix-matched glass standards were prepared. In addition, glass samples were analyzed as powders thus reducing the accuracy and precision and increasing the sample preparation time. A truly useful analytical technique should provide the composition of all the major elements present because the major elements will affect the chemical durability of the final waste form.

An alternative approach for XRF quantification, called Fundamental Parameters (FP), uses mathematical equations with fundamental constants to convert the element intensities into composition (5). In this analysis, no standards are required but the accuracy depends upon knowledge of certain individual fundamental constants for each element such as absorption coefficients and fluorescence yields. The method was developed for XRF by Criss and Birks in 1968 (6) but it was of limited value due to the scarcity of accurate data on these fundamental constants. As measurements of these constants improved, so did the accuracy in the calculation of the elemental concentrations. This FP method is now in widespread usage for XRF analysis because of this improved accuracy and because it does not require standards.

Another reason that XRF has not been applied to high-level borosilicate waste glass analysis has been the difficulty in analyzing light elements such as lithium, boron, fluorine, and sodium. The fluorescent yields from these elements are diminished and their corresponding X-rays have greatly reduced penetrating power. As a result, the light element intensities are typically very low resulting in poor measurement precision. Developments in instrumentation and in the analyzing crystals have resulted in remarkable improvements in recent years in the analysis of sodium (7). Recent XRF measurements on borosilicate waste glasses showed that sodium line intensities could be determined with a high degree of precision but the boron intensity was still too low to be quantifiable (8). Boron, then, would have to be determined by difference. However, if the concentration of all the other species in the glass are accurately determined then boron could be determined by difference. This is the approach that will be employed in this study to determine boron.

In previous work in this laboratory, a rapid glass sampling technique was developed (8). It was reported that samples could be taken directly from the crucible or melter tank and prepared for analysis within 30 to 45 minutes. The intensities from replicate samples were measured and the precision was found to be better than one percent relative standard deviation. This paper reports the quantitative elemental analysis of waste glass samples using a wavelength dispersive XRF spectrometer and the FP method. These results are then compared to results from a reference analysis utilizing conventional wet chemical glass dissolution techniques and atomic spectroscopy.

EXPERIMENTAL

Glass Preparation

Seven waste glasses were prepared from platinum crucibles according to a procedure described elsewhere (9). Each melt was cast into graphite molds to produce three disks 40 mm in diameter by about 5 mm in height. The disks were then annealed at 450C for 30 minutes and then ground to prepare a flat surface using 1000 grit SiC paper on a Buehler polishing wheel. Next, the glass was carefully abraded manually using 600 grit paper, first in one direction, then orthogonal to the original direction. The disks all had a uniform 8-inch surface finish.

Glass Analysis

The reference analysis of these glasses was performed by wet chemical dissolution techniques followed by flame emission spectroscopy (FES) or inductively coupled plasma-emission spectroscopy (ICPES) by Corning Engineering Laboratory Services of Corning, New York. Three separate digestion techniques were employed to dissolve the glasses. Sodium was then determined by FES following digestion in hot hydrofluoric acid. Boron was determined by ICPES following digestion in cold hydrofluoric acid. The remainder of the elements were determined by ICPES following digestion with an alkali fusion technique. Instrument calibration was carried out with matrix-matched standards to compensate for matrix interference. Each determination was carried out in triplicate and averaged.

The XRF analysis of the disks was performed with a Rigaku Model 3271

wavelength-dispersive XRF spectrometer using 30 mm sample apertures and the instrument conditions listed in Table I. Samples were rotated at 30 revolutions per minute. An end window X-ray tube with a rhodium target was used to generate the X-rays. Among the analyzing crystals were three Ovonyx Multilayers for B, Na, and Si obtained from Osmic, Inc. of Troy, MI.

Conversion of the elemental intensities into concentration was carried out by an FP procedure developed by Rigaku. Element sensitivities were determined from Glass Sample #MMHO using the wet chemical analyses from Corning Laboratories.

RESULTS and DISCUSSION

The precision in the XRF analysis of these waste glasses was determined by comparing the measured elemental intensities from three replicate glass disks. The relative standard deviations (RSD) for each element from each of six glasses are shown in Table II. Boron intensity data were not included because it was too low to be quantifiable. The results show that the RSD is less than 1.0 percent for all measured elemental intensities. This level of precision for quantitative elemental determination is far better than is normally achievable with either ICPES or AA techniques (2). The precision reported here is representative of both the sample preparation technique and instrument data collection. Such a high degree of precision reduces the number of replicates necessary for an accurate quantitative analysis. In fact, this high degree of precision suggests that one sample might be sufficient for the elemental analysis of waste glass carried out in a process control environment, if the glass is sampled from a homogeneous melt.

The accuracy of the elemental analysis of these silicate waste glasses by XRF has been assessed by comparison with a reference method. The reference method employed here involved several wet chemical dissolution techniques, depending upon the analyte of interest, followed by one of several atomic spectroscopic techniques. The XRF analysis, on the other hand, involved the measurement of X-ray intensities from each element with a sequential wavelength dispersive spectrometer. The element intensities were converted into weight percent composition with a fundamental parameters routine utilizing a single glass standard (MMHO) to determine the element sensitivities. The composition of the standard was also determined by the wet chemical procedure.

The results of the elemental analyses from XRF are compared to the analyses from the reference method in Table III. Boron could not be analyzed directly due to the low intensities, therefore, its composition was calculated by difference. The results for B_2O_3, Al_2O_3, SiO_2, CaO, and Fe_2O_3 all agree within 0.65 percent of the reference values. Furthermore, Na_2O, NiO, BaO, and PbO all agree within 0.10 percent. It is clear that the elemental analyses of these wastes glasses by XRF are in excellent agreement with the results from the more conventional wet chemical method.

The time required to perform the XRF analysis was considerably shorter than that for the wet chemical reference analysis. The analysis of each sample required only 10 minutes while the reference wet chemical methods required many hours of lab work. Instrument calibration which involved the determination of elemental sensitivities using sample MMHO also took only 10 minutes, as well. The reference wet chemical method, on the other hand, required that each sample first be milled to a powder and then subjected to three separate digestion procedures. Each of the three resulting digestates was then analyzed by either ICPES or FES. Furthermore, separate calibration curves were developed for each digestate from each glass from a set of matrix-matched standard solutions. This wet chemical analysis was both labor and time intensive.

Potential limitations of this XRF technique are discussed briefly here. Actual waste glass samples will likely differ from these surrogate glasses in several ways. Waste glasses may likely contain more than the nine oxides in these test species, and concentrations of some species will be below one percent. The analysis of more complex matrices including trace elements should still be achievable with XRF but further work is required to demonstrate this capability. Furthermore, mixed waste glasses will contain one or more radioisotopes which may affect background count rates. An example of a low-level mixed waste scheduled for vitrification is the wastewater treatment sludge currently in storage in M-area at the Savannah River Plant in Aiken, SC (10). The resulting waste glass will include about 20 oxides ranging from boron through uranium.

This XRF technique should be capable of quantifying all elements from sodium through uranium. Elements lighter than sodium present special problems due to their

inherently low count rates. Further testing is required before the precision and accuracy of fluorine analysis can be established. Only certain specialized wastes, however, are likely to contain fluorine. Oxygen does not need to be calculated if one assumes all cations to be present as the oxide and the oxidation states are known. For determination of elements lighter than oxygen, direct analysis appears unlikely and quantification must be carried out indirectly by difference. This study has demonstrated that boron can be quantified by difference if it is the only light element present. If the glass contains lithium as well as boron, as is the case with certain high-level waste glasses, then this method would not be capable of distinguishing between the two.

Radioisotopes, present in low-level mixed wastes, could potentially cause spectral interference due the intrinsic radiation from the glass itself. Defense wastes can contain a variety of radioisotopes including 90Sr, 99Tc, 137Cs, 238U, and 242Pu. Further testing needs to be carried out in order to determine the maximum tolerable radioactive levels in glass without adversely affecting the analysis.

XRF could play an important role in vitrification quality control. It is known that chemical durability is largely a function of glass composition. Composition limits for the glass product would be predetermined and then compared to the actual product composition to ensure product durability. In all likelihood, trace oxides would not play a role in product acceptability because of the minimal effect that they would have on chemical durability.

The actual chemical composition of the glass product could be determined periodically by XRF and then compared to the predetermined limits with the aid of statistical process control charts (11). The key steps in the elemental determination will include the following:

1) rapid, representative glass sampling;
2) precise measurement of X-ray intensities; and
3) accurate determination of glass composition using "standardless" FP.

The glass product could be sampled directly as it flows from the melter by collection in a graphite mold. In this way the sample would be ready for analysis within 30 to 45 minutes. Even though the glass disks were polished in these experiments, previous studies on waste glass have shown that precision does not greatly suffer when disks are analyzed without prior polishing (8). The quantitative chemical analysis could then be available within one hour from the time the glass flows from the melter.

CONCLUSION

This study demonstrates that rapid, precise, and accurate XRF analysis can be obtained for waste glasses containing Al_2O_3, B_2O_3, SiO_2, Fe_2O_3, Na_2O, CaO, NiO, PbO, and BaO. Replicate sample precision is better than one percent relative standard deviation for all elements except boron which cannot be determined directly. The accuracy of this XRF procedure is comparable to the wet chemical methods traditionally employed. Boron, which must be determined by difference, has been shown to be within one percent, absolute, of the reference method.

This technique can potentially be used for both laboratory quantitative chemical analysis of waste glass or as a tool in vitrification process control by providing elemental analysis of the glass shortly after it is sampled from melter.

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37-23

APPLICATIONS OF ATOMISTIC SIMULATION TO RADIOACTIVE AND HAZARDOUS WASTE GLASS FORMULATION DEVELOPMENT

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ABSTRACT

Glass formulation development depends on an understanding of the effects of glass composition on its processibility and product quality. Such compositional effects on properties in turn depend on the microscopic structure of the glass. Historically, compositional effects on macroscopic properties have been explored empirically, e.g., by measuring viscosity at various glass compositions. The relationship of composition to structure has been studied by microstructural experimental methods (nuclear magnetic resonance, x-ray spectroscopy, etc.). More recently, computer simulation has proved a fruitful complement to these more traditional methods of study. By simulating atomic interaction over a period of time using the molecular dynamics method, a direct picture of the glass structure and dynamics is obtained which can verify existing concepts as well as permit "measurement" of quantities inaccessible to experiment.

Atomistic simulation can be of particular benefit in the development of waste glasses. As vitrification is being considered for an increasing variety of waste streams, process and product models are needed to formulate compositions for an extremely wide variety of elemental species and compositional ranges. The demand for process and product models which can predict over such a diverse composition space requires mechanistic understanding of glass behavior; atomistic simulation is ideally suited for providing this understanding. Moreover, while simulation cannot completely eliminate the need for treatability studies, it can play a role in minimizing the experimentation on (and therefore contact handling of) such materials.

This paper briefly reviews the molecular dynamics method, which is the primary atomistic simulation tool for studying glass structure. We then summarize the current state of glass simulation, emphasizing areas of importance for waste glass process/product modeling.

At SRS, glass process and product models have been formulated in terms of glass structural concepts (1). These models are now being extended for application to both low-level mixed waste glass formulation (2) and to high-purity actinide stream immobilization (3). Atomistic simulation is providing input for the development of the process and product models for these new applications. Recent results are described in this paper.

INTRODUCTION

"Atomistic simulation" refers to a variety of computational techniques for determining structural and dynamic properties of materials. The most powerful of these, through its ability to provide dynamic as well as structural data, is the molecular dynamics method. Through this calculational process, the positions and

velocities of all particles are calculated for a succession of timesteps, thus defining the atomic structure and dynamics of the material. Through graphical representation of the calculational results, the glass network can be viewed "directly" as well as being analyzed statistically.

The visualization capabilities of the method alone provide valuable insight into the behavior of glass; the human eye is still the best computational tool for "discerning the presence of collective motions" (4). However, the wealth of properties which can be calculated from simulation can be correlated to macroscopic behavior as well, providing direct input to models suitable for process control.

MOLECULAR DYNAMICS SIMULATION METHODS

In the molecular dynamics simulation technique, Newton's equations of motion are solved for each of a collection of particles (atoms, ions, or molecules) as it interacts with all others. The numerical solution of Newton's equations is straightforward; thus, the key to solving this problem is the specification of the forces between the particles or, equivalently, the interatomic potential function between the particles. Time averages of various quantities and their variances can be related to macroscopic thermodynamic quantities such as pressure, heat capacity, enthalpy, etc., through well-known statistical mechanical developments (5).

Simulations are most commonly done holding fixed the volume of the simulation sample, the number of atoms, and the total energy (the "microcanonical ensemble" of statistical mechanics). Periodic boundary conditions are used to hold the number of atoms fixed and to eliminate surface effects in simulation of bulk materials. Computer technology limits the size of the simulated sample and the simulation time interval; most current published work is based on no more than about 1000 atoms, simulated over periods of the order of nanoseconds.

Many quantities of interest in glass processing, such as viscosity and chemical durability, have been modeled as functions of atomic phenomena such as bond length, bond strength, non-bridging oxygen content, etc. Atomic simulation provides a means of "observing" such phenomena directly, thus providing input to models couched in such terms. The positions of the particles at any given time can also be represented graphically (See Fig. 1), thus providing a powerful aid to conceptualizing the structure of a glass.

Fig. 1.

The veracity of the simulation must be checked by comparison to microstructural experimental data. Once this is done, simulation can be used to obtain additional quantities which can be obtained by experiment (sometimes with far more difficulty), as well as quantities for which no experimental technique has been developed. An example of the latter is the pair distribution functions, i.e., the radial distribution of atoms of type j around an atom of type i. These functions describe the environment of an atom in terms of each of the other types of atoms in the system. Experimentally, only the total radial distribution function can be obtained, i.e., the sum of all the individual pair distributions. The simulation results therefore show how to correctly deconvolute the experimental quantity to reveal the details of the local environments of each atom type.

Other quantities which can be calculated from the simulation results include:

Sizes, energies, and distribution of "holes" in the glass network. These quantities are related to macroscopic quantities such as electrical conductivity, gas solubility, and activation energies. Significant progress has been made on the well-known mixed-alkali problem in glass science using atomistic simulation (6).

Non-bridging oxygen (NBO) characterization. This includes the fraction of oxygens which are non-bridging, and the Qn distribution, where Qn denotes the fraction of silicon atoms (in a silicate glass) having n bridging oxygen atoms (i.e., Q4 denotes the fraction of completely-polymerized silicon tetrahedra, Q0 the fraction which are completely isolated from the network). These quantities have been related to macroscopic parameters such as viscosity (1).

Ring statistics. These characterize medium-range order in the glass (7). The presence of rings larger or smaller than the optimum for the network (e.g., six-membered rings for pure silica crystal) represents "defects" in the structure which have been used to characterize viscosity (8) and diffusion (9).

ATOMISTIC SIMULATION OF GLASS

At present, most of the simulation work which as been performed is on bulk glasses of one or a few components. The primary attention has been applied to silicate glasses, although studies of phosphate glasses (10) and borates (11) have also been

performed. Numerous studies have been made of vitreous silica (12-14), whose properties have been well-characterized experimentally and whose behavior, which is in several respects atypical of oxides, provides a rather stringent test of interatomic potential functions. Thus, the focus of many of these studies is on developing and verifying a proposed interatomic potential function. Potential functions which model silica gas structure well provide a good basis for extension to multicomponent silicate glass systems.

Oxide structure is uniquely determined by its mixed covalent-ionic character. Locally, the bonds are directional, a reflection of the covalent nature of the constituents. In this way the oxides are similar to very covalent materials like silicon. On the other hand, the bonding is also ionic, leading to long-range electrostatic interactions which are similar to the case of materials like NaCl. These aspects are interdependent. The available interatomic potentials which appear most suitable for glasses generally treat only the long-range, electrostatic interactions between pairs of atoms. A typical functional form for a two-body potential is the Born-Meyer-Huggins, i.e.,

Eq. (1)

where the first term represents the Coulomb potential and the second term represents the repulsion potential. The success of such ionic potentials in representing the major features of the silicate glass network is remarkable. Recently, improvement in results has been obtained by using interatomic functions which attempt to include the covalency via the introduction of explicit 3-body terms (10,13). One of a number of such potential forms is that due to Garofalini (13):

Eq. (2)

where the Coulomb potential term of equation (1) is modified to account for long-range Coulomb forces, and

Eq. (3)

if rij <ric and rik <ric, or equals zero otherwise. Here rij is the distance between atoms i and j; qjik is the angle between atoms i, j, and k whose vertex is at atom i; and qjikc denotes the unconstrained, minimum energy value of the angle. Other symbols denote constants.

Boron-containing glasses have proved challenging to model. The conversion between trigonal and tetrahedral boron-oxygen units, and the reproduction of the boroxyl unit, are stringent tests of interatomic potentials for borate and borosilicate glasses. At present, only one potential model has successfully reproduced the boroxyl grouping (11). This is important to waste glass modeling, since many waste glasses are of the borosilicate family.

Interatomic potentials for heavy elements have received little attention in the literature. There is currently a need to develop good potentials for both transition metals and actinides, in order to model waste glasses which typically contain such species.

Despite enormous advances in computing technology, simulations are of very small (in macroscopic terms) samples of material and over very short spans of time. Since numerical solution requires timesteps of the order of picoseconds, even a long simulation mimics only a small interval of "real" time. For glasses, the simulation is typically started from an arbitrary arrangement of atoms (often a crystal structure, if the composition being simulated has a crystalline form), is then randomized by simulation at high temperatures (e.g., 6000K) and is then "quenched" by a series of temperature decreases interspersed with equilibration periods. The short duration of the simulation thus imposes extremely high quench rates which are realized in practice only in a few physical processes, such as splat cooling of metals. These aphysical quench rates have some influence on the resulting structure, although the major features of the structure are accurately represented.

Spatial limitations of simulation impact the study of phase separation and of multicomponent systems. The characteristic wavelength for phase separation is of the order of 30-100 nanometers (15), while typical simulation sizes are considerably less than this. Phase separation has been studied, nonetheless, by exploiting the ability to calculate thermodynamic quantities from simulation results (15). Inhomogeneity in composition has been observed on the scale of the simulations themselves (16), which seems to imply that phase separation (or some precursor) could be directly observable via simulation.

The impact of size limitations on the study of multicomponent systems is one of statistics. This is, in simulating a system in which an element represents only a

trace quantity, a sufficient number of atoms of that element are required to be present, in order to obtain a statistically sound representation of the behavior of that element. For example, if 20 atoms of a given type are needed to achieve good statistics for a given property, all components must be present in quantities of at least two mole percent, for a 1000 atom simulation.

APPLICATION TO WASTE GLASS MODELING

Atomistic simulation has been used to simulate many phenomena of interest in waste glass modeling. Glass homogeneity is key in assessing the durability of waste glass (1); phase separation studies have already been noted above (15). Another key issue in waste glass durability is that of water attack on the glass surface. Water-silica interface simulations by Garofalini (17) represent an important step in this area.

Atomistic simulation is currently being used as part of the developmental extension of SRS product and process models, originally formulated for high-level waste glass. In-house simulation software, originally developed for studying gas-metal interactions (18), has been adapted for the study of glass. A unique feature of this software is that simulations can be conducted in a variety of statistical mechanical ensembles; alternate ensembles provide a means of calculating, or calculating more expediently, many thermodynamic quantities not easily computed in the standard microcanonical ensemble (i.e., where the simulation volume, number of atoms, and system energy are all conserved). Currently, two- and three-body interatomic potential functions are being used, although efforts have begun to develop alternate potential functions which should <allow a more accurate representation of multicomponent and heavy element-containing systems.

One of the SRS process models is for melt viscosity, which is characterized as a function of temperature and non-bridging oxygen content (1):

Eq. (4)

NBO, the number of non-bridging oxygens per silicon, is computed as

Eq. (5)

where M' is an alkaline element and M is an alkaline earth element. This reflects the conventional glass science view that each alkali cation produces one non-bridging oxygen, while the divalent alkaline earth cations each produce two such non-bridging oxygens. It was found (2) that the fit to experimental data was much improved by reducing the assumed relationship between alkaline earth cations and NBO, i.e.,

Eq. (6)

Aj = 1.2 for [MO]j] 0.45

Atomistic simulation (19) of a series of calcium silicates (xCaO(1-x)SiO2 with x = 0, 0.1, 0.2, 0.3, 0.4, and 0.5) was used to verify the NBO relationship for alkaline earth cations. The simulations confirmed a slight reduction of the NBO content below the theoretical value of two per Ca2+ cation, but the reduction was too small to be consistent with the Aj values of Eq. 6). These results were checked by simulating corresponding sodium silicate systems, where the expected NBO relationship obtained. From comparison of the simulation results of the xCaO(1-x)SiO2 series with those of the sodium silicate glass, we infer that the increase in viscosity effect of alkaline earth versus alkali (which is implied by the reduction in the NBO factor shown in equation (6) is due to the higher degree of order that the alkaline earth cation imposes on the relatively disordered glass network. This is shown in Fig. 2, where sodium silicate and calcium silicate compositions having equal proportions of cations have been compared. This figure shows the number of oxygen neighbors seen by the Na or Ca cation. For both cases, the sodium ions have a much broader distribution of oxygen neighbors. That is, although there may be a slight preference for one or two configurations (as indicated by the locations of the peaks), the sodium ions occupy a variety of local environments in the glass. In contrast, the calcium cations show a more sharply peaked distribution, implying that the calcium ion has a more definite "reference" for its environment. It therefore imposes more structure on the network, resists changes to that structure, and hence increases the viscosity.

Fig. 2.

These results suggest that an additional term is needed in equation (4) to represent both the non-bridging oxygen content and the strength of the cation's ability to order its environment. Future simulation work will focus on characterizing this phenomenon in terms of a simple parameter which can be incorporated into the viscosity model. Possible choices of this parameter are the width of the neighbor

distribution, or the energy required to alter the local cation environment. These are quantities which are directly obtainable from the simulation results.

CONCLUSIONS

The molecular dynamics method of simulating the behavior of atoms can provide insight into the structure of materials. A review of the recent literature shows that important waste glass modeling issues are currently being investigated by these means. Simulation is assisting glass process/product model development by SRS for novel applications such as mixed and low-level waste glass and actinide waste glass formulation. Simulation studies of sodium and calcium silicates show that the alkaline earth cation increases the local ordering of the glass network, thus increasing the structural rigidity of the glass. Future work will focus on characterizing this difference in terms of a simple parameter which can be incorporated into an improved process model for glass melt viscosity.

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37-24
AN INNOVATIVE, REAL-TIME MONITOR FOR
AIRBORNE ALPHA EMISSIONS
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ABSTRACT
The public is demanding real-time, high-sensitivity, continuous monitoring of any facility or operation that can potentially release radionuclides to the environment. In response to this demand, Los Alamos National Laboratory (LANL) is developing a new technology for on-line, real-time monitoring of off-gas stacks for low levels of airborne alpha activity. In this paper we briefly describe the requirements for such a system, our approach to meeting these requirements, and the present status of the development and fielding of this technology.
INTRODUCTION
Across the DOE complex, monitoring airborne emissions is one of the most pressing of the public's concerns. In particular monitoring for potential radioactive emissions is an issue that affects nearly every operation from handling radioactive material to site clean-up. As the public becomes more involved, limitations in current monitoring technologies are being identified and targeted for technology development. While the technology exists for long-term monitoring to ensure total emission compliance, most systems are poorly suited to monitoring at low levels on a real-time, fast response basis. Specifically, there currently exists no commercial system to monitor for alpha emitting radionuclides that works in real-time (i.e. seconds) on an entire off-gas stack or air effluent stream.
At Los Alamos we are developing a new detector technology that can monitor and measure the quantity of alpha emitting nuclides that may be present in a large volume of moving air. This detector is unique in that it can measure on-line, in real-time, a complete effluent stream without having to use sampling techniques or long measuring times. This technology provides a robust and fault-tolerant real-time safety monitor (i.e. a "smoke alarm") that can signal the need to shutdown a malfunctioning operation before the public is exposed to significant amounts of radioactive material.
Background
Our development of this monitoring system is an out growth of our long history of innovation in radiation detector technology. Several years ago, members of the Rocky Flats Fluidized Bed Incinerator Project approached our group about solving their need for advanced monitoring technologies. More specifically they were concerned about the lack of a real-time alarm for low-levels of radionuclides potentially present in the off-gas stream.
An extensive survey of available commercial products revealed that, with some minor variations, nearly all commercial alpha monitoring techniques are based on a common approach. This approach consists of extracting a sample of the gas being monitored and passing it through a filter placed close to a detector sensitive to alpha decay radiation. Particulates bearing alpha emitting radionuclides are entrained on the filter and subsequent alpha decays are measured by the detection system. This approach, while useful in many applications has severe limitations in real-time applications. Primarily, this limitation arises because these detectors extract a small sample which is measured for a relatively long period of time. Additionally, the small volume of gas measured is assumed to be "representative" of the remainder of the stack. Sampling a relatively small fraction of the total flow in the stack limits the useful sensitivity of the instrument.
With these limitations in mind, we took a "clean sheet of paper" approach to determine what else could be done. Based on concepts being developed for a different application, we proposed the development of a new monitoring technology with a large measurement volume, and thus high sensitivity, that can be inserted directly into the effluent stream
APPROACH
This new technology was originally called the Large Volume Flow Through Detector System (LVFTDS). Now called the Flow-Through Alpha Monitor (FTAM), this approach is designed to quantitatively detect, in real-time, low concentrations (in the range of picoCuries/liter) of alpha-emitting radioactive materials potentially present in an off gas stream. The key obstacle to overcome in making this type of measurement is the short range of the alpha particles. For typical decay energies alpha particles travel only a few centimeters in air, making their detection difficult. Our detector

overcomes this difficulty by using multiple, alpha-sensitive, scintillating panels spaced closely together, but covering a large volume. Figure 1 illustrates this approach. With this arrangement approximately 75% of the alpha particles from radioactive decay in the detector active volume can reach a panel and generate a detectable light pulse. The light pulse is transmitted to a set of photomultiplier tubes by optical light guides.

A key feature of the FTAM detector is that the radioactive materials measured are not filtered, sampled, or otherwise removed from the primary gas stream for detection. The radioactive materials are detected directly when they emit an alpha particle. For most flow conditions, the entire gas stream, or a substantial fraction of the stream, can be monitored directly with the FTAM. The large active volume of the detector permits direct detection with high sensitivity without using concentration techniques (such as filtering) that require substantial collection times and thus limit the real-time capability of the system. For short response times of a few seconds our detector has a sensitivity more than an order of magnitude higher than commercially available systems. Filtering techniques also lead to a reduction in sensitivity over time because of the accumulation of radioactive materials on the filter. With its unique, real-time capability the FTAM system can respond essentially instantly in the event of airborne contamination, at levels that remain constant over time.

Fig. 1. The flow-through alpha monitor approach.

FTAM ADVANTAGES

The FTAM technology offers a number of advantages. First, as discussed earlier, the entire volume of gas flows through the detector, eliminating the need for sampling trains. Second, the detector's large active volume may permit as much as an order of magnitude improvement in sensitivity over current technology at the short (less than one minute) integration times needed for real-time monitoring and control. Third, without a filter, the maintenance requirements are greatly reduced. The detector sensitive elements are easily cleaned, and with a modular design can be replaced easily should they be damaged. Additionally, the inherent redundancy of the multi-plate design offers built-in fail safe operation. Since the detector is intended to provide an alarm in the event of an emergency, the FTAM is designed to respond over a very wide dynamic range.

Although designed for monitoring alpha radiation, experiments are being conducted to extend the detector into beta and gamma monitoring applications. While the ability of the detector to distinguish between alpha and some beta emitters may be poor, the detector should be able to provide the alarming feature on both alpha and beta radiation simultaneously. This "total radiation monitoring" feature is an attractive possibility, providing system operators comprehensive radiation alarming in a single device.

DEVELOPMENT PROGRAM

Two versions of the detector have been built and tested to date, and a third is being designed at the time of this writing. The first version was a laboratory scale prototype unit designed to allow us to test the detector concept, using a radioactive gas, on a scale large enough to be indicative of the design issues, but small enough to be easily fabricated, assembled, and modified. This version used a stack of five scintillation plates, with a 2 cm inter-plate spacing, giving a total detector volume of 10,230 cm3 (~0.4 ft3). A number of closed loop tests were conducted in which radon (Rn^{220}) from a thorium source was introduced into the detector box for a short time and then stopped. The radon concentration during these tests was typically less than 4 nCi/liter. We continued to circulate the air in the box while the contained radon decayed, allowing us to confirm the radon half-life and the detector's long-term stability. During these tests we typically set the data acquisition system to integrate the detector count rate over 5 second intervals.

From this series of tests we reached three important conclusions. First, we demonstrated that the overall detector concept was sound and workable. Second, we were able to measure the detector efficiency to be around 73%, very close to our expectations. Finally, we found that the background in the detector was higher than we had hoped for based on the early work we had done.

The second version of the detector took the development to an intermediate step, beyond the lab-scale prototype, but not as large as the unit to be field tested. The development and testing of this detector allowed us to concentrate on improving the detection sensitivity by both increasing the size of the detector and decreasing the

background. Engineering issues such as the airflow were also addressed. Table I shows a comparison between the prototype, second generation, and field test systems. The most significant change between the prototype and second generation systems is the reduction in the detector background of around a factor of 37. This achievement is the result of a combination of passive and active shielding. The passive shielding consists of a layer of steel and a layer of lead, which together reduce the background by around a factor of 3. The balance of the reduction comes from the active shielding. This technique uses the detector elements themselves to provide veto signals whenever cosmic rays or other highly penetrating external radiation passes through the detector. A data processing algorithm processes signals from each detector plate, looking for plate-to-plate coincidences. Knowing that an alpha particle can only excite a single plate allows us to veto background events exciting more than one plate simultaneously. This process is aided by the use of guard plates on the top and bottom of the plate stack. These thicker plates are optimized to ensure nearly 100% detection efficiency for the cosmic rays that can contribute to the alpha detection plate background.

The increase in sensitivity brought on by the background reduction and the increase in size has allowed us to test the second generation detector in several interesting ways. Rather than testing the detector solely on radon recirculated in a closed loop around the detector we configured the system to monitor ambient levels of naturally occurring radon in room air pulled through the detector in a single pass. In a second set of tests we were able to detect short bursts of radon (Rn220) injected into the single pass airstream just ahead of the detector. Again in these tests we integrated the detector signals for short (i.e. 5 second) intervals and demonstrated real-time response to airborne alpha activity in the 10s of picoCuries per liter range.

FIELD DEMONSTRATION PLANS

We have selected the Incineration Research Facility in Arkansas as the most suitable site for the initial field tests. Testing is planned for June 1995, and will consist of three phases. The functional test phase, the first of the three, will verify the detector background and alpha detection performance under nominal incineration off-gas conditions. The second phase will consist of a series of challenges such as temperature and humidity excursions. During this phase we will most likely include one or more plates designed to fail at certain temperatures to test the fail-safe operability. The final phase will consist of long-term exposure of the detector to the off-gas environment for life-cycle testing. At the close of FY 95 we plan to have successfully field tested and documented the system performance in anticipation of installation at a mixed waste treatment facility.

BEYOND INCINERATORS

While monitoring incinerator off-gasses is certainly a pressing need, the base technology of FTAM has a wide range of exciting applications. Examples of applications within the DOE complex include monitoring of gas/ventilation systems such as those on high-level radioactive waste storage tanks or other storage areas, and the monitoring of ventilation systems in buildings with combined labs and offices. Monitoring during site remediation activities is an application of particular interest. Only the FTAM technology offers the ability to monitor in real time the very large (hundreds to thousands of cubic feet per minute) volumes of air needed to do credible monitoring during large scale site remediation or decommissioning.

Commercial applications include some segments of the radon monitoring market, including large facilities or a trailer mounted mobile monitoring service. Another market which falls under this general category is the monitoring of mines and mine shafts. Given the dynamic nature of mining operations a single sampling type monitoring system is insufficient. The FTAM would be used in the mine ventilation system, thus monitoring much more effectively the worker's exposure. As a real time monitor, this detector could be very effective in shutting down dangerous operations (ones that result in a high airborne radon exposure) quickly.

Although poised for field testing within the year, a series of improvements and enhancements have been identified. These include a dramatic cost and complexity reduction, extension into higher temperature and harsher environments, and simultaneous alpha, beta, and gamma monitoring.

SUMMARY

The development of this new technology is being driven by the need for improved

on-line monitoring technology for alarming in real-time at low levels. This technology directly addresses the public concern over limitations in present monitoring capabilities. It must be emphasized that this new technology is intended to compliment, rather than entirely replace, current monitoring techniques. The sampling and long integration times of conventional detectors are necessary for the ultimate sensitivities required for regulatory compliance. The fast response, large active volume, and complete stack gas measurement of the FTAM technology is necessary to adequately provide fast real-time alarming.
With the promise of providing the incinerator site a comprehensive, fast responding, on-line alarm , the FTAM technology will soon establish the state of the art of incinerator monitoring for radioactive materials.
We wish to acknowledge and thank Paul Hart and Nina French of the DOE Mixed Waste Integrated Program, and Paul Williams of the Rocky Flats Fluidized Bed Incinerator project for their support of the development of this technology.

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MERCURY REMOVAL FROM LIQUID AND SOLID MIXED WASTE

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ABSTRACT
Large volumes of mercury-contaminated aqueous and solid mixed wastes are stored on Department of Energy (DOE) sites. These mixed wastes contain sufficient radioactivity and mercury that they need to be handled both under the Atomic Energy Act and the Resource Conservation and Recovery Act. Removal of volatile mercury from the aqueous waste is desired before potential water evaporation and thermal treatments are implemented. Vitrification and plasma hearth technologies are being tested at DOE facilities as thermal treatment methods for solid mixed waste. Due to mercury's volatility, treatment methods such as mercury leaching must be developed as pretreatment methods for mercury-contaminated solid mixed wastes.
Laboratory work has been conducted to investigate the feasibility of using inexpensive sulfur-impregnated activated carbon beads for mercury removal from aqueous waste. Laboratory studies have shown that mercury can be removed to levels below the current Environmental Protection Agency toxic characteristic level (0.2 mg/L). When comparing the use of inexpensive carbon beads with a more expensive ion exchanger, carbon beads had favorable process economics.
Clean, synthetic soil material was artificially contaminated with 20,000 mg/kg total mercury (Hg) prior to treatment in laboratory bench-scale experiments. The resulting surrogate waste was treated with KI/I2 solutions at concentrations in the range of 0.33 M KI/0.033 M I2 to 2.5 M KI /0.25 M I2 at temperatures of 25, 40, and 55C. It was observed that KI/I2 leaching solutions could effectively reduce the mercury concentration in soil by as much as 99.8%. It was also demonstrated that the mercury removal efficiency achieved with KI/I2 leaching was independent of mercury speciation or initial mercury concentration for the conditions studied.
A group of laboratory experiments was conducted with crushed fluorescent bulbs doped with small amounts of elemental mercury to simulate contaminated waste glass. The following leaching solutions were used to treat the prepared glass: NaOCl, KI/I2, HCl, HNO3, NaBr, NaBr + acid to pH 2, NaCl, and NaCl + acid to pH 2. The three most effective agents for leaching mercury from the crushed fluorescent bulbs were KI/I2, NaOCl, and NaBr + acid, which removed 98.8, 98.1, and 57.5% of the dosed mercury respectively.

INTRODUCTION
Large volumes of mercury-contaminated aqueous and solid mixed wastes are stored on Department of Energy (DOE) sites. These mixed wastes contain sufficient radioactivity and mercury that they need to be handled both under the Atomic Energy

Act and the Resource Conservation and Recovery Act. Removal of volatile mercury from the aqueous waste is desired before potential water evaporation and thermal treatments are implemented. Vitrification and plasma hearth technologies are being tested at DOE facilities as thermal treatment methods for solid mixed waste. Due to mercury's volatility, treatment methods such as mercury leaching must be developed as pretreatment methods for mercury-contaminated solid mixed wastes.

This goal of this project was to develop treatment processes for the removal of mercury from mixed waste. Processes that are applicable to aqueous waste include sulfur-impregnated activated carbon, ion exchange resins, and ion exchange membranes. To focus development activities on realistic process variables, target wastes were selected from the U.S. Department of Energy (DOE) mixed waste inventory. Inventoried wastes were evaluated based on seven criteria ranging from waste characteristics to regulatory drivers. The aqueous target waste selected was the Idaho Chemical Processing Plant (ICPP) sodium-bearing acid waste. The solid target wastes selected consisted of soils/sediments stored at the Oak Ridge Y-12 Site and crushed fluorescent tubes/lamps found at all DOE sites.

MERCURY REMOVAL FROM LIQUID MIXED WASTE

The specific objective of the laboratory studies was to investigate the ability of sulfur-impregnated activated carbon to remove mercury from acid solutions. Effects of pH, sorbent-to-liquid ratio, mass transfer versus kinetic control, competing ions, and mercury bead loading on the adsorption were examined. In addition, the sorbent capacity was determined in batch tests. After completion of batch experiments, column studies were performed to investigate design parameters.

Methods

Two types of surrogate wastes were used in this study. The first was an aqueous solution of mercuric nitrate [Hg(NO3)2] prepared from either 1,000 or 10,000 mg/L National Institute of Standards and Technology standards diluted with deionized water. In some cases, crystalline Hg(NO3)2 was directly dissolved in deionized water. The acidity of the surrogate waste was adjusted with nitric acid (HNO3) or sodium hydroxide (NaOH) to achieve the desired pH. The second surrogate waste, which simulated the sodium-bearing liquid waste, was prepared based on a formulation obtained for the ICPP waste. The average acidity (from HNO3) of the ICPP waste was 1.4 mol/L (M), and mercury levels were in the range of 100 to 860 mg/L (as Hg2+) with an average value of 476 mg/L. Other metal contaminants were present in the following average concentrations (in mg/L): aluminum, 15,000; cadmium, 184; iron, 1,475; lead, 213; manganese, 744; and nickel, 140.

Sulfur-impregnated granular activated carbon was obtained from Nucon International, Inc. (Columbus, Ohio). The Mersorb pellets had a diameter of 3 mm and a length of 3 to 8 mm. The data sheet obtained from the pellet manufacturer listed the sulfur content to be 13% and the mercury capacity to be 85% of the theoretical value. This translates to a "real" capacity of 0.71 g mercury/g beads.

Jar mill studies were conducted using a US Stoneware (Mahwah, New Jersey) variable-speed jar mill using 1-L Pyrex glass bottles as jars. Column studies were performed in Kontes (Vineland, New Jersey) liquid chromatography columns with a diameter of 48 mm, and the length could be varied by adjustable plunger-type end caps. The feed was supplied to the bottom of the column with a variable speed/flow Masterflex pump (Cole Parmer Instrument Co., Niles, Illinois). Except for pump tubing, all surfaces in contact with the liquid were glass or Teflon.

Mercury concentration in the aqueous solutions was measured using an 1100B atomic absorption spectrometer (Perkin-Elmer, Norwalk, Connecticut) equipped with a cold vapor injection system (Model FIAS 400, Perkin-Elmer) and automatic sampler (Model AS 90, Perkin-Elmer). Argon was used as the gas carrier and 0.36 M hydrochloric acid (HCl) was used as liquid carrier. The reducing agent was 0.30 M stannous chloride (SnCl2) in 0.36 M HCl. Typically, 4 mL of the aqueous sample was filtered, and 1 mL of the filtrate was diluted with 4 mL preservative solution [3.4 mM potassium dichromate (K2Cr2O7) in 0.80 M HNO3]. Serial dilutions of the preserved sample were made with preservative solution to a final mercury concentration in the range of 1 to 20 g/L. Standards were prepared by serial dilution with preservative solution from a 1000-mg/L primary standard to final concentrations of 1, 10, and 20 g/L.

Results and Discussion

Initial batch experiments were carried out in a jar mill in which Mersorb pellets were contacted with an acidic solution containing mercuric nitrate. Tests were made to investigate the effects of jar mill speed on the uptake rate of mercury from a

100-mL solution at a pH of 2. Jar mill speeds tested were 11, 22, and 38 revolutions per minute (rpm) with an excess of pellets (5 g). The Mersorb pellets were capable of reducing mercury concentrations to <0.2 mg/L. The initial mercury uptake rate followed first-order kinetic behavior (which was expected) and is described by the equation

$$\log[Hg2+] = -0.4343kt + \text{constant} \qquad \text{Eq. (1)}$$

where k is the apparent reaction rate constant and t is time. Based on the result of the initial experiments, it was determined that a jar mill speed of approximately 22 rpm would be used for the remainder of the study to avoid external (to the pellets) mass transfer limiting conditions.

The effect of pH on mercury uptake was investigated for three initial pH levels (pH 2, 3, and 4) using 100 mL of a 10 mg/L mercuric nitrate solution and 5 g Mersorb pellets. Mercury levels decreased to less than 0.5 mg/L in a 30-min contact time, and the overall mercury uptake was approximately the same for each of the pH conditions studied.

Mersorb saturation loading (or capacity) at a low and neutral pH was evaluated at various mercury and sorbent concentrations. The measured saturation loading at pH 2 was on average 0.04 g mercury/g dry pellets. At a pH of 8, the loading increased to 0.12 g/g. These correspond to 5.6% and 16.9% of the theoretical capacity respectively.

Surrogate waste (to simulate ICPP waste) was prepared from a waste description obtained from Idaho National Engineering Laboratory. Mercury uptake from a 100-mL solution was monitored for the ICPP surrogate waste and for a well-defined mercuric nitrate solution using two liquid/Mersorb ratios (100:0.9 and 100:0.5 mL/g). Compared with the mercuric nitrate solution, mercury loading on the beads was reduced by 50% when the ICPP surrogate waste was used (see Fig. 1). This was not unexpected because the ICPP surrogate waste contained competing ions such as iron, lead, cadmium, and nickel.

Fig. 1. Mercury loading of 0.5g of pellets from two simulated wastes.

Large-scale separation of mercury from solutions would most likely be accomplished by passing the contaminated stream through columns packed with Mersorb pellets. A small laboratory column was designed and constructed to obtain mass transfer data and mercury capacity for columns. The column length (and thus the amount of sorbent) could be adjusted, and the solution was pumped through the bed by a variable-speed pump.

An absorption column in operation has three distinctive zones: 1) the saturated zone which is close to the feed inlet; 2) the mass transfer zone (or absorption zone), in which active absorption of the solute occurs; and 3) the unsaturated zone, which is essentially free from the solute and the sorbent retains its original absorption capacity. The mass transfer zone is mobile during the course of the operation and is positioned at the inlet of the column at start-up and moves toward the outlet with time. The length of the mass transfer zone is dependent upon fluid flow and absorption kinetics. The mass transfer zone length is calculated from breakthrough curves (obtained by monitoring effluent concentration during continuous column operation). Typically, absorption columns are designed to operate until 5% breakthrough is noticed in the effluent. In general, the shorter the mass transfer zone length, the more efficiently the absorption column is used.

An initial column experiment was conducted with a bed depth of about 2.5 cm. The feed solution concentration was made from $Hg(NO3)2$ to contain approximately 1000 mg/L mercury and had a pH of 2. The feed flow rate of 7 mL/min was a factor of ten lower than typical process flow rates but was chosen as a conservative first attempt. The first fraction of effluent, consisting of 6.5 bed volumes, had a mercury concentration of 222 mg/L, indicating immediate breakthrough. This also indicated that the length of the mass transfer zone was longer than the total length of the column.

To improve mercury removal and to find the mass transfer zone length, a second column experiment was performed with bed depth of 7.6 cm under the conditions mentioned above. Based on the results from this experiment, it was concluded that the mass transfer zone length was approximately 6.2 cm at a liquid superficial velocity [(volumetric feed rate/(cross-sectional column area)] of 0.0068 mm/s. A mass balance revealed that at exhaustion the column had reached 5.8% of the theoretical mercury capacity, which is consistent with results from jar mill tests at this pH. As previously reported, capacity increases with solution pH. This result

may be used to design large-scale columns operating at similar liquid superficial velocities. The initial effluent that exited the column contained 20 mg/L Hg, which is 100 times higher than the Environmental Protection Agency's (EPA's) limit (0.2 mg/L). The column length was subsequently increased (to 12.7 cm) in the next experiment, which resulted in a tenfold decrease in the initial effluent concentration from the column. The results of the three experiments are shown in Fig. 2.

Fig. 2. Effect of column length on mercury removal from a feed stream containing approximately 1000 mg/L mercury.

EPA's toxic characteristic level (0.2 mg/L) was reached in a column test with a bed length of 56.5 cm. This column experiment worked so well that it was shut down after 127 bed volumes of feed (350 mg/L, pH 3) had passed through the column. The effluent at shutdown contained 0.025 mg/L mercury. A variety of liquid superficial velocities were tested throughout the experiment with essentially no change in the effluent concentration. A mass balance over the column indicated that 45 g mercury was absorbed in the column.

Process economics will be much more favorable for Mersorb when applied for solutions at the higher pH. Even at the lower pH, Mersorb may be competitive with ion-exchange resins. This may be illustrated by comparing sorbent costs with a typical ion exchanger (Duolite GT-73) used for mercury removal. Mersorb costs on the order of $3/lb, and even when loaded to only 3% of theoretical capacity, the cost for Mersorb is about $0.28 per g of mercury absorbed. Duolite GT-73 costs about $17/L with a capacity of about 30 g Hg/L resin(1). This gives a cost for Duolite GT-73 of about $0.56 per g of mercury absorbed. It is assumed that neither sorbent is regenerated. A complete economic analysis would consider many other process costs, but the sorbent costs cited here indicate that Mersorb may find some applications for DOE mixed wastes.

MERCURY REMOVAL FROM SOLID MIXED WASTES

The second focus of this research was to evaluate chemical leaching as a technique to treat soils and glass contaminated with either elemental mercury (Hg0) or a combination of several mercury species, including elemental mercury (Hg0), mercuric oxide (HgO), mercuric sulfide (HgS), and mercuric chloride (HgCl2). Potassium iodide/iodine (KI/I2) solution was investigated as chemical leaching agents for contaminated soils, and in addition to KI/I2 solution, several halide solutions and dilute acids were evaluated as leaching agents for glass wastes.

For a leaching solution to be effective, it must be capable of both solubilizing solid mercury compounds and of forming mercury complexes that will remain in solution, making separation from the treated solids possible. Halides, including bromide (Br-), chloride (Cl-), and iodide (I-), have been shown to form soluble complexes with mercury (II) species. According to chemical data2, halogen ions such as I-, Br-, and Cl- have large formation constants (Kf) with mercury and low oxidation potentials, thus allowing the formation of stable mercury-halide complexes.

Researchers at the General Electric Company (GE) have developed a mercury removal process in which I2 is used to oxidize mercury to the 2+ state and KI is used as a source of I-. When KI/I2 leaching solutions are applied to a mercury-contaminated solid, the various solid mercury species are transformed to the soluble complex HgI42- according to the following reactions:(3)

$$HgO + I2 = HgI2 \quad (2)$$

$$HgI2 + 2\ I- = HgI42- \quad (3)$$

$$HgO + H2O + 4\ I- = HgI42- + 2\ OH- \quad (4)$$

$$HgS + I2 + 2\ I- = HgI42- + S \quad (5)$$

A review of the literature suggested that halide solutions and some acids should be effective leaching agents for mercury-contaminated glass. Radian Corporation (Oak Ridge, Tennessee) conducted a soil treatability study in 1992 with several leaching agents, such as ammonium hydroxide, sodium hypochlorite (NaOCl), bromic acid (HBr), KI/I2, thiourea, acetic acid, ethylenediaminetetraacetic acid (EDTA), synthetic acid rain, and deionized water. It was reported that KI/I2, HBr, and NaOCl were the most effective in leaching mercury-contaminated soils. Because KI/I2 and NaOCl were found to be effective in leaching mercury from soil, it was to decided to evaluate their effectiveness on glass waste. Sodium bromide (NaBr) was evaluated as a leaching agent instead of HBr to avoid safety hazards associated with HBr. Wang and co-workers used sodium chloride (NaCl) to remove mercury from freshwater

sediments(4). Sodium chloride was evaluated in this study because if it were found to be applicable, its use on a larger scale would be very cost-effective. Because both Br- and Cl- form complexes primarily with Hg2+, adjustment of pH to a value of 2 was investigated as a means to improve the leaching efficiency of the NaBr and NaCl solutions. Also, dilute HNO3 and hydrochloric acid (HCl) solutions were investigated because it is well known that most forms of mercury are soluble in these acids.

Methods

For the soil leaching experiments, EPA synthetic soil material5 was contaminated by adding measured amounts of HgO, HgS, or HgO to aliquots of 10 g soil in 125-mL Erlenmeyer flasks. In all experiments, the target pretreatment total mercury concentration was 20,000 mg/kg. Surrogate radionuclides CeO2 and CsCl (approximately 0.025 g of each) were also added to the soil in some of the experiments. The flasks were capped after mercury addition and agitated for 24 h at 40C on a rotational shaker to allow the soil and the mercury to equilibrate.

Deionized water and reagent-grade KI and I2 were used to prepare KI/I2 solutions at least 12 h prior to soil treatment. Thirty milliliters of leaching solution was added to each flask containing soil to be treated. Deionized water that had been pH adjusted was added to control flasks instead of KI/I2 leaching solution. A blank consisting of surrogate soil without added mercury was also treated with the KI/I2 leaching solution. The flasks containing soil and leaching solution were either capped or covered with parafilm (depending on treatment temperature) and placed in preheated environmental shaking chambers (Lab-line Instruments, Inc., Melrose Park, Illinois).

After the 4-h reaction time had elapsed, the contents of the flasks were vacuum filtered through 0.45-m membrane filters. The residuals remaining in the flask and the filter cake were washed with an additional 30 mL of deionized water. The volume of the supernatant and wash water was measured, and the liquids were combined in 80-mL glass sample jars. The filter and filter cake were returned to the Erlenmeyer flask for digestion. All samples were stored at room temperature in glass containers prior to analysis.

For glass leaching experiments, 30.5-cm-long by 2-cm-diameter (8-W) GE fluorescent bulbs were used. The following leaching solutions were used to treat the prepared glass: NaOCl, KI/I2, HCl, HNO3, NaBr, NaBr + acid to pH 2, NaCl, and NaCl + acid to pH 2. The fluorescent bulbs were prepared for leaching by first removing the aluminum end cap with pliers and then breaking the glass into small pieces (less than 1.3 cm) inside a sealed plastic bag. The broken glass and phosphor material were then weighed and transferred to an Erlenmeyer flask for treatment. The average mass of crushed glass in each flask was 24 0.9 g. A small amount of HgO was added to the glass in each flask to achieve mercury contamination levels similar to those found in larger bulbs, such as those used at DOE facilities. The target mercury contamination level was 250 mg/kg HgO. After glass contamination, approximately 75 mL of prepared leaching solution was added to each flask. Blank and control samples were also prepared. The flasks were then covered and placed on the rotational shakers. All of the glass treatment experiments were conducted at room temperature (25C).

The mercury concentration in the post-treatment samples was measured as described above. The presence of iodine in concentrations greater than 5 mg/L significantly interfered with the analysis of Hg because of the competition for the reducing agent. Therefore, it was necessary to dilute samples in the range of 50,000- to 200,000-fold. Using this method, Hg recoveries in excess of 93% were observed. A flame atomic absorption spectrometer (Perkin-Elmer, Model 5000) was used to measure the cesium concentrations, and cerium concentrations were measured by an outside analytical laboratory using inductively coupled plasma (EPA method 200.7).

Results and Discussion

Results of the mercury leaching studies from contaminated soil and glass are summarized in Figs. 3 and 4. The mercury removal percentages presented may reflect more than one mercury removal mechanism, including the aqueous solubilization of the mercury compounds and the chemical leaching of mercury with KI/I2 or other leaching solutions. Because the overall removal of mercury from solid mixed waste was of most interest in this study, the reporting of these combined removal efficiencies was deemed appropriate. Control studies in which mercury-contaminated soils were leached with deionized water instead of KI/I2 leaching solution revealed that less than 1%

of the mercury was removed as a result of the aqueous solubilization of the mercury.
Fig. 3. KI/I2 leaching of mercury-contaminated EPA synthetic soil.
Fig. 4. Mercury removal from crushed glass contaminated with Hg0. (Data for treatment with NaCl, NaCl + acid, NaBr, dilute HCl, and dilute HNO3 are not shown; no mercury removal was observed for these solutions.
The KI/I2 leaching solutions were able to reduce the mercury concentration in the contaminated soil by as much as 99%. It was observed that KI/I2 leaching is applicable to multiple forms of mercury, including Hg0, HgO, HgS, and HgCl2, and that treatment efficiency is independent of temperature for the KI/I2 leach solution concentrations evaluated. The experiments conducted with the surrogate radionuclides revealed that CeO2, which was a surrogate for uranium, did not solubilize and remained with the solids following treatment. Cesium, in contrast, was found in both the solid and liquid phases following treatment.
Of the eight different leaching solutions evaluated for the crushed fluorescent bulbs, only three, 1.0/0.1 M KI/I2, 0.8 M NaOCl, and 0.5 M NaBr + acetic acid, were able to remove any mercury from the glass. The percentages of removal reported were calculated using the mass of mercury added to the glass and the measured mercury concentration in the glass after treatment. Of the three treatments that worked for the crushed glass, KI/I2 removed the greatest amount of mercury and resulted in the lowest Hg0 residual concentration. NaOCl was almost as effective as the KI/I2, but the NaBr + acid removed less Hg0 than the either KI/I2 or NaOCl. The post-treatment mercury concentration in the glass treated with either KI/I2 or NaOCl was less than 10 mg/kg. The lower Hg0 residuals were probably obtained with glass compared with soils because of the much lower initial mercury concentration and the simplicity of the glass matrix.

CONCLUSIONS

Based on the bench-scale laboratory experiments, the following conclusions have been reached:
1. Sulfur-impregnated, activated carbon pellets (Mersorb) can be used to remove mercury (Hg2+) from aqueous solutions to below EPA's toxic characteristic level (0.2 mg/L).
2. Mersorb works under acidic conditions (pH of 2), but its capacity is reduced by approximately 50% compared with neutral conditions.
3. Competing ions present in the target waste stream reduced the Mersorb capacity by 50%.
4. Mersorb appears to be economical compared with leading ion-exchange resin.
5. KI/I2 leaching solution can be used to remove up to 99% of the mercury in contaminated soil and glass.
6. KI/I2 leaching solution worked well with several mercury species, including Hg0, HgO, HgS, and HgCl2.
7. KI/I2 leaching solution worked well with a wide variety of initial mercury concentrations.
8. Radionuclide surrogate studies suggested that uranium will not partition into KI/I2 leaching solutions. Cesium may partition into the KI/I2 leaching solution because of the high solubility of cesium salts.

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CONTINUOUS MONITORING OF VOCS USING TUNABLE DIODE LASERS*

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ABSTRACT

In this paper we describe our ongoing project to develop and apply a family of on-line process monitors based on tunable diode lasers (TDLs). We focus on a continuous emission monitor for volatile organic compounds (VOCs) likely to be present in trace amounts in the off-gas of thermal waste treatment processes. Partnering with PSI Environmental Instruments Corp. of Andover, MA, we have demonstrated TDL-based technology for the real-time monitoring of ammonia at ppm levels in the harsh environments of power plants and municipal incinerators. We are extending this technique to the detection of VOCs through a combination of laboratory spectroscopic measurements and field trials.

INTRODUCTION TO TUNABLE DIODE LASER-BASED CONTINUOUS EMISSION MONITORS

There are a number of industrially-produced, gas-phase species for which reliable real-time monitors are currently not available. The Department of Energy's Mixed Waste Integrated Program has identified the need for such instrumentation in order to provide for rapid and cost-effective regulatory compliance. In addition, there is demand in numerous industries for real-time information about important process variables that can be incorporated into process-control algorithms. We are investigating the use of tunable diode lasers (TDLs) for the purpose of both continuous emissions monitoring and process control.

Essential characteristics of a useful continuous emission monitor (CEM) are as follows: 1.) The CEM must be robust and sensitive; it must detect the species for which it is designed, but not suffer from interferences due to other species in the effluent. 2.) The CEM must be reliable and rugged; it should function continuously with very little down-time and maintenance required. The CEM should be sufficiently sturdy to survive in the hostile environments that must be probed to provide the desired information. 3.) The CEM response should be rapid, providing information sufficiently quickly that the information can be used (if desired) for process control. 4.) The CEM should be affordable, providing clear benefit for the money invested in its purchase, preferably recovering its own cost in a reasonable time period.

There are additional features that are attractive in CEMs. In many cases in situ measurements are desirable. User interaction, such as calibration, should be minimal and preferably automatic. In addition, versatility is an advantage: a single piece of hardware that can do the job of several monitors is preferable. Unfortunately, all or most of these requirements are rarely met in a single instrument.

The technical basis of TDL-based monitors is the detection of molecular gas-phase species through optical absorption of laser radiation by molecular vibrational transitions in the mid-infrared (mid-IR) or near-infrared (near-IR) wavelength region. TDLs emit extremely narrow, single-mode radiation that can be tuned across molecular absorption lines by varying injection current or device temperature. In general, the operating temperature of the diode is held constant while the current supplied to the diode is modulated rapidly to vary the wavelength.

Single-mode tuning of diode lasers can be accomplished over a small spectral range for a given temperature at rates faster than 1 KHz. The wavelength range over which a TDL will lase is determined by the composition of the semiconductor laser; appropriate choices of diode material and temperature/current combinations lead to detection of different molecular gas-phase species. While the small spectral tuning range of these lasers is a liability for the simultaneous detection of many gas-phase species, the extremely narrow laser line width permits the resolution and unambiguous identification of specific molecules in contrast with other optical

methods such as Fourier transform infrared (FTIR) spectroscopy (see below).
TDLs are extremely compact, robust solid-state devices. These lasers have been extensively developed for other industries (such as the communication and entertainment industries) and have been demonstrated to be low-cost, reliable, physically compact, and easily adaptable to spectroscopic applications. Since they are semiconductors, they can be manufactured by mass production methods, thus greatly reducing the cost of an individual laser component.
The implementation of TDL-based monitors is simple. Laser radiation traverses the volume of interest while the laser wavelength is repeatedly scanned. The degree to which the radiation at each wavelength is absorbed is related to the presence of specific absorbers in the sample. Quantification of the absorption can be straightforward and provides a measure of the concentration of the absorbing species. The sensitivity of this technique to molecular species is greatly enhanced by high-frequency modulation of the laser beam and phase-sensitive detection. This can be a great advantage when monitoring species in a high-temperature, particle-laden environment where blackbody radiation interferes with measurements. Since background signals are invariant over the small wavelength region scanned by high-frequency modulation, application of lock-in amplification suppresses the contribution of background radiation to the absorption signal thereby improving the signal-to-noise ratio. Contributions to the signal from other nearby, molecular-absorption features can also be minimized by the use of frequency modulation, which improves the selectivity and sensitivity of the absorption sensor. Detection limits lower than 1 ppm-meter of absorbing pathlength have been demonstrated for small molecular species by this method (1).
Other analytical techniques, such as gas chromatography coupled with mass spectrometry (GC-MS), have also been applied to monitoring off-gas streams from industrial processes. In general these methods are able to provide molecular identification, but are necessarily extractive in nature. Extensive sample pretreatment may be required prior to analysis, and real-time information is unavailable due to time lags associated with extraction, particulate removal and separation on the chromatographic column. In addition, the process gas-temperature information is also lost.
Alternate methods employing optical techniques for gas-phase molecular detection are being actively pursued by others. A leading candidate is long-path optical absorption in the mid-IR region by FTIR spectroscopy (2). While FTIR has been used extensively in the analytical laboratory, developing this technology into a field monitor is difficult for many applications due to the limitations imposed by the characteristic low spectral resolution of hardened, field instruments.
The great strength of FTIR spectroscopy is its inherent wide spectral bandwidth, which allows for the simultaneous detection of many molecular species in the mid-infrared spectrum. Spectral resolution for real-time monitoring purposes, however, is on the order of 1 cm-1 or worse, which is more than ten times the line width of typical gas-phase absorption transitions. This makes the speciation and quantitative measurement of many VOCs very difficult due to distorted and strongly overlapping absorption features.
Laser-induced fluorescence (LIF) is another diagnostic technique for the detection of trace amounts of specific small molecules and atomic species. Although this approach is well suited for the measurement of extremely low concentrations of absorbing species, serious problems impede its utilization as an CEM monitor for complex effluent streams. While LIF is a suitable laboratory for a few small molecules, the necessary laser systems tend to be bulky, sophisticated, expensive, and not well suited to rapid wavelength switching which would be necessary for the rapid detection of more than one molecular species. With suitable calibration, broad-band excitation and fluorescence spectroscopy may be useful for quantifying certain classes of VOCs (aromatics, halogenated hydrocarbons, etc.).
In summary, many of these analytical techniques offer complementary capabilities to TDL spectroscopy for continuous monitoring. Requirements for a CEM are highly application-dependent and, as a result, an instrument that combines one or more of these methods may offer the most comprehensive approach to real-time monitoring of molecular emissions and process variables in off-gas streams.
TDL-BASED CEMS: TRACE GAS DETECTION FOR COMPLIANCE MONITORING
Direct laser absorption coupled with high-frequency modulation and lock-in detection can be very sensitive. Successful detection of trace species present in

concentrations less than ppm with TDL-technology has been demonstrated in the field (3). This can be of great value in industries where it is desirable to monitor chemical emissions on a real time-basis for the purpose of regulatory compliance. An important example of a need for a CEM is ammonia monitoring. Ammonia is used as a reactant or released as a by-product in the Selective Catalytic Reduction (SCR) or Selective Non-Catalytic Reduction (SNCR) methods for controlling NOx emissions from large combustion facilities. Ammonia that is not consumed by NOx is released into the atmosphere ("ammonia slip"). Excess ammonia slip not only wastes costly chemicals, but leads to plugging and fouling of heat exchange surfaces, especially when sulfur is present.

There are numerous obstacles to monitoring ammonia in boiler environments. For example, electrical and mechanical noise abounds, the operating temperature is high, and particulates are pervasive. Because current EPA regulations demand extractive sampling, and since fast response times are desired, there must be a high rate of gas exchange within the sampling cell. Both the sampling cell and tubing between the sampling point and the cell must be heated to prevent formation of ammonium sulfate powders. All surfaces that contact ammonia must be constructed using stainless steel in order to withstand temperatures as high as 260C (500F) and a corrosive gaseous environment.

Other techniques for measuring ammonia have drawbacks in their use as CEMs. FTIR spectroscopy cannot rapidly provide 1 ppm sensitivity. Wet chemistry techniques require about 30 minutes and human participation. Ion-mobility spectroscopy requires about 10 minutes in order to achieve 10 ppm sensitivity. Ultraviolet spectroscopy suffers from interferences due to the presence of SO2 in the gas stream. Gas correlation spectroscopy suffers H2O and CO2 interferences.

PSI Environmental Instruments Corp. has developed a TDL-based ammonia monitor (the trade name is SpectraScanTM) that overcomes each of these barriers (4). Figure 1 provides a schematic of this monitor. The modular design of this instrument is well-suited to field measurements. The heated, multi-pass gas cell is located near the process gas stream and is connected to the duct via a heated pipe with appropriate filters to remove particulate matter. The laser and associated electronics are mounted in a standard 19" equipment rack which can be remotely located, for example in a control room. An important feature of this monitor is the use of a near-IR TDL that can be operated near room temperature. This obviates the need for cryogenic cooling as is required for mid-IR TDLs. In addition, there are inexpensive and robust fiber optics available to transmit the near-IR laser beam long distances, improving the ease of remote operation.

Fig. 1.

PSI Environmental performed several series of field trials of the ammonia monitor in commercial power plants in Massachusetts and New Jersey. For these tests, continuous monitoring of ammonia was performed in conjunction with the operation of NOx suppression hardware. Results demonstrated ammonia detection in the ppm range when compared with independent wet chemical analysis. One set of results from these tests is shown in Fig. 2. The large variation in the ammonia content of the gas stream at the sampling point is clearly evident.

Fig. 2.

EXTENSION OF TDL-BASED MONITOR TO VOC DETECTION

We are partnering with PSI Environmental to extend this TDL-based technology to monitor other molecules. Of particular interest are volatile organic compounds (VOCs) in the off-gas streams from thermal waste processes. DOE's Mixed Waste Integrated Program has identified the need for such instrumentation in order to provide for rapid and cost-effective regulatory compliance and assurance of process control of a number of developing thermal waste treatment technologies. We are identifying wavelength regions where molecules of interest absorb radiation. The required diode laser is specified, and high-resolution absorption and calibration measurements are made to appropriately configure the monitor. After laboratory testing is complete, the prototype monitor will be tested at thermal waste treatment sites and its performance will be compared to other CEM methods and standard analytical results. Our TDL-based monitor is well suited to easy modification. Due to its designed modularity, laser wavelengths can be changed simply by substitution of TDL modules in the instrument. Future plans call for the incorporation of multiple TDL modules into a single instrument, allowing real-time monitoring of multiple species.

Unfortunately, the near-IR absorption signature of many VOCs and light inorganic species have not been well determined at sufficiently high resolution for our purposes. In order to identify optimal wavelength regions for analysis by TDL spectroscopy, we have surveyed the near-infrared absorption spectrum for a variety of species using FTIR spectroscopy at moderate spectral resolution. Certified mixtures of individual species at low concentrations in nitrogen or in permeation tubes were obtained for calibration purposes. Results for several chlorinated VOCs for the wavelength 1.58-1.74 mm region are displayed in Fig. 3.
Fig. 3.
The improved sensitivity attainable with the TDL-based absorption technique is demonstrated in Fig. 4 by comparing a portion of the near-infrared absorption spectrum of ammonia recorded with our FTIR spectrometer at moderate resolution (0.1 cm-1), and the TDL absorption signal for a "single" feature in that spectrum. The TDL signal clearly indicates that multiple absorption lines are actually present in the unresolved FTIR peak. Frequency modulation has been shown to improve the sensitivity of absorption measurements by several orders of magnitude, and is an important feature of the TDL monitors under development.
Fig. 4.
We are currently in the process of characterizing a tunable diode laser that emits in the 1.65-mm wavelength region. Figure 3 shows that a number of VOCs absorb in this region, and a CEM based on this laser will be tested in the laboratory shortly.
SUMMARY
Sandia and PSI Environmental have completed developmental work on a TDL-based ammonia monitor that serves as the basis for a CEM for VOC detection. Spectroscopic survey work using a laboratory FTIR spectrometer revealed absorption signatures for several VOCs in the 1.58-1.74 micrometer wavelength range, and a prototype TDL-based monitor was fabricated and is currently being tested. The FTIR measurements are used to specify the TDL wavelength range required for a given VOC. Future work includes: detailed spectroscopic analysis with the TDLs to determine optimum wavelengths for detection of target VOCs; laboratory calibration of the monitor for each VOC; development of a protocol for field trials including on-line calibration; and tests of the VOC monitor under field conditions.
ACKNOWLEDGMENTS
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37-29
PROGRESS REPORT - MICROWAVE VITRIFICATION OF ROCKY FLATS HYDROXIDE COPRECIPITATION SLUDGE, BUILDING 774
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ABSTRACT
This report describes the first set of experiments performed on transuranic (TRU) precipitation sludge produced in Building 774, to determine the operating parameters

for the microwave vitrification process. Toxicity Characteristic Leach Procedure (TCLP) results of the raw sludge showed concentrations of lead, silver and cadmium which were in excess of land disposal restrictions (LDR). Crushed, borosilicate, glass was used as a frit source to produce a highly desirable, vitrified, product that required less energy to produce. TCLP testing, of microwaved samples, showed favorable results for 40 and 50 percent waste loading. The results of this study are encouraging and support the development of microwave vitrification technology for the treatment of various mixed waste streams at Rocky Flats Environmental Technology Site. However, additional experiments are required to fully define the operating parameters for a production-scale system.

INTRODUCTION

The Department of Energy's (DOE) Rocky Flats Environmental Technology Site was a production facility for the nuclear weapons complex for more than forty years. As a by-product of production at Rocky Flats, a variety of hazardous and radioactive wastes were generated, stored, and treated. Most of the stored wastes do not meet the current Department of Transportation (DOT) shipping requirements or the waste acceptance criteria (WAC) for land disposal. The WAC requires that the waste form passes the TCLP testing as required by the Resource Conservation and Recovery Act (RCRA) for land disposal.

The microwave vitrification process has been in development at RFP since 1986 to set forth the critical operating parameters for the vitrification of wastes suitable for land disposal (Petersen, R. D., 1990). The objective of the microwave vitrification bench scale testing is to determine if microwave energy is capable of melting actual TRU precipitation sludge generated in Building 774 to reduce its volume, and produce a waste form acceptable to the Waste Isolation Pilot Plant. The hydroxide precipitation sludge from Liquid Waste Treatment operations in Building 774 was one of the waste streams treated during the experiment. This sludge is produced during a coprecipitation process in the liquid waste treatment facility (Building 774) and is dewatered using a vacuum drum filter precoated with a filter aid media. The TRU precipitation sludges are currently treated in a cementation process. The negative aspects of this process are:

1) Increase in waste volume
2) Process did not produce a consistent repeatable product
3) Product did not meet the requirements of DOT or LDR.

The experimental test plan and design matrix, used for the hydroxide precipitation sludges, was developed from previous testing surrogate wastes using a bench scale microwave in Building 701 (Eschen, Fenner, Splett, and Sprenger, 1994). The testing of the surrogate waste produced a window of operation which was validated by the microwave vitrification testing of the actual hydroxide precipitation sludges in Building 774. This report discusses the results of the tests.

EXPERIMENTS

The following subsections describe the procedure for performing the experiments and the results of the experiments.

Procedure

Hydroxide sludge was not available for the experiment due to curtailment of the coprecipitation process in the liquid waste treatment facility (Building 774). Residue Treatment Technology (RTT) personnel developed a procedure to sample one of the feed tanks (tank 40) to extract enough liquid and particulate material to filter for conducting the microwave vitrification experiment. Twenty four 4-liter bottles of liquid were sampled from the feed tank 40. The liquid was then vacuum-filtered through a Buchner funnel, which was precoated with the same filter aid media used on the vacuum drum filter of the coprecipitation process. The sludge was scraped from the filter and collected for microwave vitrification testing. Approximately 2 kilograms of material was collected. This filtering process yielded material with a composition of sludge to filter aid media that was similar to the current production vacuum drum filter process. A sample of the filtered sludge was sent to the Plutonium Operations Support Laboratory in Building 771 for TCLP analyses to determine if any element that would restrict land disposal were present. Table I, TCLP Analyses of Raw Sludge from the Coprecipitation Process of Liquid Waste Treatment Facility (Building 774), provides the results of these analyses. Shaded boxes exceed land disposal restrictions.

TABLE I

Procedure RTT986277-50-2, Operating Procedure, Microwave Solidification - Building

774, is the operating procedure used for microwave vitrification testing. The microwave vitrification developmental equipment used in Building 774 is in room 210, and consists of a 6 kW, 2450 MHz Cober microwave generator, waveguides, tuner, resonant cavity, glovebox, turntable, screwfeeder and stand alone chiller. The generator has a variable output of 0 to 6 kW of microwave energy. The waveguide is separated from the glovebox with a 1/4-inch-thick teflon window.
The following operating parameters were controlled during the experimental test runs:
  Waste loading (S/F) -- Weight percent ratio of waste sludge (S) to the frit (F).
  Temperature (oC) -- Measured with an infrared thermometer or a thermocouple.
A range of waste load ratios was selected based on previous testing of surrogate waste (Eschen, Fenner, Splett, and Sprenger, 1994). Table II, Test Matrix for Microwave Solidification of Hydroxide Coprecipitation Sludge (Building 774), provides the matrix of waste loading and temperature per test run.
TABLE II
For this series of tests, crushed, borosilicate glass was used as a frit source to produce the desired vitrified product. In each test run, 150 grams of the feed material (sludge / frit) was vitrified in a ceramic crucible, insulated on all sides, and held within a commercially available 3 liter, stainless steel, container. The time required for each test run was very consistent, approximately 60 minutes, but did vary slightly due to the composition of the feed material and the amount of reflected power produced.
The final waste forms produced from the tests were prepared for TCLP analyses in accordance with Environmental Protection Agency (EPA) protocol. Material from the vitrified samples was removed from the ceramic crucibles and fractured with a hammer to collect a representative sample. No further size reduction was required for the TCLP analyses. Table III, TCLP Results of Microwaved Vitrification Samples, provides the volume of each vitrified sample collected and sent to the Plutonium Operations Support Laboratory in Building 771 for TCLP analyses.
TABLE III
TCLP analyses was performed in accordance with SW 846 Method 1311, Toxicity Characteristic Leaching Procedure. The samples were radiologically "hot" (>70 pCi/gm activity), so TCLP extracts were prepared within A-box containment. TCLP Extraction Fluid No. 1 was determined to be the proper extraction fluid by performing pH tests on the samples. The solid extracts were prepared and agitated in a standard rotating TCLP extractor at 30 2 rpm for 18 2 hours, following EPA SW-846 Method 1311. The solids extracts were immediately filtered through 0.7 micron borosilicate glass fiber filters to obtain the final TCLP extracts. Particle size reduction was not performed on the samples since they were visually determined to be smaller then the 9.5 mm mesh size reduction criterion stipulated in Method 1311.
The samples then underwent total metals digestion following EPA SW-846 Method 3010A, Acid Digestion of Aqueous Samples and Extracts for Total Metals by Flame Absorption or ICP Spectroscopy. The final nitric/hydrochloric acid sample digest solutions were analyzed for the requested RCRA metals using a Varian Model SpectrAA 30/40 atomic absorption spectrometer.
Results
The results of the TCLP analyses for the raw sludge obtained from the coprecipitation process of the liquid waste treatment facility (Building 774) are provided in Table I. The concentrations of cadmium, lead and silver exceeded the WAC for land disposal.
The results of the TCLP analyses for the microwave vitrified sludge samples are provided in Table III. These results showed the concentrations of silver, cadmium and lead, in four of the samples (774-RAW, 774-TPS-1, 774-TPS-6, and TPS-16), exceeded the WAC for land disposal. The shaded boxes in Table III identify where LDRs are exceeded.
However, the test samples with the sludge to frit ratios of 40/60 and 50/50 produced acceptable results. Since only one sample per test were prepared, the confidence level of the TCLP analyses results is low. The 774-RAW sample was a raw sludge sample submitted to verify the TCLP results in Table I. However, the TCLP analyses results are encouraging because samples with sludge/frit ratios of 40/60 and 50/50 did pass the TCLP tests.
CONCLUSIONS
The Microwave vitrification experiment was performed to determine if microwave

energy is capable of melting the TRU precipitation sludge, immobilizing the hazardous constituents, reduce its volume, and produce a waste form acceptable to the Waste Isolation Pilot Plant.
The following conclusions have been determined:
The TCLP analyses results concluded that the 40/60 and 50/50 sludge to frit ratio samples are acceptable to DOT and WAC requirements.
Although acceptable, TCLP analysis on the limited samples produced a scatter of the data. Therefore, specific operating parameters for the microwave vitrification system cannot be defined, necessitating further testing.
The results of the TCLP analyses are encouraging and support the continued development of microwave vitrification technology at the Rocky Flats Environmental Technology Site.
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37-31
POLLUTION PREVENTION AND SAFE VENTILATION OF NUCLEAR, HAZARDOUS AND MIXED WASTE CONTAINERS USING ACTIVATED CARBON-CARBON CARTRIDGES AS POROUS ADSORBENTS: A MATHEMATICAL AND EG&G ROCKY FLATS PLANT FIELD REVIEW
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ABSTRACT
Nuclear, hazardous and mixed waste containers are exposed to extreme environmental atmospheric conditions resulting in head space vapor expansion. The waste generator or TSD facility is then faced with a dilemma: Either risk violation of RCRA Legislation (40 CFR Section 264, 265) by allowing the expansion gases to freely vent to atmosphere or, if sealed, risk drum rupture and the uncontrolled release of VOC pollutants and liquids. Nuclear Filter Technology, Inc., has developed an organic vapor adsorption system, the NucFilTX200, for use on nuclear / hazardous, or mixed waste containers. The NucFilTX200 utilizes a regenerable, activated carbon cartridge housed in stainless steel, that offers an economical solution to this dilemma. EG&G Rocky Flats Environmental Technologies Site, recently retro-fitted three existing containers of VOC waste with the NucFilTX200. Upon installation, a significant decrease in fugitive emissions within the drum storage area was measured and it was noted that drums no longer become pressurized. Prior to use at the Rocky Flats Environmental Technologies Site, an estimate of the activated carbon-carbon cartridge life was needed. A mathematical equation that predicts the volatile organic mass to challenge the cartridge was derived and, given several known variables, a computer application designed to model cartridge life.
INTRODUCTION
Drums that contain volatile organic compounds (VOCs) may become over-pressurized if sealed. If the drum is not sealed, gasses will vent to atmosphere causing increased fugitive emissions. A low cost solution to this dilemma is to vent the drum head space gasses through an adsorbent activated carbon media. Allowing the ventilation of head space gases to atmosphere through an activated carbon media assures ambient pressure within drum head space, and eliminates fugitive emissions. One device manufactured by Nuclear Filter Technology, Inc., is the NucFilTX200 Hazardous Waste Adsorption System, utilizing an activated-carbon-bonded-carbon cartridge, shown in Fig. 1.
Fig. 1.
The NucFilTX200 housing is designed to fit inside the container to allow for drum stacking, and to protect the filter housing from physical damage. The housing may be manufactured from a variety of materials to suit almost any adverse storage condition. Although this paper address a very specific application of the NucFilTX200, it should be noted that the filter can be adapted for general use on hazardous waste containers.

Computer Model

Use of the NucFilTX200 at the Rocky Flats Environmental Technology Site required an accurate estimate of cartridge life since the activated carbon-bonded-carbon cartridge has a limited life. An estimate of cartridge life was first established using an equation derived from Ideal Gas and corresponding state arguments, and then by applying a computer to model cartridge organic vapor loading.

Adsorption of VOCs onto granular carbon is well understood, and that adsorptive capacity is finite. The active sites available for adsorption of gases eventually become fully loaded with VOCs, depending on several known variables. Useful adsorptive life of the activated carbon-bonded-carbon cartridge is a function of diffusion rates and time, changing atmospheric conditions, head space available within drum, and the nature of the organic constituent. The two primary atmospheric variables include temperature and atmospheric pressure variation, of which temperature, is the largest driving mechanism for vapor expansion within the drum head space. For example, it is often observed in the field that drums exposed to bright sunshine will pressurize and in some cases show signs of drum deformation.

The amount of organic vapor to challenge the activated carbon-bonded-carbon cartridge should follow the general form:

Eq. (1)

where the total adsorptive challenge to the activated carbon cartridge, AT, is the sum of the adsorptive challenge from gas expansion at constant pressure, AEP, and at constant temperature, AET, and the adsorptive challenge from gas diffusion, AD.

Solving the equation yields*:

Eq. (2)

where VHS (liters) is the volume of the head space within the drum, Wi, and gi, are constants relating to vapor pressure as a function of temperature of some VOC species i. The initial or low temperature of a diurnal cycle is given by TL (Kelvin), and the high or maximum temperature of the diurnal cycle is TH (Kelvin). The term Pi is a constant relating to barometric pressure for some VOC species i. The initial or low barometric pressure in a 24 hour cycle is given by PL (Torr), and the high or maximum barometric pressure of the cycle is PH (Torr).

A vent tube of length LT (cm), and cross sectional area aT (cm2), provides a path for gasses to diffuse from the drum head space to the activated carbon cartridge. Time, in hours, is t. The constant, gi, relates diffusion characteristics of some VOC species i in air.

The term AET, the adsorptive component at constant temperature resulting from a decrease in atmospheric barometric pressure, is assumed to be negligible if the geographic site of drum storage does not change. Indeed, in any region, even given a substantial barometric pressure decrease, e.g. from 31" Hg to 28" Hg over a 24 hour period, the pressure component challenge is an order of magnitude less than the diffusional component over the same period of time.

The constants gi, Wi, Pi and gi, for five commonly used VOCs is presented in Table I

TABLE I

Using Eq. 2 and a computer model, one can calculate the total amount of organic vapors to challenge the activated carbon cartridge AT. Table II below demonstrates model predicted values of total organic vapor to challenge adsorbent, AT, for 1,1,1 trichloroethane (TCA), acetone, isopropanol, and methyl ethyl ketone (MEK). Various head space volumes are given, as are two 24 hour, diurnal temperature swings: One from 60F (288 K) to 65F (291 K), typical storage room conditions, and one from 65F to 85F (302 K), extreme storage room conditions. It should be noted that a 55 gallon drum filled, leaving a six inch head space, provides a head space volume, VHS, of 40 liters.

Table II

Estimate of Cartridge Change-Out Cycle

The NucFilTX200 utilizes a 200 gram activated carbon-bonded-carbon cartridge. Bruaner, Emmet and Teller (BET) surface area analysis conducted internally at Nuclear Filter Technology, Inc., verifies adsorptive surface area of about 700 m2/g. Desiccator studies of activated-carbon-carbon composites soaked in a continuous vapor phase of acetone, carbontetrachloride, or 1,1,1 trichloroethane indicates the cartridges' adsorptive capacity is 20% of its mass, or about 40 grams.

In order to estimate the number of days until the cartridge needs to be replaced or regenerated, assume a 'typical' storage area where temperature cycling occurs between 60F and 65F, and an 'extreme' storage area where temperature cycling occurs

between 65F and 85F. Assume that each consecutive day's temperature cycle is precisely the same. Assume two different drums are stored in the same geographic area, each of standard 55 gallon capacity but with different degrees of used capacity; one is filled with liquid organic solvent to within six inches of the top providing about a 40 liter head space, and a second drum that is partially filled to within 30" of the lid top (VHS =200 l) but maintains a saturated vapor condition. Then, given the two different drum's, one can quickly see that not only the temperature range, but also the available head space, or void volume, is important to estimating the cartridge change-out period. Table III below demonstrates the usefulness of the computer model given a full drum and a partially full drum given a series of typical and extreme storage conditions for five common volatile organic solvents.

Table III

BENCH SCALE LABORATORY TESTING OF COMPUTER MODEL

Bench scale laboratory testing was conducted to determine reliability of computer model predictions. In this test, over 1300 hours of activated cartridge exposure to isopropyl alcohol demonstrated that organic vapor loading may be predicted with about 75% accuracy given known environmental conditions.

Test Procedure

In order to test reliability of Eq. 2, and the computer model, a test was designed where a test vessel of known volume, containing 200 milliliters of isopropyl alcohol, was cycled though controlled simulated changes in temperature.

An activated carbon cartridge was off-gassed for 3 hours at 150C, weighed at 34.215 grams, then fitted to aluminum end caps, one end of which was specially fashioned so that it could be easily threaded on and off a special flange that attaches to a vent tube. The flange is equipped with a needle valve. The vent tube extends about 7 centimeters into the test vessel and has an overall length of 13 centimeters. A thermocouple is fitted to the test vessel so that temperature measurements can be made of the head space.

Fig. 2.

The steel test vessel with VHS equal to 19.40 liters, contained 200 milliliters of isopropyl alcohol (IPA) at the start of the test. The ventilation tube is 13 cm long and has a cross sectional area of 0.114 cm2. The carbon sample is then securely attached to the flange of the test vessel. The test vessel was placed in a refrigerator where the temperature was kept at a constant -1C. After a period of time, the needle valve was closed and the sample detached from the vessel, weighed, and the mass, and elapsed time recorded.

Similarly, the test vessel was placed in a warm environment and kept at a constant 24C. The time of each phase was varied from 25 to 378 hours. Since barometric pressure variation is impossible to control in the laboratory setting, for simplicity, it is assumed to be negligible. With the known time and temperature for each cycle, the computer model predicted the adsorptive challenge, AT, which was compared to actual measured values of adsorbed isopropanol.

RESULTS

Test results are summarized in Table IV and compared with predicted values for total adsorptive challenge, AT. Predicted and measured results are shown graphically in Fig. 3.

Table IV

Results indicate that model predictions explain about 75% of adsorbed isopropyl alcohol on the activated carbon-bonded-carbon cartridge. The very large error noted in the first two cycles may in part be because the sample had been completely off-gassed prior to use. Upon exposure to air, the highly active surface may have adsorbed additional contaminates from the air in the laboratory. Nevertheless, given normal temperature ranges and shorter periods of time between sample measurements, results may further improve. Likely causes of error include: Measuring error; effects of barometric pressure variances; mathematical error in the computer model; the fact that some desorption of the cartridge occurs; and, although slight, the fact that the cartridge is about 99% efficient as an adsorber.

Fig. 3.

FIELD USE OF NUCFILTX200 AT ROCKY FLATS ENVIRONMENTAL TECHNOLOGIES SITE

The Rocky Flats Environmental Technologies Site recently had a problem with pressurization of three liquid waste drums. The waste was generated as a result of laboratory processes and primarily consists of mixtures of organic solvents

including, carbontetrachloride, isopropyl alcohol and acetone. The drums were vented under controlled conditions whenever signs of pressurization were present. This practice raised concerns about employee safety and compliance with the RCRA legislation.
The NucFilTX200 met all requirements; it offers a low cost treatment alternative, the housing is constructed of stainless steel and fastens at the 2" bung of the drum, it is designed to reduce the possibility of liquid infiltration to the adsorbent, and it utilizes the highly activated carbon-bonded-carbon cartridge that provides longevity and chemical resistance from the harsh solvent atmosphere. Most importantly, the cartridge eliminates pressurization, and has sufficient adsorbent capacity to cost-effectively reduce volatile organic emissions significantly below important levels including the Lower Explosive Limit (LEL) and the Immediate Danger to Life and Health (IDHL). The LEL and IDHL regularly exceeded levels of 1000 ppm. Although statistically significant data is not available at this time, preliminary measurements suggest a 95% reduction in VOCs emitted to atmosphere. Emissions which would normally read above the LEL and IDLH (greater than 1000 parts per million) at the 2" bung, have been reduced to quantities well below the limits. The following data provides evidence of the filter's adsorptive efficiency.
Table V
Since installation of three NucFilTX200 at the Rocky Flats Environmental Technologies Site, there is no longer drum pressurization. Measurements of VOC concentrations within the storage room are now well below LEL and IDLH levels. Safety and environmental issues have been resolved since personnel are not venting pressurized gasses from the drum to atmosphere. After over five months of use, cartridge breakthrough has not been detected. According to the computer model, capacity (breakthrough) should occur after about 200 days. With organic reductions at greater than 95%, the NucFilTX200 could qualify as a closed vent system and control device per 40 CFR 265.1087, although slight engineering modifications may be required to facilitate cartridge change-out. Although additional data is required, it may be noted that measurements reproduced in Table V are well below established RCRA detection limits for detectable organic emissions which is equal to or greater than 500 ppm.
CONCLUSION
As pollution prevention measures become tightened, fugitive VOC emissions from existing containers of waste will become targeted for control. Tightly sealing drums of VOCs increases the risk of drum rupture allowing the uncontrolled release of VOC emissions and liquids. The NucFilTX200 assures pollution prevention and safe ventilation of drums that contain nuclear, hazardous or mixed waste constituents. It has also been shown that cartridge life may be predicted with reasonable accuracy to efficiently gauge cartridge change-out cycles.

37-32
UPFRONT DELISTING OF F006 MIXED WASTE (U)
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ABSTRACT
The U.S Department of Energy at the Savannah River Site (DOE-SRS) will petition the Environmental Protection Agency (EPA) to "upfront" delist treatment residues generated from the vitrification of approximately 650,000 gallons of a Resource Conservation and Recovery Act regulated mixed (hazardous and radioactive) waste. The upfront petition, based on bench-scale treatability studies and pilot-scale system data, will exclude the vitrified wasteform from hazardous waste management regulations. The EPA encourages the use of the upfront delisting method as it allows applicants prior knowledge of waste specific treatment standards, which when met will render the waste non-hazardous, before generating the final wasteform.
To meet the EPA performance based treatment standards, the waste must be stabilized to control the leaching of hazardous and radioactive constituents from the final wasteform. SRS has contracted a vendor to stabilize the mixed waste in a temporary

Vitrification Treatment Facility (VTF). The EPA has declared vitrification as the Best Demonstrated Available Technology for high level radioactive wastes (1) and the DOE Office of Technology Development has taken the position that mixed waste needs to be stabilized to the highest degree possible to ensure that the resulting wasteform meets both current and future regulatory specifications. Treatability studies conducted on a VTF pilot-scale system unit indicates that the mixed waste can be converted into a highly durable glass form, which exceeds the projected EPA performance based criteria. Upfront petitions can be processed by the EPA concurrently during facility construction or permitting activities; therefore, the SRS VTF will be capable of producing wastes which are considered non-hazardous sooner than otherwise expected. At the same time, EPA imposed conditional testing requirements to verify that the delisting levels are achieved by the fully operational VTF, ensures that only non-hazardous wastes are removed from hazardous waste management regulations.

Vitrification of the above waste will result in a volume reduction of approximately 75%. Volume reduction combined with excluding the waste from hazardous waste management regulations through delisting will significantly reduce overall management and disposal costs, resulting in an approximate $2.5 million cost savings.

INTRODUCTION

Fuel and target assemblies for the Savannah River Site (SRS) nuclear reactors were fabricated in metal finishing and aluminum forming facilities within the 300 M-Area. 300 M-Area manufacturing operations ended in 1992 as a result of the cessation of the cold war. Liquid effluents generated from past 300 M-Area manufacturing operations consisted primarily of metal finishing and aluminum forming process effluents, including nickel electroplating process effluents. These process waste effluents contained rinsewaters, stack acid scrubber effluents, and spent process solutions. Area support facilities (e.g., degreasing activities, laboratory effluents) contributed halogenated organics to the above wastestream to a lesser degree.

Since July 1985, area process wastewaters have been discharged to the M-Area Liquid Effluent Treatment Facility (LETF). The LETF utilizes the Best Available Technology Economically Achievable for the aluminum forming and metal finishing industries. The LETF is composed of three closed-coupled treatment facilities: the Dilute Effluent Treatment Facility (DETF), an Industrial Wastewater Treatment Plant; the Chemical Transfer Facility (CTF); and the Process Waste Interim Treatment/Storage Facility (PWIT/SF). Wastewater treatment sludges generated from both CTF and DETF operations are characterized as a Resource Conservation & Recovery Act (RCRA) listed F006 mixed waste (contains both hazardous and radioactive constituents). Approximately 1.5 million gallons of F006 mixed wastes were generated, with 650,000 gallons remaining in the PWIT/SF awaiting stabilization. The remainder of the waste, a supernate which separated from the residual precipitate, was treated in the DETF and released via a National Pollutant Discharge Elimination System permitted outfall.

The above M-Area Plating Line Wastes (MPLW) are subject to the Land Disposal Restrictions (LDR) in 40 CFR 268. The LDRs prohibits any storage of a land disposal prohibited waste except for the purpose of the accumulation of such quantities of hazardous waste as are necessary to facilitate proper recovery, treatment, or disposal. F006 wastes were restricted from land disposal on May 8, 1990. However, due to a national capacity extension for mixed wastes granted by the EPA, disposal and or storage was not prohibited until May 8, 1992. Subsequently, the PWIT/SF waste was added to the Land Disposal Restriction - Federal Facility Compliance Agreement (LDR-FFCA) between EPA and DOE, on March 13, 1991 to allow continued storage until a stabilization/solidification facility became operational.

Stabilization of this waste is not only required for compliance with hazardous waste regulations, but for the M-Area sludges, the LDR-FFCA specifies that the sludges will be treated in a vendor-supplied temporary Vitrification Treatment Facility (VTF). The LDR-FFCA sets strict milestones for completing the vendor stabilization program. SRS has contracted a vendor to stabilize the MPLW in a temporary VTF located in the 300 M-Area immediately adjacent to the PWIT/SF. The resultant treatment residues will be stored in a RCRA permitted container storage facility while awaiting confirmation that analyses are below the EPA imposed regulatory thresholds.

UPFRONT DELISTING OVERVIEW

The EPA recognizes that listed wastes may not be hazardous due to differences in feed stocks or industrial processes. Therefore, 40 CFR 260.20 & 260.22 contains a procedure whereby anyone can petition the EPA to "delist" or exclude such a listed waste from hazardous waste regulations. Originally, the intent was to ease the regulatory burden of the hazardous waste management of listed wastes improperly captured by the broad listing definitions. Since then, delisting has evolved to include those wastes sufficiently treated such that they no longer pose a threat to human health or the environment.
The DOE-SR will request the EPA for an upfront exclusion for certain wastes yet to be generated, as described in the "derived from" rule [40 CFR 261.3(c)(2)(i)] based on: bench-scale treatability studies; pilot-scale system data; the untreated waste characteristics, and detailed facility process descriptions. To be successful, the analyses of the treatment residues from the bench-scale waste treatment process must show that the waste no longer meets the criteria for which it was originally listed; must not exhibit a hazardous waste characteristic; and must not exhibit any other factors including additional 40 CFR 261 Appendix VIII constituents above regulatory threshold levels.
Regulatory threshold levels are calculated by the use of appropriate fate and transport models, for example the EPA's Composite Model for Landfills (2). The EPA believes that the primary pathway for a contaminant to adversely affect human health is through the ingestion of contaminated groundwater from the leaching of chemicals from land filled waste. Therefore, delisting petitions are evaluated by comparing leachate concentrations for specific contaminants contained within the petitioned waste against health-based levels, usually the Maximum Contaminant Levels for safe drinking water. The maximum allowable leachate concentration for specific contaminants are a function of waste volume, the smaller the volume of petitioned waste the larger the allowable dilution factor. Site-specific disposal conditions are not considered by the EPA in the evaluation of an upfront delisting petition since the generator is not required to dispose of the excluded wastes in a specific facility once the exclusion is granted.
To determine leachate concentrations for upfront delisting petitions, the EPA requires that treatment residues be analyzed by the Toxicity Characteristic Leaching Procedure (TCLP) and the Multiple Extraction Procedure (MEP)(2). The MEP is a test developed by the EPA to assist in predicting the long-term leachability of stabilized wastes. This procedure consists of the TCLP extraction, followed by nine sequential extractions on the same sample using synthetic acid rain to simulate multiple washings of percolating rainfall in an improperly designed sanitary landfill. It is estimated that these extractions simulate approximately 1,000 years of rainfall. In addition, total concentration analyses of all the 40 CFR 261 Appendix VIII constituents will be conducted to verify their absence.
Once the EPA grants an exclusion based upon the above information, the treatment residues would receive an upfront delisting with imposed verification testing requirements, which must be met through an EPA approved statistical sampling plan when the VTF becomes operational. The full-scale verification testing requirements may involve more then one round of waste characterizations to address any concerns regarding the potential for waste variability.
M-AREA PLATING LINE WASTES
The M-Area Plating Line Wastes (MPLW) are primarily wastewater treatment sludges generated from electroplating operations, since the wastewaters originated from electroplating
processes involving nickel plating of aluminum-clad, depleted-uranium slugs. The majority of the wastestream is considered a listed F006 mixed waste as it is specifically listed as hazardous per RCRA, Title 40 of the Code of Federal Regulations (CFR) 261.3(a)(2)(ii), and also contains a radioactive component consisting of source material (i.e., depleted uranium) as defined by the Atomic Energy Act Title 10 CFR 20.3(a)(15)(i). The primary hazardous constituent of concern is nickel, while the primary radioactive constituent of concern is depleted uranium (i.e., 238U). The sludges also contain significant levels of nitrate.
To a lesser degree, additional wastes generated from the electroplating process (i.e., Watts Type Nickel Bath Plating Line Solution, Spent Plating Bath Sludge, Mark 15 filtercake) as well as treatment residues generated from bench-scale treatability studies will be incorporated with the PWIT/SF F006 sludge prior to vitrification.
The MPLW partially occupies three 500,000 gallon tanks, six 35,000 gallon tanks, and

approximately one hundred and eight 55 gallon drums (i.e. Mark 15 filtercake, and Nickel Plating Line Solution). The quantity of MPLW (i.e., 650,000 gallons) combined with tank capacity limitations (i.e., largest tank capacity equivalent to 500,000 gallons) requires homogenization of the MPLW into two macro batches. The homogenization is designed to ensure that each macro batch will approach each other compositionally to the greatest extent possible. Table I provides a detailed chemical composition of the homogenized MPLW(3).

BENCH-SCALE DATA

Vitrification involves the exposure of hazardous materials to molten glass and related process conditions to effect the destruction, removal, and/or permanent immobilization of hazardous constituents. Vitrification is defined as the conversion of such solids into a glass residual form through the application of heat to the point of fusion. This process forms a molten, vitreous mass, and produces a glass-like residual product upon cooling. The residual solid is a solid (super-cooled liquid) containing an amorphous mixture of oxides (primarily silica and alumina) with little or no crystallization present (4).

Vitrification was chosen as the desired stabilization technique for the MPLW due to: destruction of hazardous organic constituents by pyrolytic decomposition and/or oxidation; removal of inorganic constituents in the residual glass product through chemical incorporation and/or encapsulation; and the chemical composition of the MPLW is amenable for vitrification as it already contains high levels of glass forming compounds (i.e., Si, Al, Na, Ca). The resulting glass residue exhibits excellent structural, weathering, and biotoxicity characteristics making it suitable for long term environmental exposure and is therefore capable of surpassing the EPA TCLP threshold levels. Vitrification will also result in an approximate 75% volume reduction resulting in reduced waste management costs.

Representative samples of the MPLW were submitted for bench-scale studies. Proportional amounts were homogenized thoroughly prior to vitrification. The resultant treatment residues were analyzed by the TCLP for inorganics only (i.e., Ba, Cr, Pb, Ni, U), as organics compounds are completely destroyed at the high temperatures at which vitrification occurs. All of the treatment residues passed the projected upfront delisting petition regulatory threshold levels by a wide margin (5) (Table II).

CONCLUSION

Bench-scale waste treatment studies using the process of vitrification have been successfully performed. The success of vitrification is attributable to a number of factors including:

the powerful solvating properties of glass melts and their ability to incorporate a wide range and large amounts of hazardous inorganics and radioactive components;

the complete destruction of organic compounds at the high temperatures at which vitrification occurs;

a stable, relatively homogeneous waste form that is highly resistant to aqueous corrosion; and

relatively high density waste form resulting in an accompanying large volume reduction.

The upfront delisting petition will show that the resultant treatment residue generated from process vitrification will not: meet any criteria for which it was listed; exhibit any hazardous waste characteristics as identified in 40 CFR 261 Subpart C; and will not exhibit any other factors or additional constituents which would render it to be considered hazardous.

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37-34
AN INTEGRATED CHEMICAL/THERMAL TREATMENT SYSTEM FOR MIXED WASTE
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ABSTRACT
Lockheed has developed an integrated technology train that incorporates existing technologies used in the petroleum, mining, and hazardous waste treatment industries for use in treating mixed radioactive and hazardous wastes, using a Retech-developed plasma vitrification process to 1) completely destroy the organic constituents, 2) result in a final waste that meets RCRA land disposal restricted criteria, and 3) permit permanent storage or disposal of the encapsulated radionuclides. The treatment system is very robust, in that wastes having a variety of chemical radiological, or physical characteristics can be successfully processed. No comprehensive pre-processing characterization is required; therefore costs and the potential for personnel exposure and contamination release (either chemical or radiological) are dramatically reduced. The only pre-processing measurement is the assay for the radionuclide concentration, needed for treatment decisions and criticality safety.
Volatile organic compounds (VOCs) in waste and soil are removed using low temperature solvent extraction with triethylamine. Radionuclides and other heavy metals in soil are removed using a nitric acid leach. The radionuclides combine with the nitric acid into primarily a nitrate form. This solute is removed using a counter-current ion exchange system, which also removes heavy metals. The clean soil is transferred from the leach circuit after dewatering to a rotary kiln to remove residual nitrates. The soil fines plus the concentrated residues from the triethylamine extraction and nitric acid leach are transferred to a plasma melter for stabilization as a glassy slag.
The non-soil wastes and residual concentrate from the soil treatment system are sent to the plasma melter which destroys any remaining organics and produces a virtually non-leachable cast slag that immobilizes both heavy metals and radionuclides. Significant volume reductions of the final waste to be disposed can be realized. Tests using various combination of soil and waste with variable organic and heavy metal constituents have shown that a volume reduction of greater than 90 percent will be achieved in its initial application for remediating Pit 9 at the Idaho National Engineering Laboratory.
INTRODUCTION
An integrated technology train to treat mixed and radioactive and hazardous wastes has been developed by Lockheed Environmental & Advanced Technology (LESAT) and its subcontractors (Retech, Inc., Merrick & Co., Sonsub, MSE, Inc., S. M. Stoller and Jason Associates) for remediating Pit 9 at the Idaho National Engineering Laboratory (INEL). The wastes are retrieved in containment inside a movable retrieval building. Ionization counting of boxes or drums of waste determines whether treatment is needed. Lumpy wastes are shredded and fed to a plasma melter, while rocks and soil are washed with triethylamine to remove organics and leached with nitric acid to remove actinides. Organics and actinides also go to the melter, where all organics are destroyed, and inorganics are melted into a non-leaching slag. Off-gases are cleaned before discharge. Process liquids are concentrated and recycled, while oxide particulate caught in the off-gas cleanup is recycled to the melter. Figure 1 is a process functional diagram.
Fig. 1.
The stabilized waste form made by the plasma melter is a synthetic basalt which easily passes the TCLP and meets INEL TRU waste acceptance criteria. This waste is containerized in fifty-five gallon drums for compatibility with long-term storage materials handling. The other solid output is soil and metal containing less than 10 nCi/g, meeting LDR disposal criteria, which will be placed back into Pit 9.
The treatment train avoids any liquid effluent, and the exhaust gases will comply with federal and Idaho air emission standards.
PIT SOLIDS RETRIEVAL

Retrieval will be performed within a double-walled building that spans the pit crosswise and that will be moved lengthwise on rails along the pit. A mobile gantry within the retrieval building supports all retrieval tools, which are remotely operated to prevent worker exposure to pit waste hazards. A negative pressure, oxygen deficient atmosphere is maintained in the retrieval building to prevent escape of contamination to the environment and to suppress combustion of any pyrophoric material uncovered.

To meet program completion objectives, it is planned to excavate 600 pounds per hour (30 cubic feet) of waste and 8000 pounds per hour (70 cubic feet) of soil. Since an average of 1000 pounds per hour of water mist will be sprayed at the dig face to suppress dust, 9000 pounds per hour of wetted soil will be excavated.

Soil less than one inch in size and waste with soil greater than one inch in size are separately excavated. The two separate material streams will be transported in steel boxes to/from the retrieval building. Since it is estimated that half the waste and half the soil will have less than 10 nCi/g TRU activity and therefore will not be processed, the retrieval subsystem will handle the return and emptying of boxed non-treated material to Pit 9 at the rate of 300 pounds per hour of waste and 4500 pounds per hour of damp soil. Clean rocks/metal at 90 pounds per hour and clean soil at 3900 pounds per hour will also be returned to the pit in this manner.

The retrieval building is a movable steel structure with three isolated layers of steel skin over a disposable internal liner. It fully contains all excavation work in a relatively small containment volume that is easy to control and decontaminate. The structure is moved along the pit as the dig face advances. Remotely adjustable segmented steel skirts, with sealing between segments, extend down to the bottom of the pit. The skirts control and protect the pit liners which are deployed to cover the cut surface as the retrieval building advances.

The block diagram in Fig. 2 shows retrieval and size reduction operations.

Fig. 2.

The proposed bridge crane retrieval system offers a unique advantage in that the operator can observe the nature of the waste as it is retrieved and call for more frequent use of the dig face monitor when suspect materials are encountered.

Retrieval of interstitial soil begins after survey with the dig face monitor. The two lowest bridges are fitted for this task after overburden removal is complete. Soil retrieval tools include modified street sweeper brushes which can clean down to the irregular basalt surface if necessary to retrieve waste that an excavator bucket would leave behind. Bristles with stiffness chosen to suit soil conditions scratch away the surface of the soil and sweep it through 1-inch slots. Loose items larger than an inch are swept past the slots and deposited behind the brush.

The tool cleans a path five feet wide but is built in eight-inch wide segments to allow it to conform to irregular surfaces. A shroud around the tool contains the dust. An air intake inside the shroud creates a negative internal pressure to scavenge fugitive dust. A fine mist is sprayed under the skirt to dampen the soil and help reduce the dust load in the air.

Soil, after being swept through the grating, is dampened for dust control and periodically transferred to a box for removal from the retrieval building. Waste items are retrieved by tools which have been used and refined on a DOE mixed waste retrieval project. Soil and waste are retrieved in alternating passes over the excavation face.

Items too large to be boxed are size-reduced by a portable shear. Sonsub used this approach extensively at Kerr Hollow Quarry. Pipes and other large items were sized in this manner during retrieval at the INEL cold test pit. The throat of the shear is 24 inches deep and opens 22 inches wide. The blade of the shear develops 110 tons of cutting force. The shear is suspended from its center of gravity. Controls on the remote console can be used to shift the shear from vertical to horizontal and rotate it about a vertical axis. The shear can therefore approach an oversized item from the most effective angle.

Loads of up to 15 tons can be lifted by a retrieval bridge. Items such as reactor vessel segments may be too heavy to lift and too thick to be size reduced by the portable shear. Any such items will be surveyed by an instrument on the maintenance and intervention system and left in place in the pit. If necessary, special tools and procedures can be developed and the gantry can return to process these items at a later date.

COUNTING, SHREDDING AND SEPARATION

This subsystem selects for processing boxes of soil and boxes of waste from Retrieval with greater than 10 nCi/g TRU activity, and returns boxes with less than 10 nCi/g TRU activity to Retrieval. This subsystem also shreds TRU waste to a manageable size and separates magnetic metal (mostly carbon steel) from other TRU waste to control the iron content of the final TRU waste slag product.
The TRU activity of boxed material is measured with a high-performance box counter. Boxes of soil and waste from Retrieval with less then 10 nCi/g TRU activity will be returned to Retrieval. Waste with greater than 10 nCi/g TRU activity will be shredded into pieces no larger than 4 inches size. This shredded waste stream goes to a high-intensity magnetic drum separator that removes more than 95 percent of the magnetic ferrous metal. Both the magnetic and non-magnetic shredded waste streams will be boxed and sent to Box Storage and Feeding, along with the boxes of soil with greater than 10 nCi/g TRU activity.
The boxes of treated rock/metal (from Pre-Leach Treatment) and processed soil (from Concentration and Conversion) are also measured for TRU activity to verify that these cleaned materials have less than 10 nCi/g TRU activity. Validated boxes of these materials will be returned to Retrieval, while boxes that fail the TRU activity test will be reprocessed.
Drums of treated material from the plasma melter are also returned to the counter for measurement prior to disposition.
Inside the transfer room, empty boxes are lowered into waiting carriers. Box lids are remotely unlatched and stay with the lid operators to maintain the boundary between the gantry interior and the transfer room as the boxes, inside box carriers, are lowered and then moved horizontally onto trolley cars. The box carrier keeps the outside of the box clean while it is in the retrieval area. This helps minimize the decontamination time and effort when filled boxes are transferred back to the truck. The box carriers stay in place to maintain the boundary between the gantry and the transfer room while a box is moved between the retrieval area and the transfer room.
Soil (material less than one inch in size) is retrieved and handled separately from the waste. Retrieved soil is dampened to prevent generation of dust and transferred to a conveyor which runs the length of the retrieval building. The conveyor moves the soil to a metering hopper at the west end of the retrieval building.
Each time it is cycled, the metering hopper dispenses a quantity of soil which fills a standard box not more than half full. As shown in Fig. 3, this transfer of soil from the hopper is done through a double lid transfer system which keeps the outside of the box clean.
Fig. 3.
CHEMICAL SEPARATIONS
These operations separate actinides from the soil so that essentially all the radioactivity in the material processed can be bound by the melter into a leach-resistant slag.
Pre-Leach Treatment
This subsystem treats a mixture of minus 1-inch soil and magnetic metal waste. It removes water and liquid organic waste from this mixture, cleans radioactive and heavy metal contamination from rocks and metal, and separates the clean rocks/metal, then separates floatable solid organics and +400 mesh soil, and generates a slurry of -400 mesh soil in acidic leach solution to the Leaching and Stripping subsystem. The system can process an average of 4500 pounds per hour. Soil and shredded magnetic waste is fed to a triethylamine (TEA) solvent wash. This EPA-approved BDAT process dissolves water and all liquid organics into TEA, separates the TEA/water/organics single-phase liquid from the soil/metals, and separates this liquid into TEA, water, and liquid organic streams. The TEA is reused, the water is reused as cooling or process water, and the oily liquid organics stream is sent to the plasma melter.
The dry solids stream is then mixed with nitric acid leach solution in a pre-leach tank. Agitation of the resulting slurry in the pre-leach tank scrubs radioactive and heavy metal particulates from the rocks and metal. The rocks and metal sink to the bottom of the pre-leach tank, and this underflow is washed clean of soil and leach liquor and fed into boxes as clean product material. Floatable solid organics rise to the top of the pre-leach tank and are removed there. Metal carbonates (primarily calcite) in the soil react with the nitric acid to form metal nitrates in solution.
The slurry of soil and acidic leach liquor passes through a sizing separator that creates a small quantity +400 mesh soil solids stream and a larger quantity -400

mesh soil slurry. This slurry is sent to the Leaching and Stripping subsystem, while the +400 mesh solids are combined with the float from the top of the pre-leach tank and put in boxes for feed to the plasma melter.
Chemical leaching and stripping is sufficient for removal of TRU metals from soil to below 10 nCi/g TRU activity.
Leaching and Stripping
This portion of the mixed waste remediation process is used to dissolve radioactive/heavy metal particulates into the liquid portion of the soil slurry from Pre-Leach Treatment, and to extract these dissolved metals from the soil slurry. About three hours of slurry residence time is provided at 80-90C temperature, sufficient for complete rad/heavy metal dissolution. The leached soil slurry is thickened, then diluted with metal nitrate solution to adjust the slurry chemistry for anion exchange. The slurry will then be thickened and diluted with clean water for cation exchange. The soil slurry, now free of rad/heavy metal contamination, and the contaminated ion exchange strip solutions will then go to Concentration and Conversion for further processing.
Concentration and Conversion
In this subsystem, contaminated ion exchange strip solutions are transferred from Leaching and Stripping into a concentrated radioactive/heavy metal sludge for plasma melter feed and a metal nitrate reagent for recycle to Leaching and Stripping. The treated soil slurry from Leaching and Stripping is transformed into a clean soil product, strong acid reagent for recycle to Pre-Leach, and fresh ion exchange strip solutions for recycle to Leaching and Stripping.
Spent strip solutions are concentrated by water removal and treated to make a rad/heavy metal precipitate, which is then separated from the resulting metal nitrate recycle solution. This rad/heavy metal sludge is closely monitored to ensure criticality safety. Clean soil slurry is thickened, calcined at about 400 C temperature to decompose metal nitrates to solid oxides and nitrogen dioxide off-gas, and the remaining solid is cooled with water to make a clean soil product which is boxed for return to Pit 9. Spent leach liquor, separated from the soil slurry, goes to Off-Gas Treatment and returns enriched in nitric acid. It is then concentrated by water removal to a strong (azeotropic) acid and metal nitrate solution. Strong acid is separated by distillation, leaving a metal nitrate sludge. The strong acid is recycled, and a little is mixed with clean water to make fresh anion strip solution. Some of the metal nitrate sludge is dissolved in water to make fresh cation strip solution. The remaining nitrate sludge, representing the metal carbonates originally dissolved from the soil, is returned to the soil.
PLASMA MELTER TREATMENT
Retech's unique plasma melter (1,2) transforms all material fed to it into glassy basaltic slag and off-gas. The primary chamber of the melter contains a centrifuge rotating at 10 to 50 revolutions per minute. The floor of the centrifuge has an axial throat used for periodic tapping of the molten slag into molds. Water and organics are volatilized in the refractory-lined primary chamber, and organic material is oxidized to carbon dioxide, water vapor and acid gases (mostly nitrogen oxides with some sulfur dioxide and hydrochloric acid vapor).
Wastes to be treated are fed to the melter at up to 1000 pounds per hour. The feed material will contain all organics extracted from the soil, all actinides extracted from the soil, all shredded waste and a small amount of INEL soil to ensure good leach properties in the slag formed. Any oxide particulate waste separated from the off-gas downstream of the melter will also be recycled into the melter.
A 1200 kW plasma torch with the arc current flowing between the torch and the rotating slag bath transfers energy to the bath to melt the input material and vaporize the water and organics. Oxygen is added to ensure complete oxidation of organics and desired slag properties. The processing chamber is a double-walled, water-cooled vessel containing the rotating crucible (centrifuge). The chamber has ports for feeders, off-gas viewing systems, pouring and the torch. The chamber is hermetically sealed and operates at 25 to 50 millibars below atmospheric pressure. A surge tank and appropriate valves ensure that the chamber pressure can never exceed the design pressure of 15 psig.
The water-cooled plasma torch mounts on the process chamber lid in a vacuum tight ball joint allowing the torch to tilt up to 15 about the vertical centerline, and permits lowering and raising the torch nozzle to and from the slag bed. Horizontal and vertical torch motion is by servo controlled hydraulic actuators. Torch motion

over the slag bed, its position for starting, material feeding and slag pouring will be pre-programmed. Remote manual positioning of the torch by the operator is permitted for the control of unusual events. A pylon provides repeatable positioning for automated removal and replacement of the torch from the chamber lid to a glove box for routine maintenance. All water, gas and hydraulic lines pass through the axis of the pylon such that torch removal repair and replacement is accomplished in a timely manner and under containment. This remote maintenance capability was demonstrated in the Lockheed Pit 9 Interim Action Proof of Principle Test for Maintenance-In-Containment (3).

The slag collection chamber receives and transfers the slag safety mold and the slag receiving drums. The safety mold is kept in place under the processing chamber throat during a melt, and is shuttled aside when a slag receiving drum is positioned for a pour. The slag collection chamber operates at a negative pressure and has air-lock valves for moving slag receiving drums in and out. This chamber is also ported to allow the safety mold to be moved out for emptying or as required. The safety mold is sized to accept an entire centrifuge load of slag (3 to 4 drums full) should there be a power failure or some other event that slows or stops the rotation of the centrifuge.

When the level of the molten slag in the centrifuge reaches the appropriate level, as determined by the operator, the slag safety mold is translated to the side and another mold with a slag receiving drum positioned in its center is positioned under the pour throat of the centrifuge, and raised hydraulically to effect a seal with the separation chamber.

At this point the rotation of the centrifuge is slowed in a pre-programmed manner so as to allow the molten slag to pour through the throat into the slag receiving drum. Sensors determine the fill level/rate and signal the hydraulic rotation mechanism to spin the centrifuge back up to operating velocity thus stopping the slag flow.

When the slag receiving drum is filled, it and the mold in which it rests are translated to the side and the empty safety mold is repositioned under the throat. The filled slag drum is moved through an isolation air lock valve into the slag storage system by conveyor. After cooling, the slag receiving drum and containment bag are sealed into a 55 gallon drum and transported by a manually operated, battery driven fork lift through the isolation air lock out of the melter bay. New, clean slag receiving drums are brought into the melter by conveyor through an isolation air lock.

Figure 4 is a cross-sectional view of the processing, separation and slag collection chambers.

Fig. 4.

Downstream of the melter chamber is a secondary combustion chamber (SCC) which ensures that off-gas temperature remains above 2000F for at least two seconds. At startup, the SCC is preheated with a propane/air flame for several hours prior to feeding any waste into the primary chamber. When starting a campaign, the plasma torch is also operated for a few hours to melt any slag remaining in the primary chamber prior to feeding waste.

OFF-GAS AND PROCESS LIQUID TREATMENT

Low oxygen off-gas streams, containing dust but no acid gases, are produced from the oxygen-deficient atmosphere in the shredder/separator enclosure and the TEA solvent wash process. The TEA wash process off-gas also contains noncondensable organic vapors separated from the feed soil. These gas streams are merged and dust is removed by a fabric bag filter.

Contaminants to be removed from the melter off-gas are mainly nitrogen oxides, plus some sulfur oxides, halogen acids and ozone. The two streams are merged, then dust, humidity, and heat are removed by a venturi wet scrubber and a condensing water separator. The dust slurry is thickened in a rotary microscreen filter, then mixed with dust from the bag filter to form a dust sludge for melter feed. Filtrate water supplies the venturi scrubber, with excess water (from humidity removal) supplying the plasma melter off-gas quench.

The gas stream is now treated by an acid gas removal sequence. A packed-bed caustic wet scrubber removes sulfur oxides and halogen acids, with the caustic soda blowdown going to Leach and Strip for acid neutralization. A noble metal catalyst bed then decomposes ozone and oxidizes nitric oxide to the nitrogen dioxide required by the acid scrubber. Oxygen is added to the gas stream to maintain the five-to-one oxygen-to-nitrogen-dioxide molar ratio required by the acid scrubber. This packed-bed

catalytic wet scrubber absorbs nitrogen dioxide and oxygen into spent leach liquor to recover nitric acid solution for concentration and conversion feed. Finally, the cleansed gas goes to Air and Exhaust Management for backup (safety) charcoal/HEPA filtration before return to the ambient atmosphere.
The concentration and conversion subsystems transform dilute contaminated strip solutions and weak strip solutions into fresh reagents for recycle and radioactive/heavy metal sludge for stabilization. This results in minimal chemical consumables cost and no liquid effluent.
SUMMARY
This integrated system will transform the contaminated buried waste and surrounding soil in Pit 9 into clean material and highly stable TRU waste. The treated waste form is very leach-resistant with excellent long-term durability. The estimated volume reduction of contaminated feed material is over 97 percent.
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37-35
SOIL*EXTM PROCESS DESIGN BASIS FOR
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ABSTRACT
Rust's patented SOIL*EXTM process is designed to process and treat difficult mixed waste forms including soils, sludges, and contaminated debris that contain hazardous and radioactive contaminants. The treatment of these waste forms is a national environmental priority since there are numerous sites contaminated with mixed waste that require safe, cost-effective remediation technologies.
SOIL*EXTM has been designed to remove radioactive and other heavy metal contamination while also removing and destroying any volatile organic compounds in the solid waste forms. The process consists of two subsystems labeled ACT*DE*CONSM and PO*WW*ERTM. ACT*DE*CONSM is designed to extract the contaminants from the solid waste forms using constituent-specific, aqueous-based chemistry. PO*WW*ERTM receives the extractant from ACT*DE*CONSM and concentrates the inorganic materials in the waste stream while destroying the volatile organic compounds (VOCs) in the vapor phase using catalytic oxidation.
The SOIL*EXTM process has been demonstrated on a pilot plant scale. This paper discusses the pilot plant test criteria and results along with the salient design features for the SOIL*EXTM system.
INTRODUCTION
The remediation of difficult-to-treat waste forms containing radioactive and hazardous constituents is one of the most challenging tasks confronting the American technology sector. Large volumes of soil and debris that are contaminated with radioactive and hazardous constituents exist at government and industrial locations. To answer this challenge, Rust International Inc. has developed and patented a unique, innovative mixed waste treatment scheme called the SOIL*EXTM process. The process removes the radioactive and hazardous metals from the solid waste while destroying the volatile organic compounds VOCs often associated with soil and debris.
The SOIL*EXTM treatment process incorporates constituent-specific, aqueous-based chemical extraction with solids separation, evaporation, and catalytic oxidation of the VOCs in the vapor phase. The configuration discussed utilizes the extraction

ACT*DE*CONSM chemistry in conjunction with Rust's demonstrated PO*WW*ERTM technology. In addition to destroying the VOCs, PO*WW*ERTM is used to concentrate the inorganic contaminants and subsequently reduce the extractant volume from ACT*DE*CONSM.

PROCESS OVERVIEW

Figures 1 and 2 are schematics for the ACT*DE*CONSM and PO*WW*ERTM subsystems which constitute the SOIL*EXTM treatment process.

Fig. 1. ACT*DE*CONSM Process Flow Diagram.

Fig. 2. PO*WW*ER Process Flow Diagram.

ACT*DE*CONSM basically consists of a multi-stage, counter-current flow extraction system equipped with continuous stirred tank reactors (CSTR) and conventional settlers for solid/liquid separation. Located at the "liquid" end is a membrane filter that removes the suspended solids from the extractant, which contains the dissolved actinide and heavy metal contaminants. The membrane filter is used to prevent the treated solids from flowing with the solubilized contaminants to PO*WW*ERTM. This process step minimizes the secondary waste volume produced in the evaporator for the PO*WW*ERTM treatment system. A conventional filter, located at the "solids" end of the ACT*DE*CONSM process, removes the treated solids from the system. The type of filter used depends upon the physical nature of the solids treated.

The ACT*DE*CONSM chemistry consists of a chelating agent, Ethyl-Diamine-Tetra-Acetic acid (EDTA), carbonate, and other solution conditioning agents. Since the chelant forms a stable ring structure with strong co-ordinate bonds to the metal contaminant, its presence assures that once the metals are dissolved they remain solubilized in the liquid phase and do not re-precipitate onto the solids. In cases when there are significant organic contaminants in the waste, a surfactant can be added to assist in extracting them from the solids.

The ACT*DE*CONSM process is a versatile mixed waste treatment system. The treatment process operates most efficiently on a continuous solids feed basis. However, it can be operated in a batch mode to clean debris such as metals and plastics. The mixed waste solids are fed to a hydropulper where a slurry containing 5 to 15 weight percent solids is produced. The hydropulper is a high shear mixing device that mechanically provides intimate contact between the contaminants in the solid phase and the aqueous chemistry. The slurry flows from the hydropulper into the membrane filter feed tank designed as a settler. The concentrated slurry is pumped from the bottom of the tank into the first CSTR. The extract containing the contaminants is fed from the tank to a membrane filter designed to remove virtually all of the suspended solids before it is sent to PO*WW*ERTM for volume reduction. The solids in the slurry flowing from the membrane filter feed tank are processed through a series of CSTRs and settlers. Each paired CSTR and settler is an equilibrium-stage operation. The solids from the previous unit or stage is mixed with the liquid from the succeeding unit, and the resulting mixture is sent to the settler where the solids are thickened to produce a 20 to 30 weight percent slurry. The underflow from the settler is then transferred to the next succeeding CSTR while the liquid in the settler overflow flows to the previous CSTR. As the liquid flows from unit to unit, it becomes enriched with the solubilized radionuclides and heavy metals, and as the solids flow in the opposite direction, they are depleted of the contaminants. The proficiency of the extraction depends upon the amount of chemicals and the number of equilibrium stages used. Since fewer equilibrium stages require more chemicals, a comparison study between operating expenses and capital costs should be made for each application. Given the required decontamination factors and operating conditions, the number of stages is determined using a McCabe-Thiele diagram. Any number of stages can be used, but the typical number is in the range of 3 to 6. Following the final extraction stage, the treated solids, i.e. the filter cake, are filtered and washed with water recovered from the PO*WW*ERTM process. Thorough filter cake washing is necessary to remove trace amounts of the solubilized contaminants from the solids.

The extractant, rich in contaminants, exits ACT*DE*CONSM 's membrane filter and is fed to the PO*WW*ERTM process. The dissolved metals and organics are pumped to a forced circulation, single- effect evaporator designed to vaporize the water and VOCs. The evaporator produces a concentrated waste product containing the radionuclides, inorganic salts, and heavy metals extracted by ACT*DE*CONSM. The evaporator overhead stream, containing the vaporized components, flows to the

catalytic oxidizer where the VOCs are destroyed to produce carbon dioxide and acid gas. The acid gas is removed from the vapor stream by a conventional scrubber located down-stream of the oxidizer. The salts formed from the neutralization of the acid gas are fed back to the evaporator and combine with the concentrated waste product. Following the scrubber, a condenser is used to recover the water vapor from the scrubber overhead stream. This water is reused as filter cake wash and for chemical makeup in the ACT*DE*CONSM treatment system.

PILOT-SCALE TESTING

Background

A pilot-scale SOIL*EXTM system was designed, fabricated, and tested at the Rust Clemson Technical Center (CTC). The pilot plant was fully instrumented with computer control. Computers were also used for real-time data acquisition and management. To preclude the need and associated expense for building a pilot plant capable of processing mixed waste containing transuranic components, the treatment process was tested using surrogate metals in the solids feed. The test program involved treating soils and oily sludges spiked with Neodymium (Nd), Thorium (Th), Manganese (Mg), and Bismuth (Bi). These metals mimicked the responses to the ACT*DE*CONSM chemistry for Americium (Am), Plutonium (Pu), and the hazardous metals Chromium (Cr), and Lead (Pb), respectively. The surrogate decontamination factors obtained in the pilot plant study were then compared to those established in a bench-scale surrogate validation test program. The pilot plant test program also included treating sludges spiked with VOCs, Trichloroethylene (TCE) and Carbon Tetrachloride (CCl4). These tests were performed independently from the surrogate metal tests to preclude the possibility of mixed waste production.

Test Criteria

The performance of the SOIL*EXTM pilot plant was measured using the following criteria: a) the treated solids, i.e. the filter cake from ACT*DE*CONSM, must represent at least 90 percent of the solids fed to the system; b) with an average waste feed activity of 100 nanocurries per gram (nCi/g) TRU equivalent, the treated material's activity must comply with applicable low-level waste acceptance criteria, i.e. the treated material's activity shall be less than 10 (nCi/g) TRU equivalent; c) the TRU waste product from PO*WW*ERTM, i.e. the concentrate from the evaporator, must satisfy applicable TRU waste acceptance criteria; and d) the on-stream or operability factor must be at least 60 percent.

Test Procedures

The pilot plant was operated and evaluated as a continuous, steady-state process. A computer simulation program, developed by Rust Engineering, was used to establish the time required to reach steady-state solids concentrations throughout the ACT*DE*CONSM process for each test operating condition. The effectiveness of the SOIL*EXTM treatment process was tested by tracking the fate of each hazardous and radioactive surrogate in the soil/sludge feed streams. Chemical equilibrium and performance were determined from scheduled sampling of the major process streams which were analyzed for the spiked contaminants and process chemistry.

Test Results

The pilot plant test program was successfully accomplished from July 1994 to November 1994. During this time, nearly 20 tons of soil and sludges were processed to demonstrate SOIL*EXTM 's ability to treat soils and sludges contaminated with mixed waste. Approximately 600 samples, 5000 analyses, and 17,000 pages of analytical data were produced in accordance to EPA Category II Quality Assurance Criteria.

A series of tests were performed to show that the treatment process is capable of meeting or exceeding the test criteria.

During Test 1, the waste feed to ACT*DE*CONSM included soil mixed with sludges containing only the VOCs. The test results demonstrated that all the VOCs volatilize in the ACT*DE*CONSM system since none were detected in the filter cake or the extract at a Minimum Detection Limit (MDL) of 0.005 parts per million (ppm). An addendum test was performed to demonstrate PO*WW*ERTM's effectiveness for destroying the VOCs. During this test, the extract feed to PO*WW*ERTM was spiked with TCE and CCl4 to concentrations corresponding to the most conservative values. The TCE and CCl4 feed spike concentrations of 500 and 1000 ppm, respectively, were chosen by assuming that all the VOCs are extracted into ACT*DE*CONSM's chemistry rather than volatilized. TCE was detected in some product concentrate samples ranging in concentration from non-detectable to 0.12 ppm. The maximum concentration is an order

of magnitude below the LDR standard of 5.6 ppm. CCl4 was not detected in any concentrate sample. Neither VOC was found in the PO*WW*ERTM off-gas system (MDL=0.139 mg/L) demonstrating the system's ability to destroy the compounds. The data clearly demonstrate SOIL*EXTM's ability to process and destroy volatile organics while producing a residual material (filter cake) meeting applicable Low-Level Radioactive Waste Acceptance Criteria and a concentrate meeting applicable TRU Waste Acceptance Criteria lacking RCRA hazardous characteristics for VOCs.
Tests 2 and 3 were designed to establish SOIL*EXTM ' s response to a waste feed containing the actinide and RCRA metal surrogates without the VOCs present. The treated material, recovered as filter cake, represented 96.5 percent of the test feed material. This result substantially exceeded the 90 percent threshold.
Table I lists the surrogate background concentrations naturally occurring in the soil along with the spiking and final concentrations in the treated material (i.e., filter cake) for each test. All values are reported on a dry weight basis. The results show that the ACT*DE*CONSM process successfully removed the surrogate constituents from the soil/sludge mixtures. The mean final Th and Nd concentrations for both tests demonstrate that the process was able to dissolve and remove the entire spike amount as well as a portion of the naturally occurring background constituent. In Test 2, the entire U spike was extracted from the solids matrix while Test 3 results indicated that 99.7 percent of the spike was removed.
Combining these test results with the decontamination factors authenticated during the surrogate validation tests, yields the TRU equivalent specific activities for the treated solids as listed in Table II. These activities, reported on a dry weight basis, range from 2.2 to 9.0 nCi/g with an average value of 4.7 nCi/g. This is less than 50 percent of the requisite 10 nCi/g assigned as the maximum activity for a low level TRU waste.
The TRU waste product generated in PO*WW*ERTM 's evaporator met the applicable TRU waste acceptance criteria. The volume reduction ratio, defined as the volume of extractant feed to PO*WW*ERTM divided by the volume of the waste product generated, averaged 30 to 1.
To demonstrate the pilot plant's mechanical dependability, a 100 Hour Operability Test was conducted. The test was designed to last for a maximum of 167 hours to achieve the 100 hours of operation. The system proved to be very reliable, and the test was completed in less than 108 hours of clock time, providing an availability factor of 93 percent. Six hours of the downtime were caused by filtration system failure. With the exception of the filter press, the SOIL*EXTM treatment system proved to be simple to operate and maintain throughout the test program. A significant advantage for using the SOIL*EXTM process is that the processing equipment used is commercially available from established vendors. This results in short downtimes for required maintenance and repair.

APPLICABILITY

As demonstrated in the pilot plant studies, the SOIL*EXTM process uniquely combines a highly selective dissolution for radionuclide and heavy metal removal from a solids media with a concentration/volume reduction technology for treatment of residues. This combination provides a state-of-the-art approach for treatment of hazardous and radioactive mixed waste. The waste can be in the form of sludges, soil, or debris. In addition, SOIL*EXTM is a simple, cost effective, low temperature process that can be modularized and installed at the contaminated site with minimal effort.
The treatment system can be readily modified to further diminish the secondary waste product volume. This is accomplished by either regenerating the chelant or removing it entirely from the extractant prior to PO*WW*ERTM treatment.
The SOIL*EXTM technology is ideally suited to treat wastes containing any combination of the contaminants listed in Table III.

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37-36
THE MODELING OF CONTAMINANT FLOW DURING PROPOSED TREATMENT OF U.S. DEPARTMENT OF ENERGY LOW-LEVEL RADIOACTIVE MIXED WASTES*

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ABSTRACT

Estimations of waste materials throughput and the potential radiological and chemical releases resulting from the proposed treatment of U.S. Department of Energy (DOE) low-level mixed wastes (LLMWs) were used to support analyses of risks and costs associated with various waste management alternatives outlined in the Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS). The modeling of material flow and contaminant releases through a consolidated waste management flowchart was performed by the WASTE_MGMT computational model developed by Argonne National Laboratory. This paper 1) briefly describes the process used to model estimated material and contaminant flow through the proposed treatment scenarios for the EM PEIS, 2) discusses the key site- and/or waste-stream-dependent factors involved in the determination of radiological and chemical emissions, and 3) explains the assumptions used to integrate the available LLMW database with the computational model.

INTRODUCTION

The November 1994 internal draft of the U.S. Department of Energy (DOE) Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS) addresses the environmental and health risks and the costs associated with managing DOE radioactive and hazardous wastes. An important part of this effort is the identification and quantification of the risks and costs resulting from the treatment, storage, and disposal (TSD) of DOE low-level radioactive mixed wastes (LLMWs). LLMW is considered both hazardous under the guidelines of the Resource Conservation and Recovery Act (RCRA) (Title 40, Part 261 of the Code of Federal Regulations [CFR]) and meets the definition of low-level waste (LLW) under the 1954 Atomic Energy Act (codified under 10 CFR, Part 61).

To support the EM PEIS effort to determine the risks and costs of LLMW management, Argonne National Laboratory (ANL) estimated source terms for chemical and radiological contaminant emissions and for the waste material flow during the proposed processing of LLMW. The source terms were estimated by applying waste input data derived from various DOE LLMW databases to a computational model that tracks the flow of materials and contaminants through the proposed waste-processing facilities. The model does not track contaminant emissions from waste disposal facilities.

The source term results from the model have been utilized in the overall EM PEIS program to 1) calculate chemical and radiological risks to site workers and the general public, 2) estimate waste-processing facility costs and resource requirements, 3) project the risks associated with transportation of LLMW and disposal products, and 4) provide comparative estimates of LLMW management activities across sites under various alternatives.

This paper briefly 1) describes the model used to estimate material and contaminant flow under the various proposed treatment scenarios for the EM PEIS, 2) discusses the key site- and/or waste-stream-dependent factors involved in the determination of radiological and chemical emissions, and 3) explains the assumptions used to integrate the available LLMW database with the computational model.

METHODOLOGY

The modeling of proposed LLMW treatment is based on a consolidated waste management flowchart (CWMF) developed by the Mixed Waste Treatment Project (MWTP) (1). The flowchart consists of a series of waste-processing modules common to all sites; the modules are designed to process the major LLMW streams identified in the 1994 Mixed Waste Inventory Report (MWIR-2) (2). Each module delineates one step in waste processing and may include one or more distinct process technologies. The complete treatment sequence for each waste type is called the waste "treatment train." The tracking of material and contaminant flow through the waste-processing modules identified in the CWMF is accomplished by the WASTE_MGMT computational model, an analytical tool that quantifies the throughput of LLMW and estimates the emissions of radionuclides and hazardous chemicals during the operation of proposed

waste-processing facilities (3).
The overall modeling process requires 1) the development of data on LLMW amounts, composition, and shipping routings that are used as inputs to the computational model; 2) determination of the waste-processing module operational parameters used as multiplication factors within the computational model; and 3) operation of the model.
The ANL WASTE_MGMT computational model is run on an IBM-compatible personal computer using the Microsoft FOXPRO relational database system (4). The model is run using information from three types of data files: a waste volume and contaminant characterization file, a waste-processing module characterization file, and a waste-shipping configuration file.
The computational model tracks the flow of contaminants through each treatment module through the entire treatment train for each waste treatment category. The model utilizes the volume partitioning for each treatment module and the densities of both primary and secondary waste-processing streams to establish a consistent mass balance flow through the entire treatment train for a given waste. The model conserves the waste input mass; added solidification materials (i.e., grout cement) are counted in the final disposal product, but added treatment materials (e.g., process water and incinerator fuel) are not tracked by mass.
In addition to partitioning by volume, the computational model also tracks the partitioning of radionuclide and chemical contaminants through treatment at each waste-processing module. At each step in the treatment train, a contaminant may partition into the air, liquid, or solid portions of the process. Summing over all the waste streams, the model calculates the mass (and/or concentration) of each contaminant in air, water, and solid (including final disposal product) discharges.
Five major types of LLMW are followed in detail in the model; aqueous liquids, organic liquids, solid process residues, soils, and debris waste. These wastes constitute over 90% of the inventory and projected generation volume of LLMW that will require waste processing as described in the EM PEIS. The current CWMF does not include the processing of the remaining 10% of special category LLMW (e.g., contaminated lead, beryllium, and certain reactive wastes). To facilitate the comparative analyses of waste management costs and risks at each site, the consolidated flowchart was assumed to be site-independent. A particular waste treatment train may contain as few as four or as many as 10 distinct processing modules; moreover, each module may be a part of the treatment sequence for several waste types. The integrated flowchart contains approximately 25 individual treatment modules. The CWMF applies to both Waste Management (WM) inventory and operations LLMW and to LLMW derived from Environmental Restoration (ER) activities. However, material and emissions source terms are modeled separately for the processing of WM- and ER-derived LLMW.
As wastes are transferred between modules in a treatment train, treatment processes may alter the volume and/or mass throughput and the chemical/radiological composition of the waste. The flow of contaminants through the model is driven by how the contaminant chemistry interacts with the specific operational conditions in each processing module. The movement of bulk waste mass is a function of the physical changes in material volume and density through each module. Thus, the flow of contaminants is independent of the bulk mass throughput in the flowchart. Changes in bulk mass between modules are computed by multiplying the processing-induced changes in waste volume by the expected changes in waste density. Changes in the contaminant mass between modules are calculated by multiplying contaminant-specific partitioning coefficients by the input mass of each contaminant. The partitioning of contaminants is dependent on the solubility and volatility of the specific chemical/radiological species, the waste stream matrix, and the operating conditions (e.g., temperature) of each module (technology).
LLMW INPUT DATA
Input data to the computational model consist of 1) site waste stream volumetric data, 2) radioactive waste handling characteristics, 3) waste-treatment-category-specific chemical profiles, 4) site-specific radionuclide profiles, and 5) the waste-shipping configuration that is dependent on the management alternative considered. This section briefly discusses the derivation and limitations of the model input data.
Approximately 128,000 m3 WM LLMW is in storage (1994 inventory) at 44 DOE sites; another 146,000 m3 of additional LLMW is expected to be generated through 2014. (An

additional 140,000 m3 of waste from the Hanford and Oak Ridge sites is considered as LLMW but was not modeled for the EM PEIS). Under the semi-restricted treatment case, ER-derived LLMW is expected to total approximately 7.3 million m3 through 2030 (5). Contaminant emissions are modeled separately for WM and ER LLMW. For WM LLMW, processing of the total waste volume (inventory plus generated waste) is annualized into an expected 10-year processing period. For ER wastes, the model calculates waste treatment conducted over 30 years.

LLMWs are organized into one of four radiological handling categories depending on the nature and activity of the radioactivity in the waste: contact-handled (CH) non-alpha, CH alpha, remote-handled (RH) alpha, and RH non-alpha. By far the largest volume of waste (>99%) is CH. In the computational model, CH wastes are tracked separately from RH LLMW. The type of radiation emitted also determines LLMW classification and affects emissions modeling. The majority of the modeled LLMW (approximately 75%) is non-alpha waste (less than 10 nCi/g transuranic radionuclide alpha activity). The remaining 25% of LLMW is alpha LLMW (between 10-100 nCi/g transuranic alpha activity) (over 99% of ER LLMW is projected to be CH non-alpha). Because of regulatory concerns about commingling of waste types, contaminant flow and emissions modeling are performed separately for alpha and non-alpha wastes.

For the purpose of the model, the majority of the 2,000 individual MWIR-2 waste streams were grouped into 23 general waste treatment categories (each category is a subset of one of the five major waste types described in the previous section) (6). The 23 waste treatment categories were grouped according to similar chemical/physical treatment characteristics, which allows for common waste processing. Some of the more significant LLMW treatment categories include aqueous waste waters (acidic, basic, neutral, or cyanide), halogenated organic liquids, non-halogenated organic liquids, inorganic particulates and sludges, salt wastes, halogenated organic particulates/sludge, non-halogenated organic sludge, contaminated soils with debris, soils without debris, inorganic non-metal debris, metal debris, combustible debris, heterogeneous debris, and various lab pack wastes. Each waste treatment category follows a distinct treatment train.

Estimation of emission source terms for LLMW requires quantification of the composition and concentration of hazardous chemicals, radionuclides, and certain physical parameters (e.g., ash content) in each waste stream at a site. However, because the current empirical information is insufficient to develop complete chemical/radiological profiles for each waste stream/site combination, the model uses data that assume 1) site-independent chemical profiles based on the generic waste treatment category and 2) site-based radiological profiles that are independent of the particular waste treatment category. The development of LLMW chemical/radiological profiles is detailed in other documents (6, 7, 8).

Chemical profiles for the 23 treatment categories of WM-derived CH non-alpha LLMW were developed from compilation of the chemical concentration data presented in MWIR-2 and an engineering assessment of the industrial processes that generated the respective LLMW streams. Hazardous chemical profiles for ER LLMW were primarily derived from compilation of chemical data from the secondary waste streams expected to be sent to WM for treatment as outlined in the Automated Remedial Assessment Methodology (ARAM) database provided by Pacific Northwest Laboratory (PNL) (5, 9). Information about historical DOE site operations, industrial processes, and waste generation, suggests that the waste streams of a given treatment category classification at sites across the DOE complex are similar. Thus, the model input data assume that chemical profiles for both WM and ER-derived LLMW are waste-treatment-category-specific and are independent of the site that generated the waste. To provide a consistent assessment of chemicals that is comparable across waste treatment categories, the over 100 individual chemical species identified in DOE LLMW were condensed into 16 distinct chemicals and/or grouped into classes of chemicals that are common to most waste streams. These include six toxic metals (silver, barium, cadmium, chromium, lead, and mercury); three inorganic chemicals (arsenic, selenium, and cyanide); and seven classes of organic chemicals that are grouped according to important treatment parameters such as density, solubility, volatility, and degree of chlorination. These classes are 1) acetone, butanone, and methanol; 2) toluene, xylene, and benzene; 3) trichlorethanes; 4) tetrachloroethanes; 5) dichloroethanes; 6) methylene chloride; and 7) chlorofluoro-hydrocarbons.

The input ash content of a waste stream is related to the amount of particulate

emissions from the incineration treatment module. The input ash content is an important component of the LLMW compositional profile because most air releases of hazardous heavy metals and non-soluble radionuclides tend to follow the flow of particulate emissions through the model.
Radiological profiles for LLMW were estimated from the information about the concentrations and activities (Ci) of various radionuclides in LLW compiled in the Integrated Data Base (IDB) (10). The data on LLW radiological profiles were applied to LLMW because it was assumed that the particular radionuclide profiles of LLW and LLMW at a site are dependent on the site operation(s) that generated the radionuclides. The IDB outlines five radiological source profiles that represent the groupings of radionuclides according to the source and time of generation: 1) fission products, 2) induced activity, 3) uranium/thorium, 4) transuranics, and 5) tritium. Each of the DOE sites has a distinct proportion of these five radiological source profiles that is dependent on the site's operational history. A single mission site would be expected to have a radionuclide profile derived primarily from one radiological source category (e.g., uranium-235 concentration facilities like Portsmouth or Paduch have radiological profiles derived almost 100% from the uranium/thorium source profile). Multipurpose sites like Oak Ridge with uranium-235 concentration facilities, reactor operations, and specific isotope production would have an estimated profile that is a proportional combination of the five source profiles (e.g., for Oak Ridge, 30% fission products, 1% uranium/thorium, 1% tritium, and 68% induced activity).
The relative intensities (activity per unit volume in Ci/m3) of LLMW radionuclides were adjusted from the baseline LLW profiles by "aging" the radionuclides to account for the greater age of LLMW at the assumed time of treatment relative to LLW data in the IDB. The age of LLMW radionuclides is site-specific and depends on the site operational history and whether the LLMW is in current inventory or has yet to be generated. The model uses LLMW radiological profiles aged to the median proposed treatment year of 2008.
The ER LLMW radionuclide profiles correspond to those used for WM LLMW with two adjustments: 1) the mean time for generation of radionuclides in all ER wastes was assumed to be 1965, which would produce a different adjustment to aging for the ER wastes compared with most WM LLMW at each site, and 2) the intensity of ER LLMW was assumed to be 10% of the intensity for the equivalent WM LLMW at the site. Both the increased time for radioactive decay and dilution of ER wastes by nonradioactive external materials, such as soil and old packaging materials, account for the lower intensity.
Input volumes of LLMW were also adjusted according to one of seven waste-shipping configurations described in the EM PEIS. Each shipping scenario represents a treatment alternative that delineates the sites for LLMW processing. The alternatives range from each site treating its own waste to requiring shipment of all LLMW to the Hanford site for processing. The input information on LLMW volumes, chemical/radiological profiles, and radiation treatment category was adjusted according to the shipping configuration prior to input to the computational model.
FLOWCHART MODEL
The CWMF covers five basic processes: front-end waste handling, pretreatment, primary treatment, secondary treatment, and tertiary (final disposal form) processing (Fig. 1).
Fig. 1.
The waste input is represented by the five waste types (and their 23 subset waste treatment categories). Each of the processing steps consists of one or more modules. Each module may contain one or more specific process technologies. A particular series of modules represents the treatment train for a specific waste treatment category. The arrows in Fig. 1 represent the contaminant/waste mass flowpaths through the CWMF. These generic flowpaths apply across sites and for each of the 23 waste treatment categories. The bulk mass partitioning factors along the flowpaths were determined separately for each waste treatment category. The contaminant partitioning coefficients along the flowpaths were determined by how module conditions affect each contaminant and are not dependent on the waste treatment category.
As currently proposed, waste-processing options rely on flame treatment (incineration) for most combustible wastes and grouting as the preferred final waste disposal form. Except for a few existing facilities (e.g., the Liquid Effluent

Treatment and Disposal Facility at the Idaho National Engineering Laboratory [INEL]), the processing facilities outlined in the flowchart have yet to be designed. Specific facility operational parameters will be designed on a site-by-site basis. For the computational model, common module operational parameters are used for each generic module, irrespective of site. The flowchart and model are flexible enough to allow for modification of individual waste-processing modules as new waste data, technology information, or DOE waste management guidance are introduced.

The assumed operational conditions of each waste-processing module used in the CWMF are based on empirical data and engineering assessments of similar existing technologies. Proportional waste volume fractions (volume splits) for each module were provided by the MWTP. Waste densities were estimated from data in MWIR-2, along with engineering judgement based on comparison of LLMWs with wastes generated in similar industrial processes. The chemical/physical partitioning factors are specific for each of the 16 chemical species and depend on the technology used in the module, the waste stream matrix, species solubility/volatility, destruction efficiency, and the residual media (i.e., air emissions, water releases, or incorporation into solid product). For example, the air residual of a particular species (e.g., tri-chlorinated organics) is different in the incinerator module (9.90 E-08 of input is released) than in the wet oxidation module (1.0 E-06 of input). In this case, the factors differ as a result of the effect of the higher incinerator operational temperatures on species volatility and destruction (i.e., the incinerator destroys 99.99% of the tri-chlorinated organics compared to a 99.9% destruction for wet oxidation). A list of the module operational parameters is given in Wilkins et al. (6).

The initial waste mass inputs were determined by multiplying the LLMW volume data with the empirical waste stream gross densities derived from wastes at INEL. The INEL data were used because they represent the most complete information currently available on LLMW densities.

ASSUMPTIONS/UNCERTAINTIES

As described in this paper, two primary factors contribute to uncertainty in the LLMW processing model: 1) the existing DOE data on LLMW are incomplete, and 2) the operational requirements for the CWMF remain more conceptual than specific. To estimate emissions and waste flow source terms, several assumptions were applied to the modeling process. The primary purpose of the assumptions is to provide consistent baseline data that, when run through the computational model, furnish comparative source term results across the DOE sites. The more important assumptions and some of the uncertainty introduced by each are reviewed in this section.

The input data assume different origins for the chemical and radiological contaminant profiles for the LLMW waste streams; chemical profiles are waste-treatment-category-specific, while radionuclide profiles are site-specific. In reality, each of the 2,000 LLMW streams may vary in chemical composition by site and in radiological composition by waste treatment category compared to the generic profiles; however, the current incomplete data do not allow for quantification of these potential differences. In the future, more refined LLMW input data may alter modeled emission output at each site.

The condensing of over 100 chemical contaminants into 16 distinct chemical species and/or chemical groups may oversimplify the differences inherent in how each chemical partitions in a given waste-processing module. For a waste that may contain an unusual species of a given chemical, the model may not provide accurate output data. For example, barium, one of the six toxic metals, is assumed to be water insoluble throughout the model; however, if a particular waste actually contains a soluble form of the chemical (e.g., barium chloride), the partitioning between air and water fractions could be different than that modeled. The generalizing of related compounds into chemical groups can also introduce uncertainty. The model uses volatility and solubility coefficients that represent averaged values for a group of related compounds (e.g., four-chlorine halogenated organics). If a waste contains a preponderance of one particular species in the group, and if the volatility/solubility values of that species differ from the assumed group averages, the model may give an inaccurate emissions output. With more complete data on LLMW, the model could be refined to account for such differences.

The assumption that waste input density adjustments for all sites are based on data from INEL may introduce uncertainty if the waste composition and, thus, densities of

the equivalent LLMW from other sites vary from the INEL data. Because this density adjustment is applied at the front end of the computational model to determine waste input mass, inaccurate density values would affect the mass flow throughout the model. At present, not enough data are available from other sites to make a consistent refinement of the input density adjustment.

The model only tracks the mass balance of input contaminants; emissions from the addition of treatment chemicals are only minimally quantified. For example, the amount of fuel (e.g., fuel oil) needed in the incinerator module is dependent on the heat content of the input waste. The amount of added fuel will affect the emission source terms of such gases as sulfur dioxide (SO2), nitrogen oxide (NOx), and carbon monoxide (CO).

Tracking the flow of contaminants independent of the bulk waste mass flow works best when the contaminant mass is a small percentage of the waste mass. For the majority of non-special category LLMW, the contaminant mass constitutes less than 1% of the bulk mass (for inorganic contaminants the value is usually less than 0.01%; certain organic waste streams have organic contaminants concentrations that exceed 1%). A high (greater than 10%) contaminant concentration (especially of an inorganic contaminant that is not destroyed in the treatment process) in a waste stream would result in an inaccurate mass balance through those modules where the chemical partitioning factors differ from the bulk mass partitioning. This would not significantly affect the modeling of the contaminant emission source terms; however, the sum of the contaminant emission mass and the bulk mass output would appear to be greater than the waste input mass. Currently, wastes with known high concentrations of inorganic contaminants (e.g., lead bricks and beryllium wastes) are not run through the model. High concentrations of organic contaminants are less likely to upset the model mass balance because these contaminants are largely destroyed in the treatment process.

The LLMW radiological profiles were assumed to be similar to the profiles for corresponding LLW at each site (adjusted for the age of the waste), as adapted from the IDB (10). This assumption is probably reasonable for the radionuclide profile of LLMW. However, the assumption that the initial intensities (in Ci) of the two waste types are the same may be less accurate. If the initial radiological intensities differed between LLW and LLMW, the level of activity for each radionuclide projected for the model's air emissions, water releases, and residual solids would differ from the current result.

CONCLUSION

The modeling of contaminant and waste material flow and the estimation of emission source terms are important in the effort to understand the risks and costs associated with the proposed alternatives to manage DOE LLMW. The modeling of waste and contaminant throughput by use of the WASTE_MGMT computational model is one possible method of estimating emission source terms. Clearly, current model results are uncertain because of the assumptions applied to fill in gaps in the existing LLMW data and the conceptual (as opposed to designed) nature of the proposed waste-processing flowchart. Since the proposed waste-processing operations will not be fully realized for several years, changes in management alternatives and regulatory requirements can also affect the validity of the currently modeled data. The modeling process as outlined here is dynamic and can be adapted to future changes in LLMW data, waste-processing module operational criteria, and management alternatives.

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BENCH- AND PILOT-SCALE DEMONSTRATION OF THERMAL DESORPTION FOR REMOVAL OF MERCURY FROM THE LOWER EAST FORK POPLAR CREEK FLOODPLAIN SOILS

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ABSTRACT

Thermal desorption is an innovative technology that has seen significant growth in applications to organically contaminated soils and sludges for the remediation of hazardous, radioactive and mixed waste sites. This paper will present the results of a bench- and pilot-scale demonstration of this technology for the removal of mercury from the Lower East Fork Poplar Creek floodplain soil. Results demonstrate that the mercury in this soil can be successfully removed to the target treatment levels of 10 milligrams per kilogram (mg/kg) and that all process residuals could be rendered RCRA-nonhazardous as defined by the Resource Conservation and Recovery Act. Sampling and analyses of the desorber off-gas before and after the air pollution control system demonstrated effective collection of mercury and organic constituents. Pilot-scale testing was also conducted to verify requirements for material handling of soil into and out of the process.

This paper will also present a conceptual design and preliminary costs of a full-scale system, including feed preparation, thermal treatment, and residuals handling for the soil.

INTRODUCTION

Thermal desorption is being considered by the U.S. Department of Energy (DOE) as a potential treatment process for floodplain soils from the Lower East Fork Poplar Creek (LEFPC), a tributary which drains the Y-12 site in Oak Ridge, Tennessee. The LEFPC soils are being addressed under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA); the technology demonstration was in support of the feasibility study to select an appropriate remedy. Successful treatment would provide the opportunity for considering replacement at the site.

Thermal desorption is a low to moderate temperature separation process that removes contaminants from soil and other solids. It has been proven on a wide variety of organic chemicals, including polynuclear aromatic hydrocarbons (PAH), polychlorinated biphenyls (PCB), chlorinated pesticides, and others having low volatility (1). The technology can also be applied to solids containing mercury, due

to mercury's relatively high vapor pressure, although experience in this application has been very limited. Thermal desorption consists of an indirectly heated desorber unit and an off-gas treatment system, which collects the volatilized contaminants and water from the soil using condensation, scrubbing, adsorption and other conventional control devices. For treatment of mixed waste, thermal desorption offers the ability to separate the hazardous chemicals (RCRA) or toxic constituents (Toxic Substances Control Act [TSCA]) from the radioactive constituents, which remain in the bulk solids (2).

Based on the positive results of bench-scale treatability testing (3), Martin Marietta Energy Systems subcontracted with IT Corporation (IT) to conduct pilot-scale thermal desorption demonstration tests. The scope of work was to evaluate the various major process components of a complete treatment facility through a test program that confirmed that cleanup goals could be achieved, determined that desorbed contaminants could be effectively collected/controlled, and provided key scale-up data for preparing a conceptual full-scale design and cost estimate.

SOIL CHARACTERISTICS

LEFPC soil was excavated by hand from an area that had been identified during extensive site assessment to have the highest concentration of mercury (3). Ten drums of soil were shipped to IT's Environmental Technology Development Center in Oak Ridge, Tennessee. The soil was prepared for testing and analysis by sieving to remove oversize material, partial air drying, and thorough manual blending. This preparation is routinely done for treatability tests to ensure a uniform feed material for all testing, and to achieve material handling characteristics suitable for the pilot desorber system.

Chemical analysis included selected metals, volatile and semivolatile organics, and PCBs. Mercury concentration of the composite sample was 283 mg/kg. Eight grab samples were analyzed for mercury to show that the soil was well homogenized; the relative standard deviation was 11 percent. Trace levels of various PAHs were identified; these were the only semivolatile organic compounds (SVOC) detected. Arochlor-1260 was the only PCB detected. No volatile organic compounds (VOC) were detected. Total activity was 1.5E-4 microcuries/gram and total uranium of 48 picocuries/gram. The total organic carbon (TOC) value for the soil was 4.7 percent. Physical properties were also measured since these affect materials handling as well as thermal processing. The prepared soil particle size distribution showed approximately 60 percent silt, 20 percent clay (less than 2 micron), and 20 percent fine sand. The moisture content of the soil as received was 24.4 percent; the Atterberg limits were 32 for the liquid limit and 50 for the plastic limit. Air drying of the soil in the trays for several days resulted in the moisture dropping to about 15 percent for use in the pilot tests.

BENCH-SCALE THERMAL DESORPTION TESTING PROGRAM

Bench-scale testing was performed to confirm the effect of primary treatment variables on reduction of mercury. The tests also enabled an assessment of the characteristics of the residuals collected from the off-gas, which was an important step in preparing for the pilot tests. The treatment goals for the bench-scale tests were to reduce mercury to 10 mg/kg; other constituents were not investigated.

The test program involved two types of thermal desorption systems: the static tray test and the rotary thermal apparatus (RTA). Testing began with the RTA because previous laboratory thermal desorption testing directed by Martin Marietta Energy Systems (3) provided a basis for selecting soil temperature and residence time values. The first series of RTA tests were to determine the effectiveness of 350C for removing the mercury from the soil. The second series of tests were for investigating, at one residence time, the effect of air, steam, and nitrogen as different purge gases. The third series consisting of four tests was to determine the effect of particle size of the RTA feed on mercury removal. The feed soil was screened through wire cloth to produce a very coarse (0.5- to 1-inch) material. This material was treated in the RTA at the same conditions as the fine soil. The final series of RTA tests were conducted at selected conditions to fill in specific additional data needs.

The original test plan was modified because residual mercury concentrations were above the 10 mg/kg goal. Supplemental static tray screening tests were performed at a range of temperatures from 350oC to 650oC. Ten tests were conducted; all used nitrogen as a purge gas, and nine used a time-at-temperature of 10 minutes.

Bench-Scale Equipment Description

Tray Desorption Test Apparatus - IT's "tray test" apparatus is identical to that described in EPA's 1992 guidance document (4). Typically, a 30- to 100-gram (g) sample of soil is spread in a thin (2-3 millimeter [mm]) layer in an open metal tray, which is placed in a standard muffle furnace. The temperature is rapidly adjusted to the desired desorption condition, while a purge gas is passed through the furnace interior. Temperatures of the soil layer and furnace interior adjacent to the tray are measured.

Rotary Thermal Apparatus - IT's RTA consists of a batch-loaded, indirectly heated rotating tube and associated off-gas collection/treatment train. The rotating tube portion of the system is 12.7 centimeter (cm) in diameter by 30.5 cm long; it rotates at five revolutions per minute (rpm) and is externally heated with a custom-made electrical oven. Temperatures of the soil bed, the gas in the rotating cylinder, the oven, and off-gas system temperatures are recorded. The off-gases (purge gas, water vapor, and desorbed contaminants) from the cylinder exit through a heated quartz tube into a collection/treatment system that is configured according to the chemicals being desorbed and the data objectives of the testing. Typically, the off-gas first enters a heavy glass-walled spray tower scrubber where a caustic solution is recirculated through the spray tower. Uncondensed gases passing the spray tower are pulled through a carbon bed by an air-aspirated venturi pump before being discharged into a high-efficiency particulate air (HEPA)-filtered laboratory fume hood.

Bench-Scale Thermal Desorption Results

The results for the RTA tests are summarized in Table I. Only one test achieved a residual mercury concentration below 10 mg/kg; this test had the longest residence time at temperature (1 hour). There was no significant difference between all the RTA results considering the typical variability to be expected in sampling and analytical accuracy for multiple tests from the same prepared soil batch. Residence time (between 10 and 20 minutes) and purge gas type did not have a definite influence on mercury removal. The oxygen concentration in the RTA vent gas from the first four experiments was higher than expected due to leakage through the seals, which were replaced; this variance is noted in the table (*). The use of nitrogen and steam purge reduced the oxygen level to approximately 2 to 5 percent for Tests 5 and 7. These results are typical of what could be achieved in a full-scale desorber with an inert gas purge.

The tray test results were comparable to the RTA results. Residual mercury values ranged from 16.1 to 6.3 mg/kg over the range from 350 to 650oC. All the results were below 10 mg/kg at temperatures above 400oC. Increased temperature appeared to have a minor influence on residual mercury values, although a definite correlation could not be established due to the limited number of tests and the small range of mercury values. Increased residence time did not have a definite effect on treatment.

PILOT-SCALE THERMAL DESORPTION TESTING PROGRAM

The pilot program consisted of several tasks that were conducted separately. Initial objectives were as follows:

- Reduce mercury in the soil to below 10 mg/kg.
- Demonstrate accountability for mercury throughout the process.
- Generate treated soil that is RCRA nonhazardous.
- Demonstrate the effectiveness of off-gas treatment.
- Minimize residuals and treat any RCRA residuals.
- Evaluate soil material handling/preparation characteristics.
- Obtain sufficient engineering data to enable full-scale conceptual design and budgetary costing for use in feasibility study.

After the bench-scale desorption test results had been evaluated, the first objective was modified to demonstrate less than 20 mg/kg residual mercury. It was decided by the project team that it was preferable not to increase the desorption temperature substantially in order to achieve a residual mercury concentration of 10 mg/kg because this increase was expected to have a deleterious effect on the treated soil properties relative to replacement/reuse at the site.

Desorption Pilot Test Plan

A series of three pilot-scale tests were performed on LEFPC soil. The first test was to establish operating conditions for the desorber in order to obtain the throughput and treatment goals desired; soil residence time and temperature were selected based on the RTA results.

Test 2 was the official verification test, performed at 350oC. The operating conditions developed during Test 1 were used to treat 330 kg of soil. Samples of soil feed, treated soil, desorber off-gas, scrubber liquids and solids, activated carbon, demister pad, and final vent gas were sampled and analyzed for various chemical parameters including mercury, selected other metals (e.g., cadmium, arsenic), VOCs, SVOCs, and PCBs. Treated soil was also analyzed for chlorodibenzodioxins/chlorodibenzofurans (CDD/CDF). Scrubber water was analyzed for total organic carbon and pH. Off-gas and vent gas were also sampled and analyzed for particulate, acid gases, ammonia, oxygen and carbon dioxide.

Test 2 was also used to generate sufficient soil for two other types of subsequent testing. Forty-five kg of treated soil was used by the project team to evaluate its characteristics relative to being replaced in the floodplain (e.g., ability to support plant growth). Another portion of treated soil (197 kg) was used for back-end material handling tests.

Test 3 was performed at a higher desorption temperature of 600oC; all other operating conditions were the same. This test was to demonstrate that the improved treatment to below 10 mg/kg mercury that was measured in the tray tests could be accomplished at the larger scale. The sampling and analysis program was similar to Test 2 with the exception of the gas sampling, which was limited to Orsat measurements of oxygen and carbon dioxide.

Table II lists the key process parameters for both the 350oC verification test and the 600oC test.

Materials Handling Tests

The materials handling test program consisted of physical property measurements, feed chute drop tests, reagglomeration tests, and wet drag flight conveyor tests. The physical property tests were performed to assess handling characteristics such as adhesion, cohesion, and properties important to the design of soil preparation and feed systems. The "as received" soil was tested for initial moisture content and Atterberg limits.

Lime addition, calcium hydroxide, was also evaluated in an effort to produce a material that had less tendency to agglomerate and stick to surfaces. An addition of 1 and 2 percent lime by weight to two soil samples caused no significant change in Atterberg limits. Therefore, lime addition was not pursued as a method of preparing the soil feed for the thermal desorber.

The feed chute drop tests were designed to evaluate the tendency of the feed soil to stick to surfaces upon impact. The simplest method of introducing solids to a rotary desorber is by dropping the material onto an inclined chute, which directs it onto the soil bed. A pilot chute system was set up to test the effect of moisture content and chute angle on the degree of soil deposition on the chute surface. Moisture content was varied from 17 to 39 percent by adding water to the prepared soil batch; chute angle was varied from 45 to 71 degrees. A drop height of 13 feet was used for all tests; this was considered representative of a full-scale system.

The reagglomeration test was designed to observe any tendency of the soil to reagglomerate from tumbling and mixing in the rotating desorber. Soil was placed in a rotating drum and observed to see if a ball or "sausage" was formed. Soil moisture content was varied from 9 to 40 percent to determine at what point agglomeration began and at what point soil stuck to the rotating surface. Both phenomenon were considered detrimental to proper desorber performance.

To simulate a wet quench system with drag-flight conveyor, a slurry of treated soil from the 350oC verification test was prepared in a shallow tank. An inclined metal plate was placed in the tank at a 35 degree angle as a "dewatering beach"; the bottom section was submerged but the majority of its length was above the water level. A modified garden hoe was used to pull wet soil up the dewatering beach at a rate of 3.0 meters (m) per minute. The angle of the beach and the drag rate and corresponding drainage time were selected to be representative of a full-scale system. The dewatered soil was stockpiled and weighed. The amount of soil retained by the "drag flight" and its moisture content were measured. The test was repeated until sufficient drained material was collected to form a pile for determination of the angle of repose before and after vibration. This characteristic is important in determining the practicality of stockpiling the wet soil prior to transport. The test was repeated using a longer drainage time (1 versus 0.3 minute). Two sets of additional tests were performed using higher water-to-soil ratios.

Residuals Treatment Tests

All liquid and solid residuals generated from the pilot testing were evaluated for RCRA hazardous characteristics (toxicity characteristic [TC]), both to determine if this was an issue for applying the technology and to enable return of these materials to Martin Marietta Energy Systems. If a residual had a constituent exceeding TC limits, it was treated in an appropriate manner based on initial bench-scale testing. For the liquid residuals, a treatment system was assembled consisting of a modified open-top drum, stir paddle, diaphragm pump, and specially made laboratory glass filtration vessel. Treatment chemicals were weighed, added to the tank, and blended manually. The pH was monitored continuously using a meter. The final settled solids "flock" was dewatered using a two section glass vacuum filtration vessel with filter media pad. Solid residuals from the desorption tests that contain mercury above toxicity characteristics leaching procedures (TCLP) limits were treated in the RTA or by acid rinsing in a small vessel.

Pilot-Scale Thermal Desorption System

The pilot testing of LEFPC soil was performed at IT's Environmental Technology Development Center in Oak Ridge, Tennessee. The pilot thermal desorption system has been used for dozens of thermal desorption tests beginning in 1985 (5,6,7). In addition to thermal desorption tests, pilot-scale materials handling and residuals treatment testing was performed. This section describes the pilot facilities, with emphasis on the thermal desorption system.

IT's pilot thermal desorber process is essentially a scaled-down version of a full-scale system. The pilot system has in past studies given comparable results to bench-scale tests. The thermal desorber is a standard-design pilot calciner that consists of a continuously rotating tube partially enclosed with a gas-fired furnace shell. A flow diagram of the pilot system is shown in Fig. 1.

Fig. 1.

The tube, constructed of Castalloy H-H, has a 16.5-cm internal diameter and is 4.3 m long; the heated section is 2.3 m long. Small flights are placed at intervals within this tube to provide soil agitation. A stationary thermowell extends from the discharge end into the tube, with six thermocouples to monitor the soil temperature and three to monitor the gas temperature along the tube length. The soil residence time is a function of the soil bed volume and the soil feed rate. Soil bed volume is controlled by the rotational speed and inclination of the desorber tube. The soil feed rate is controlled by the rotational speed of a feed-screw conveyor. The furnace is a refractory-lined chamber with 14 equally spaced burners (propane) controlled by a standard burner control system with appropriate safety features. Burner firing rate is manually controlled to give the desired soil temperature profile.

The discharge end section of the desorber tube is enclosed by an air-cooled casing. Soil exits the desorber tube through a gas plenum/transition section and a rotary air-lock valve into a metal receiver can. Purge gas is introduced at a low flow rate at the soil discharge end to help flush desorbed contaminant vapors and to maintain the proper atmosphere for the treatment process. The off-gas flows from a plenum at the soil feed end of the desorber containing the purge gas, volatilized contaminants and water, and some particulates. The off-gas is transferred through a short electrically-heated, insulated duct to the off-gas treatment system.

The off-gas treatment system was comprised of a hot cyclone, a spray scrubber with demister pad, a wet electrostatic precipitator (WESP), a chilled, noncontact condenser, a mist eliminator, a HEPA filter, a two-stage carbon adsorber, and an induced-draft fan.

Pilot-Scale Test Results - LEFPC Soil

Thermal Desorption Results, Soil and Residuals - A summary of the analytical results for mercury for both the 350oC and 600oC verification tests is given in Table III. Mercury was reduced to approximately 14 and 8 mg/kg for the 350oC and 600oC desorption temperatures, respectively. A series of temporally related feed and treated soil mercury results showed a relative standard deviation of 20 percent or less.

Table III shows the accountability/distribution for mercury throughout the system. Mercury was recovered primarily in the scrubber system; it concentrated in the scrubber solids and as elemental (liquid) mercury. During inspection of the off-gas treatment system after each test, elemental mercury was found throughout the scrubber system. The greatest amount was in the demister pad located on top of the scrubber. Elemental mercury was recovered from the demister pad and scrubber system

at 12 and 13 g for the two tests, respectively. A small amount of floating semisolid material collected from the scrubber system tank for the 600oC test had a very high concentration of mercury. The elemental mercury values in the table represent the total combined liquid mercury isolated from all sources, including as a result of residuals treatment to achieve nonhazardous criteria. The carbon adsorber also collected some mercury. The data for 600oC test show less mercury collected in the scrubber and more collected in the carbon. In fact mercury broke through the primary adsorber, as indicated by the separate analysis of the primary and secondary carbon. Approximately 45 percent of the total mercury (95 and 70 g) calculated to be contained in the feed soil for Tests 2 and 3, respectively, was accounted for in all the process output streams. Despite very thorough removal of all liquid and solid materials from the scrubber system, the accountability was low, presumably due to the size (surface area) and complexity of the pilot system and the tendency of the condensed mercury to spread and adhere to surfaces.

Organics found in the feed soil (PAHs and PCBs) were reduced in the treated soils to below the analytical method detection limit. Several other SVOCs, mostly tentatively identified compounds (TIC), were found in the treated soil from the 350oC test at concentrations below 1 mg/kg. For the 600oC test, the only organic compound detected in the treated soil was naphthalene at 39 micrograms per kilogram (mg/kg) (parts per billion). Various CDD/CDF were detected at levels under 1 mg/kg for the treated soil from the 350oC test; only octachlorodibenzodioxin was detected at less than 1 mg/kg, for the 600oC treated soil. Treated soil from both tests passed TCLP for metals and organics.

The organic analytical data for the residuals showed PAHs and PCBs, as well as VOCs and SVOCs, that were not detected in the untreated soil. The type of compounds found in the residuals were comparable to those measured in the off-gas sampling, which is discussed in the next section, except for a few notable exceptions. Highly volatile compounds, such as chloromethane, were not detected in the residuals; pyridine, which was found at relatively high levels in the scrubber liquid and sludge, was not found in the off-gas analysis. The type of organics present in the residuals from the 600C test were very similar to those found in the 350C test, but the concentrations were higher. For the 350C test, PCBs concentrated in the scrubber sludge to slightly greater than 50 mg/kg, the TSCA limit; the PCB content was lower in the sludge from the 600oC test.

Less soluble compounds tended to concentrate in the scrubber sludge, whereas the phenols and pyridine distributed between the scrubber liquid and sludge. The activated carbon from both tests had VOCs, particularly the more volatile ones; naphthalene and several other lower molecular weight SVOCs were also found in the carbon from the 600oC test.

Total uranium and gross alpha/beta values measured for the untreated soil and various residuals indicate that there is no significant partitioning of radionuclides to any residual. The residuals are generally lower in uranium than the untreated soil. However, final determination of the potential for concentration of radioactive constituents in certain residuals could not be made from the limited results of this testing.

Off-Gas Treatment Results - The off-gas (downstream of the cyclone) and final vent gas were sampled and analyzed to demonstrate the removal efficiency of the off-gas treatment system. Parameters that were evaluated include particulate; metals (mercury, arsenic, cadmium, and uranium); volatile and semivolatile organics; acid gases and ammonia; and for the vent gas only, oxygen, carbon monoxide, and carbon dioxide.

The total particulate loading from the desorber, determined from combining the mass of solids collected in the hot cyclone and scrubber liquid, represented approximately 1.5 percent of the soil feed; the majority of these solids collected in the scrubber.

VOC results for the off-gas indicate aromatic hydrocarbons, several aliphatic hydrocarbons (paraffins and olefins), several oxygenated paraffins (e.g., ketones), and furans. In addition, chlorinated hydrocarbons and carbon disulfide were detected. The most significant VOCs included benzene, toluene, acetone, 2-butanone, furan, 2-methyl furan, chloromethane, and carbon disulfide.

A large number of SVOCs were found in the off-gas, including phenolics, PAHs, and other cyclic hydrocarbons and some oxygenated paraffins (e.g., aldehydes). The total SVOC content was similar to the total VOC content. Phenol and methyl phenols

contributed most of the confirmed SVOCs; the major TIC contributed approximately 40 percent of the total SVOCs. This data corresponds with the residuals data, except no pyridine was detected in the off-gas.
The off-gas contained only trace levels of most acid gases based on analysis of anions. However, ammonia was found in the off-gas at a relatively high level (790 g/m3). The source of the ammonia was not determined. It could have originated from decomposition of nitrogenous soil organic matter.
The off-gas treatment system proved very effective in controlling all parameters, which was expected because of the variety and number of control devices. Table IV shows the off-gas and vent gas concentrations, as well as overall removal efficiency for each parameter.
Materials Handling Test Results - The data indicates that as the moisture content of soil approaches the plastic limit (32 percent) of the untreated soil, the soil tends to fail both the drop test and agglomeration test. To avoid difficulties, the moisture content should be maintained at less than 25 percent; excavated soil having higher moisture could be partially air dried or blended with dryer material in the storage area.
The back-end materials handling data demonstrated that the treated soil can be processed effectively using a drag-flight conveyor and can be stockpiled.
Residuals Treatment Results - The purpose of the residuals treatment was to ensure that none of the residuals were classified as a RCRA hazardous waste. The aqueous scrubber solution from the verification pilot tests exceeded TC limits for mercury and pyridine. The liquid was treated using chelation/chemical precipitation followed by carbon adsorption. Mercury in the scrubber liquid was reduced from several milligrams per liter (mg/L) to 0.008 mg/L, well below the TC limit of 0.2 mg/L. Pyridine was reduced from 120 mg/L to less than 1 mg/L. Other aqueous residuals which were above the mercury TC limit were also successfully treated by chelation/chemical precipitation.
The sludge collected in the filter bags from all three pilot tests was air dried and placed in the RTA for thermal desorption treatment. Mercury was not detected in the TCLP extract of the treated solids; the total mercury had been reduced to 20 mg/kg. The precipitated and dewatered solids generated from treatment of the scrubber liquids and decon water were composited and successfully treated in the RTA in the same manner.
The liquid elemental mercury (approximately 30 g) collected from all sources was combined and treated with powdered zinc and powdered sulfur to produce an amalgam. This mixture was then stabilized using cement and water. The resulting TCLP mercury concentration was 0.008 mg/L.
The synthetic fabric filter bags, demister media, and HEPA filter media were treated by washing with dilute nitric acid which reduced the mercury (and other metals in the case of the HEPA) to below TC limits.
ENGINEERING AND COST ASSESSMENT
The information from the pilot tests was used to assess the application of a full-scale thermal desorption system to treat LEFPC soil. A preliminary process design was developed, which included an overall material and energy balance, list and sizing of major equipment, and operating requirements. The basis was 32 metric tons per hour (90 percent on-stream factor) of soil containing 300 mg/kg mercury and 25 percent moisture. The facility scope included all process components, from desorber feed system to treated soil material handling system; residual treatment prior to off-site disposal and effluent treatment and recycle was included.
The budgetary capital cost estimate was $12.3 million (plus or minus 25 percent), with the desorber and off-gas system representing 58 percent of this total. Estimated major operating and maintenance costs including labor, utilities, consumables, maintenance materials, and off-site disposal represent a total processing cost of $45 per metric ton. Other remediation costs, including excavation, transport, replacement, and other site activities, would be additional to this processing cost.
CONCLUSIONS
The LEFPC pilot testing demonstrated that thermal desorption is applicable to remediate the LEFPC floodplain soils. The process removed mercury from 300 mg/kg to below 15 mg/kg at treatment conditions of 350oC for 10 minutes at temperature, and to below 10 mg/kg at 600oC for 10 minutes at temperature. The desorber off-gas can be cleaned using a series of conventional air pollution control devices to

approximately 99 percent or greater removal of all measured contaminants. Mercury emissions were controlled by greater than 99.99 percent. More than 85 percent of the mercury recovered was collected in the wet scrubber, with approximately 60 percent of this isolated as elemental liquid mercury. Several process residuals were generated that could be RCRA hazardous. These residuals can be treated using conventional processes to render them nonhazardous. The capital cost of a high capacity desorption facility is estimated to be $12.3 million with operating costs of $45 per metric ton.

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ENGINEERING DESIGN AND TEST PLAN FOR DEMONSTRATING DETOXSM TREATMENT OF MIXED WASTES

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ABSTRACT

DETOXSM is a cocatalyzed wet oxidation process in which the catalysts are a relatively great concentration of iron ions (typically as iron(III) chloride) in the presence of small amounts of platinum and ruthenium ions (1,2). Organic compounds are oxidized completely to carbon dioxide, water, and (if chlorinated) hydrogen chloride. The process has shown promise as a non-thermal alternative to incineration for treatment and/or volume reduction of hazardous, radioactive, and mixed wastes (3-5).

Design and fabrication of a demonstration unit capable of destroying 25. Kg/hr of organic material is now in progress. This paper describes the Title II design of the demonstration unit, and the planned demonstration effort at Savannah River Site (SRS) and Weldon Spring Site Remedial Action Project (WSSRAP).

BACKGROUND

Oxidation of materials in the DETOXSM process is at present conducted in a stirred tank containing an acidic (3. - 4.% HCl) aqueous solution of the cocatalysts at temperatures of 400. - 473. K. System operating pressure ranges from atmospheric at 400. K to approximately 700. kPa at 473. K. Organic compounds are oxidized in the catalyst solution by iron(III) ions. In the oxidation process, iron(III) is reduced to iron(II). Oxygen introduced into the solution oxidizes iron(II) back to iron(III). Figure 1 illustrates the process.

Fig. 1.

Laboratory tests have determined the apparent rate constants and activation energies for oxidation of a wide variety of organic materials, including cellulosic materials, hydrocarbon oils, chlorinated and non-chlorinated aromatic and aliphatic solvents, energetic compounds, plastics, and chlorinated aromatic compounds such as polychlorinated biphenyls (PCBs). Measured rates indicate that at a temperature of 423. K approximately 0.5 to 10. moles/L-hr of organic material can be destroyed at organic material/catalyst solution contact areas of 300. to 6000. cm2/L.

There are many low level and mixed wastes which are organic in nature and could be treated using DETOXSM as an alternative to incineration. Mobile treatment units could be used to destroy the organic components of wastes on site, thus eliminating the costs and concerns of transportation and other forms of disposal. Heavy and radioactive metals remain in the DETOXSM process solution. The process solution can be converted to an iron oxide residue by boiling off HCl, and the residue stabilized for ultimate disposal.

TITLE II DESIGN

Design Basis

A site selection evaluation was conducted in parallel with the start of Phase II design. Ten (10) DOE sites were considered as possible demonstration sites due to the types of wastes available for the demonstration effort. The list was reduced to seven sites based on expressions of interest, and these seven sites were ranked according to evaluation criteria provided by the DOE. The two most highly ranked sites, Savannah River Site (SRS) and Weldon Spring Site Remedial Action Project (WSSRAP), were recommended for the proposed demonstration. The wastes at these sites helped define the requirements for the demonstration unit.

Waste types available for the demonstration include high and low boiling organic liquids (including chlorinated solvents), some paint sludges and similar solids, tributyl phosphate, and polychlorinated biphenyl (PCB)-contaminated liquids. For the purposes of mass and energy balance calculations, three organic compounds were chosen which would represent compositions typical of the range of waste types in the demonstration. These were a completely saturated aliphatic hydrocarbon (i.e., -CH2-), tributyl phosphate, and a PCB having 60.% by weight chlorine content.

Further requirements for the demonstration unit were provided in discussion with DOE EM-50 representatives. The requirements were for the most versatile unit possible consistent with a reasonable cost. It was decided in these discussions that the unit would have a method of feeding both liquid and solid organic materials to the reaction vessel and that the unit would have a system for semi-continuous removal of inert particulates or particulates generated during waste processing. These features at least doubled the unit size and cost over a liquid-only unit without solids removal, but it was felt important to demonstrate as wide a range of waste treatment ability as reasonably possible. Transportability of the demonstration unit was also an important consideration in design.

After consideration of the types and amounts of wastes at the demonstration sites, and particularly at WSSRAP, in combination with what was considered a reasonable throughput consistent with unit size and cost, a nominal capacity of 25. kg/hr was chosen for the demonstration unit. Once the reaction vessel is at temperature, the unit will operate 24. hrs per day.

Critical issues in the design effort were: operator and environmental safety and health; process effluents of minimum environmental concern; and ability to achieve the planned capacity.

Special considerations in the design effort were: modularity and transportability of the demonstration unit; ability to treat both liquid and solid waste materials; ability to remove solids from the process solution during waste treatment; design conformance with demonstration site requirements; compatible materials of construction for components contacting the process solution or its vapors; and secondary or tertiary containment in critical areas.

The design effort also took into account the requirements of 40 CFR Parts 264 and 265, the OSHA Standards for General Industry, 29 CFR Part 1910, and all applicable sections of DOT regulations, 49 CFR.

Title II Design Format

Jacobs Applied Technology (JAT), a subsidiary of Jacobs Engineering Group (JEG), was chosen for the Title II design effort due to JAT's extensive experience in modular chemical process design and construction, the resources and experience in chemical process design and environmental applications which were available through JEG, and the estimated cost and schedule for designing and constructing the unit.

The Title II design effort was conducted in a logical progression of preliminary PFD and P&ID preparation and review, revised PFD and P&ID preparation and review, HAZOP review, final review, and construction cost estimate.

Title II Design

PFD. Figure 2 is the final PFD for the demonstration unit. Flows 1 and 2 are liquid and solid organic feeds to the reaction vessel. Flow 3 is oxygen feed to the

reaction vessel. Flow 4 is product gas and water vapor exiting the reaction vessel, and flow 5 is water return to the reaction vessel from the reflux condenser. The reflux condenser serves to reduce the amount of water vapor in the product gases to the overhead condenser, provide some cooling for the reaction vessel (the reaction vessel is also cooled with an oil-filled jacket), and knock down any heavy organics in the product gases. Flow 6 is product gas to the overhead condenser, and flow 7 is condensed water to a receiver tank. Flows 20 and 21 are product gas, which contains a small amount of excess oxygen, through a carbon bed and to vent. Analyzers for oxygen, carbon dioxide, volatile organic compounds, HCl, and radiation are on the product gas lines.

Fig. 2. Demonstration Unit PFD.

In order to remove inert solids or inorganic solids formed during treatment of organics, a flow of process solution is taken from the reaction vessel, cooled, filtered, and returned to the vessel. Flows 8 and 9 are process solution through a solution cooler and to the filter. Flows 10 and 11 are filtered process solution to a surge tank and returned to the reaction vessel. Flow 12 is solids removed from the process solution, which have been rinsed thoroughly and are fed into a drum for stabilization and disposal. The solids are rinsed with a dilute citric acid solution (Flow 22) and water (Flow 23) before the are removed from the filter. Flow 29 is the solids rinses fed to an evaporator, where excess water is removed in Flows 14 and 15 to a receiver tank. Flow 26 is water from the evaporator receiver tank used to make up new citric acid solution or for rinsing collected solids. Flow 16 is the bottoms from the evaporator returned to the reaction vessel.

Water is produced in the destruction of most organic compounds. The water collected in the overhead receiver is this product water. Although most of the HCl in the product vapors will have been removed by the reflux condenser, it is expected that some HCl will be in the collected product water. Flow 24 is caustic solution to a neutralization tank to neutralize the acid content of the distillate. The neutralized distillate can then be fed to the evaporator (Flow 13) to return non-product water to the reaction vessel, or it can be taken directly to the evaporator overhead receiver (Flow 27) and from there to water treatment by Flows 17 and 19.

Materials of Construction

The 1000. L reaction vessel is specified to be tantalum-lined titanium. This construction provides excellent corrosion resistance (6) plus a strong and corrosion resistant pressure shell. The tantalum liner is expected to last the estimated 10. year lifetime of the unit, but even should it be breached the titanium shell will provide effective double containment until the process solution can be cooled and removed so that the liner can be replaced or repaired. A potentiometer connected between an electrically isolated counterelectrode in the process solution and the pressure shell will be able to detect any breach in the tantalum liner.

Other areas where the process solution will contact materials are in the solution cooler, filter, evaporator, evaporator bottoms pump, and evaporator bottoms return line. The solution cooler will be tantalum on the process side. Once the process solution has been cooled, it is compatible with Grade 7 titanium, which will be the material of construction for the solution filter, surge tank, evaporator, and associated plumbing.

Due to the HCl content of the hot gases exiting the reaction vessel, the reflux condenser and the overhead condenser will have process sides of tantalum. Once the gases have been cooled, Teflon-lined carbon steel will be used for the vent gas and condensate.

Design Issues

Reliably feeding solid materials to the pressurized reaction vessel is a challenge. Solids will be shredded and slurried with water, then fed to the reaction vessel through redundant pinch valves using a progressive cavity slurry pump. Shredding will also increase reaction rate by increasing the surface area of the solids.

Effective agitation is vital to achieving good organic destruction rates, since the oxidation rate for most organic compounds is dependent on the surface area of contact between the organic compound and the process solution. Achieving effective agitation in the demonstration units reaction vessel is complicated by the range of waste forms. More volatile materials will be in vapor form, which will require good gas/liquid contact. Less volatile materials will be in liquid or solid form. Effective oxidation of liquids and solids will require good liquid/liquid and

solid/liquid contact. In consultation with agitator companies, one of the more versatile arrangements was determined to be the combination of a standard 45. pitched blade turbine mounted somewhat below solution level to pull vapor from the headspace of the reaction vessel into the process solution, and a disk/paddle turbine near the bottom of the reaction vessel to provide high shear. Baffles in the vessel will help provide good turbulence.

The combination of organic compounds and oxygen presents the potential for formation of explosive mixtures in the reaction vessel headspace. Computer simulations, conducted at Sandia National Laboratories (SNL), have indicated that below approximately 10.% by volume oxygen content in the reaction vessel headspace gases there is no flammability possible regardless of organic concentration. This is illustrated in Fig. 3, which is presented here as a generic figure only and is not meant to represent actual data. Design parameters for the process specify operation at 5.% by volume oxygen in the vent gases, which translates to less than 1.% by volume oxygen in the reaction vessel headspace gases at operating conditions in the reaction vessel. These operational parameters give a large safe operating range (approximately 10x) for the process. Redundant, calibrated oxygen sensors on the vent gas line will ensure correct measurement of oxygen levels. Additional tests are planned at SNL to identify the flammability limits for some representative volatile organic compounds at the operating conditions of the process.

Fig. 3.

An important issue in the unit's ability to destroy TBP was the ability of the filter system to remove ferric phosphate precipitate from the process solution at a rate sufficient to allow operation at the unit's design capacity of 25. kg/hr. Results of laboratory particle size measurements and filtration/rinsing experiments using ferric phosphate precipitated under the conditions of the process were used to size the filter. A sealed, Nutsche type or helical filter with a scraper/agitator will be used, so that filter cake rinsing and discharge will not normally require handling of the solids by the unit operators.

A preliminary layout of the equipment is shown in Fig. 4. The demonstration unit is to be assembled as two transportable process modules plus a module for the oil cooling/heating system (not shown). A vertical module approximately 9.75 m tall x 3.66 m x 2.44 m will include the solids feed, reaction vessel, reflux condenser, solution cooler, particulate filter, surge tank, process solution return pump, evaporator and evaporator overhead condenser, evaporator bottoms return pump, and reaction vessel quench tank. A horizontal module approximately 7.32 m long x 2.44 m x 3.66 m high will include the remainder of the pumps, the overhead condenser, overhead receiver, evaporator overhead receiver, distillate neutralization tank, vent knock out tank, and vent gas carbon bed. A third module 7.32 m long x 2.44 m x 3.66 m high will contain the oil cooling/heating system. The oil module can be placed away from the process modules and must in any case be placed in an exterior location so that excess process heat may be rejected.

Fig. 4.

Fabrication and assembly of the demonstration unit is scheduled for fiscal year 1995.

DEMONSTRATION PLAN

The demonstration plan is a progression from tests with minimally hazardous or non-hazardous organic materials through treatment of hazardous organic wastes, and finally to demonstration of the process on a variety of mixed wastes. Non-hazardous and hazardous waste treatment is planned at SRS. Mixed waste treatment is planned at WSSRAP. The demonstration effort is planned to take place in Fiscal Years 1996 and 1997. A successful demonstration will establish the DETOXSM process as a versatile, mobile on site mixed waste treatment method.

Non-Hazardous Materials Testing

After installation of the unit at SRS, functionality tests will be conducted to determine the ability of the unit to operate as designed, to identify any obvious flaws in control algorithms, equipment, materials, or construction, and to run through safety and emergency response procedures. Functionality tests will follow a progression from operation with cold water to operation with hot water and finally to operation with the process solution.

Following functionality testing the filter system will be tested for its ability to remove inert solids from the process solution, the solids feed will be tested separately for its ability to reliably deliver solids slurry to a pressurized

environment, and the unit will be used to destroy a selection of minimally- or non-hazardous organic materials.
The liquid organic materials to be oxidized will be white petroleum oil and tributyl phosphate. The solids to be used in the solids feed will be a mixture of polyethylene, paper, latex rubber, and poly(vinyl chloride). The process vent gas, condensate, and solution will be monitored and analyzed to determine the levels of any volatile organics and to establish destruction efficiencies. After these tests, the unit will be inspected thoroughly to determine equipment compatibility and performance.

Hazardous Waste Treatment

Hazardous wastes to be treated include contaminated oils, non-halogenated solvents, halogenated solvents, and, if time and funding permit, combustible solids. The test regimen will most likely include benzene, trichloroethylene, tetraphenyl borate in water solution, and mineral oils contaminated with lead and/or mercury.
Vented product gas will be monitored for levels of volatile organic compounds, and the condensate product and process solution will be sampled and analyzed for organic residues to determine destruction efficiencies. When wastes containing toxic metals are destroyed, the process solution will be analyzed for the metals as well.
At the conclusion of hazardous waste treatment, the process solution will be boiled to evaporate HCl until approximately 95.% of the iron is in the form of the hydrated iron oxide. The iron oxide, which may contain toxic metals, will be filtered from the solution as it forms, and may be stabilized if SRS desires. The HCl will be condensed with the boiled water, neutralized with caustic, and disposed.
Following the hazardous waste campaign, the demonstration unit will be cleaned, thoroughly inspected, maintained, and any desired changes in design made before it is shipped to WSSRAP.

Mixed Waste Treatment

Mixed wastes to be treated include ethylene glycol, oils, non-halogenated solvents, halogenated solvents, paints and sludges, PCBs and PCB-contaminated liquids, and tributyl phosphate. The demonstration may be extended to other materials including combustible solids.
The unit will be installed, checked out, and an Operational Readiness Review conducted before mixed waste treatment is begun. The order of waste treatment will be as given above.
Calculations indicate that at least two process solutions will have to be used over the course of waste treatment to ensure that radioactivity is kept at low levels in the process solution residue. Vented product gas will be monitored for volatile organic compounds and radioactivity. Product water will be analyzed for volatile organic content and radioactivity, then sent to the on-site water treatment plant. The process solution will be analyzed periodically for organic compounds, radionuclide content, and toxic metals. The process solution residue and the ferric phosphate residue from treatment of tributyl phosphate will be stabilized and disposed at WSSRAP.

Demonstration Requirements

Requirements for the demonstration tests include: operating, safety, and maintenance manuals for the demonstration unit; formal test plans and procedures; hazardous materials handling and emergency response training for operators; lock out/tag out procedures; site-specific safety and operations training; NEPA documentation; a RCRA Part B R&D permit for tests with hazardous wastes at SRS; review of estimated water and air emissions by the cognizant regulatory agencies; and preparation of Radcon procedures, Safe Work Plans, and Task-Specific Safety Assessments for operations at WSSRAP.

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DESIGN OF A HYDROTHERMAL OXIDATION TEST BED AND VESSEL REACTOR*

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ABSTRACT

A test bed capable of evaluating several different hydrothermal oxidation (HTO) reactor concepts and other HTO components, has been designed and is planned to be constructed at the Idaho National Engineering Laboratory (INEL). A vessel reactor will be designed, fabricated, installed, and tested with a series of simulated wastes selected to represent the types of mixed radioactive wastes currently stored and/or generated by the U.S. Department of Energy (DOE). The major HTO component in the process is a chemical reactor into which simulated mixed waste, water, and gaseous oxygen are injected. The process pressures and temperatures are above the critical point of water. In this environment, organics become highly soluble and are transformed into benign chemicals or elements. The test bed is to be built and initially perform 2 years of vessel reactor testing. After the initial testing, other reactor types such as a transpiring wall, may be installed and evaluated.

INTRODUCTION AND BACKGROUND

To support the development of a full-scale Hydrothermal Oxidation (HTO) mixed waste plant for treating U.S. Department of Energy (DOE) mixed wastes, a 189 liters per hour (lph) (50 gallons per hour [gph]) HTO test bed has been designed and is planned to be constructed at the Idaho National Engineering Laboratory (INEL). The test bed has been designed so several different HTO reactors and associated components may be tested. The first reactor planned for testing is a vessel reactor. The test bed will be located in an existing building, the Water Reactor Research Test Facility (WRRTF), located at the north end of the 890 square mile INEL. During FY-95 procurement of long lead materials will begin, and construction will start in early FY-96 with start up planned for mid to late FY-96. The DOE is currently storing a large volume of organic wastes mixed with radioactive constituents (mixed waste). Various methods are being explored to reduce the volume of wastes being stored to protect the environment, and to reduce costs. HTO may be a technology that can be employed to effectively and economically destroy organic compounds in certain mixed waste streams that include hydrocarbons, Polychlorinated Biphenyls (PCB's), fuels, lubricating oils, solvents, and paints.

HTO involves heating and pressurizing water above the critical temperature of 374C (705F) and critical pressure of 2.21 x 107 Pascals (Pa) (3,208 pounds per square inch [psi]). This condition is referred to as supercritical. Water above the critical point does not boil or condense when heat is added or removed, rather it undergoes a continuous change in density. When water is above the critical point, the solvating characteristics change significantly. The bonding force between

hydrogen and oxygen atoms is also reduced resulting in increased mobility of the hydrogen atom and, therefore, reduced molecular polarity. Because of this change, many organic compounds that are only somewhat soluble in water at subcritical conditions become highly soluble in supercritical water. When water loses its normal polar behavior, many ionic salts are less soluble than under standard conditions. As a result, many salts in aqueous solution precipitate out of solution as the critical temperature of water is neared. This property is useful because a large portion of metals present in a waste stream can be removed as oxides or precipitated salt. Another important feature of the technology is that gases can be added to supercritical water in all proportions, and specifically, oxygen can be added to a supercritical water and organic waste mixture in sufficient quantities to ensure complete oxidation of the organics. When oxidation is complete, complex organic wastes are transformed into carbon dioxide, water, nitrogen, or other innocuous compounds.

There are several key advantages for using HTO as a means to destroy organic wastes. First, the HTO process destroys organic wastes by converting them into harmless substances such as water, nitrogen, or other nonhazardous chemical compounds. Second, emissions from the HTO process are completely controlled and can be sampled and tested before discharge and, if required, can be further treated to ensure safe emission to the environment. Most of the gaseous emissions from a HTO process are carbon dioxide, excess oxygen, nitrogen, and small amounts of oxides. Third, and the most pronounced advantage of the process is the reduction of volume of liquid wastes. Especially in the case of mixed wastes, the HTO process will destroy the organic constituents and separate the water leaving only a small fraction of the original volume of the waste radioactively contaminated. This concentration of dilute waste streams will help with the final disposal of the residue by means of vitrification, encapsulation, or other methods. A final advantage of the process is the potential for thermal energy, electrical energy, and clean water as process byproducts.

Technical challenges in using HTO technology include the generation of severely corrosive acids during the oxidation process. Developing materials for use in HTO process systems which resist the effects of corrosion over the long term, has been difficult. To combat the effects of corrosion, a neutralizing agent is often added to the process. The addition of the neutralizer forms salts that precipitate near the critical point of water and tend to deposit on the interior of process equipment. The deposition is often well adhered to equipment surfaces, difficult to remove, and requires frequent cleaning. If not removed from the process equipment, deposition will impede system performance. Finally, control of HTO systems is complicated by wide and varying operating ranges.

The HTO process is illustrated in Fig. 1. A mixture of waste and water is pressurized, heated, and injected into the reactor. Water, also pressurized and heated to supercritcal temperature and pressure, is injected into the reactor. Oxygen at ambient temperature and pressure is injected and mixed with the water and waste. The oxidation process occurs inside the reactor. The reacted fluid exits the reactor where it is cooled, and the pressure and temperature are reduced. The gas is then separated and released to the atmosphere (after appropriate filtering and monitoring). The effluent is further cooled, depressurized, and the solids separated. The water from the effluent can be either recycled or discharged to an evaporation pond.

Fig. 1.

OBJECTIVES

The primary objective of the HTO test bed is to provide the capability for testing and evaluating reactors and HTO system components and to determine the applicability of the test bed in treating mixed wastes. Primary evaluation criteria include demonstrating that effective Destruction/Removal Efficiency (DRE) rates of at least 99.99% can be achieved for the organic portion of mixed wastes, eliminating or mitigating corrosion and deposition, and operating a facility that demonstrates reasonable longevity and safe operation. To accomplish these objectives it was proposed that the test bed be built so several different reactor types can be installed, tested, and evaluated to determine which configuration best meets the stated goals. Reactors that could be tested include a transpiring wall, tubular, and vessel. Initially, a vessel reactor will be designed and installed in the test bed, with plans to design, build, install, and test other reactor types in later years.

The vessel reactor appears at this time to have the most promise to obtain the DRE rates while limiting deposition and corrosion for DOE mixed wastes.
Long-range objectives of the proposed testing program are to prove that HTO technology is both a technically and economically feasible approach for eliminating DOE hazardous and mixed wastes.
DESCRIPTION OF TEST BED AND REACTOR DESIGN*
The following is a description of the HTO process design. Definitions of key terms follow:
Waste Constituents-Undiluted materials such as salts, hydrocarbons, heavy metals, and chlorides.
Waste-Simulated waste constituents such as salts, hydrocarbons, heavy metals, chlorides, and deionized water.
Deionized Water (or "Water")-Water with ions of dissolved salts removed.
Reactor Mixing Water-Heated deionized water mixed with the waste.
Cooling Water-Deionized water that is added to cool the reacted fluid after oxidation.
Vessel Makeup Water--Deionized water heated and added to the vessel brine pool.
Gaseous Oxygen-(GOX)GOX added to the process to provide adequate oxidation of organic material in the waste.
Supplemental Fuel-Hydrocarbon-based fuel added to wastes to provide adequate BTUcontent for oxidation, or used to heat up the reactor during startup.
Neutralizing Agent-Chemical that neutralizes acids in the reactor effluent.
Reacted Fluid-Liquid and gas that remains after oxidation, is not entrained in the brine pool, and has not been cooled.
Reactor Effluent--Liquid and gas that remain after oxidation, is not entrained in the brine pool, and has been cooled.
Brine-A mixture of injected vessel makeup water and salts (generated after the neutralizing agent reacts) that collects in the bottom of the vessel reactor. Brine may also contain small amounts of metals, corrosion products, or other solids.
The test bed is designed to handle 189 lph (50 gph) of waste and reactor mixing water, will run continuously for up to 100 hours (most testing will be from 10 to 20 hours), and has a design life of 5 years. Figure 2 illustrates the HTO process.
Fig. 2.
Demineralized Water Supply
The demineralized water supply consists of a 11,355 liter (3,000 gallon) storage tank, a mixed bed resin designed to strip free ions of salt and mineral contaminants from a raw water supply source, and acid and caustic are used to regenerate the resin. The conductivity (resistivity) of the effluent is monitored to control water quality. The demineralized water is of a purity satisfactory for use in the HTO process. The system is capable of supplying 45.4 liters per minute (lpm) (12 gallons per minute [gpm]) of water (although greater maximum flows are available because of the storage capacity). The feed line from the demineralized storage tank supplies two booster pumps. One pump supplies water to the waste mixing tanks at a rate of 189 lpm (50 gpm), 1.03 x 106 Pa (150 psi), and ambient temperature. The other booster feeds a variable speed high-pressure pump that delivers reactor mixing water at 189 lpm (50 gpm), and 2.41 x 107 Pa (3,500 psi). After passing through a direct current electric heater, the temperatures of the water from the booster pump is 593C (1,100F), and the water is injected into the reactor. A branch line from the booster feeds a cooling water pump that operates at 2.41 x 107 Pa (3,500 psi) and supplies 22 lpm (6 gpm) of water at ambient temperature. After the water exits the cooling water pump, the line branches again. One branch passes through a heater at 200C (418F) to supply makeup water to the reactor brine pool. The other branch is used for cooling water to reduce the temperature of the reacted fluid after it leaves the reactor.
Waste Mixing and Delivery
Waste mixing is achieved by flowing demineralized water into two 22,171 liter (600 gallon) waste mixing tanks. Waste constituents are measured and poured manually into the tanks or fed by small metering pumps. The concentration of waste constituents in the water will vary depending on the specific test to be conducted; however, the maximum concentration of waste constituents with respect to the waste and reactor mixing water is 5 to 10% by weight. A paddle type mixer is supplied with each mixing tank to homogenize the waste. Two mixing tanks are provided because as waste is being mixed in one tank, the other tank can be used to feed the reactor. Waste is

fed from the tanks to a high-pressure variable speed pump, which operates at 2.41 x 107 Pa (3,500 psi). Just before the waste enters the pump, supplemental fuel, such as methanol, may be added to the waste. The supplemental fuel is added in varying quantities depending on the waste to ensure the BTU content of the waste is sufficient to initiate and sustain oxidation inside the reactor. After the waste is pressurized, it passes through a direct current electric heater that elevates the temperature of the waste to a maximum of 300C (598F) (the amount of heat added to the waste stream varies depending on the BTU content of the waste). A neutralizing agent is also added to the waste in varying quantities depending on the simulated waste. The waste stream is then injected into the reactor.

Oxygen Supply

Because of the volume of GOX required to support the test bed, and for reasons of economy and convenience, a storage tank is used to store liquid oxygen (LOX) and a vaporizer converts it to GOX. The LOX storage tank is 22,710 liter (6,000 gal) capacity. The LOX is drawn out of the storage tank by a high-pressure pump 3.1 x 107 Pa (4,500 psi) and is then vaporized. The temperature of GOX is ambient as it enters the reactor. Flow rates into the reactor vary from .142 to 1.98 cubic meters per minute (5 to 70 standard cubic feet per minute [scfm]) depending on the specific type of waste being tested. A maximum stoichiometric ratio of 3:1 can be achieved to ensure complete combustion of organics (the normal stoichiometric ratio is expected to be 1.5:1 or less).

Vessel Reactor and Cooling

Figure 3 shows the process that occurs inside the reactor during HTO. The waste, water, and GOX are connected to a header at the top of the reactor. Each line is routed separately through the header, and mixing of the water, waste, and GOX occurs in the reactor nozzle as shown in Fig. 3. As the oxidation process proceeds, the neutralizing agent combines with chlorides and salt is formed. The temperature at the reaction is approximately 600 to 800C (1,138 to 1,498F). At this temperature and pressure, the salt solidifies and complete combustion of organics with minimal residence time is likely. The salts, metals, and other solid particles drop into the brine pool. The brine pool is fed with makeup water at a temperature of 200C (418F). At this temperature, the salts redissolve in the brine pool and the volume of flow in the pool is maintained so that salt deposition on the walls of the reactor is minimal. The brine is fed to the reactor effluent discharge line, just upstream of the first stage letdown vessel. After oxidation, the reacted fluid leaves the reactor and is fed into a mixing tee or other device to reduce the temperature of the fluid to approximately 200C (418F). Cooling water is introduced into the cooling device to reduce the temperature of the reacted fluid. As the effluent leaves the cooling device, it passes through a series of orifice plates that reduce the pressure from 2.41 x 107 to 1.37 x 104 Pa (3,500 to 2,000 psi). The effluent then passes through a control valve that further reduces the pressure to 6.89 x 106 Pa (1,000 psi). The control valve receives input from a pressure indicator located on the upstream side of the cooling device. The effluent is then ready to enter the first stage let down vessel.

Pressure Let Down

Once past the control valve located on the downstream side of the reacted fluid cooling device, the reactor effluent enters a first stage let down vessel. As fluid enters, there is some flashing and gas separation. During this first stage letdown approximately 90% of the oxygen and 50% of the CO2 and liquid separate (these are the two primary gases entrained in the fluid; there are trace amounts of other gases). The gas exits the vessel and is routed through the filtering and emission control system. The volume of liquid in the vessel is maintained at about half the vessel volume. The liquid is discharged from the vessel and passes through a series of orifice plates to reduce the pressure from 6. 89 x 106 to 4.82 x 106 Pa (1,000 to 700 psi). A control valve linked with a level controller on the vessel helps maintain a constant level within the vessel and further reduces the pressure from 4.82 x 106 to 4.83 x 105 Pa (700 to 70 psi). The liquid then enters the second stage let down vessel. A manifold located inside of the vessel sprays the liquid onto the vessel walls and helps liberate the remaining gas from the liquid. After separation, the remaining gas is routed to the emission control and filtering system. A control valve on the discharge line and a pressure controller on the vessel help to maintain a constant pressure and volume of gas inside the vessel. The liquid occupies about half the vessel volume and is discharged from the vessel to a heat exchanger. The

heat exchanger reduces the temperature of the liquid from 149C to about 38C (300 to 100F). A control valve on the discharge side of the heat exchanger is linked with a level controller on the vessel to regulate the level of fluid inside the vessel. Fig. 3.

Gas Filtering and Emission

As the gas is discharged from the first stage let down vessel, it is reduced in pressure through a series of orifice plates and then travels through a control valve. The control valve is linked with a pressure indicator located on the upstream side of the orifice plates. The control valve controls the pressure (and therefore the flow) of gas through the line. The flow of gas from the second stage let down vessel tees into the line from the first stage letdown vessel. The gas is then routed through a series of charcoal filters designed to remove trace amounts of wastes that may be present in the gas stream. The gas is then discharged to atmosphere. The quantity of gas emitted to the atmosphere is estimated at about 145 kilograms per hour (320 lb per hour). An additional 272 kilograms per hour (600 lb per hour) of steam is released. The gas is mostly oxygen, CO_2, and trace amounts of other elements. All emissions are within regulatory guidelines.

Fluid Filtering and Evaporation

As the fluid (reactor effluent) leaves the heat exchanger, it passes through a bag filtering system. Two bag type filters are provided, but only one is used at a time, so one may be bypassed and changed out while the process continues to operate. The filters are designed to filter particles as small as 1 micron and are primarily used to remove metals and salts that may have solidified during pressure letdown. The fluid is then routed to two 11,355 liter (3,000 gal) storage tanks. The fluid is stored in these tanks, sampled, tested, and a determination is made if it can be discharged to the evaporation ponds. If certain constituents are below permit levels, then the fluid will be routed to the ponds. If the constituents are higher than permit levels, the fluid will be routed to an evaporator where it will be concentrated. The residue left in the evaporator will be collected and disposed of as hazardous waste. It is anticipated that a small volume (less than several gallons) of residue would be left after evaporation. The residue would be comprised of salts and heavy metals.

Control, Data Acquisition, and Sampling Systems

The control system will be comprised of off-the-shelf hardware and software, configured as a process data acquisition, display, control, and archival system. An operators console will be located remotely from the test bed (no personnel will be allowed near the test bed when the system is pressurized or at temperature). The control system will use the National Instruments LabVIEW application. This software is a graphically oriented user interface that allows simple user configuration of the hardware to perform acquisition and control applications. The application communicates to process interface hardware via general purpose interface bus to two remotely located cabinets. The cabinets contain low and high analog inputs, voltage and current outputs, digital I/O and serial communication interfaces fed from instrumentation located in the test bed.

The sample system includes a series of drains (on low pressure, low temperature lines) from which samples of fluid can be obtained manually. A series of sample bombs will be used to collect gas just after the effluent leaves the reactor and at other locations along the gas filtering and discharge system. A Residual Gas Analyzer (RGA) will be used to obtain on line samples from certain process streams (e.g. reactor effluent). The RGA is capable of giving nearly instantaneous feedback on the composition of samples. This information shall be used to make adjustments in the control and operation of the test bed.

PLANNED TEST PROGRAM

Initially a series of eight waste types are planned for testing in the INEL HTO test bed and vessel reactor. The test series have been selected to be representative of a wide variety of hazardous elements found in DOE mixed wastes (the test series will contain surrogates of radioactive materials but no actual radionuclides). The waste series are

Series 1. Aromatics with Mercury
Series 2. Nonchlorinated Solvents with Metals
Series 3. Paper, Cloth, and Plastic
Series 4. TRIMSOL
Series 5. Mixed Oils

Series 6. Activated Carbon
Series 7. Scintillation Fluid
Series 8. Tributyl Phosphate.
A total of 40 tests using the eight waste series are planned over a period of 2 years. Most tests will average about 10 to 20 hours in length with at least one 100-hour test planned. In addition to monitoring for DRE and deposition, other data will be collected during the testing to include information on power consumption, efficiency of various process components, erosion and corrosion of process components, material characteristic studies, temperature and pressure of certain process hardware, flowrates, consumption of neutralizing agents and supplemental fuel, and other aspects of testing that will help gain understanding of the overall process. This information will be the basis for deciding the configuration of a full scale HTO plant to be used for the destruction of hazardous and mixed wastes. A detailed testing and operations plan is currently being prepared as the basis for conducting the field testing, addressing how requirements will be met, and defining information and data to be gathered and analyzed during the testing that will be used for future design work.

37-40
RESULTS OF DETAILED CHARACTERIZATION ON CH-TRU MIXED WASTE AT ARGONNE NATIONAL LABORATORY-WEST
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ABSTRACT
Argonne National Laboratory-West and Lockheed Idaho Technologies Company have jointly participated in the Department of Energy's (DOE) Waste Isolation Pilot Plant Experimental Test Program since 1990. A new facility at Argonne was developed to improve the effectiveness and efficiency of contact-handled transuranic mixed waste characterization and to decrease the potential for facility contamination and personnel exposures. This new facility, the Waste Characterization Area, was approved for radioactive operations in March 1994. Between April and September 1994, forty-two waste drums containing mixed debris waste were characterized to support a study being performed to evaluate volatile organic compound concentrations in the void volume headspaces of waste drums. This paper presents the results of characterization performed at Argonne, emphasizing parameters important from a facility standpoint. Specifically, information is presented on drum surface dose rate, fissile content, number and type of gas samples, volatile organic compound concentration, and facility contamination levels. Actual values are compared to enveloping conditions assumed in the safety assessment for the characterization facility. Argonne-West is one of the first DOE sites to perform detailed waste characterization under the DOE's Transuranic Waste Characterization Program. The information presented herein could aid other storage and generator sites in developing characterization procedures and facilities.
BACKGROUND
Argonne National Laboratory-West (ANL-W) and Lockheed Idaho Technologies Company (formerly EG&G Idaho) have jointly participated in the Department of Energy's (DOE) Waste Isolation Pilot Plant (WIPP) Experimental Program (1) since 1990. Detailed characterization of contact-handled transuranic (CH-TRU) mixed waste was performed on thirty-two drums of waste which were loaded into seven test bins between March 1991 and July 1993. The detailed characterization included non-destructive and destructive examinations. Non-destructive examinations were performed at the Idaho National Engineering Laboratory Stored Waste Examination Pilot Plant (SWEPP); they included real-time radiography, passive/active neutron fissile assay, and ultrasonic container integrity measurements. The destructive examinations were performed in Argonne's Hot Fuel Examination Facility (HFEF); they included collection and analysis of gas samples in most of the layers of confinement within the drum, as well as removal, categorization, measurement or estimation of various physical parameters, and repackaging of the waste contents. The description of these operations and results of characterization were presented in previous Waste Management conferences (2,3).
Since the last of the 32 drums was characterized in July 1993, a new facility within

HFEF at ANL-W has been developed to improve the effectiveness and efficiency of CH-TRU mixed waste characterization and to decrease the potential for facility contamination and personnel exposures. This new facility, the Waste Characterization Area (WCA), was approved for radioactive operations in March 1994 (4). Between April and September 1994, forty-two additional waste drums containing mixed debris waste were characterized at Argonne to support a study being performed to evaluate volatile organic compound (VOC) concentrations in the void volume headspaces of waste drums. This VOC study was conducted by EG&G Idaho to demonstrate compliance to Environmental Protection Agency requirements for drums destined for WIPP (5,6,7).

SUMMARY OF CHARACTERIZATION PROCEDURES

Following nondestructive examination of drums at the SWEPP, which includes fissile assay by passive and active neutron methods and gamma scanning, they are transported 32 kilometers to Argonne in a TRUPACT-II cask. Drums are unloaded in the HFEF truck lock and transported to the preparation room within the WCA. One drum at a time is then transferred by cart into the transfer room beneath the waste characterization chamber. The drum is raised to one of two drum ports and physically attached. After the drum is mechanically locked in place and a plastic bagging sleeve is in place, the drum port cover inside the waste characterization chamber is opened to initiate characterization operations. Prior to characterization, a number of quality control samples are collected for the gas sampling system, including field blanks, manifold blanks, and reference standards. The first gas sample collected from the drum is a drum headspace sample. The sampling needle is inserted through the filter on the drum lid before the lid is removed. Then the lid is removed and the plastic rigid liner lid is removed. A gas sample is then collected from the large drum bag. The drum bag is then opened and any inner bags which are not breached and which contain at least 100 ml of void volume are sampled. Following gas sampling, the drum contents are individually removed and various physical parameters are either measured or estimated. The waste package shapes and contents are described and then the waste is placed in a new drum which is interfaced to the other drum port. All of the characterization operations are videotaped, and a data report is written. Drums, waste packages, and chamber surfaces are smeared often during the characterization process to monitor contamination levels. Decontamination is performed often to keep the chamber as clean as practical.

RECENT CHARACTERIZATION DATA

This section summarizes characterization data obtained from 42 drums of CH-TRU debris waste. Waste code descriptions and the number of drums within each waste code are presented in Table I. Parameters presented in Table II include minimum, maximum, and average values for surface dose rate, plutonium and americium content, waste weight, number of gas samples collected by type, and contamination levels encountered. Where applicable, the corresponding maximum allowable or assumed value used in the WCA safety assessment is included for comparison. Finally, Table III lists the highest VOC concentrations measured.

CONCLUSION

Including the first 32 drums designated for the WIPP Bin Tests, Argonne has completed detailed characterization and repackaging of 74 CH-TRU, mixed, debris drums. The characterization and repackaging was performed in accordance with DOE and EPA quality assurance programs. The 42-drum campaign was completed in 22 weeks from the initial hot startup of the WCA. The drum throughput increased over this period from approximately 1 drum per week to 3 drums per week. This improvement was partly due to a change in shift length; from 8 hours per day 5 days per week, to 12 hours per day on 3 days and 8 hours on one day per week. Another factor in the improved efficiency was operator experience and proficiency, which improved as the campaign progressed. Activities in the WCA were completed without any contamination incidents. This can be attributed to the design and operational philosophy implemented at Argonne; confinement is paramount to success.

At least six major DOE sites in the United States store and/or generate CH-TRU waste. Before waste from these sites can eventually be disposed at WIPP, waste characterization must be performed. Argonne-West is one of the first DOE sites to perform detailed waste characterization under the DOE's TRU Waste Characterization Program (8). The information presented herein could aid other storage/generator sites in developing characterization procedures and facilities.

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37-41
HEAT SOURCES IN HANFORD'S TANK C-106
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ABSTRACT
A calculation of the heat generation in nuclear wastes due to the decay of cesium-137 and strontium-90, using a simple parametrization to determine the energy partition between betas and neutrinos, is applied to evaluate the missing heat in Hanford's tank C-106. The possibility is examined of accumulating stored energy in the form of lattice dislocations in solid waste layers via highly non-relativistic daughter recoils from heavy alpha emitters. This work was prompted by the observation that calculations converting beta decay activities into heat deposition within nuclear waste have in the past been based on a variety of widely varying conversion factors. In order to eliminate such unnecessary uncertainties, these factors are here calculated ab initio from fundamental principles. Use is made of a closed-form Fermi function which is capable of reproducing the shapes of beta spectra to better than one percent. Application of these results to tank C-106 confirm that 2/3 of the heat source is not accounted for by the activity measured in sludge samples. The possibility is examined that the missing heat source, comprising fission fragments as well as transuranics, may be located in hard deposits formed on the tank bottom early on in the operation history and estimates are made of the density of stored energy that might have accumulated in such solids as a result of displaced atoms.
AB INITIO CALCULATION OF HEAT GENERATION FROM 90Sr AND 137Cs DECAY
The calculation of heat generation due to weak radioactive decays requires a knowledge of energy escape via neutrino leakage. In large volumes of radioactive material the energy carried by gamma photons is absorbed, except at the surface. In the case of large waste storage tanks, the energy of gamma rays escaping from the waste surfaces is almost completely absorbed and converted to heat in the tank walls. Although heat generation from beta emitters like 90Sr and 137Cs can be determined calorimetrically, such measurements have low sensitivity. In this paper the energy loss via neutrino leakage is calculated ab initio using weak interaction theory and experimentally determined nuclear energy levels, which are known to high accuracy and are readily available from the published literature (1).
Five separate decays are taken into account in this analysis (we have neglected the 0.01% g branch in 90Y.
Eq.

The energy sharing between the b- and the neutrino can be expressed in a simple parametrization (2), which is more than adequate for a 1% calculation. In terms of the ratio K of the kinetic to rest energy, the b- distribution can be written:
Eq.
where * corresponds to zero neutrino energy and z Zh c/e2
with Z being the daughter nuclear charge. The parameter ki represents shape-factor sealing and takes the following values:
Eq.
Eq.
The value of k2 for the 5.4% branch in the decay of 137Cs has been determined experimentally (3). Numerical integration yields the neutrino energy leakage fractions for the various transitions:
Expressing ground-state and excited-state transitions by subscripts 0 and 1 respectively, we get the following:
Eq.
where b denotes the branching ratio.
The energy deposited as heat for each decay is (in units of the electron rest mass):
Eq.
Taking the electron rest mass as 0.511 MeV we get:
Eq.
The above particle decay heats are equivalent to:
Eq.
We estimate these heat conversion factors to be accurate to better than one percent, based on the knowledge of energy levels and the parametrization of the beta spectra involved. Comparison with the factors (4) applied, for example, to Hanford's tank SY-101, shows the following discrepancy:
Eq.
APPLICATION EXAMPLE:
We now consider some possible implications of discrepancies between measured heat generation and the concentration of radionuclides in Hanford's tank C-106.
In tank 106-C only one third of the thermodynamically calculated heat source term is accounted for by the radioactivity present in sludge samples (see WHC-SD-WM-PSE-010, Rev. 0):
Eq.
We test here the assumption that the missing heat source term Q3 = Q1 - Q2 = 20.6 kW may be due to the a-decay of plutonium in heavy metal waste residues left on the tank floor (see WHC-SD-W320-ANAL-001, Vol II, Rev. 0). Taking the average a-particle energy from plutonium decay to be 5.156 MeV we get a daughter nucleus recoil energy per atomic mass unit: 373 Ev/amu.
These recoils are thus highly non-relativistic since b = 2.8 x 10-4, so that energy dissipation via lattice dislocations is strongly favored. We assume, as a bounding extreme, that the entire daughter recoil energy is stored as atomic displacement energy. The contribution from a-particle stopping is neglected. Now we have stored power: Eq.
and stored energy since 1947 ES = QSt = 0.52 TJ.
Taking the latent heat of vaporization of water to be LV = 4.07 x 104 J/mol, the release of stored energy ES would be able to generate a volume of steam:
Eq.
If we take the heel thickness to be 30 cm then the volume of the solid layer is calculated to be 50,000 l (WHC-SD-W320-ANAL-001, Rev. 0, p. AAtt 19-3).
This implies a stored energy density of order 10 MJ/l. If it is assumed that the specific heat of this material is similar to Cp for bismuth, the release of ES as a result of the dissolution of the solid would raise the temperature by:
Eq.
This result should, of course, be seen in perspective. First, it should be noted that a heat generation discrepancy by a factor of order 3 would be apparent using older conversion factors. Nevertheless, we consider it desirable to base such calculations on the most accurate parameters available. Secondly, the stored lattice energy result is no more than an illustration of the upper limit in the magnitude of such an effect. The assumption that the missing heat is entirely due to plutonium is not credible; rather it is expected that fission products would be responsible for the bulk of any heat generation found in the heel layer. It is, however, prudent to consider potential hazards associated with the dissolution of solids that have the

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37-42
ACHIEVEMENT OF PROCESS CONTROL, SAFETY, AND REGULATORY COMPLIANCE IN A MIXED WASTE EVAPORATOR SYSTEM AT THE HANFORD SITE
USING DATA QUALITY OBJECTIVES
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ABSTRACT
The Data Quality Objectives (DQO) Process was applied to the operation of the 242-A Evaporator at the Hanford Site. A team consisting of representatives from process engineering, environmental engineering, regulatory compliance, analytical laboratories, and DOE utilized the step by step DQO process to define the issues, variables, and inputs necessary to develop the decision rules which govern plant operations. The sampling and analyses required to make these decisions was then optimized concerning factors such as sample number, total analyses, cost, radiation exposure, quality assurance, and deliverables.
PROCESS DESCRIPTION
The 242-A Evaporator process (see Fig. 1) employs a conventional forced circulation, vacuum evaporation system designed to concentrate mixed waste solutions. Candidate waste feed tanks are characterized, then pumped to the Evaporator feed tank. This feed stream is then separated into three process streams: concentrated slurry, process condensate, and a gaseous process exhaust stream. Two non-hazardous effluent streams, steam condensate and cooling water, are also produced and discharged to an evaporative pond.
The concentrated bottoms, or slurry, contains the majority of the radionuclides and inorganic constituents. This stream is recycled back through the plant until it is concentrated to target levels, then is pumped to double shell tanks to be stored for further treatment. The condensed boiloff, or process condensate, contains primarily water as well as trace organic material and a greatly reduced concentration of radionuclides. This stream is stored at the Liquid Effluent Retention Facility awaiting treatment capacity in the Effluent Treatment Facility. The process exhaust, or vessel vent stream, consisting primarily of non-condensible gasses drawn from the condenser system, is filtered and discharged through an exhaust stack. The Evaporator process is continuous with typical feed flow rates of 303 L/min (80 gal/min) to 454 L/min (120 gal/min) resulting in process condensate flow rates from 114 L/min (30 gal/min) to 227 L/min (60 gal/min).
Fig. 1.
PROBLEM STATEMENT
The primary goal of the Evaporator program is to operate the facility for the purpose of reducing the volume of mixed waste received and stored in the underground tank system. In today's environment, the requirements to achieve that primary goal have become considerable and complex. The Data Quality Objective Process, as applied to the Evaporator program, was used in the planning and design of data collection activities in order to support the decisions related to the successful operation of the Evaporator facility. Successful operation is defined as meeting the campaign objectives while operating within established requirements.
DECISION VARIABLES
Each subsystem of the facility was evaluated with respect to three issues, process control, safety, and environmental compliance, to identify data requirements for each of the process streams. Process control issues are those that are directly related to maintaining the Evaporator system within acceptable process parameters; safety issues are related to operation of the plant within acceptable safety parameters; and environmental compliance to those parameters defined by federal,

state, and local regulations.
The primary issues were further subdivided into secondary issues in order to adequately address specific requirements for each of the process streams (see Table I).
TABLE Ia
TABLE I, cont'd
DECISION LOGIC
A decision rule integrates a particular parameter of interest to its action level into a single statement that describes a logical basis for choosing among alternative actions. For the Evaporator process, the decision components were divided into the various streams to address each of the parameters of interest. The decision process for all of the process streams was portrayed graphically in a set of decision logics (see Fig. 2 for an example of a decision logic diagram). This compilation of rules into a single controlling document, the Evaporator DQO Document (2), is a major advantage to engineering personnel in planning the processing campaigns. Prior to the DQO Process, the requirements for plant operations were contained in many different documents, often without basis and in many cases incompatible with one another. These problems have been corrected through utilization of the DQO Process.
Fig. 2.
DQO IMPLEMENTATION DOCUMENTATION
Many organizations support the Evaporator program and have varying areas of concern. The Evaporator DQO document forms the basis for all of the implementation documents. These lower level documents contain the details of the analyses, sampling requirements, and specific decisions that are necessary to support the operation of the 242-A Evaporator. The documents used by supporting organizations are as follows:
Tank Characterization Plan - Provides sampling and analysis requirements for candidate Evaporator feed solutions. The data is used to evaluate process control, safety, and compliance decisions prior to plant operations.
Waste Analysis Plan - Provides sampling and analysis requirements for Evaporator streams subject to Resource Conservation and Recovery Act (RCRA) requirements.
Quality Assurance Project Plan - Provides the Quality Assurance and Quality Control necessary for sampling and analysis of Evaporator streams listed in the Waste Analysis Plan.
Process Control Plan - Provides a systematic evaluation of each decision rule in the DQO (prior to starting plant operations) resulting in a control strategy for a discrete Evaporator campaign.
Facility Sample Schedule - Provides sampling and analysis requirements for process control and safety decisions while the Evaporator is operating.
CONCLUSIONS
The application of the DQO process at the 242-A Evaporator has proven to be a very successful venture. The implementation of the DQO has resulted in focused decision-making, better management control as well as savings in sampling and analysis. Table II illustrates in detail the major benefits of implementing the DQO Process.
TABLE II
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37-44
REMOVAL OF REGULATED ORGANIC COMPOUNDS FROM A TARLIKE MIXED-WASTE MATRIX
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ABSTRACT

Closure of a Department of Energy low-level radioactive waste facility under the Resource Conservation and Recovery Act resulted in the generation of mixed-waste sludge. This material had the consistency of tar and contained extractable organic halides (EOX) and volatile organic compounds (VOCs) in concentrations above regulatory criteria. Typical examples of EOX compounds are carbon tetrachloride; chloroform; 1,4-dichlorobenzene; and vinyl chloride. These organic compounds in the feed material were treated using a combination of technologies. The objective of this treatment was to meet the toxicity characteristic leaching procedure regulatory levels, land disposal restrictions criteria, and the waste acceptance criteria for a commercial land disposal facility.

The initial EOX concentration in the waste feed was 6,880 ppm. The initial VOC concentration, primarily tetrachloroethene (PCE), was in the order of magnitude of 1,000 ppm. A total quantity of approximately 110 kg of mixed-waste sludge was oxidized using concentrated nitric acid. Two small batches (1 kg each) were processed concurrently because of the tarlike consistency of the feed material, the exothermicity of the oxidation reaction, and the limited amount of waste.

In the oxidized product, the EOX concentration was reduced to 756 ppm, and the VOC concentration, primarily PCE, was reduced to 150 ppm. The efficiency of the oxidation process in removing the EOX and VOCs, primarily PCE, was approximately 53 percent and 36 percent, respectively; EOX and VOC (primarily PCE) concentrations in the oxidized product were adjusted for product mass increase. The pH of the oxidized product was less than 2.0 because of unreacted acid in the product matrix.

The acid remaining in the oxidized product was neutralized with concentrated sodium hydroxide solution. The purpose of the neutralization process was to raise the pH. It was noted from visual observation, analytical data, and thermodynamic data that the exothermicity of the neutralization process helped "steam-strip" the remaining EOX and VOCs. In this oxidized, steam-stripped product, the concentration of EOX was further reduced to 366 ppm, and the VOC concentration, specifically PCE, was reduced to 8.4 ppm (estimated value). The efficiency of the steam-stripping process in removing PCE was approximately 92 percent (adjusted for product mass increase). The final treated product had a matrix of low viscosity and was amenable to solidification with portland cement. The solidified product met all the above-mentioned standards for land disposal.

BACKGROUND

During the closure of the Colonie Interim Storage Site (CISS), a Department of Energy (DOE) facility in Colonie, New York, radioactively contaminated sludge waste was treated for disposal. This waste was regulated under the Resource Conservation and Recovery Act (RCRA).

The sludge had a component of tarlike consistency that contained extractable organic halides (EOX) and volatile organic compounds (VOCs) at concentrations above regulatory levels (1,2). EOX is essentially the same as halogenated organic compounds, which are analyzed following an extraction method. A thermal treatment process was used in an effort to desorb the VOCs and EOX from the sludge matrix. However, heating the sludge to a temperature of 393-423K caused the tarlike component in the sludge to become wet, sticky, and oily, consequently hindering the desorption process. As a result, the thermal treatment approach was unsuccessful in meeting treatment standards. The tarlike component was separated from the sludge; the subsequent removal of organic compounds from this component of the sludge using a two-phased treatment approach is the subject of this paper.

The mixed-waste sludge was listed under RCRA as F001, F007, F008, and F009 waste. The regulatory guidelines governing the treatment goals are the toxicity characteristic leaching procedure (TCLP) regulatory levels, land disposal restrictions (LDRs), and the waste acceptance criteria (WAC) for a commercial land disposal facility. The treatment standards are reported in later sections of this paper.

The site is a facility primarily contaminated with low-level radioactive waste (LLRW) and is being prepared for decommissioning. Because of radiological controls, waste treatment had to be performed within the facility. Considering the limited quantity of the waste to be treated, it was not cost-effective to procure a ready-made system. Therefore, equipment was fabricated onsite, and a system was set up to treat the waste.

TREATMENT METHODOLOGY

A two-phased treatment methodology was designed to treat the EOX and VOCs in the

waste matrix:
Phase 1: Nitric acid (67 percent by weight) was used as the oxidizing agent to remove EOX and VOCs, primarily tetrachloroethene (PCE), from the waste feed. This process was effective in breaking up the tarlike consistency of the material, in removing approximately 53 percent of the initial EOX content, and in meeting the treatment standards for EOX. Approximately 36 percent of the initial VOC content (primarily PCE) was removed from the waste matrix. The primary VOC, PCE, may have been broken down into its oxidation by-products, but it was probably not desorbed because the temperature attained by the reactants was 377K, which is lower than 394K, the boiling point of PCE.
Phase 2: The unreacted acid in Phase 1 was neutralized using sodium hydroxide solution (50 percent by weight). The exothermic neutralization process and consequent steam generation caused the removal of 92 percent of the Phase 1 residual VOC content, primarily PCE. It also further reduced the EOX concentration.
It is postulated that the removal of organic compounds (EOX and VOCs) resulted from the decomposition or steam-stripping of the compounds. Steam-stripping is considered to be the primary mechanism in the removal of organic compounds, primarily PCE, because of the copious amounts of steam generated as a product of the neutralization process.
PHASE 1: OXIDATION
Nitric acid was used as the oxidant to break down the EOX and VOCs. Typical examples of EOX compounds are carbon tetrachloride; chloroform; 1,4-dichlorobenzene; and vinyl chloride. A specific oxidation reaction mechanism could not be established because of the variety of halogenated organic compounds in the waste matrix. The effluent gas composition indicated that nitric acid helped fragment the organic compounds into smaller molecules.
Process Setup, Operation, and Maintenance
The initial EOX and PCE concentrations of the waste feed were 6,880 ppm and in the order of 1,000 ppm, respectively. Approximately 110 kg of mixed-waste sludge was treated. Because of the tarlike consistency of the feed material, the exothermicity of the oxidation reaction, and the limited quantity of waste, the amount of waste to be treated was limited to 1 kg for each batch.
Two 1-gal (3.8 10-3 m3) reactor vessels were placed in the work area under the fume hood. Each reactor was sealed with a lid that had an effluent gas emission port, pressure vent, dial thermometer port, and nitric acid dispensing port. Each mixer was connected to a "knockout" drum, an alkaline scrubber, and an activated carbon drum, all connected in series. The outlet from the activated carbon drum was connected to a vacuum pump and finally to the high-efficiency particulate air (HEPA) system. Fig. 1 shows the system setup for the oxidation process.
Fig. 1.
The knockout drum was a 55-gal (0.2-m3), 17C drum. A cooling jacket constructed around the drum consisted of an overpack filled with blue ice. The knockout drum was cooled to condense the steam generated by the process and minimize the flow of steam to the subsequent units. Entry of steam into the scrubber could dilute the scrubber liquid.
A 55-gal (0.2-m3) drum containing 40 gal (0.15 m3) of 35 percent sodium hydroxide solution served as the alkaline scrubber unit. A diffuser was fabricated and installed at the bottom of the unit to sparge effluent gas into the scrubber liquid at a rate that was sufficient to scrub the gas. The diffuser frame was fabricated from 1.3 10-2-m diam, 0.3-m-long polyvinyl chloride pipes joined in a square configuration. The gas inlet was fabricated on the side using a "T" connector. Twenty-four diffuser holes (0.0625-in. or 1.6 10-3-m diam) were drilled on each of the other three sides. The holes faced the bottom of the drum so that the effluent gas bubbles would have a longer contact time with the scrubber liquid. A metal rod (anchor) tied diagonally to the diffuser frame weighed it down and kept it from floating. These design details of the scrubber unit were determined after a series of trial runs. Additionally, stoichiometric calculations were performed to determine the specifications of effluent gas lines. Fig. 1 includes a sketch of the diffuser.
The 55-gal (0.2-m3) activated carbon drum was a vendor-supplied unit. When monitoring instruments indicated imminent breakthrough of the activated carbon, the drum was replaced.
Based on bench-scale study results, approximately 2.1 10-3 m3 of 67 percent nitric acid was required to treat 1 kg of feed material; however, 20 percent additional

nitric acid was used to ensure completion of the oxidation treatment. Therefore, approximately 2.5 10-3 m3 of nitric acid was used for 1 kg of feed material. Two batches of feed material were processed concurrently in two 1-gal (3.8 10-3 m3) mixer-reactors.
Nitric acid was dispensed slowly into each mixer-reactor through a separatory funnel while the mixer was operating. The effluent acidic gases (nitrogen dioxide, carbon dioxide, chlorine, and hydrochloric acid) were neutralized in the alkaline scrubber. The temperature was monitored closely. The completion of reaction could be determined by the stabilization of the temperature and a visual inspection of the treated product. The temperature ranged between 339 and 377K for different batches, and treatment duration for each batch was approximately 35 minutes (2,100 s). The oxidized product was transferred into a lined 55-gal (0.2-m3), 17C drum. Engineering controls were used to ensure that the entire process was conducted safely.
Engineering Controls
The work area was lined and bermed to contain any potential spills of waste or chemicals. Workers used chemical-resistant suits and gloves to handle nitric acid. The reactor drum was placed under a fume hood connected to a 9.44 m3/s HEPA vacuum system. This system vented any fugitive emissions from the immediate work area. Any gases escaping from the alkaline scrubber were also captured onto the activated carbon canister connected to a HEPA vacuum system. Class A, B, C, and D extinguishers were kept nearby.
The effluent gases that were expected to be generated during the oxidation were primarily nitrogen dioxide, chlorine, hydrochloric acid, and carbon dioxide. Generation of carbon monoxide by the reaction of hot nitrogen dioxide gas with activated carbon in the carbon adsorption unit was also possible. These gases were drawn into the units by a vacuum pump operating at a flow rate of 7.1 10-3 m3/s. Continuous monitoring of effluent gases was performed to ensure that levels in the breathing zone did not exceed health and safety limits prescribed by the National Institute for Occupational Safety and Health and the Occupational Safety and Health Administration. Additionally, organic vapor analyzer (OVA) monitoring was performed during the process operation; VOCs were not detected.
Results
The analytical results for the target organic compounds in the oxidized product are summarized in Table I. The oxidation process was successful in meeting regulatory criteria for EOX, and the efficiency in removing the EOX was approximately 53 percent (adjusted for product mass increase). The efficiency in removing the PCE during the oxidation process was approximately 36 percent (adjusted for product mass increase); the treatment standard for PCE was not met.
PHASE 2: STEAM-STRIPPING
Unreacted nitric acid (67 percent by weight) caused the pH in the oxidized product to be less than 2. To make this product amenable to final disposition, the acid had to be neutralized with a suitable alkali. Also, the heat generated during the neutralization process would help desorb the VOCs (primarily PCE). The physical, chemical, and thermodynamic properties of PCE are presented in Table II.
Reaction Chemistry
Sodium hydroxide solution (50 percent by weight) was selected for the neutralization process for the following reasons:

- Equimolar amounts of nitric acid and sodium hydroxide solution are required to effect the neutralization.
- Nitric acid and sodium hydroxide solutions are both strong electrolytes and hence generate substantial heat during neutralization. This heat, if properly distributed within the waste matrix, would help desorb the VOCs.
- Neutralization of nitric acid with sodium hydroxide can be accomplished with minimal generation of nitrogen dioxide gas.

Neutralization of nitric acid with sodium hydroxide produces an environmentally benign, water-soluble salt (sodium nitrate) and water (Reaction #1, Table III). This reaction is exothermic and generates 89.6 kJ of enthalpy for every g-mole of nitric acid neutralized. Thermodynamically, approximately 50 percent of this enthalpy is absorbed by the water in Reaction #1 (Table III) to be vaporized to steam in Reaction #2 (Table III). This steam helps in stripping the VOCs (primarily PCE) from the waste matrix.
The thermodynamic values calculated for the nitric acid-sodium hydroxide neutralization reaction are summarized in Table III.

Process Setup and Operation

The product of Phase 1 was the feed of Phase 2. The feed mass (471 kg) was made up of acid-digested sludge (133 kg) and unreacted acid (338 kg). The EOX concentration in the feed was 756 ppm, and the PCE concentration in the feed was 150 ppm.

The feed was divided into nine approximately equal volume batches. Because of the corrosivity of the feed material (pH less than 2), each batch of feed material was put into a 90-mil, high-density polyethylene (HDPE) liner, contained within a 55-gal (0.2-m3), 17C steel drum. The neutralization process was carried out in these lined drums. The batches of feed material weighed an average of 52 kg, and each batch occupied approximately one-sixth of the reaction drum. Feed batches were kept small because of the exothermicity of the reaction and consequent steam generation and to allow room for expansion of the contents during the reaction.

Sodium hydroxide solution was added at an average rate of 1.7 10-6 m3/s. The reactants were manually mixed continually with a four-blade paddle agitator. The gradual addition of alkali and the continual mixing of the reactants ensured a uniform reaction characterized by gradual generation and distribution of heat and steam. The temperature readings were taken outside the wall of the drum; the temperature was approximately 422K. Fig. 2 shows the process setup.

Fig. 2.

The reaction drum was connected to the activated carbon drum. The effluent gases from the reaction process (primarily steam and gaseous PCE) were drawn out by a vacuum pump operating at a flow rate of 7.1 10-3 m3/s. Additionally, OVA monitoring was performed during the process operation; VOCs were not detected. From qualitative industrial hygiene data, it is postulated that steam-stripping is the mechanism of VOC (PCE) desorption from the matrix. Phase 2 process parameters and a summary of the stoichiometric and thermodynamic calculations are presented in Table IV.

Result

PCE(l) + 10-3)Hrxn,T' PCE(g) + PCE (l)

(1.00 wt. fract.) (0.92 wt. fract.) (0.08 wt. fract.)

It was concluded that a substantial amount of enthalpy is generated during the acid-base neutralization process. After 50 percent loss of this heat is accounted for by conduction and radiation and with a heat transfer efficiency of 10 percent, only one one-thousandth of this heat was required to desorb about 92 percent of the PCE from this waste matrix.

Although the regulatory treatment standard for EOX was met during Phase 1, the EOX concentration was further reduced by 29 percent during Phase 2. The Phase 2 steam-stripping analytical results are summarized in Table V.

The Phase 2 treatment result for PCE (8.4 j ppm) is not the final result; the concentration of this compound was further reduced during the solidification process (see "Final Disposition").

HEALTH AND SAFETY

Before the treatment work began, a hazardous work permit was issued, and workers participated in a prejob briefing. The action level protective gear was equivalent to EPA level C (7). The respiratory protection included National Institute for Safety and Health-approved full-face respirators with GMC-H cartridges. The protective clothing included a chemical-resistant suit, hard hat, chemical-resistant gloves, and chemical-resistant boots. Additional protection included a buddy system and two-way radio communications systems. Industrial hygiene monitoring of the workers and breathing zone in the work area was conducted during the work.

Both the treatment phases were exothermic, resulting in dissipation of heat to the surroundings. Workers donning the protective clothing and working in the hot surroundings were vulnerable to heat stress. Therefore, workers were monitored for heat stress, and work assignments were rotated to alleviate this potential health effect.

FINAL DISPOSITION

The oxidized, steam-stripped product was a matrix of low viscosity that was amenable to solidification. A bench-scale test was performed using portland cement. The pH of the waste was adjusted close to 11. Approximately 25 ml (2.5 10-5 m3) of a plasticizer and 1.4 kg of portland cement were used to solidify 1 kg of the waste. The material was allowed to set at least 24 hours.

The solidified material was analyzed for isotopic uranium, thorium-232, radium-226,

total volatiles, EOX, ignitability, total cyanides, percent solids, pH, paint filter test, reactivity for cyanide and sulfide, and TCLP-total.
Based on successful completion of a bench-scale test, full-scale solidification was performed, and the solidified material was devoid of free liquid in the waste matrix. The solidification results are presented in Table VI and Table VII a, b. The solidification test results met the LDR standards, TCLP regulatory levels, and all WAC for the commercial disposal facility contracted for final disposition of the waste.

DISCUSSION

Figure 3 presents the percentages of residual organic compounds in the waste stream following Phases 1 and 2 and the final disposition.

Fig. 3.

It is reported in published literature and corroborated by the present work that destructive oxidation of organic compounds with nitric acid is energetically favorable. The oxidation of organic compounds is usually initiated by the production of organic radicals generated by dissolved nitrogen dioxide and nitric oxide in solution. The oxidation rate for nonaliphatic organic compounds is surface-area dependent and relatively fast compared to the aliphatic organic compounds (10). The removal of EOX from the subject waste matrix was achieved primarily by the nitric acid oxidation method.

Nitric acid could be regenerated in a standard air-driven acid recovery system by reoxidizing the released nitric acid, nitric oxide, and nitrogen dioxide. A small amount of waste was treated; therefore, this recovery system was not implemented in the Phase 1 treatment because this oxidant regeneration process would not have been cost-effective for this project.

Steam-stripping can be used to desorb VOCs (like PCE), which have vapor pressures in the range of 1.3 10-2 Pa to 2.4 104 Pa at 298K (11). In the past, acid-base neutralization has been used successfully to treat the corrosivity characteristic of hazardous waste. In the present case, it is postulated that the exothermicity of the neutralization process helped in making the waste matrix flowable, and the consequent generation of steam aided in stripping the VOCs (primarily PCE) from the matrix.

ACKNOWLEDGMENT

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37-45

PROCESSING OF STEAM GENERATOR CHEMICAL CLEANING SOLVENT AT THE PALO VERDE NUCLEAR STATION

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ABSTRACT
Arizona Public Service Company (APS) and B&W Nuclear Technologies (BWNT) recently completed chemically cleaning the steam generators at Units 2 and 3 of the Palo Verde Nuclear Generating Station. Over 1,433,000 liters of low-level radioactive solvents that were generated during these cleanings are currently being processed on-site. Chemical cleaning solutions containing high concentrations of organic chelating wastes are difficult to reduce in volume using standard technologies. Normal methods for evaporating low-level radioactive waste solutions often use high maintenance evaporators that can be costly to maintain and have heat transfer surfaces that are easily fouled. To address the volume reduction of spent, low-level radioactive, chelating-based chemical cleaning solutions, BWNT developed the ECOSAFER Liquid Volume Reduction System (LVRS).
The process being used at Palo Verde involves three distinct processing methods*:
The evaporation step uses BWNT's submerged combustion evaporator (SCE) that has been successfully used at Arkansas Nuclear One-Unit 1, Three Mile Island-Unit 1, and Oconee on similar waste.
The Synthetica DetoxifierR reduces the concentrate from the SCE to its elemental form (i.e., metals and metal oxides only) for the best possible volume reduction.
The polishing step of the distillate uses filtration and reverse osmosis (RO) technology that has been used extensively by Ontario Hydro to assist in their processing of chemical cleaning solvent.
INTRODUCTION
Due to corrosion mechanisms associated with the secondary side deposits of the Palo Verde steam generators (SGs), APS implemented a program to chemically clean Units 1, 2 and 3 steam generators. Units 2 and 3 were cleaned in early 1994 with excellent results. Early indications show that the cleaning has had a positive effect on reducing the effects of the corrosion mechanisms, and should therefore, result in a longer life for these steam generators. The Unit 1 cleaning is scheduled for April 1995.
Unfortunately, in early 1994 some of the waste solvent remaining at the completion of each cleaning was determined to be "mixed" waste, i.e., it was "low-level radioactive" waste and contained chromium (a hazardous metal) at levels just above the regulatory limit of 5 ppm. On-site interim processing successfully treated the Unit 2 waste to reduce the chromium concentration to less than 5 ppm and thus remove the solvent from the "mixed" waste classification. All remaining wastes were successfully blended to a uniform concentration of organics and metals with a chromium concentration of less than 5 ppm, in preparation for future volume reduction. The volume reduction process reduces the remaining solvents to a dry residue suitable for long-term storage as low-level radioactive waste. Planning and equipment to process the waste to be generated in April 1995 from the Unit 1 cleaning are being formalized at this time.
BACKGROUND
Unit 2
Based on previous steam generator chemical cleanings conducted in the U.S. prior to 1994, APS anticipated that the waste solvent remaining after the cleaning of Unit 2's steam generators would be low-level radioactive waste with less than 5 ppm chromium. When a portion of the actual waste solvent was analyzed to have a chromium content of approximately 10 ppm, the initial waste processing plans of volume reducing the solvent by evaporation, and then incinerating the concentrated waste needed to be modified. Current incinerator facilities can only accept low-level radioactive waste (not licensed for the treatment of hazardous or mixed waste streams).
The chromium source has since been determined to be cumulative corrosion (during the iron solvent step application) of the SGs stainless steel components. Even though the corrosion levels were minute, the surface area of stainless steel components is sufficiently large to significantly contribute to, and increase, the chromium level above the 5 ppm hazardous material classification limit.

APS evaluated options and decided to use a chromium precipitation approach implemented by an experienced vendor. This method involved blending the "mixed" waste iron solvent with the waste rinse solvent to lower the iron concentration, raising the pH with sodium hydroxide, adding large volumes of lime, adding a flocculent, filtering the solution through a filter press to remove the precipitated metals (including chromium), and then lowering the pH of the remaining solution. The lime cake containing the chromium and a portion of the iron was packaged and sent off-site for volume reduction after acceptable chromium leachability was verified using the Toxicity Characteristics Leaching Procedure (TCLP). Though the remaining solvent was verified to have chromium levels below 1 ppm, it still remained a low-level radioactive waste. Processing all the hazardous waste was completed within 60 days from the time of generation.

Unit 3

The Unit 3 steam generators were cleaned soon after Unit 2. Due to the large volume of the radioactive lime cake that was generated during the Unit 2 processing, APS immediately starting investigating more environmentally sound, economical and technically superior processes that could be used. Many potential technologies indicated that a uniform waste stream composition would be required to ensure proper and efficient operations. Based on this knowledge, APS blended the waste solvent from the Unit 3 cleaning with the waste rinse solvent from the Unit 3 cleaning. All of the resultant uniform waste from Unit 3 was verified to be less than 5 ppm chromium.

Due to the unique nature of the waste involved; requirements to minimize the final waste volume for long-term, on-site storage; and the zero-release criteria at the Palo Verde site, all potential technologies involved "first-of-a-kind" equipment or processes. APS decided to implement the process with the lowest environmental risk and the highest probability for success. The chosen technology would allow for all processing to be conducted on-site, under APS control, with no waste being shipped off-site. This technology and process would destroy the chemical organics, allow the release of the purified water to the on-site evaporation pond and result in the lowest overall dry radioactive waste volume to be stored on-site.

PROCESSING EXISTING UNIT 2 AND 3 WASTE

BWNT uses field-proven technology for the reduction of chemical cleaning waste volumes, including treating and releasing of gaseous and liquid effluents. This system volume reduces cleaning waste solvent and rinse solutions by more than 99%. The solvent waste and rinse water waste are evaporated with a Submerged Combustion Evaporator (SCE) to achieve an initial volume reduction of 70% to 90%. The distillate from the SCE undergoes filtration and then reverse osmosis to meet the required secondary side discharge limits at Palo Verde. The concentrates from the SCE are fed to a high temperature detoxifier that destroys all organics and further reduces the waste volume leaving a dry end product.

This unique approach provides a straightforward, flexible and efficient evaporation, purification and detoxification system for processing these low-level radioactive waste solutions.

The presence of chromium in the solvent adds a complexity to the processing. A test program was completed to show that the residue from the detoxifier process will result in a waste that will pass the TCLP test. No significant amount of chromium is seen in the distillate since less than 1% of the chemical constituents in the SCE carry over to the distillate.

The flow diagram for the waste processing task at Palo Verde is shown in Fig. 1. Each process is fully explained in the following sections.

Fig. 1.

Evaporation Step

All of the Palo Verde waste is first fed to the SCE. Air from a blower and propane are mixed and enter the burner tube of the SCE, which is below the level of the waste liquid. Combustion of the propane/air mixture takes place in the burner tube after ignition by an electric element. The 1 MW of heat generated leaves the burner as sensible heat in the combustion gases. The hot gases are released, under the liquid level, as a tremendous number of bubbles, creating the maximum surface area for heat transfer. Since these gases are bubbling through the waste solvent and are therefore, directly in contact with the solution, the heat is transferred immediately to the liquid. The combustion gases in transferring their heat energy, are cooled, and escape the solution at the same temperature as the liquid. The

temperature of the solution is raised to its boiling temperature and the water vapor leaves the solution, intimately mixed with the off-gases of propane combustion. Since the SCE does not have heat transfer surfaces that could become fouled during operation, it requires only minimal maintenance.

Bubbling combustion gases through a liquid lowers its boiling point. Therefore, under submerged combustion conditions, the solvent boils at a temperature appreciably below its atmospheric boiling point. For example, when burning propane with 5% excess air, the boiling point of water is approximately 89C. As the amount of excess air is increased, the boiling point decreases accordingly (1).

The waste-liquid feed to the SCE passes through a magnetic flow meter with a flow totalizer that is used to meter and control the feed to the evaporator. An operator sets the evaporator feed control valve at an equilibrium flow rate of approximately 15 liters per minute. When the evaporator concentrate reaches the desired concentration (determined by specific gravity measurement), a process valve is opened to allow pumping of the bottoms to the detoxifier system. The operator controls the evaporator level by adjusting the feed to, and/or discharge from, the evaporator. When the system reaches equilibrium, the rate of evaporator concentrate removal plus the rate of distillate production is equal to the rate of feed.

The exhaust gases from the SCE are drawn into the vent trailer. The gas stream is scrubbed with a unique narrow-gap venturi scrubber to remove particulates. These gases are cooled to remove any excess moisture and then reheated in the stack to vaporize any residual moisture and minimize fouling the HEPA filters.

Detoxification Process

The concentrated bottoms from the SCE are fed to the Synthetica Detoxifier process. The Synthetica DetoxifierR involves several steps. In the first step, the liquid organics are vaporized in a heated screw evaporator (HSE) by continuously feeding the liquid. The dry residue at the outlet of the HSE collects in a 55-gallon drum that is periodically changed. Within the HSE, the organic vapors (ie. ethylenediaminetetraacetic acid, ethylenediamine, hydrazine) are first exposed to superheated steam. In the HSE, steam reforming chemistry starts to occur at temperatures from 315 to 593oC. The steam reforming gases enter the second step (approximately 1370oC), which takes place in the Detoxification Reactor. Here, all the organics that were vaporized continue to undergo steam reformation and are converted to non-toxic synthetic vent gas (syngas) (2). The dry residue from the HSE contains the elemental metals or associated metal oxides removed from the Palo Verde steam generators and is stored in stainless steel 55-gallon drums. Testing indicates that any chromium in this state passes the TCLP test and that the final residue will fall within the Class A radioactive waste category.

Reverse Osmosis

The final major processing step is the purification of the distillate from the SCE to ensure it can be discharged to the secondary side evaporation ponds. The treatment train consists of filtration followed by single- or double-pass reverse osmosis (DPRO).

Table I shows the characteristics of the components of concern in the distillate stream that is fed to the membrane filtration system. Table II lists the effluent criteria for the final permeate stream released from the RO system to the Palo Verde evaporation pond. A volume reduction factor of 90% has been achieved using this treatment train. The remaining 10% concentrate is either fed back to the SCE or to the Detoxifier.

TABLE I

TABLE II

SUMMARY

APS is utilizing the Evaporator/Detoxifier/Reverse Osmosis system to process all waste generated from chemically cleaning all Palo Verde SGs. This process is believed to be the option that produces the least amount of long-term storage solids, is the most economical, and provides for the most environmentally sound method for dealing with the large amount of existing or future spent cleaning solvents.

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CONVERSION OF ELEMENTAL AND AQUEOUS MERCURY-BEARING MIXED WASTES TO NON-HAZARDOUS FINAL WASTE FORMS

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ABSTRACT

A variety of radioactively-contaminated, mercury-bearing wastes have been generated at DOE facilities. These wastes are classified as D009 characteristic and, therefore, are considered as mixed wastes. Disposal of these special wastes in a licensed radioactive landfill requires that the final waste form be non-hazardous. Advanced Recovery Systems, Inc. (ARS) has developed technology to convert a variety of mercury-bearing mixed wastes to final non-hazardous waste forms. This paper presents the results of two successful treatability studies to render mercury-bearing mixed waste as non-hazardous per requirements in 40 CFR.

The first treatability study was conducted on elemental mercury contaminated with low levels of tritium. The study successfully demonstrated the effectiveness of the amalgamation treatment process to immobilize elemental mercury to meet the RCRA treatment standard for D009 wastes. The amalgam produced in proving tests using two kilograms of elemental mercury had TCLP leach mercury concentrations averaging 0.06 mg/l, below the regulatory limit of 0.2 mg/L. Several significant findings were derived from this study. The first is that the waste mercury only comes in contact with the ultimate disposal container and does not contact processing equipment. The second is that expensive blending/mixing equipment is not required to convert the waste to a final non-hazardous waste form.

The second treatability study was performed on samples of aqueous and organic-bearing mercury solutions contaminated with trace levels of tritium. The mercury treatment process selected for this study was demonstrated as an effective treatment for the mercury-bearing solutions. Both the organic and aqueous phases of the treatability samples were treated to produce test filtrates containing 0.02 mg/L mercury and solid residues having TCLP leachate mercury concentrations averaging 0.003 mg/L. This result is nearly three orders of magnitude less than the RCRA treatment standard of 0.02 mg/L mercury in TCLP leachate. An additional finding of this work was that the final waste form was a solid, free-standing monolith with good handling and storage characteristics.

In summary, the above studies have successfully converted several different forms of mercury mixed wastes to non-hazardous waste forms that fully achieve land disposal requirements. A treatment system has been designed and is to be permitted for treating mercury bearing debris generated at a fuel production facility. This technology is applicable to other DOE mercury wastes and may have merit for mercury mixed wastes containing other hazardous metals such as cadmium, silver, and lead.

INTRODUCTION

A variety of mercury-bearing, radioactively-contaminated wastes have been generated at DOE facilities. These wastes have been classified as D009 mixed wastes. Disposal of these special wastes in a licensed radioactive landfill requires that the final waste form be nonhazardous. Advanced Recovery Systems Inc. (ARS) has developed technology for removing the hazardous characteristic of mercury from a variety of radioactive waste materials. This paper presents the results of two successful treatability studies to render D009 mixed wastes as nonhazardous per 40 CFR treatment standards.

The first study was conducted on elemental mercury contaminated with low levels of tritium. The objective of this study was to demonstrate the effectiveness of an amalgamation process for immobilizing bulk liquid mercury to meet the RCRA treatment standard for D009 wastes.

The second study was performed on mixtures of organic and aqueous solutions containing dissolved mercury and trace quantities of tritium. The objective of this study was to demonstrate the effectiveness of a chemical stabilization process for removing mercury from the waste solutions while producing a solid waste matrix meeting RCRA D009 treatment standards. This process utilizes a proprietary reagent (patent pending) which has been shown to destroy mercury complexes and stabilize the mercury in a non-leachable solid matrix.

DISCUSSION AND RESULTS

Mercury Amalgamation

Scoping Tests. Scoping Tests were performed in order to estimate the required processing parameters for treating the elemental mercury. The primary requirement of the process was to produce amalgam having TCLP leachate mercury concentrations below the 40 CFR 268.41 treatment standard of 0.2 ppm for D009 waste. In the Scoping Tests an arbitrary limit of 0.1 ppm mercury was chosen as a target to insure results were safely below the 0.2 ppm standard. Testing was performed per the requirements of 40 CFR 261, Appendix II, Method 1311 and SW-846 analytical and characterization guidelines. The parameters of the Scoping Tests were selected such that the data could be evaluated in a 23 Factorial Experimental Design Matrix. Table I details the layout of the test matrix. The three variables studied were:

TABLE I

1. Metal quantity;
2. Acid strength;
3. Acid quantity.

Table II contains the data from the Scoping Tests. Evaluation of this data in the experimental design matrix indicated the following:

1. Metal quantity was the most significant variable studied. Increasing the metal to mercury molar ratio from 3 to 10 decreased the average TCLP leachate concentration from 0.47 to 0.07 ppm.
2. Acid strength was a marginally significant variable. Increasing acid strength from 1N to 5N decreased the average TCLP leachate mercury concentration from 0.30 to 0.24 ppm.
3. Acid quantity was not a significant variable over the range studied.
4. There were no significant interactions between the variables.

Figure 1 is a visual representation of the effectiveness of amalgamation for stabilizing elemental mercury. The bars on this chart represent the average TCLP leachate mercury concentrations from the Scoping Tests and the solubility of elemental mercury in water. The solubility of elemental mercury in water was selected as an approximation of a TCLP performed on an untreated sample of liquid mercury (0:1 reagent to Hg molar ratio).

TABLE II

Fig. 1.

Proving Tests

Three Proving Tests were performed to evaluate the processing parameters established as a result of the Scoping Tests on larger samples of elemental mercury. Each test was performed using about 2000 grams of elemental mercury. In addition to TCLP evaluations, the amalgam produced in these tests was subjected to a centrifuge test to ensure that no free liquid mercury existed in the final stabilized form. The centrifuge tests consisted of centrifuging 200 gram samples of the amalgam for 30 minutes at 2000 RPM.

Table III contains the data from these tests. The first two tests, Tests 17 and 18, were simply scale-ups of the successful Scoping Tests. The final test, Test 19, was a demonstration of a proposed actual treatment process.

The data from Tests 17 and 18 demonstrated that the amalgamation process described in this report is an effective treatment for elemental mercury contaminated with radioactivity. The upper limit of the 80% confidence intervals of the TCLP leachate mercury concentrations for Tests 17 and 18 were 0.10 and 0.05 ppm respectively. These are both safely below the RCRA regulatory limit of 0.2 ppm mercury. The centrifuge tests for these amalgams produced no detectable liquid mercury. The treated material from these tests had an average pH of 6.37 (per Method 1311 TCLP protocol) and therefore do not exhibit the hazardous characteristic of corrosivity.

TABLE III

Test 19 was performed as a demonstration of a proposed treatment process. The purpose of this test was to demonstrate the production of homogeneous amalgam without the use of intrusive mixing. The basis for the test was the observation that a globule of liquid mercury would disperse as small droplets throughout the amalgamating metal with minimal mixing intensity. Addition of the acid caused an immediate reaction of the mercury and metal to produce a uniform matrix.

The reagent quantities and mixing times for Test 19 were basically the same as those used for Tests 17 and 18. The primary difference between Test 19 and the other Tests was the method of mixing. Tests 17 and 18 were mixed intrusively using an electric

mixer. Test 19 was mixed by rotating the reagents and mercury sample end-over-end in a closed 2-liter polyethylene bottle. As the bottle was rotated, the liquid mercury flowed through the bed of metal and after about 5 minutes was uniformly dispersed. Addition of the acid resulted in the production of uniform amalgam. The upper limit of the TCLP leachate mercury concentration for this material was 0.04 ppm. The centrifuge test did not produce any detectable liquid mercury.

The significance of this demonstration was the fact that amalgam can be produced without the use of intricate and expensive processing equipment. A secondary benefit from this technique is that the mercury does not contact any equipment or containers except for the container it will be ultimately disposed in. The final 2-liter package weighed about 9 kilograms. Packages of this size and weight could be easily packaged in larger containers, such as 55-gallon drums, for disposal.

Treatment of Mercury Bearing Solutions

Sample Characterization. The treatability sample consisted of five bottles of various types of mercury waste solutions. Table IV is a description of these solutions prior to characterization. The treatability sample was prepared by compositing the five bottles of waste and mixing. This produced a milky-white emulsion which was stable for about one minute. Twenty milliliters of a non-ionic surfactant were added to the sample and the solution mixed. The resulting emulsion was stable for about two hours. While in one phase, the sample was characterized for mercury content by cold-vapor atomic absorption analysis. The sample was found to contain 3965 ppm mercury.

Table IV

Scoping Tests

Scoping tests were performed to estimate the required processing parameters for treating the mercury waste using the ARS treatment process. The objectives for this study were to produce solid residues having TCLP leachate mercury concentrations below the RCRA treatment standard of 0.2 ppm for D009 wastes while producing process filtrates containing less than 0.05 ppm mercury. This was achieved by performing tests to determine the correct balance of proprietary reagent, ferrous sulfate, and basic oxide.

Table V is a summary of the tests performed for this study. The following were the significant findings resulting from the Scoping Tests:

1. Due to interference from the organic compounds contained in the waste, greater than anticipated quantities of treatment reagents were required to produce precipitates meeting TCLP requirements.
2. The waste could not be treated as one stream. Addition of the treatment reagents coalesced the waste emulsion into two phases. This could present a significant concern at full-scale processing.
3. An appropriate absorbing medium, when slurried with the separated aqueous phase, decolorized the solution and removed nearly all organic odor.
4. The separated organic phase could be treated by absorbing the liquid into an appropriate substrate and treating the resulting paste with the ARS process.
5. Sufficient pH adjustment is required in order for the pH's of the process filtrate and TCLP leachate to be greater than 9.
6. The large quantities of pH adjustment chemicals and iron sulfate required to treat this waste created a solid waste product. The quantity of sulfate compounds contained in the test precipitates would cause the residues from the test to set up as a solid mass. This created a final waste form with characteristics similar to what would be obtained by solidifying the stabilized mercury in a plaster-of-paris type compound.

Table V

Proving Tests

Two proving tests were performed to evaluate the parameters established as a result of the Scoping Tests using larger aliquots of the waste samples. Test 5 was performed using 2000 mL of the aqueous sample. Test 6 was performed using 250 mL of the organic phase. Figure 2 is a visual representation of the effectiveness of the ARS mercury stabilization treatment for treating this difficult waste stream.

The filtrate from Test 5 contained 0.02 ppm mercury and the average TCLP leachate mercury concentration was 0.02 ppm. These results were far below the RCRA regulatory limit of 0.2 ppm mercury. The final volume of solids generated from Test 5 was slightly less than 4 liters. This is only about twice the initial waste volume of 2 liters. As in Test 4, the precipitants from Test 5 set up into a solid mass.

Fig. 2.
The filtrate from Test 6 contained 0.02 ppm mercury. The mercury concentration of the TCLP leachate was 0.003 ppm. As in Test 5, these results were safely below the RCRA limit and the test precipitants solidified overnight.

CONCLUSIONS

Mercury Amalgamation

Amalgamation was demonstrated as an effective treatment for the tritium contaminated elemental mercury. Two-kilogram samples were treated to produce amalgam having TCLP leachate mercury concentrations averaging 0.06 ppm. These results were all safely below the RCRA treatment standard of 0.2 ppm for D009 wastes.

The amalgam from these tests set into solid masses with total weights of about 9 kg and volumes of about 2 liters for each 2 kg of mercury treated. The amalgam from these tests was subjected to a centrifuge test to ensure that free liquid mercury did not exist in the stabilized matrix. All samples tested passed this evaluation. The process tested did not generate any excess solutions. Such solutions, if produced, would create handling and possibly secondary treatment concerns.

Retesting of the amalgam after storage for two weeks indicated improvement in the sample matrices. The average TCLP leachate mercury concentrations for the samples retested decreased from 0.08 to 0.04 ppm.

The final two-kilogram test was a demonstration of a process for producing amalgam without the use of intrusive mixing or blending. The upper limit of the 80% confidence interval of the TCLP leachate mercury concentration for the amalgam from this test was 0.04 ppm. This is significant from the following standpoints:

1. Can work with small quantities of mercury at a time. This reduces occupational exposure concerns and reduces the potential for a large mercury spill.
2. Would not contaminate processing equipment. The mercury only comes in contact with its ultimate disposal container. Will not have to decontaminate processing equipment after completing mercury treatments.
3. Expensive blending/mixing equipment not required.

Stabilization of Mercury-Bearing Solutions

The mercury treatment process selected for this study was demonstrated as an effective treatment for the tritium contaminated mercury bearing-solutions. Both the aqueous and organic phases of the treatability samples were treated to produce test filtrates containing 0.02 ppm mercury and solid residues having TCLP leachate mercury concentrations averaging 0.003 ppm.

The treatment process was modified to include a pretreatment of test solutions with absorbing media. This was required to bind the organic constituents which were apparently re-extracting mercury out of the stabilized matrix. The absorbing media was slurried with the test solutions and ultimately became part of the final waste matrix. The use of absorbing media in this manner did not adversely affect the performance of the solids when evaluated by TCLP protocol.

Greater quantities of reagents were needed than anticipated based on the mercury content of the samples. This may have been due to interference by the organic constituents of the waste. The use of such amounts of reagents produced a larger volume of treated solids than anticipated. The volume of waste was increased from 4 liters to about 12 liters. The quantity of reagents used caused the final waste to solidify. This produced improved waste forms with favorable handling and storage characteristics.

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PLASMA ARC HEATED SECONDARY COMBUSTION CHAMBER*

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ABSTRACT

This paper describes a secondary combustion chamber (SCC) for hazardous waste treatment systems that uses a plasma arc torch as the heat source. Developed under a

cooperative research and development agreement (CRADA) between Retech, Inc. and Lawrence Livermore National Laboratory (LLNL), the unit is intended primarily to handle the off-gas from a Plasma Arc Centrifugal Treatment (PACT) system. It is designed to heat the effluent gas which may contain volatile organic compounds, and maintain the gas temperature above 1000C for two seconds or more. The benefits of using a plasma arc gas heater are described in comparison to a conventional fossil fuel heated SCC. Thermal design considerations are discussed. Analysis and experimental results are presented to show the effectiveness in destroying hazardous compounds and reducing the total volume of gaseous emissions.

INTRODUCTION

The Plasma Arc Centrifugal Treatment (PACT) system uses heat from a plasma torch to treat a wide variety of hazardous waste materials. Figure 1 is a diagram of a typical PACT system. To process waste, material is fed into a sealed centrifuge where it is exposed to intense heat from a transferred-arc plasma torch. Organic materials are thermally decomposed soon after entering the primary chamber and become an effluent gas. When treating hazardous substances, this off-gas must pass through a secondary combustion chamber (SCC) to assure complete oxidation of hydrocarbons and destruction of toxic compounds. Nonvolatile inorganic materials are melted in the primary PACT chamber and reduced to a molten slag. The centrifugal motion of the refractory lined tub and the pressure of the plasma arc serve to uniformly heat and mix the material.

Fig. 1.

An oxygen lance is used to provide supplementary oxygen that gets consumed by oxidizing metals and organics. By slowing the centrifuge, molten slag is discharged through the throat region into a collection mold in the collection chamber below. The final product is a glass-like solid that easily passes toxicity characteristic leachability procedures (TCLP) and may therefore be disposed of safely.

RATIONALE FOR A PLASMA HEATED SCC

The purpose of the SCC is to ensure that all of the volatile organic and inorganic material leaving the PACT unit has been completely oxidized and effectively destroyed. To achieve destruction and removal efficiency (DRE) of 99.999% or greater for organic material, the standard design for a secondary chamber keeps the gas at 1000 C in an oxygen rich environment for at least two seconds. The conventional method to achieve this temperature is to use a combustion burner fueled by oil or natural gas. This adds a substantial amount to the gas stream. A plasma torch heated SCC is more efficient in heating gas and can provide more heat for a given volume of added gas than a fossil-fuel burner. For example, a 200-kW plasma torch operates with a typical gas flow rate of 8.5 Nm3/h. Assuming a nominal loss of 40 kW in the torch head, the plasma torch system can provide 18.8 kWh/Nm3 of process energy at temperatures above the needed 1000C. In comparison, a standard natural gas burner consuming a stoichiometric amount of oxygen derived from air, at best provides only 5.3 kWh/Nm3 for the same volume flow rate. Typically, for an equivalent heat input, a plasma torch will introduce only one eighth the gas flow added by a natural gas burner. This increased efficiency reduces the volume of the SCC as well as any downstream gas scrubbing equipment and auxiliary systems.

PACT-1.5 OFF-GAS MEASUREMENTS

To gain a better understanding of the process conditions that the SCC must contend with, we measured and analyzed the gaseous output from a representative PACT unit. Measurements were made for selected feed materials under various operating conditions. LLNL developed a diagnostic module that could be inserted into Retech's existing PACT-1.5 experimental testbed. The PACT-1.5 is a small (1.5-foot diameter) unit suitable for batch processing. The gas sampling section included thermocouples to measure gas and wall temperatures, pitot tubes to measure flow rates, and sampling lines to extract hot gases for on-line or subsequent off-line chemical and particulate analysis.

The basic feed material for all of these experiments was soil from Mendocino County, California. The composition of the soil is approximately 64% SiO_2, 15% Al_2O_3, 8% Fe_2O_3, 4% MgO, and 3% CaO, along with minor amounts of other trace elements. The total organic content in the soil was determined to be approximately one percent. The water content is estimated to be between five and ten percent. Diesel fuel was used as a representative organic contaminant material. Polyethylene (PE) and polyvinyl chloride (PVC) were added in modest quantities to the feed soil for some tests. Nitrogen was used as the primary torch gas, but helium and air were also

tested.
Figure 2 represents typical data from one series of runs. It shows the gas temperature (Tg2) rising rapidly after starting the plasma torch and increasing as the system begins to warm up. The gas temperature drops dramatically when the torch is turned off to make adjustments to control settings. The inner wall temperature (Ts4) of the gas sampling section is lower than the gas temperature and indicates the amount of heat loss through the thermal insulation as well as the inefficient convective heat transfer from the gas to the wall.
Fig. 2. Plot of data from PACT-1.5 off-gas measurements.
Figure 2 also shows the amount of gas (in scfm) introduced by the plasma torch, oxygen lance, and the pinhole camera viewport for these runs, as well as the times when gas samples were drawn into cylinders (C-x) or bags (B-x). The comb-like trace indicates when and how material was fed in using an Archimedes-screw feeder. Each spike represents one turn of the screw feeder which introduces approximately 350 g of soil containing 10% diesel fuel. For this series of runs the feed was added incrementally at a rate of one turn every minute. A sharp temperature excursion can be seen each time material is fed in, while there are slower changes in average temperature as equilibrium conditions are reached for different processing conditions.
With these experiments we were able to quantify the amount and chemical composition of gaseous effluent the SCC has to handle for typical PACT operating conditions. Table I has summary information for some of the experiments. Each column represents a different set of run conditions. It is worth noting that in four of these runs there was insufficient oxygen introduced for full combustion of all the hydrocarbons. The last column contains data from the run where enough oxygen was added for complete combustion. With an SCC there will be flexibility in adding supplemental oxygen either in the primary or the secondary chamber and adjusting the process heat source accordingly.
TABLE I
Chemical analysis of the gas samples showed good agreement with the expected gas makeup for the process conditions and gas inputs. While time-averaged gas sampling and chemical analysis can provide sufficient information to determine destruction and removal efficiency (DRE) of targeted compounds, time-resolved chemistry measurements are important in understanding the chemical kinetics and ways to optimize the SCC performance.
Calculations of kinetic chemical reactions were performed by LLNL for several gas mixtures resembling the chemical composition of the effluent gas from a PACT system treating contaminated soils. The chemical kinetics code follows a Lagrangian element (bubble of gas) in its path through a chemical reactor. This time-dependent, zero-dimensional code can calculate the chemical reactions and chemical composition for any number of chemical species. The objective of these calculations is to find the optimum conditions of temperature and residence time for the destruction and oxidation of any remaining hydrocarbons and carbon monoxide in the PACT effluent gas.
Calculations show that for a temperature of 1000 C, and an abundance of oxygen 30 % greater than the stoichiometric amount, all the hydrocarbons exiting the PACT-1.5 are oxidized below the levels currently measurable. Carbon monoxide (CO) is the most difficult compound to oxidize, especially to levels below 10 ppm. Figure 3 shows a series of curves indicating the CO abundance verses time for different gas temperatures. These calculations were run for 20 seconds. In this particular calculation the temperature was kept constant as a function of time. While higher temperatures (1200C) reduce the CO abundance quite rapidly, the final asymptotic value is relatively high. With lower temperatures (700C) the reaction takes longer to complete, but reaches a much lower final level. These results imply that a preferred temperature profile would heat the effluent gases very rapidly to 1200C and then allow the gases to cool down to 800C in two seconds while passing through the SCC. This scenario is certainly possible with a plasma arc gas heater in the secondary combustion chamber. Experiments are needed to to confirm these predictions.
Fig. 3. Calculations of kinetic chemical reactions in SCC.
SCC DESIGN CONSIDERATIONS
A prototype secondary combustion chamber was designed and constructed, and has been operational since July of 1994. Whereas the PACT unit typically employs a

transferred-arc plasma torch, the SCC uses a non-transferred-arc plasma torch. Schematic diagrams of both types of plasma torches are shown in Fig. 4. These swirl-flow, hollow-electrode plasma torches can operate with a wide variety of gases at moderate flow rates.

Fig. 4. Plasma torch types.

The basic chamber design for a plasma heated secondary chamber is a refractory lined cylinder incorporating an inner baffle to divert the gas flow. A schematic of the chamber is shown in Fig. 5. The chamber is water cooled utilizing the double jacketed steel construction. This insures all seal surfaces are maintained at or near room temperature. This type of design minimizes the possibility of leaking unprocessed off-gas from the secondary chamber, and minimizes the possibility of air leakage into the chamber. The plasma torch is also water cooled and is further protected by a surrounding layer of refractory material.

Fig. 5. Plasma-arc heated secondary combustion chamber.

Maintaining the exhaust gas stream at 1000C for two seconds requires a well insulated vessel. High temperature refractories, having melt temperatures ranging from 1500C to 2200C, are used for insulation. The insulation material and thickness was chosen to limit the heat losses but keep the inside wall temperatures below the melt rating. A temperature gradient exists along the axial direction of the inner walls because the torch preferentially heats the region nearest it.

The insulation in the SCC stores a significant amount thermal energy. A typical warm-up time was estimated to be approximately five hours based on an instantaneous temperature rise at the inside refractory wall to 1500C . Once steady state operation is reached in the SCC, the plasma torch only needs to provide enough power to heat the effluent gases from the PACT along with any supplemental oxygen, and to compensate for any heat losses. The primary heat loss is by conduction heat transfer through the insulated walls of the chamber and the jacket surrounding the water cooled torch. The exhaust gas exiting the SCC also represents a heat loss to the system. Estimates of these heat losses are given in Table II, along with the required torch power, assuming an 80% torch efficiency.

TABLE II

PRELIMINARY RESULTS WITH PROTOTYPE SCC

A series of test runs were conducted during July of 1994 in which 2500 grams of polyvinyl chloride (PVC) was fed into the PACT-1.5. Under ideal conditions, PVC (C_2H_2HCl polymer) will convert completely to carbon dioxide, water vapor, and hydrogen chloride. The expected concentrations of O_2, CO_2, H_2O and HCl were calculated based on the PVC feed rate and rates of gas flow into the PACT and SCC, assuming complete burning. Continuous emissions monitoring during these experiments recorded the quantities of these same compounds in the effluent stream before the gas scrubber.

Table III compares the calculated estimates and the measured quantities of gas concentrations when PVC is being fed into the PACT unit. The measurements during run times are in agreement with the predicted values to within a few percent. This shows that a simple analysis can be used to estimate the amount of oxygen to feed into the PACT unit for a particular waste stream. To handle waste streams with widely varying amounts of organic material, the SCC should have on-line monitoring of oxygen levels and proportional control of oxygen feed to insure conditions for complete combustion.

TABLE IIIa

TABLE IIIb

TABLE IIIc

During this same series of test runs, calorimetry measurements were taken on all the SCC cooling circuits. The readings from the SCC chamber circuit show that chamber heat losses had not reached steady state levels by the end of the run. This is as expected since the operating time for the SCC during any one of the tests was less than 80 minutes. The greatest heat loss from the SCC chamber circuit was 8 kW.

The average torch power into the SCC during this series of tests runs was 130 kW. An average of 90 kW was removed from the SCC by the torch cooling system. During these tests, the torch was fully extended into the hot center of the chamber and was not fully covered with insulating refractory material, therefore becoming a sizable heat sink. Approximately 40 kW was available to heat the off gas and chamber walls. Previous analysis had predicted that 16 to 18 kW of torch power would be needed for steady state operation. Measurements made with thermocouples placed at selected

points within the SCC showed that temperatures never exceeded 1500 C; however, it was apparent that substantially more power was being applied than was needed for steady state operation. Further testing must be done to gain operational experience and provide a thorough evaluation of the SCC performance.

SUMMARY

A plasma-arc heated secondary combustion chamber is being developed under a CRADA between Retech, Inc. and LLNL. The unit is being designed to handle the effluent gas from Retech's PACT system. Experiments have been conducted to characterize the off-gas flows from a PACT system processing contaminated soils. Design analysis shows the unit will meet the requirements of heating the effluent gas and maintaining its temperature above 1000C for two seconds. Calculations of kinetic chemical reactions are being performed to investigate optimal processing conditions. A prototype unit has been in operation since July 1994, and work is proceeding to demonstrate and validate its performance in actual processing runs as part of a PACT-2 system.

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THERMOSET MACROENCAPSULATION OF SURROGATE BERYLLIUM FINES WASTES

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ABSTRACT

Several drums of debris wastes contaminated with beryllium fines are included in the Rocky Flats mixed waste inventory. These drums contain a variety of debris wastes, including HEPA filters, a vacuum cleaner, and soft wastes such as Kimwipes and coveralls. Recovery of the beryllium fines is not feasible because the waste is co-contaminated with uranium. Moreover, the debris wastes contain very small amounts of beryllium fines and any attempt to recover these fines is likely to generate additional debris wastes contaminated with fines. As an alternative to recovery, Rocky Flats is investigating solidification of entire drums of this debris waste using an epoxy resin system.

The encapsulation system tested is a mixture of epoxy resin, curing agent, and flyash filler that has been successfully demonstrated on lead wastes at Rocky Flats. This system was selected because of its low viscosity, low peak exotherm, and relative safety (the resin does not contain RCRA hazardous components and does not have the regulatory characteristic of flammability). Other factors influencing this selection decision included toughness, durability, impact resistance, and relative ease of chemical processing.

The epoxy resin encapsulation is a viable treatment option for the beryllium contaminated debris wastes and other debris wastes in storage at Rocky Flats and elsewhere in the DOE complex. This option has several advantages over recovery of the beryllium fines, including reduced cost, reduced worker exposure, reduced complexity, and compressed treatment schedules relative to other options.

INTRODUCTION

Fifteen drums of debris wastes contaminated with beryllium fines are included in the Rocky Flats stored waste inventory. The waste was generated from production machining operations and from a research application that used a high temperature, high pressure process to form beryllium powder into a beryllium casting (1). The spilled or unused beryllium powder from this forming process is a discarded commercial chemical product and thus an EPA listed P-series hazardous waste. The beryllium fines waste stream is classified as a low level mixed waste because of uranium contamination in addition to the beryllium fines.

An examination of real-time radiography (RTR) tapes showed that the drums contain a

variety of debris wastes, including HEPA filters, a vacuum cleaner, and soft wastes such as kimwipes and coveralls. The waste is packaged in 55-gallon carbon steel drums with two polyethylene bag liners.

The EPA hazardous waste number is P015, beryllium, for these wastes. In the June 1990 Rule on Third Third scheduled wastes, the EPA promulgated "Recovery" as the "Method of Treatment" for all forms of P015 wastes (2). In the Rule, the EPA stated that "recovery is a viable and preferred treatment method in light of the high economic value of the recovered beryllium." The EPA also noted that Congress expressed a strong preference in the land disposal ban legislative history for recovery as opposed to treatment followed by disposal. The Rocky Flats beryllium waste, however, also contains uranium contamination, which greatly reduces the economic viability of recovering the beryllium. Moreover, the debris wastes contain very small amounts of beryllium fines and any attempt to recover these fines is likely to generate additional debris wastes contaminated with fines.

The beryllium fines waste meets the definition of hazardous debris as specified in the August 18, 1992 Debris Rule (3). One of the immobilization technologies that can be used as an alternative treatment standard for hazardous debris is macroencapsulation. This requires application of surface coating materials such as polymeric organics (e.g., resins and plastics) or use of a jacket of inert inorganic materials to substantially reduce surface exposure to potential leaching media. The EPA recently provided clarification by stating that using a sealed, preformed container into which the waste has been placed does not meet the definition of macroencapsulation because it would contain void spaces between the debris and container (4).

As an alternative to recovery, Rocky Flats is investigating macroencapsulation of beryllium fines debris waste using an epoxy resin system. This would remove the beryllium fines characteristic and would render the waste LDR compliant.

EXPERIMENTAL RESULTS

The encapsulation system selected for these tests was an epoxy resin/amidoamine curing agent/fly ash filler that has been successfully demonstrated on lead wastes (5). This system was selected because of its low viscosity, low peak exotherm, and relative safety (the resin does not contain RCRA hazardous components and does not have the regulatory characteristic of flammability). Other factors influencing this selection included toughness, durability, impact resistance, and relative ease of chemical processing.

Three surrogate wastes were evaluated using the epoxy thermoset system. Small scale (5-gallon) testing was performed on two surrogate wastes, a small tube-shaped vacuum cleaner with a hose and two 8" X 8" HEPA filters. After the initial success of the two small scale tests, a large scale (55 gallon) test was performed on nine 8" x 8" HEPA filters.

Small Scale Test Results

The vacuum cleaner and HEPA filters were separately bagged in polyethylene bags and taped closed. The bagged vacuum cleaner and HEPA filters were placed in separate 5-gallon metal drums with 0.100 inch thick high density polyethylene (HDPE) liners. Two type K thermocouples were placed in fluoroethylenepropylene (FEP) type sleeves with a sealed end and one thermocouple was placed inside each drum next to the surrogate waste. This allowed for temperature monitoring and thermocouple retrieval after the final cure. The thermocouple output was monitored using a strip chart recorder to identify the initial onset of the gel state as well as the peak exotherm temperature.

The epoxy resin and curing agent were mixed in a ratio of 70:30 (by weight) using a powered hand mixer. After the resin/curing agent were adequately mixed (approximately two minutes), 40% by weight fly ash filler was added to the mixture and power mixed. The epoxy charge was then poured into the drum containing the vacuum cleaner. Due to trapped air in the polyethylene bag, the vacuum cleaner and hose floated to the top of the drum. The bags were pierced, the air was vented, and the items were manually pressed to the bottom of the drum. The process was repeated for the HEPA filters; however, a metal screen was placed over the top of the filters to prevent them from floating to the top.

The curing of the castings was monitored with the thermocouples and by visual and tactile observation. The gel state commenced approximately 90 minutes after the initial mixing based on tactile and visual observations. A peak exotherm of 137C was recorded after 2.5 hours in the drum containing the vacuum cleaner. The drum

containing the HEPA filters recorded a peak exotherm of 104C approximately 2.5 hours after the initial mixing. The higher peak exotherm in the drum containing the vacuum cleaner is probably caused by the higher void volume in the drum.
After curing overnight, the encapsulated wastes were removed from their metal drums and visually examined. The waste form containing the vacuum cleaner was a hard, dark grey monolith with some cured foam on the top surface, probably resulting from the venting of excess air in the polyethylene bag. On one side of the waste form, a large crack formed where the vacuum cleaner had contacted the drum liner.
The top of the waste form containing the HEPA filters was relatively smooth, probably due to the fact the bagged HEPA filters were held in place by a screen, thus eliminating the need to vent the bag. On one side of the monolith, a very small portion of the wooden frame of one of the HEPA filters showed through the epoxy. Otherwise, no cracks or apparent weaknesses in the monolith were noted.
Several possible explanations for the crack in the waste form containing the vacuum cleaner were considered. One hypothesis was that the crack was caused by excess heat build-up due to the metal vacuum cleaner touching the drum liner. The epoxy surrounding the contact point may have cooled quicker than the epoxy at the contact point. This may have created a stress point, resulting in the crack.
An alternate explanation was that the trapped air inside the polyethylene bags expanded as the epoxy cured and contracted as the epoxy cooled. This appears less likely because the waste form containing the bagged HEPA filters did not crack. Also, the air was released from the bags soon after the epoxy charge was placed in the drum in order to keep the waste from floating.
The most probable explanation is that the crack was caused by interferences during the curing process. As the vacuum cleaner began to float to the top during the gel stage, it was manually pushed back into the container. This probably disrupted the waste matrix, resulting in the observed crack.
In order to identify the source of the crack and, therefore, eliminate it, the experiment with the vacuum cleaner was repeated with a few modifications. The vacuum cleaner was simulated using a metal cylindrical filter cartridge housing. The simulated vacuum cleaner was bagged and placed in a 5-gallon drum as before. A metal screen was placed inside the drum to keep the vacuum cleaner from floating above the epoxy. The epoxy, curing agent, and fly ash filler were mixed and poured as in the previous experiments. A peak exotherm of 149C was observed approximately three hours after the epoxy was poured.
After curing overnight, the waste form was removed from the drum and visually examined. It was light grey, very smooth on top, with no cracks or other imperfections detected on the outside.
These results support the hypothesis that the crack was caused by manual interference during the gel stage in an attempt to keep the bagged waste from floating to the top of the epoxy. This problem was effectively eliminated by the placement of the metal screen on top of the bagged waste. The theory of the crack being caused by excess heat build-up was eliminated because the second waste form containing the vacuum cleaner had a similar packing configuration, a slightly higher peak exotherm, and no crack.
Large Scale Test Results
The large scale 55-gallon drum test was conducted on nine 8" x 8" HEPA filters which were placed in three columns of three filters each, double bagged, and put into a 55-gallon standard drum containing a standard rigid drum liner (nominally 0.1 inch thick, carbon black-filled HDPE). Type K thermocouples were located in various areas of the waste, between the rigid drum liner and the inside drum wall, and on the outside surface of the drum. A 1-inch metal standoff allowed resin to flow between the bottom of the bagged HEPA filters and the inside bottom surface of the rigid liner. A metal cross was placed on top of the surrogate waste, about two inches below the top of the rigid drum liner to keep the bagged HEPA filters from floating. Approximately 25 gallons of the epoxy encapsulant was used to fill the void volume in the drum. A peak exotherm of 152C occurred approximately 5.5 hours after mixing. During the mixing of the epoxy resin, curing agent and flyash filler, a more rigorous, high shear mixer was used which provided improved mixing but also resulted in more entrained air in the mixture. As the mixture approached gelation, the additional entrained air, coupled with the relatively high surface tension of the epoxy system, resulted in excessive foam formation on the top surface of the drum. After complete cure the encapsulated monolith was removed from the drum and several

inches of closed-cell foam structure material were removed from the top surface. A "cap" charge of the epoxy system was then added to the monolith to fill in void space left by removal of the foam material.
After curing, the rigid liner was cut away and the exterior surface of the monolith was examined for excessive cracks and areas where low compressive strength foam might exist. Only one small surface crack was observed. Even though the closed-cell structure of the cured material appeared to exist several inches below the level of the epoxy "cap", the apparent compressive strength of this closed-cell material relative to the remaining areas of epoxy was adequate. In future large scale tests, a different mixing system will be evaluated to avoid excessive foam formation.
After examining the exterior surface, a small section of the monolith was cut away and removed. The section revealed a portion of one HEPA filter. The degree of encapsulation of the HEPA filter was qualitatively judged to be more than adequate to meet regulatory guidelines.
Future large scale surrogate tests will evaluate the following:

a new curing agent which has over three times the pot life of the present curing agent resulting in lower peak exotherms,

lower shear mixing system to minimize foam formation, and

an improved mechanical system to position and hold down the buoyant HEPA filters.

These evaluations will precede tests performed on low level radioactive waste drums containing debris wastes contaminated with beryllium fines.

NON-DESTRUCTIVE TESTING OF WASTE FORMS

In previous investigations, epoxy/fly ash castings of lead wastes were destructively examined by sectioning in order to determine the penetration of the epoxy into the waste material and to look for internal cracking and large voids. Although enumerated standards have not been established for macroencapsulated waste forms, researchers at Rocky Flats are pursuing methods for nondestructive testing in order to facilitate certification of the final waste forms. Of particular interest would be any crack pathways penetrating into the casting that could lead to possible escape routes for the beryllium particles. The current nondestructive method for inspection of packaged waste uses real time radiography (RTR) equipment. Drums, crates, and other containers are examined by X-ray exposure in real time. A monitor allows for direct viewing and a videotape is used to store the inspection record for future retrieval.
The small scale HEPA filter casting and the original vacuum cleaner casting with the surface crack were selected for RTR evaluation. Two very different results were obtained on these castings. The HEPA filter casting was viewed first and, under X-ray exposure, the HEPA filters were visible inside the casting. The individual metal screws used in the outer wood frame of the HEPA filter were also visible, as were the aluminum separators and filter media. No large cracks or voids were detected in the casting.
Under RTR examination, no internal features could be visually resolved in the cracked vacuum cleaner casting. The reason for this lack of X-ray resolution was the additional amount of epoxy/fly ash encapsulant present in this casting relative to the HEPA filter casting. The vacuum cleaner casting had a larger void volume than the HEPA filter casting since the vacuum cleaner occupied less volume than the HEPA filters. Therefore, there was more fly ash for the X-rays to penetrate in the vacuum cleaner casting compared to the HEPA filter casting. The fly ash composition includes some higher atomic number (Z-number) elements that act as attenuators to the transmission of X-rays through the casting. This results in poor resolution and a lack of discernible features in the X-ray image.
In spite of the mixed results with the RTR examinations, this technique is still a viable option for nondestructive testing of the thermoset castings. In the near future, RTR examination of the large scale HEPA filter casting will be performed. Depending on the outcome of these tests, other nondestructive testing techniques, such as digital radiography and computed tomography, may be evaluated.

CONCLUSIONS/RECOMMENDATIONS

The epoxy resin/curing agent/fly ash filler encapsulation system is a viable option for treating the fifteen drums of beryllium contaminated debris wastes in storage at Rocky Flats. This option has several advantages over recovery of the beryllium fines, including reduced cost, reduced worker exposure, reduced complexity, and compressed treatment schedules when compared to other options. Moreover, recovery of the beryllium fines is not feasible due to the paucity of fines in the drums and the

likelihood of generating additional beryllium contaminated debris during the recovery process.

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COMBINED PHOTOOXIDATION/PHOTOREDUCTION USING TIO2 PHOTOCATALYSTS TO TREAT ORGANIC/INORGANIC METAL-LADEN WASTEWATERS

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ABSTRACT

A technology has been developed that employs new photocatalysts that have the capacity of both sequestering and converting heavy metal ions from aqueous solution to their less toxic, readily recoverable metallic forms. The technology exploits a strategy to enhance the reactivity of semiconductor photocatalysts for selected toxic compounds by derivatization of the photocatalyst surface with metal-binding ligands. Titanium dioxide (TiO2) colloids prepared from titanium chloride (TiCl4) were used as photocatalysts in the experiments. Cysteine was selected as the model derivative for the surface modification of the TiO2 nanoparticles. Testing of TiO2 photocatalysts conducted in the absence of organic compounds demonstrated that adsorption of lead ions (Pb2+) occurred in the system with either untreated (virgin) or treated (modified using cysteine) TiO2 photocatalysts. Adsorption rates of the metal ions in the systems that used treated TiO2 photocatalysts were about three times faster than those where untreated TiO2 photocatalysts were used. Lead ion concentrations in the irradiated solutions decreased as the UV irradiation time increased; greater metal removals were achieved at longer irradiation times. The rate of decreasing lead ion concentrations in the system that used treated TiO2 photocatalysts was about two to three times faster than that in the system using untreated TiO2 photocatalysts.
Experiments were also performed in which organic compounds (naphthalene or phenol) and heavy metals (Pb2+) were simultaneously treated with TiO2 photocatalysts. The presence of lead ions did not interfere with the photo-degradation of the selected organic compounds from solution. The presence of phenol compound appeared not to affect the photoreduction of lead ions, while addition of naphthalene compound resulted in a "lag-time" effect on the photoreduction of lead ions from solution. All of the experimental results showed that the cysteine-modified TiO2 resulted in faster and more effective removal than that for the untreated TiO2 system. This technique has resulted in the simultaneous photocatalytic removal/recovery of organic and inorganic compounds in the system.

INTRODUCTION

Toxic organic and inorganic contaminants in waste streams generated by chemical, metallurgical, and other industries must be destroyed or removed before the waste streams are released to the environment. Waste streams containing toxic organics and heavy metals have also been generated by DOE activities. It has been estimated (1-3)

that more than 200,000 m3 of aqueous mixed waste, contaminated with both toxic organics and metals, will have to be treated at DOE sites over the next 20 years. In the last ten years, extensive studies of photocatalytic methods for the removal of organic and inorganic contaminants from aqueous waste streams have been undertaken (4-13). Such semiconductor particles as TiO2 can act as microelectrodes that consist of cathodic and anodic parts. These materials are excellent light-harvesting materials and mediators for photochemical degradation of various environmental contaminants. The principle behind semiconductor-assisted photocatalysis involves the photoexcitation of the semiconductor with light energy greater than the semiconductor band gap, followed by generation of electron-hole pairs. The electrons and holes diffuse to the semiconductor particle surface and can be exploited for the various REDOX processes analogous to those of an electrochemical cell.

Application of semiconductor-assisted photocatalysis is primarily focused on TiO2, because it is cost-effective, non-toxic, insoluble under most conditions, and photostable. It is effective for oxidative destruction of the most resistant organic compounds, such as polychlorinated aromatics (14) or polyaromatic hydrocarbons (such as those prevalent in oil spills) (15), and for reduction of several heavy metals (Hg2+, Pt4+, Au4+ , Rh4+). The principal advantage of these systems is the potential for accumulation of charges on the particle surface, resulting in simultaneous injection of multiple charges if a suitable scavenger of the opposite charge is present on the surface. However, the reduction of toxic heavy metal ions such as Pb2+ or Cd2+ with negative REDOX potentials of the first one-electron transfer process was not reported in aqueous suspensions of TiO2.

Because the charge carriers created on an illuminated semiconductor are trapped on the surface, and the REDOX reactions occur at the solid/solution interface, the surface of the particles plays an important role in electron/hole transfer reactions and can be engineered to improve both light-harvesting and/or REDOX characteristics of a semiconductor.

It has been recently shown (16) that the modification of the particle surface of colloidal TiO2 enhances the reduction properties of conduction band electrons and enables effective removal of lead and mercury ions from aqueous solutions. These surface-modified TiO2 colloids form the basis of an engineering study whose objective is to develop an industrial process for selective heavy metal recovery and for simultaneous removal of organic and metal contaminants from aqueous waste streams.

This paper briefly describes the mechanism of lead reduction initiated by colloidal TiO2 photocatalyst, in which the particle surface has been modified by cysteine and presents the results of bench-scale engineering studies in which surface-modified TiO2 colloids have been used as photocatalysts for the combined removal of organic compounds and lead from aqueous solutions.

UNDERLYING CHEMISTRY

Unmodified and Modified TiO2 Catalysts

Electron-hole pairs generated during the illumination of TiO2 particles (Eq. (1)) can be exploited for various REDOX processes. Conduction-band electrons that have the potential of -0.3 V (vs. NHE) at pH 317 are trapped on the particle surface in about 30 picoseconds, and valence-band holes with potential of +2.9 V at pH 3 are trapped in the hundred-nanosecond range (18). As the energy levels of the surface traps lie within the band gap, the actual reduction and oxidation properties of photogenerated charges are reduced and are dependent on the REDOX potential of surface trapping sites. EPR investigations of surface trapping sites in colloidal TiO2 suggests that electrons are trapped at metal centers (Ti3+), while the holes are trapped on the surface OH groups as a (TiO2)nTi(IV)O radical intermediate (19,20).

Eq. (1)

Eq. (2)

where (TiO2)n represents bulk material.

We have found that in the presence of the tridentate ligand cysteine, the characteristic oxygen radical signal in the EPR spectrum disappears, and an axially symmetrical, lattice-defect-type EPR signal with g=2.004 is observed (21). These results suggest that cysteine is strongly bound to a colloid surface and passivates the surface states, which act as hole traps. Infrared spectroscopy results indicate that cysteine is bound to a surface Ti atom through the carboxyl group, with

formation of the corresponding salt. As oxygen is covalently linked to the lattice, it can easily accept the photogenerated hole and transfer it to a cysteine molecule. Indeed, we found that at 200K, the electron transfer process occurs, the holes are transferred from the TiO2 particle to a cysteine molecule, and a sulfur-centered radical is formed:

h+ + (TiO2)nTi(IV)-O2C-CH(NH3+)CH2SH

(TiO2)nTi(IV)-O2C-CH(NH3+)CH2S +H+ Eq. (3)

The EPR signal of the TiO2 colloid itself at 200K was not observed (21). The REDOX potential for the oxidation of cysteine into a sulfur radical intermediate requires the potential of the hole to be greater than +0.92 V vs. NHE (22). The sulfur-centered cysteine radicals most probably undergo dimerization with free cysteine molecules and form the cystine radical (RSSR-; E(RSSR-/2RS-) = +0.65 V vs. NHE) (23) at room temperature. Illumination of a cysteine-modified TiO2 colloid thus results in oxidation of cysteine molecules into cystine and accumulation of trapped electrons.

Photodeposition of Lead

Upon addition of lead ions into solutions with surface-modified TiO2 colloid, a new complex of cysteine with heavy metal was observed with IR spectroscopy. Lead ions bind with the cysteine molecule in a strong 1:1 complex (pK=11.6), in which cysteine acts as a tridentate ligand (24). IR spectroscopy of the cysteine-modified TiO2 colloid containing lead ions also suggests that lead is chelated with cysteine ligand, with only the carboxyl and mercapto groups. The carboxyl group bridges Ti and Pb ions (Ti-O-C-O-Pb), and in this way Pb becomes covalently linked to the particle surface as a continuation of the particle lattice. Illumination of this system did not result in the formation of a sulfur-centered radical at 200K, probably because the lone pair of electrons from sulfur that participate in oxidation are bound to the lead ion. However, upon increasing the temperature to room temperature for several minutes, all photogenerated electrons were scavenged by metal ions,

Eq. (4)

and the metal lead was observed to precipitate.

As the reduction of Pb2+ ions in TiO2 aqueous suspensions was not observed previously (25,26), our results suggest that surface modification of TiO2 particles with cysteine molecules is responsible for lead reduction. There are four ways that cysteine can modify the system properties that would lead to the reduction of Pb2+ ions: 1) electron accumulation, 2) strong adsorption of lead ions, 3) modification of lead ion reduction potentials after complexation, and 4) the modification of the REDOX properties of TiO2 itself. Our results indicate that all four factors are responsible for enhanced reduction properties of the TiO2 colloid.

We have found that the yield of trapped electrons is further enhanced after addition of methanol, which is an effective hole scavenger (E(CH3OH/CH2OH)= +1.2 V) (27). It has been shown previously that electrochemical oxidation of methanol results in formation of electron donating species (E(CH2OH/CH2O) = -0.95 V) (23). The large negative potential of the methanol radical induces electron injection into colloidal TiO2 with formation of surface trapped electrons:

(TiO2)n Ti(IV)-O...HOCH2 (TiO2)n Ti(III)-O + OCH2 + H+ Eq. (5)

The net effect is that from one photon, two electrons are formed, and consequently the yield of electrons is doubled. Trapped electrons in TiO2 have enough potential to reduce lead ions in two electron reduction processes. It should be noted that direct reduction of Pb2+ ions was not observed in homogeneous solutions due to the negative potential of one-electron reduction of lead ions. Direct reduction of Pb2+ by CH2OH on colloidal TiO2 was not observed.

EXPERIMENTAL SECTION

Lead acetate (Pb[C2H3O2] 2) were selected as the representative inorganic contaminant to be tested while naphthalene and phenol were chosen as the target organic compounds to be treated. All of the chemicals used in the studies were reagent grade (obtained from Fisher Scientific) and were used as received. The sample solutions were prepared fresh before conducting the experiments using laboratory deionized water. All of the experiments were conducted with a quartz reactor having a total capacity of 250 mL.

The photochemical reactor (Rayonet Model RMR-600, Branford, Conn.) used in the studies had a reactor chamber 10" in diameter and 10 1/2" high, equipped with eight UV light sources surrounding the chamber. The eight UV light sources provided a

maximum of 32 watts (four watts for each UV lamp) of UV energy at a wavelength of 254 nm to the reactor. During a typical experiment, a 50-mL sample solution containing a known amount of contaminant(s) and photocatalyst were placed in the center of the reactor chamber and irradiated for a desired period of time with the UV lights. The irradiated solutions were well stirred throughout the experiment. After irradiation, the photocatalyst and elemental metals were separated from the solution by filtration. The organic compound was analyzed by using a gas chromatograph (HP 5890 series II, Hewlett Packard, Wilmington, DE) equipped with an FID detector; the lead concentration in the solution was analyzed by using an atomic absorption (AA) spectrophotometer (model 200 A, Buck Scientific, East Norwalk, Conn.). The sample solutions were preserved in a 1% nitric acid solution (pH<2) before being analyzed.

Adsorption studies for the target contaminants were carried out in the laboratory; a shaker table (Eberbach Co., Ann Arbor, Mich.) was used to provide sufficient mixing of the sample solutions. A 30-mL vial sealed with a Teflon cap was used for each sample solution. During the experiments, the sample solutions and the vials were covered with aluminum foil to avoid any exposure to light.

RESULTS AND DISCUSSION

Adsorption Rate and Adsorption Capacity of TiO2 Photocatalyst

The adsorption rate and adsorption capacity of lead, as well as the target organic compounds, over either treated or untreated TiO2 photocatalysts were investigated before any UV irradiation had been performed. Adsorption is the first reaction that occurs when the photocatalyst is contacted with contaminants. This process is reversible, does not require light, and is very sensitive to the pH. The pH of the irradiated solutions was adjusted to about four. Changing the pH value of the solution changes the surface charge of TiO2, as well as the extent of hydrolysis and species distribution of the metal ions, and as a result the adsorption capacity of the photocatalysts to metal ions changes. It has been reported that there was no adsorption of Pb2+ ions on TiO2 photocatalyst semiconductor particles in the dark and at pH 1.4. If another chemical compound, such as naphthalene or phenol, is present in the solution, the lead ions will compete with the added chemical compound for the adsorption sites on the surface of the photocatalyst, which may result in a lower adsorption capacity for the target compound.

Adsorption equilibrium between the target compounds and the photocatalysts was assumed to be reached after 72 h of contact, although it was reported that adsorption equilibrium between pyridine and TiO2 photocatalyst could be reached within one hour of contact time (28). The adsorption capacities of the two photocatalysts appear to be linearly proportional to the lead concentrations in the liquid phase. The adsorption capacity of the treated TiO2 photocatalyst is appreciably higher than that of the untreated photocatalyst. Adsorption data on lead ions also reveal that the adsorption rate of the treated photocatalyst is much faster than that of the untreated photocatalyst. The coated cysteine compound behaves as a strong active site for capturing the dissolved metal ions in solution. The adsorption rate and adsorption capacity may have significant effects on the reaction rates of the photo-induced REDOX reactions. During a typical UV irradiation with a photocatalyst in solution, the electrons (e-) and holes (h+) formed on the surface of the photocatalyst would preferentially recombine on the surface sites in the absence of surface-adsorbed lead ions or other electron acceptors. If this happens, the lead ions have to diffuse from the bulk solution to the interface where the REDOX reaction occurs. The diffusion rate, in this case, may become a significant factor in the REDOX efficiency.

In the experiments, the initial lead concentration in solution was about 91 mg/L. After mixing with the photocatalysts, the concentration decreased to about 60 mg/L and 15 mg/L within 60 min of adsorption for untreated TiO2 and cysteine-modified TiO2, respectively. Adsorption rates of the lead ions in the system using treated TiO2 photocatalyst were about three times faster than those in that using untreated TiO2 photocatalyst. The adsorption rates of TiO2 photocatalysts (either treated or untreated) remained constant within this lead concentration range.

Photocatalytic REDOX Reaction

UV-irradiation of organic and inorganic compounds in the aqueous solution results in the oxidation/degradation of organic compounds and reduction of inorganic compounds. The disappearance of the introduced lead ion and naphthalene in the irradiated aqueous solution is represented in Figs. 1 and 2, respectively. The degradation of

phenol compound with or without the presence of lead ions is not shown here but is presented in a separate poster session. However, the phenol results are discussed below.
Fig. 1. Effect of Irradiation Time - Lead Only.
Fig. 2. Effect of Irradiation Time - Naphthalene Only.
The removal of lead ions in solution was verified from the residual lead analysis (using AA) of the irradiated solution, along with the observation of black particulates (believed to be metallic lead) generated in the solution. From an the initial concentration of about 120 mg/L, lead ion concentration slowly dropped down to about 30 mg/L with 60 min of irradiation, using the untreated TiO2 photocatalyst. In other experiments, the use of treated TiO2 photocatalyst significantly enhanced the concentration-decreasing rate, down to about 5 mg/L with 25 min of irradiation; more than 95% of dissolved lead ions were removed from the irradiated solution. The removal rate of lead ions in the system using treated TiO2 photocatalysts was about two to three times faster than that using untreated TiO2 photocatalysts.
During the UV irradiation, the lead ions were recognized to be adsorbed onto the surface of TiO2 photocatalyst and then reduced to elemental lead. However, it is uncertain how many lead ions were removed from the solution through photoreduction to their metallic form and how many were removed only by adsorption on the TiO2 photocatalysts.
Figure 2 shows that naphthalene concentration decreased with increasing irradiation time. Concentrations of less than 2 mg/L of naphthalene (initial naphthalene concentration was about 11 mg/L) were obtained within 30 min of irradiation in both cases. The difference in the degradation rate between the use of treated and untreated TiO2 photocatalysts was statistically insignificant. The introduced naphthalene may be degraded through direct adsorption of UV energy or through oxidization by the radicals generated by the excited TiO2 photocatalysts or by the photocatalysts themselves.
UV irradiation of phenol compound in aqueous solution using the two photocatalysts exhibited similar results. Initial phenol concentrations of about 100 mg/L decreased to less than 20 mg/L within 30 min of irradiation and leveled out over irradiation times of 30 to 60 min. Approximately 80% of the phenol was removed within the first 30 min of irradiation. The phenol degradation rate using treated TiO2 photocatalyst was somewhat faster than that with untreated photocatalyst. With lead ions present in the irradiated solution, the phenol degradation rate decreased slightly with the use of untreated photocatalyst in the solution, but the change was insignificant. It appeared that the phenol degradation rates were unchanged when using treated TiO2 as the photocatalyst in the experiments.
Simultaneous Removal of Lead and Organic Compounds
Experiments were also carried out in which lead ions and organic compounds (such as naphthalene and phenol) were simultaneously treated with the UV-TiO2 photocatalyst. When organic compounds, such as naphthalene and phenol are present, they may compete with lead ions for the adsorption sites on the surface of TiO2 photocatalyst during UV irradiation, resulting in a decrease in the lead removal rate, or serve as additional electron donors and enhance the photoreduction rate of lead ions. Under our experimental conditions, the removal rate of lead ion in the irradiated solution seems to have a minor effect on the existing phenol compounds. The decay in lead concentration remained unchanged in this case. However, noticeable differences were obtained when naphthalene was added to the solution. As shown in Fig. 3, the presence of naphthalene resulted in a "lag-time" effect on the photoreduction of lead from the irradiated solution. The removal rate of lead ions decreased significantly within the first 30 min of UV irradiation. In both cases (using untreated and treated TiO2 photocatalysts), the use of cysteine-modified TiO2 photocatalysts resulted in faster and more effective removal/recovery than was obtained for the untreated TiO2 system.
Fig. 3. Combined Naphthalene - Lead System - Lead Concentration.
With inorganic and organic compounds in solution in the same system, the presence of lead ions would provide additional electron-acceptors, resulting in an increased removal/degradation rate for the organic compounds. However, probably due to the low organic concentrations in the irradiated solution, the experimental results (as shown in Fig. 4) showed that the presence of lead ions had little effect on the removal/degradation of naphthalene from solution. The presence of lead ions also had an insignificant effect on the photooxidation of phenol compound. The degradation

rate remained unchanged with or without the presence of lead ions in the solution.
Fig. 4. Effect of Irradiation Time - Naphthalene only
SUMMARY AND CONCLUSION
The experimental results indicate that simultaneous removal of organic compounds (such as naphthalene and phenol) and inorganic compounds (such as lead ions) in aqueous solution can be achieved by using a UV-irradiated TiO2 photocatalyst system. The removal rates of organic and inorganic compounds can be enhanced through surface modification of the TiO2 photocatalyst using an organic substance such as cysteine. The cysteine-modified TiO2 photocatalyst, enhanced the oxidation rates of organics as well as the reduction rates of heavy metals in the irradiated solution, resulting in improved treatment efficiencies for combined organic/inorganic wastestreams. The photoreduction rate of lead ions were significantly affected by the presence of some organic compounds, such as naphthalene, but had little effect on the presence of other organic compounds such as phenol. The photodegradation rates of organic compounds were not significantly affected by the presence of metal ions.
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A PORTABLE SYSTEM FOR THE TREATMENT OF WATER-REACTIVE MIXED WASTE

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ABSTRACT

Los Alamos National Laboratory and other locations in the complex of experimental and production facilities operated by the United States Department of Energy (DOE) have generated an appreciable quantity of hazardous and radioactive wastes. The Resource Conservation and Recovery Act (RCRA) enacted by the United States Congress in 1976 and subsequently amended in 1984, 1986, and 1988 requires that every hazardous waste must be rendered nonhazardous before disposal. Many of the wastes generated by the DOE complex are both hazardous and radioactive. These wastes, called mixed wastes, require applying appropriate regulations for radioactive waste disposal and the regulations under RCRA. Mixed wastes must be treated to remove the hazardous waste component before they are disposed as radioactive waste.

This paper discusses the development of a treatment process for mixed wastes that exhibit the reactive hazardous characteristic. Specifically, these wastes react readily and violently with water. Wastes such as lithium hydride (LiH), sodium metal, and potassium metal are the primary wastes in this category. Besides their tendency to react with water, the wastes also produce alkaline hydroxides and hydrogen gas as products of the reactions.

$$LiH + H_2O \rightarrow LiOH + H_2 \quad (1)$$

$$2Na + 2H_2O \rightarrow 2NaOH + H_2 \quad (2)$$

$$2K + 2H_2O \rightarrow 2KOH + H_2 \quad (3)$$

If in aqueous form and if the pH exceeds 12.5, the alkaline hydroxides must be further processed to lower the pH to the range of 2-12.5 to remove the corrosive hazardous characteristic. The hydrogen gas formed during treatment is not considered a RCRA hazardous waste, but the hydrogen poses a substantial safety hazard because it can form explosive mixtures with air. Tritium may also be substituted for hydrogen in the LiH. If tritium is present, special processing may be necessary to avoid exhausting tritium into the environment.

Because of the requirement to control environmental exposure to radioactivity contained in the wastes, the process design requires a reaction within enclosed vessels. These vessels require inert gas purging with subsequent off-gas scrubbing and high-efficiency particulate air (HEPA) filtration before discharge to the atmosphere. If tritium is present in the reaction gases, a better system to capture the tritium is necessary. The hydrogen-containing off-gas could be vented, flared, oxidized to water by combustion or electrochemical oxidation, or separated from the nitrogen purge and collected in gas cylinders or thermally regenerable hydride substrates. Any alkaline hydroxide reaction products would be neutralized and further treated for disposal.

PROCESS DEVELOPMENT

Initial process development investigations centered around the idea that a stream of humidified nitrogen could be used. This approach has many apparent advantages including

- the ability to control the reaction rate by limiting the humidity level in the nitrogen purge gas;
- the nitrogen provides a carrier for the water vapor and acts as the purge blanket; and
- the possibility to control the hydrogen content in the off-gas to below explosive

limits by maintaining +96% N2 by hydrogen/nitrogen separation and nitrogen recycle. An experimental program focused on the humidified nitrogen process. In the experiment, nitrogen was passed through a humidifier and then contacted with the water-reactive substance in a reactor. The reaction kinetics were determined for LiH by measuring the gas flows and the amount of hydrogen exiting the reactor. The primary investigation variables were temperature, percent of humidity, and particle size of the water-reactive substance. Specific results of these tests have been previously presented (1). Essentially, the results showed that the reaction follows the Unreacted Shrinking Core Model (2). Acceptable control and reaction rates were achieved with finely divided powders. Rates were prohibitively slow for larger pieces of LiH.

The humidified nitrogen process was abandoned because of these factors.

The formation of an alkaline hydroxide layer on the particle surface slowed the reaction appreciably. Therefore, there was no foolproof way to ensure that the reaction had gone to completion.

Heat removal from the reacting solid is critical because melting can occur at relatively low temperatures. Because melting would quickly plug most reactor designs, operation would be below the maximum saturated temperature. Table I shows that for LiH, operation would be fairly reasonable because saturated nitrogen at 50C would carry enough water to the reaction without consuming too much nitrogen. Sodium and potassium metals would require recycling the reactor off-gas or would consume excessive nitrogen.

The reaction rate for large particles was too slow to be practical. Grinding would be necessary to ensure complete and reasonable reaction rate.

Controlling a large volume of a dusty reactant within a reactor while contacting with a moving gas stream would be a difficult design task. This task is made even more difficult because LiH has a very low density.

An alternate contacting method was developed. This process involves directly immersing the water-reactive waste in a volume of water. Reaction rate is controlled by the rate of addition of the waste to the reactor. In this system, a nominal amount of nitrogen is introduced as purge gas, but there is no attempt to maintain the hydrogen content 4%. The possibility of explosion is avoided by excluding oxygen. All off-gas is scrubbed, filtered, and discharged. During processing, the liquid volume charged gradually accumulates alkaline hydroxide strength. After pH adjustment, the waste water is sent to a wastewater processing plant. Figure 1 shows the basic process block floor diagram.

Fig. 1.

The advantages of this approach are the following:

Either powdered or large solid pieces of water-reactive waste can be fed, with acceptable reaction rates. Simple solids introduction methods can be used.

Extraneous materials can be processed with the water-reactive waste. For instance, if lithium hydride is bonded to another non-water-reactive substance, the whole mass can be introduced to the reaction system to reliably remove the water-reactive waste from the nonreactive substrate.

The volume of water in the reaction system provides an effective heat sink for the exothermic reaction. The reaction system is sized to limit the temperature rise to 20C if the cooling system fails.

Accumulations of solid alkaline hydroxide layers that could stop the reaction are minimized because the excess water dissolves the accumulations. This factor also eliminates the need for separate dissolution and rinse procedures.

The amount of water-reactive waste treated per run is limited only by the liquid volume in the reaction system and the solubility of the alkaline hydroxide or neutralized hydroxide, and not by the volume of the reactor itself.

Experiments were performed to determine reaction characteristics and to validate the process concept. Details about the results of this experimentation are not within the scope of this paper, however, these generalizations are made.

1. Water-reactive solids larger than 1/4 in. tend to complete reaction in a reasonable time and do not react instantaneously. Tests on 1/4-in. pieces of LiH resulted in reaction times between 20 and 40 seconds.
2. Finally divided water-reactive material tends to react very quickly. Tests on LiH powders showed that reaction rates equivalent to 50% reaction within 0.5 seconds

were attainable.

3. Attempts to regulate the reaction rate of powders showed some success:

Raising the solution strength to saturation of LiOH or LiSO4 had little effect on reaction rate for small samples.

Forcing water to access the LiH through orifices controlled the reaction rate. The mechanism for control is the rate of water accessing the LiH, which is controlled by the escaping hydrogen gas. With fine orifices in metal bombs or cloth bags, the reaction rate was relatively steady for several minutes. An attendant problem is plugging the orifices with undissolved LiOH. Tuning the powder containers in the process unit will be necessary. An additional advantage of using a rate-controlling container or bag is that a delivery method controlling the powder rate is possible.

Water-soluble bags were tried as a delivery method and as a rate-controlling method. Although the soluble bags contained powders well, the rate controlling aspect did not work well. Once a spot on the bag dissolved, the whole bag would quickly empty because of hydrogen pressure buildup. Another problem encountered with the bags was they lost their ability to dissolve as the ionic strength increased in the reaction liquor. Suppression began as the concentration approached 20% of saturation of LiOH.

PROCESS DESIGN

A process design was completed in early 1995. The water-reactive waste treatment system will remove the RCRA reactive characteristic by contacting the solid waste with water. The treatment process is a skid-mounted system designed with maximum flexibility to treat the anticipated water-reactive wastes. The treatment skid consists of individual modules. Each module weights a maximum of 4000 pounds and has maximum dimensions of 5 feet wide x 10 feet long x 12 feet high. The structural system is designed so that it can be easily cleaned for removal to storage. This requirement necessitates welded stainless steel construction that greatly minimizes cracks, crevices, and corrosion products such as rust. All reasonable effort has been given to meeting this goal with standard industrial equipment. When stainless construction is not possible, epoxy painting for nonstainless metal parts is specified. Figure 2 shows a simplified process flow diagram of the process.

Fig. 2.

The skid design conforms to the standards of the National Electrical Code for explosion-proof operations in Class 1 Division 2 or Class 1 Division 1 Group B (3). Explosion-proof design uses a combination of explosion-proof enclosures, purging, and intrinsic safety barriers. The instrument and control portion of the skid design allows local and remote control of all critical portions of the process. Further, the skid system is mobile for use outside the Hazardous Waste Treatment Facility (HWTF) at Los Alamos National Laboratory or can interface with the central control system of the HWTF.

The process equipment is designed as several transportable modules that will be connected together to form an integrated skid. This allows moving of the equipment to various locations at the laboratory, including movement of the skid in and out of the HWTF. Figure 3 shows the overall appearance of the skid. The four main process modules are on the lower level. A fifth module, which holds the processing glovebox, is on the second level.

Fig. 3.

Several design features are noteworthy because of the requirements of the process:

Every attempt is made to minimize the complexity of the design.

Multiple layers of operator and process protection are provided. For example, the preliminary safety analysis has determined that no single failure of equipment or procedure can lead to a catastrophic event. Much attention has been paid to ensure that water cannot accidentally contact water-reactive waste during handing operations; operators will not be exposed to water-reactive waste; explosive mixtures cannot form within the reaction or venting vessels; and radioactive materials are not released to the environment.

A computer-based process control system is specified. This control system provides maximum flexibility for necessary future changes, simplicity in operation and design, explosion-proof design, capability for logging data, and capability for remote operation. Control system components are compatible with those used in the previously designed process skids for depleted uranium treatment (4) and chemical plating waste treatment (5).

Every attempt was made to specify 304L, 316L, or higher alloy. This specification

is for corrosion control and easy decontamination. Exterior surfaces not constructed of stainless steel or other material that cannot be decontaminated will be painted with approved epoxy paint.

Because of the explosion-proof requirement and the size and weight limitations, the number of electrical devices is minimized. For example, to avoid electric motors on the main process skids, pumps driven by nitrogen gas will be specified. The processing skids adhere to NEC Class 1 Division 2 design code for explosion protection in the liquid-handling area. Twater tohe gas-handling areas adhere to the NEC Class 1 Division 1 Group B code.

Careful component selection results in a long term operation assuming normal maintenance. The ASME piping specifications (ASME 31.3-1993) call for a 20-year piping life.

A separate utility skid (6) is provided to supply chilled water to remove the heat generated during reaction. The utility skid is designed to be set up outside potentially contaminated process areas. The utility skid also provides heated water to desaturate scrubber off-gas for HEPA protection.

Operating flexibility is incorporated into the design. Examples of this flexibility are:

Some flow directions can be changed to enable the operator flexibility. For instance, the operator can change the reactor circulation/cooling loop flow to clear inlet restrictions.

The control software is capable of supporting various operating modes such as normal interlocking or bypass selected interlocks, etc.

Three is spare capacity for additional control and monitoring equipment. The unit has at least 20% analog I/O, discrete I/O and wiring spares.

All interlocking and control functions are done within the software of the control system. With the exception of new equipment additions, any control logic changes can be accomplished by programming changes with no need for hard-wiring modifications.

TREATMENT SYSTEM OPERATION

Figure 4 below presents the simplified operating sequence for treating water-reactive mixed waste. Each step represents a unit operation or series of operations in the treatment of a generic water-reactive waste. Some steps depend on the nature of the specific waste being treated. For instance, the hydrogen treatment step is necessary only if appreciable amounts of tritium are present.

The water-reactive waste treatment system is designed to receive an inspected, sampled and sorted drum of waste. These operations are performed in a sample/sorting glovebox separate from the process. The waste drum is introduced into the nitrogen purged, sample glovebox where it is opened and inspected. If further inspection, sampling, sorting, repacking or consolidation is required, the sample glovebox is equipped with multiple ports for these operations. Once introduced into the sample glovebox, the waste drum is isolated from the atmosphere until it is empty and clean. The drum with the prepared waste is placed into a transporter vessel. The transporter is designed to transfer the waste drum between the sample glovebox and the process glovebox while maintaining a nitrogen blanket for the waste drum. The transporter is designed to survive a fall from 20 feet without rupture and subsequent release of waste.

Fig. 4.

Operators receive the waste into the process glovebox, sort the waste according to size and load the feeding device. The feeding device is an air powered bucket conveyor. The buckets on the conveyor are first manually loaded and then an automatic, interlocked system feeds one bucket every five minutes. The buckets place the feed material on a check weigh scale. If the weight is greater than 1000 grams for large solids or 250 grams for powders, the system is stopped while the operator reloads the feeder with the proper weighed feed. This system provides a check against feeding the reaction system at too high a rate.

The nitrogen purged reactor vessel is fed directly from the glovebox. Isolation valves are placed between the glovebox and the reactor. These valves are sequenced for minimum open time and to never to open simultaneously. This avoids the potential for a broken glove causing air intrusion into the reaction system sufficient to cause ignition. Additionally, the valve positions are sensed. If correct valve position is not maintained, the feeding operation ceases until the operator corrects the problem. Mechanical barriers and control system interlocking are used to assure separation of glovebox and reactor atmospheres. For instance, feeding operations

cannot commence before the nitrogen purge totalizer has reached a minimum value, the oxygen meters (4 total) all confirm the elimination of oxygen to less than 0.5% O2, all doors and valves are positioned correctly and the process itself is functioning correctly.
The reaction produces hydrogen gas and an alkaline metal hydroxide solution which is continuously neutralized with sulfuric acid. Cooling water is applied to the reactor and scrubber cooling exchangers using an intermediate heat-transfer loop. An intermediate cooling loop is used to prevent environmental contamination by insuring isolation of the main cooling system. All waste water is collected and processed at the Los Alamos National Laboratory's Waste Water Treatment Plant.
Reactor vent-off gases are scrubbed with water in a venturi-eductor scrubber before passing through a HEPA filter to the stack. The nitrogen-hydrogen mixture is vented to atmosphere after passing through water seal device. The water seal, which is located adjacent to the vent stack, is designed to allow the process gasses to exit while preventing the backflow of air into the reaction system which could potentially lead to ignition of hydrogen.
Any debris separated in the glovebox such as plastic bags, filter media, tools, etc. is stored in air-right containers until sufficient accumulations require processing of this material to remove trace amounts of water-reactive waste. This debris is processed by reintroducing to the container contents into the glovebox for "washing." Protection against ignition of any hydrogen generated includes elevated nitrogen purge rates, standard interlocking and relative trace amounts of water-reactive waste present. All transfers make use of the transporters and the sample glovebox. Any bagout operation uses the bagout port on the sample glovebox. Debris items such as depleted uranium, tools, metal pieces, reaction containers, etc. that were introduced to the reactor with the water-reactive waste, is removed from the base of the reactor after reactor draining and rinsing. This operation is done under a nitrogen blanket to prevent the potential ignition of depleted uranium. The debris is placed directly into a plastic bagout bag for immediate transport to the sampling glovebox for repacking if necessary. Any recovered depleted uranium is packed under fuel oil for later treatment.
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MIXED WASTE TREATMENT USING MAGNETIC SEPARATION AND LOW TEMPERATURE THERMAL TREATMENT
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ABSTRACT
This paper documents the treatment of a characteristic mixed waste using a combination of magnetic separation to achieve waste volume reduction and thermal

desorption to treat the hazardous organic constituent. The waste was generated during the course of performing remedial activities at a fuel fabrication facility. During the dewatering and packaging of waste from an area know to contain low levels of PCE contamination, PCE was inadvertently concentrated because of the heterogeneous nature of the contamination. With further examination, the physical properties of the fraction of waste containing the organic contamination was determined to consist chiefly of ferrous material. Using grate type magnetic separator, the PCE was removed and concentrated. The separation reduced the volume of hazardous waste by a factor of approximately 100. Following separation, the PCE was removed from the ferrous material by low temperature thermal desorption. The entire process from waste generation through treatment and disposal required less than 90 days. The paper emphasizes the need to examine the nature of contamination present in a characteristic mixed waste. Relatively simple and readily available technologies are often available to minimize the volume of waste requiring treatment.

INTRODUCTION

Decommissioning of unused facilities and surface impoundments containing mixed wastes is ongoing at a nuclear fuel fabrication facility. The decommissioning activities generate soils and sediments which must be treated to remove or stabilize RCRA constituents, dewatered, and then packaged for burial at a licensed low-level radioactive waste (LLRW) disposal site. To date over 9,000 cubic feet of sediments have been processed through on-site mixed waste treatment operations. Characteristic wastes per 40CFR261.24, Table I, D006 (cadmium) and D039 (tetrachloroethylene) have been routinely treated and disposed of at a LLRW disposal facility. The predominant radiological contaminants are isotopes of uranium and thorium.

WASTE GENERATION

One of the facilities which required demolition was a contaminated underground vault known as the scale pit. The scale pit had previously contained scales for weighting UF6 cylinders. During the removal operation, soils and rainwater were pumped from the excavation to an on-site treatment operation. This procedure generated approximately 80 drums (16,654 liters) of water, soil, and debris which could not be packaged for burial without treatment and dewatering.

An on-site mixed waste treatment process was selected as the best means for safely dewatering and packaging this material. This treatment operation is used to treat sediments and soils generated during remediation of on-site sediment ponds. The pond remediation activities are presented in a separate paper (1). The treatment process consists of four principal steps:

1. coarse material removal (knock-out tank)
2. air sparging in treatment tank
3. magnesium hydroxide stabilization
4. solids dewatering and packaging

The first processing step, the knock-out tank, is used to remove coarse and heavy material from the stream being treated. These materials are generally abrasive and the removal is primarily intended to extend the life of the in-line shredder and pumps used with the treatment and dewatering operations. It consists of a tank containing a low velocity settling section and overflow weir. Water and fines flow over the weir and are then pumped to the next processing step. Coarse material, usually greater than 200 mesh, is collected in the bottom of the tank while water and fines flow over the weir to the treatment process. The coarse material generally consists of sand, gravel, and occasionally metal debris such as wire and fasteners. During the treatment of the scale pit waste, 15 drums (3,123 liters) of coarse material was collected from the knock-out tank. This material was primarily sand. Past experience with sand indicated that flushing water through the material to remove fines would be an effective treatment since the sand typically did not retain any of the RCRA constituents requiring treatment. Following flushing, the sand was allowed to drain and then transferred into 55 gallon drums for disposal. A moisture absorbent was added to each drum to insure that no free liquid was generated during transportation and storage prior to burial.

WASTE CHARACTERIZATION

Routine sampling and analysis of the sands was performed in order to confirm that the waste did not contain any RCRA constituents. The results indicated that the flushing had not been effective in this instance. In fact the PCE in the TCLP extract was higher than other fractions of the waste tested indicating the PCE was

concentrated in the coarse material
Upon visual inspection of sample taken from the drum, three distinct material types could be identified:
1. moisture absorbent
2. sand
3. ferrous magnetic (iron oxide)
Each fraction was assayed for PCE by the TCLP extract procedure. The results are provided in Fig. 1 along with three sample results from material before separation.
Fig. 1. Knock-out Tank TCLP PCE.
From the results it was concluded that the ferrous portion of the waste from the settling tank contained levels of PCE above the regulatory limit of 0.7 ppm. Due to the large particle size, this material was not amenable to the existing air stripping process. It was also observed that the ferrous material represented less than 1% of the total waste volume. The investigation concluded the settling tank had concentrated ferrous waste along with the sand fraction. The ferrous waste also contained high levels of oil and grease.

WASTE TREATMENT

Whenever possible, mixed wastes generated during remedial activities are treated within 90 days following their generation. This procedure is consistent with 40CFR265 and greatly minimizes permitting and storage costs.
Considering the nature of the waste, thermal treatment appeared to be the most effective form of treatment. However, no form of thermal treatment was available on-site with the exception of a small oven in a permitted exhaust hood which was used for routine laboratory operations. This oven had a capacity of less than 56 liters.
Installing additional thermal treatment capacity would have resulted in additional costs and would have generated a significant delay. Such a delay could have resulted in exceeding the 90 day treatment window.
Size reduction combined with stabilization was considered, but was rejected due to the large volume increase which would have resulted.
Since the ferrous waste had been readily separated during the waste characterization effort, it was elected to duplicate this procedure on a larger scale capable of treating all 15 drums. A loading chute routinely used to fill drums was modified by securing a standard 16 cm by 16 cm magnetic grating into the discharge. A standard ceramic magnet was selected over a stronger rare earth magnet. The ceramic magnet was less expensive and provided satisfactory separation based on laboratory observations.
With this system in place, the material was poured into the chute and allowed to fall through the grate into a second drum. A photograph of the operation is shown in Fig. 2. Figure 3 is a photograph of the loaded magnet after processing one drum of waste. After each drum was processed, the grating was removed and cleaned. After one pass through the grating approximately 24 liters of ferrous material and tramp iron was collected from the 15 drums processed (approximately 2.3 metric tons).
Fig. 2. Processing waste using magnetic separator.
Fig. 3. View looking down into chute showing magnetic grating and ferrous waste.

RESULTS

All 15 drums were processed in approximately one day. Samples were taken and analyzed for PCE in the TCLP extract. The results of the analysis showed an average PCE concentration in the TCLP extract of 0.357 ppm. This value was below the limit of 0.7 ppm and the material could have been disposed of at this point. However, a close examination of the treated material revealed a small amount of ferrous material was still present.
To insure that the maximum amount of PCE bearing material was removed, the drums were passed through the magnetic grate a second time. Approximately 3 liters of ferrous waste was collected on the second pass.
Following the second pass, the material was resampled. The TCLP extract contained only 0.09 ppm PCE after the second pass. At this point the sands were sent to LLRW disposal. Figure 4 shows the initial PCE values and the values following the first and second run through the magnetic separator. A block diagram of the on-site sediment treatment process and the magnetic separation process is also shown in Fig. 4.
Fig. 4.
The 21 liters of ferrous waste were placed into trays and the trays were placed into

the laboratory oven. This material was held at 180o C for 24 hours. Samples of the material were tested and no detectable PCE was observed in the TCLP extract.

CONCLUSION

The treatment operations developed for this waste served not only as an effective and efficient process, but also saved enormous costs associated with installing on-site treatment capacity or locating off-site treatment capacity. This project demonstrates the successful implementation of magnetic separation to provide a significant volume reduction during the course of treatment. Without such volume reduction, the entire volume of the material would have remained a characteristic mixed waste. Due to the success of the separation, the amount of labor and capital equipment required for treatment was greatly reduced.

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37-52

CEMENTATION OF INORGANIC NITRATE-CHLORIDE SALT, A LOW LEVEL MIXED WASTE

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ABSTRACT

Aqueous low level mixed waste is solidified with Portland cement to immobilize hazardous metals at Rocky Flats. The waste is a caustic brine solution containing 40 wt% solids with low concentrations of hazardous metals. Problems such as mixing difficulties and waste form expansion can be encountered when the waste is solidified. The cause of the expansion problems are thought to be the formation of darapskite, a nitrate sulfate hydrate [Na3 (NO3 SO4) . H2O]. This report contains a description of the existing process and a discussion of the methods used to analyze the problems. Based on this study, a formulation of 30 wt% water, 30 wt% salt and 40 wt% salt is recommended as the optimum formulation.

INTRODUCTION

The Rocky Flats Environmental Technology Site (Rocky Flats) has a variety of waste streams that must be treated to meet the Resource Conservation and Recovery Act (RCRA) land disposal restriction (LDR), universal treatment standards (UTS) promulgated by the US Environmental Protection Agency (EPA). This paper discusses the cementation of an aqueous nitrate-chloride salt waste. This waste is generated from various sources, including laboratories, laundry, production operations, and incidental waters from runoff, and solar evaporation ponds. A brief history and a description of the process follow.

The cementation process was developed in the early 1980s as a short term corrective action to meet Department of Transportation (DOT) regulations for shipping waste. The original process went into use with a formulation of 55 wt% salt, 18 wt% cement, and 27 wt% water which resulted in a pourable waste form which met the Nevada Test Site waste acceptance criteria for disposal in effect during the mid 1980s. At that time, the waste mainly consisted of neutralized nitric acid. The cementation process consistently produced a certifiable product until late 1988 when shipping delays necessitated extended outdoor storage unprotected from the weather. During this period, the waste forms were packaged in corrugated paper containers. Subsequently, some of the containers degraded and some of the waste forms expanded, crumbled, or broke.

In June 1989, the cementation formulation was changed to 51 wt% salt, 22 wt% cement and 27 wt% water to obtain a waste form with a higher compressive strength. Improved containers were also used to reduce the expansion problems. For several months, the improved boxes and lower waste loading appeared to alleviate the swelling problems. In December 1989, the composition of the waste changed. Plutonium production was suspended at Rocky Flats, and subsequently the plant stopped producing large quantities of neutralized acid wastes. However, large quantities of waste from other areas on the plant site continued to be processed, resulting in waste containing much lower concentrations of nitrates.

In June 1990, some of the crates of cemented waste began swelling while still in the temporary storage areas. Visual inspections of these blocks revealed light and dark swirling patterns, indicating that the salt blocks were poorly mixed and not homogeneous. Bench-scale mixing tests indicated that formulations that mixed easily prior to the production curtailment were now more viscous and more difficult to mix apparently due to changes in the waste composition. To compensate for the increased viscosity, operators began adding additional brine to the formulation.
The mixing and expansion problems appeared to be solved in September 1990 as a result of a lab-scale mixability study which recommended a 30 wt% waste loading formulation. However, this waste form was not certified for disposal because of a lack of waste characterization data. It was thought that all of the waste forms would have to be reprocessed to meet EPA regulations. The waste loading was increased to 35 wt% during the fall of 1992 to help conserve storage space. Three months later, some of the newly produced waste forms began to expand and damage the containers. Concerns over how the waste forms would behave in a permanent repository and/or long storage led to the decision to determine the cause of the expansion. Historical production records were statistically analyzed and existing cemented waste in inventory was examined for clues to the cause of the expansion. A review of the production records found that the expansion was less likely when the waste loading was below 30 wt%. However, it was not possible to correlate the expansion to chemical composition of the waste. An examination of the 700 cemented waste forms in inventory revealed that approximately 15% of the waste forms expanded enough to bulge or damage the plywood containers. The bulging occurred most frequently on only one end of the 2.1 m long X 1.2 m wide X 0.6 m high containers.
The process waste water at Rocky Flats is treated to remove most of the highly radioactive constituents and is then concentrated by distillation. The aqueous stream is treated by flocculation and precipitation to remove the actinides and then it is sent to a clarifier to be concentrated. The under-flow from the clarifier goes to radioactive sludge treatment, and the overflow is sent to a holding tank where it is mixed with low specific activity wastes prior to concentration in a quadruple effect evaporator. The distillate is recycled for use as process water while the concentrated brine is waste that must be treated for disposal. This waste is a 35 to 40 wt% solids brine solution with a pH of 10.5. A portion of the brine is spray-dried to fully dry the solids in order to increase the waste loading in the final waste form. The spray-dried salt consists of hollow spherical particles with a diameter of approximately 10 micrometers and a bulk density of 0.4 g/ml to 0.7 g/ml. The spray-dried salt and the brine are then mixed together to form a salt slurry. Portland cement is mixed with the salt slurry to produce the final waste form.
The waste is batch-mixed in a fiberglass tank to produce a cemented waste form using a formulation of 35 wt% salt, 30 wt% water and 35 wt% cement. The cementation equipment consists of a 350 rpm Lightning mixer with two 35-cm A-310 impellers concentrically mounted in a cylindrical tank with a conical bottom. The impellers are evenly spaced along the vertical axis. Load cells on the tank are used to determine the amount of each component as it is added to the mixing tank. The tank is equipped with a vacuum system to remove dust and an inspection hatch for rinsing the walls of the tank and for inspecting the mixture in the tank.
To prepare the cemented waste, the tank is sequentially charged with 225 kg brine, 100 kg salt, and 220 kg cement while being continuously mixed. Initially, the tank is tared then the brine is added to the desired set point. Next, salt is pneumatically transferred to the mixing tank. After salt addition, cement is pneumatically transferred to the tank until the desired amount has been added. All of the set points are based on the total weight of the mixing tank.

EXPERIMENTAL APPROACH

The steps to solving the expansion problem were to:
1. determine the waste acceptance criteria for disposal,
2. characterize the uncemented waste,
3. analyze the cemented waste,
4. determine the limitations and capabilities of the existing process,
5. simulate the cementation process on a laboratory-scale,
6. optimize the formulation of the cemented waste form on a laboratory-scale,
7. implement the formulation of the cemented waste form at full-scale.

The rest of this report represents the efforts to solve the expansion problem.
The cemented waste form should meet the waste acceptance criteria (WAC) for a low

level mixed waste (LLMW). Since the waste form is destined for disposal at Envirocare of Utah, Inc. or the Nevada Test Site (NTS), WAC for both sites should be satisfied so that the waste can be disposed at either site. Waste bound for either site must meet the EPA treatment standards in order to be buried and the DOT standards in order to be transported. The EPA standards are based on the Toxicity Characteristic Leaching Procedure (TCLP) test, which is designed to test how well hazardous chemicals are immobilized in a treated waste form. The DOT standards pertaining to waste with less than 2 nCi/gram (the activity of the untreated waste is less than 1 nCi/gram) radioactivity state that the waste must be shipped in a "strong, tight" container. A summary of the combined WAC is given below.

TABLE I

To characterize the uncemented waste, samples of brine and spray-dried salt were analyzed for metals regulated by RCRA, common cations, common anions, and organic chemicals. A summary of the analysis (Tables II and III) shows that the waste consists primarily of nitrates and chlorides of sodium and calcium. No hazardous organic waste is allowed into the aqueous process waste by administrative control. The analytical results revealed that no organics were in the waste. Some nonhazardous organic material is present in the waste probably due to processing laundry and runoff water. A TCLP test on the uncemented spray-dried salt indicates that the untreated waste nearly meets the EPA treatment standards (Table IV).

TABLE II

TABLE III

TABLE IV

Samples of the cemented waste containing 35 wt% salt were analyzed for total concentrations of RCRA regulated metals and common anions. The TCLP was performed for comparison to the treatment standards for RCRA regulated metals, and organic compounds. As the analysis indicates (Table IV), the waste form meets the EPA treatment standards. As noted earlier, the waste loading in recent years has been between 30 and 35 wt%. By demonstrating that the waste meets the treatment standards at the higher waste loading, the newly generated waste produced with lower waste loading will also be certifiable (some random sampling will be necessary to verify this). Previous analysis showed that the waste forms produced in the late 1980s meet the EPA treatment standards at 55 wt% waste loading.

Since the cementation process immobilizes the hazardous constituents at waste loadings up to 55 wt%, the problem was to determine the cause of the waste form expansion. An examination of the historical production records and the waste forms in inventory indicated that the expansion was virtually eliminated during a 1 year period when the waste loading was kept below 30 wt%. Since the 30 wt% waste loading had been effective in the past, this waste loading was adopted as the maximum. What remained was to determine the cement type, cement concentration, and the water concentration in the cemented waste form.

The formulation of the cemented waste form was developed by simulating the process on a lab-scale, followed by testing a variety of cement types, waste loadings, and water to cement ratios. The rheological properties of typical waste and the limitations of the mixer are shown in Fig. 1. On a lab-scale, the formulations were tested for compatibility with the full-scale equipment, and subjected to durability testing in anticipation of storage in harsh weather conditions. The microstructures of the cemented lab-scale samples were examined to identify expansive crystal growth and compared to the microstructures of cemented samples produced by the full-scale process.

Fig. 1.

A variety of cement/flyash blends were tested on a lab-scale to determine whether a more durable waste form could be produced. The first objective was to find a blend that required less water because low water cements are stronger and more durable (1). In addition, excess water increases the void space in a waste form which increases the mobility of the hazardous chemicals (2). The water requirements were based on viscosity measurements for each of the blends. Type I, type I/II, and type V cements were blended 1:1 with Class F and Class C flyash. The water requirements for these blends were virtually identical. A variety of superplasticizers (dispersants for reducing water requirements) were tested, but none was able to reduce the water requirement without dramatically increasing the time-of-set. Therefore, the selection of a blend would be based on minimizing expansive tendencies of the waste form.

Lab-scale samples were subjected to freeze/thaw testing and wet/dry testing to determine which type of cement blend was least susceptible to expansion. The wet/dry tests and the freeze/thaw tests were ASTM methods modified to have fewer cycles (ASTM D 559-89, and D 560-89 respectively). The wet/dry testing consisted of placing one inch diameter by two inch long cured samples in an environmental chamber for two cycles through the following conditions: 95% relative humidity at 40 C for 48 hours followed by 20% relative humidity at 55 C for 48 hours. None of the samples expanded significantly, so it was concluded that variations in temperature and humidity were not the cause of the expansion and that no blend showed an advantage.
Other lab-scale samples were subjected to freeze/thaw testing. In this test, cured samples were placed in a freezer at -20 C for 48 hours then placed in an oven at 55 C for 48 hours. The freeze/thaw cycle was repeated. In this test, blends containing type V cement and blends containing Class F flyash did not expand but other blends expanded as much as 5% in length. At this juncture, it was thought that the expansion of the waste forms could be attributed to the formation of ettringite. Ettringite is formed when sulfate present in the waste reacts with tricalcium aluminate present in cement. Type V cement has less tricalcium aluminate than the other types of cement, which could explain why type V blends were resistant to expansion. However, ettringite was not the cause of expansion, as discussed later.
Three methods were employed to examine the microstructures of the cemented samples: optical microscopy, scanning electron microscopy (SEM), and x-ray diffraction (XRD). Optical microscopy was used to identify cement, the degree of cement hydration, porosity, homogeneity, salt crystals, and structural flaws. The SEM was used in conjunction with an energy dispersive x-ray detector (EDX) to give semi-quantitative elemental analysis of crystals observed in SEM micrographs. Cemented samples produced in the laboratory with the actual waste and cemented samples taken from full-scale waste forms were examined by these methods.
The examination of samples from the full-scale waste forms showed a correlation with the formation of darapskite and waste form expansion. Furthermore, concentrations of this mineral were higher in the bulged ends of the waste forms than in other locations. Another important discovery is that no ettringite was found in the expanded waste forms. Darapskite, a nitrate-sulfate double salt [$Na_3(NO_3)(SO_4) \cdot H_2O$], was the only chemical found in the waste forms that is known to cause expansion, although others could be present. Since darapskite is formed from chemicals that are found in the waste but not in cement, the type of cement blend does not contribute to the formation of the expansive component in the waste forms.
The lab-scale samples contained the expansive calcium aluminum chloride hydrates but did not contain darapskite or ettringite. Calcium aluminum chloride hydrates were found in the full-scale waste forms, but at low concentrations. Since the expansion in the full-scale waste forms is attributed to the formation of darapskite, but the expansion in the lab-scale samples is due to another mechanism, the lab-scale freeze/thaw testing was not a good method to predict expansion on the full-scale.
A third set of lab-scale tests were performed to determine the tolerable range of the water content in the final waste form with waste loading of 30 wt% anhydrous salt. The minimum water content is defined as the amount of water that produces a paste that is barely capable of mixing as defined by the "Mixer Limitation" line in Fig. 1. The maximum water content is defined as the highest concentration of water in the waste that will not cause free water to form on top of the waste form while it is hardening. Once the water content range was determined in the laboratory, it was tested on the full-scale equipment. The full-scale water content range was larger than determined in the laboratory. The full-scale range was found to be 28 to 30 wt% at 30 wt% anhydrous salt, and 26 to 28 wt% at 28 wt% anhydrous salt (see Fig. 2). In other words, the water to cement ratio can vary from 0.67 to 0.75 at 30 wt% waste loading, and from 0.48 to 0.64 at 28 wt% waste loading. This formulation has been successfully used on a small number of full-scale batches and will become part of the process in the future.
Fig. 2.
REMAINING ISSUES
1. Lab-scale tests to simulate the waste with reagent salts (or surrogates) were unsuccessful. The water requirements and the results of the durability testing were quite dissimilar. Simulation of the waste with "cold" material would have been convenient, but time constraints were prohibitive.
2. The time-of-set of this cemented waste form is very long and may be 10 to 25

days. A brief review of the literature and available products indicated that there is no product on the market to accelerate the set of this waste form.
3. Occasionally, the viscosity of a batch of waste will be greater than the mixer is capable of agitating, which necessitates the addition of more brine to thin the batch.

CONCLUSIONS AND RECOMMENDATIONS

1. The expansion of the full-scale cemented waste forms is due to darapskite, a nitrate-sulfate double salt.
2. Darapskite is not formed from chemicals derived from cement or flyash, so the formation of darapskite is unaffected by the type of cement/flyash blend used to solidify this waste. Type I, type I/II, and type V cements are suitable for solidifying this waste.
3. Class F flyash is a suitable admixture for this waste.
4. The expansion of cemented lab-scale samples made with actual waste (not surrogate) did not occur by the same mechanism as in the full-scale cemented waste forms.
5. The range of water content found in lab-scale simulations of the process using actual waste (not surrogate) was smaller than the range that the full-scale process could tolerate.
6. The maximum nonexpansive waste loading was determined from historical records. More thorough record keeping in the future is recommended to help solve problems that may arise.
7. The hazardous chemical constituents in this waste exist at low concentrations so meeting EPA treatment standards is fairly straightforward.
8. The waste should be treated with the following formulation for maximum waste loading and minimum expansion problems:
   28 to 30 wt% water with a waste loading of 30 wt% anhydrous salt,
   26 to 28 wt% water with a waste loading of 28 wt% anhydrous salt,
   cement or cement/flyash blend be used to make up the reminder.

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THE KINETICS OF CADMIUM BIOACCUMULATION BY TWO AQUATIC MOSSES INDIGENOUS TO QUEBEC

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ABSTRACT

One of the main problems encountered when analyzing heavy metals is their very low concentrations in the aquatic environment. In such cases it is often necessary to use bioaccumulators in order to monitor the evolution of these pollutants over space and time. The aquatic mosses selected are considered good biomonitors which can provide information on the status of the environment. In the past, a number of methods have been developed to enhance our understanding of these biological interactions; but the major impediment has always been the difficulty of maintaining the aqueous concentration constant at the desired values. The experimental system developed for the present study is a simple one: it opts for gravitational rather than a mechanical one. A manual control ensures a constant rate of input feed water in this dynamic system. Experiments were carried out at concentrations of 0,10, 25 and 50 mgL-1 both with and without "EDTA" as a chelating agent. The goal pursued was to assess the performance of two aquatic mosses -Fontinalis dalecarlica Schimp ex. B.S.G. and Platyhypnidium ripariodes (Hedw) Dix.- during the build-up of cadmium over a 28-day period. Preliminary results indicate a high level of bioaccumulation on the first day. In the absence of the chelating agent, accumulated cadmium varied between 150 and 300 mgg-1 according to the genus of bryophyta. On the basis of our results, it would seem that the mosses suffer an internal ionic imbalance if the accumulation period is too long (28 days). The use of EDTA has proven useful, 1) in maintaining a constant level of pollutant concentration, 2) in lowering the

bioavailability of the metal under study and 3) in simulating more closely the natural conditions of the environment.

INTRODUCTION

Our aquatic environment is becoming more and more stressed by the many pollutants generated by modern technology. According to Hogan and Whotton the toxic metallic substances, especially cadmium, is a potential menace for the survival of the ecosystem.

One of the main problems associated with the analysis of heavy metals is their low concentration in the aquatic milieu. Direct analysis of these metals in water is often inconclusive because their concentrations are close to the normal detection limits. In order to counter this difficulty, pertinent tracers can be used. To date a variety of bioindicators have been used and depending on the pollutant, the efficency will vary according to the type and complexity of the biocaptor chosen. Bryophyta have received much attention lately for metallic analysis (Agneta and Burton, 1990). The mosses are hardy and produce many leaves which favor their use for the monitoring of contaminants in aquatic areas (Berryman and Richards, 1992). They rapidly extract most metals by simple ionic exchange from water (Mouvet, 1987). The objectives of this laboratory study are 1) to evaluate the performance of two aquatic mosses indigenous to Quebec as potential biological indicators of cadmium, 2) to compare the accumulation kinetics of this pollutant in the presence and absence of EDTA (ethylene diamine tetraacetic acid), 3) to verify the efficiency of a semi-open dynamic system with gravitational input flow, and 4) to apply a mathematical model in order to extract the uptake constant(k1), the depuration constant (k2) and the bioconcentration factor (BCF).

MATERIAL AND METHODS

Sampling

The moss species genus Fontinalis was collected in the summer of 1993 from the River Cach, a small river in the southern part of the Laurentide Reserve. This moss lives totally submerge and is attached to the bottom rocks by their rhizoides (a type of small hook on the end of a semi-root). As for Platyhypnidium, this one was found in a small water cascade close to Loretteville. Limestone rocks dominated this habitat rich with mosses of this genus.

Once the mosses were located, they were collected by cutting the top two thirds of the plant rather than tearing them from their anchorage so as to preserve the survival of this species. A series of successive washings helped remove the unwanted debris (invertebrates, sand, etc.). Then the samples were placed on ice and transported to the laboratory for the acclimatization period. It is well known that mosses harvested during the summer season will have less difficulty adapting to laboratory temperatures (18-20C) than those collected in the fall (Saitoh et al., 1970).

The conservation of the muscinal biomass necessary for the experimental stage was done in rectangular basins made of fiber glass (1 x 2.5 meters). The temperature was controlled by a glycol filled serpentine which cooled the water in an adjacent container. This temperature adjusted water was circulated by the continuous pumping action of pressurized air in the return pipe. A "Grow Lite" Fluorescent furnished sufficient light to maintain the aquatic mosses in good health .

For the laboratory study, the cadmium concentrations had been fixed at 0, 10, 25, and 50 mgL-1. Certified standard solutions of $CdCl_2$ was used (Fisher Co). For the aquaria with EDTA, the concentration was 3 mgL-1. The ethylenediaminetetraacetic acid (EDTA) was used to simulate nature in preference to other organic materials or humic acid because of its well-known stability constants.

Experimental Design

An experimental set-up attempting to simulate natural conditions was developed. The system is simple and cheap. It uses gravitational flow rather than mechanical pumping.

Fig. 1.

A series of ten aquaria, each supplied by 220 liter reservoir, is shown in Fig. 1. Each aquaria contained 720 grams of each bryophyta initially. At each sampling time, that is 0, 1.5, 3, 6, 12, 24, 48, 96, 144, 192, 240, 288, 336, 384, 432, 480, 528, 576, 624, et 672 hours, 9 grams of each moss were retrieved per aquarium, the biomass necessary for three replicates. This sampling schedule generated a maximum of information during the rapid accumulation stage and also allowed the plant to respond over a longer period of time. In order to assure a uniform exposure to the

aqueous cadmium, each aquarium of 30 liters contained both moss species. The physico-chemical parameters (temperature, conductivity, dissolved oxygen, pH, and oxydo-reduction potential) were measured with a Hydro-Lab Surveyor II (SVR 2 model).

Chemical Analysis

Each subsample was oven-dried at 110C for 12 hours. After weighing, the samples were subjected to concentrated nitric acid (HNO3) attack for 7-8 hours until the elimination of the brown fumes; the reaction was driven to completion with 30% hydrogen peroxide (H2O2) (Berryman, 1991). When the reaction was finished, the reaction tubes were cooled and maintained at 4C. The digestions were diluted to exactly 50 mL with distilled water. They were then manually stirred to assure a completely homogeneous solution. Before proceeding to the spectroscopic analyses, it was very important to allow the solutions time to attain room temperature, otherwise a physical interference was evident in the subsequent analysis. The digested mosses were analyzed by atomic absorption spectrophotometry (Varian model AA1275) with air/acetylene flame. Certified standards for water (Standard Reference Material 1643 C) and mosses (Certified Reference Material BCR No 61) were used in the quality control scheme.

Statistical Analyses

The accumulation of metal in the byrophyta is made in two stages; a first fast stage characterized by a passive ion exchange process lasting only a few hours. The second stage requires many days and slowly reaches a plateau value of accumulated cadmium. This slow step is considered an active one where the metal penetrates to the cell interior while in the first stage, the metal is found primarily in the cell walls (Pickering and Puia, 1969; Breuer and Melzer, 1990). Since the mosses appear to accumulate in an exponential fashion, the two compartment model of van Hattum et al., (1989) was applied in an attempt to interpret our kinetic accumulation data:

Eq. (1)

Since the steady state conditions are:

Eq. (2)

eqn (1) becomes:

Ct = Css [1 - exp (- K2t)]

(3)

where

Ct = concentration in the mosses at time t (mgg-1 dry weight)

Css = concentration in the mosses at steady-state conditions (mgg-1 dry weight)

Cw = concentration in the water (mgmL-1)

K1 = rate constant for uptake from the water (hour -1)

K2 = rate constant for elimination from the mosses (hour -1)

In addition, a bioconcentration factor(BCF) can be calculated from the estimated rate constant values K1 and K2 (Walker, 1987; van Hattum et al., 1989; Tessier et al., 1994) where:

Eq. (4)

RESULTS

The original concentrations of cadmium introduced into the water in presence of the chelating agent varied very little (Fig. 2). Indeed after 96 hours, the concentration at 50 mgL-1 fluctuated around 11 mgL-1 and after that the values stabilized. The relative variations were the same for the 25 mgL-1 concentration (sample and replica) and corresponded to the same time period. No important variation was recorded at the 10 mgL-1 level and the values of the control also remained constant. In absence of EDTA, the cadmium concentrations fall radically within the first few hours. At 50 mgL-1 (both test and replica) the values dropped to 10 mgL-1 within 48 hours and varied between 15 and 20 mgL-1 during the rest of the accumulation stage. Similar radical decreases of 15 and 8 mgL-1 were noted for 25 and 10 mgL-1 respectively. The control aquaria maintained stable reference values through the experimentation.

Fig. 2. The cadmium concentration (mg L-1) in the water during the accumulation stage.

Examining the processus of cadmium accumulation of F. dalecarlica (Fig. 3) in the three environments studied(50, 25 and 10 mgL-1), the mosses in presence of EDTA shows an initial bioaccumulation of cadmium followed by a slow negative shift to an equilibrium concentrations of 20 mgg-1 (50 mgL-1) and 5 mgg-1 (10 mgl-1). In absence of EDTA, the accumulation values are much higher; the pattern is identical for the three concentrations studied: relatively fast for the first 192 hours, followed by a

slower accumulation in a discontinued manner. The highest values attained after 672 hours exposure are 350, 250 and 125 mgg-1 of cadmium when the initial concentrations were 50, 25 and 10 mgL-1 respectively. The cadmium in the controls remain constant.
Fig. 3. Accumulation pattern of cadmium by Fontinalis dalecarlica at four concentrations in the presence and absence ofEDTA complexing agent.
The accumulation of cadmium by P. ripariodes (Fig. 4) indicates that in presence of the chelating agent, accumulation is small regardless of the aqueous cadmium concentration. Indeed after 192 hours of exposition the values drop from 50 to 25 mgg-1 (50 mgL-1) and from 25 to 10 mgg-1 during the same period (25 mgL-1) (test and replicate) after which the accumulated concentration stabilized. The bioaccumulation at 10 mgL-1 is negligible. In presence of EDTA a fast bioaccumulation is evident up to 192 hours. This pattern is similar for all three concentrations of aqueous cadmium. Within the 192 hours surge, the cadmium is rapidly accumulated to attain values of 175, 125 et 100 mgg-1 for 50, 25 and 10 mgL-1 respectively. After this, the accumulation is slow with an irregular pattern which continues until the values of 275 175 and 100 mgg-1 are reached after 672 hours of accumulation. The values of the controls remain small constant.
Fig. 4. Accumulation pattern of cadmium by Platyhypnidium riparioides at four concentrations in the presence and absence of EDTA complexing agent.
The kinetic parameters of cadmium accumulation by mosses with and without a chelating agent are shown in Table I. The actual concentrations measured in water are much lower than the original values; that is 28.8, 10.1, and 3 mgL-1 instead of 50, 25 and 10 mgL-1 respectively. These lower values certainly had an impact on the equilibrium concentrations at steady-state conditions (Css). F. dalecarlica had accumulation values between 120 and 320 mgg-1while P. ripariodes was situated between 97 and 204 mgg-1. The Css values of the replicates show good repeatability of our system. The values of the elimination constant (K2) are very small for F. dalecarlica and are situated between 0.005 (10 mgL-1) and 0.010 (25 mgL-1). The calculated accumulation constants (K1) are evaluated between 194 (10 mgL-1) and 55 (50 mgL-1) for F. dalecarlica while for P. ripariodes, they attained values between 266 (10 mgL-1) and 54 mgg-1 (50 mgL-1). The highest coefficient of correlation (R2) (0,914 et 0,860) were associated with the initial concentrations of 50 mgL-1 and the lowest (0,718 et 0,733) with 10 mgL-1 for both genera. F. dalecarlica exhibited the larger bioconcentration factor values (BCF) varying from 38 800 (10 mgL-1) to 15 400 (50 mgL-1) while that of P. riparioides passed from 33 220 (10 mgL-1) to 6 750 (50 mgL-1).
TABLE I
DISCUSSION
The use of an experimental system simple and reliable has permitted the study of the bioaccumulation processus over a long period of time and to constat that the bryophyta used are good indicators which respond rapidly and perform well for the detection of cadmium.
The variations of cadmium concentration in water with and without EDTA are different. Indeed, the cadmium concentrations remained relatively stable in the aquaria where a chelating agent was used while in the solutions without EDTA, the variations were fairly large. The same phenomena was observed during the analysis of cadmium in the mosses concerning the presence and absence of EDTA. Cadmium accumulation by the bryophyta is more effective without the complexing agent. Here EDTA seriously lowers the biodisponibility of the cadmium cation to the mosses (Mouvet, 1986; 1987).
To compare the accumulation results of mosses, it seems that F. dalecarlica was a better bioaccumulator than P. ripariodes. Indeed, this is confirmed by the higher values for the equilibrium concentration at steady-state conditions (Css), the lower values for the epuration constant (K2), the larger values for the accumulation constant (K1) and a overall larger bioconcentration factor (BCF) for F. dalecarlica. Both mosses in the presence of 50 mgL-1 had smaller elimination constant values (K2) than when exposed to 25 mgL-1, this suggest the possibility of some physiological stress at higher concentrations of metal pollutant. The accumulation kinetics of cadmium in mosses in presence of EDTA indicate that here the accumulation is negligible, consequently these data do not permit the modelization of this experimental part.
The results obtained from this study confirm the special capacity of mosses to accumulate well the cadmium ion. Our observations are corroborated by Penuelas,

1984a; Berryman, 1991 and Mersch,1993 but differ with the data of Mouvet 1987 who preferred P. ripariodes over the other species.
In addition, we have conducted our experiments for a period of time twice is long as that cited in the literature thus allowing our data to attain the steady-state plateau followed by an irregular pattern of accumulation.
Our experiments have shown that because of the passive nature of metallic ionic exchange reaction with the plant cell walls, the largest part of the accumulation is accomplished rapidly in the first few hours to attain a level where the accumulation stabilizes. The major part of accumulated metal in mosses can be explain by the abundance of leaves containing negatively charged sites such as uronic, glucuronic and other carboxylic acids (Mersch, 1993). The rapid rate of accumulation constitutes a large advantage for these aquatic mosses. Consequently, they represent a good choice to use in environmental assessment of natural running water especially during an accidental discharge or spill of toxic substances.

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SCREENING OF AQUATIC PLANTS AND ALGAE FOR REMOVAL OF TOXIC HEAVY METALS

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ABSTRACT
Aquatic plants and algae are known to accumulate metals and other toxic elements from solution. Different plant and algae species, even strains of the same species, exhibit large variability in their capacity to sequester toxic elements. Bioremoval of toxic elements is also greatly affected by the presence of other ions and the physical-chemical factors during contacting. Therefore a plant/algae screening program, using simulated and actual contaminated water is the required first step in development of a bioremoval process. This paper summarizes results from a screening process used to determine the capability of sea weeds, aquatic plants and algae for adsorbing the heavy metals Cd, Zn, Pb, Ni and Cu. A successful candidate would exhibit the following characteristics: 1) Capability to reduce metal concentrations to the required regulatory levels; 2) High specific metal adsorption capability (mg metal adsorbed/kg of dry biomass); 3) Capability of removing several metal ions simultaneously; 4) High productivity in a low cost cultivation system; 5) Ease of harvesting, processing, storage and/or separating biomass from cultivation water; 6) Ability to be regenerate and recover metals.
A metal-biomass contacting protocol was established with a known amount of biomass in 50 ml de-ionized water containing the metal to be adsorbed. The mixture was shaken and then centrifuged and filtered. Both biomass and filtrate were analyzed with an atomic absorption spectrophotometer to measure metal concentration. For promising species of plants, further studies were performed to establish adsorption isotherms.
The results of the screening program demonstrated some plants had significantly higher metal uptake than others. Myriophyllum spicatum, Phormidium and Spirulina were the most promising plants, having high adsorption capability at low residual concentrations.
INTRODUCTION
Aquatic plants and algae are known to accumulate metals and other toxic elements from solution (1-7). There are very large differences in bioremoval due to species and strain differences, cultivation methodology, and the metal binding experimental procedures. Some systems have been shown to operate at low pH and may be applicable to the treatment or polishing of acid mine waste waters (8,9). However, very few commercial applications have been reported for the use of aquatic plants in bioremoval. Commercial systems developed presently (6,10) use immobilized algal biomass for the removal of U and Hg from different waste waters (11). Seaweeds also appear to have potential in bioremoval. Many metal ions can diffuse rather easily through the often rough surface area of seaweeds. However, considerable limitations to commercial application still exist, in general.
The commercially available biomass used in most studies is generally not satisfactory for specific bioremoval applications. Therefore, a plant/algae screening program, using simulated and actual contaminated waters is a required first step in development of bioremoval processes. The goal of this study was to find promising species for such a process. A successful candidate for bioremoval will bind large quantities of metal and to reduce residual metal concentration to meet USEPA water quality criteria (12) and suitable for mass cultivation.
Plants Used
The plants used for screening experiments included six different vascular plants, two macroalgae (four different strains of one genus) and five different microalgae. Gracillaria Strain G16 (red, brown, green), Ulva lactuca, and Lemna minor (Duck weed) were collected from the Aquaculture Division of Harbor Branch Oceanographic Institution (HBOI). Hydrilla verticillata (hydrilla), Pistia stratiotes (water lettuce), Myriophyllum spicatum (water milfoil), Hygrophila polysperma (hygrophyllum), Alternanthera philoxeroides (alligator weed) were collected from the University of Florida, Institute of Food and Agricultural Sciences, Agricultural Research Center, Ft. Lauderdale, FL. Duckweed (wild) and Gracillaria (wild) were collected from naturally growing areas. Spirulina was provided by Cyanotech Corp. in Hawaii, Algal Turf from the University of California, Richmond Field Station Algae Laboratory. Phormidium, Synechococcus and Nitzchia were collected from Microbial Products, Inc., culture center.
The outdoor enrichment that provided Phormidium was designed to select for attached microalgae, capable of withstanding high insolation. Resulting communities were assemblages containing single cell green algae and diatoms, mostly enmeshed in

tangles of several species of filamentous algae, both green and blue-green. The Phormidium species was by far the dominant organism.
Laboratory cultivation of Phormidium was done in glass-sided tanks with inclined planes made of roughened translucent plastic rising out of a shallow (3-5 cm) pool of media. The media was pumped over the inclined planes, starting at the top and forming a stream of flowing water. In these culture units, Phormidium sp. rapidly displaced all of the other microalgae. The loose tangles became tightly bound to each other forming a flattened mat. The cultures were grown using modified Allen media (1968).
Myriophyllum spicatum (Water milfoil) was grown both indoors and outdoors in tanks. Soil (undefined composition) was added along with the slow releasing fertilizer "Sierra" (commercially available). About 40 g of fertilizer per sq. ft. was added along with fresh water.
Contacting Experiments
Weighed amounts of biomass were added to metal solutions yielding 50 mL total volume in 125 mL Erlenmeyer flasks equipped with screw caps and incubated in a shaker for 60 minutes. All experiments were conducted in triplicate. The pH and temperature of the contact solution both before and after the experiment was measured. At the end of the experiment, the contents of the flasks were either centrifuged (microalgae) or strained (macroscopic plants) to separate the biomass from the supernatant. The supernatant was immediately analyzed with an atomic absorption spectrophotometer to measure metal concentration left in the solution. The biomass was stored for future digestion with nitric acid in a microwave digester (MDS 2100).
Phormidium was collected from the culture tank, weighed and blended at low speed for 12 sec. The blended liquid was centrifuged at 1500 rpm for 15 min. The supernatant was discarded and the remaining biomass used for the contacting experiments. The blending was necessary to disrupt the impenetrable algal mat. Water milfoil was collected from the culture tanks washed with DI water, cut into pieces (1-2 cm) and blotted dry.
Both dry weight and ash free dried weight (AFDW) analyses were performed as per AWWA Standards Methods (1993), pH of the solutions was measured using a Cole-Palmer portable pH meter and temperature was recorded with a thermometer.
Metal Analysis
All samples were analyzed using an Atomic Absorption Spectrophotometer (AAS) Perkin Elmer Model 3100 or an Inductively Coupled Argon Plasma (ICAP) Trace Analyzer Thermo Jarrell Ash Model 61E. The latter was used for determining low metal concentrations (< 0.1 ppm). Before running the samples, the instruments were optimized, and a calibration blank was run. A five point standard curve was then established. Prior to sample analysis, one of the median standards was rerun. At the end of 20 samples, the 20th sample was rerun as a duplicate. A matrix spike was also performed before starting the next run of samples. The recovery of the standards, duplicates and the matrix spikes was within 85-115%.
Data reduction and Derived Variables.
The mean values from triplicate runs of both initial and final metal concentration were used to calculate the following parameters:
  Percent metal remaining in solution
  Percent metal removed by biomass
  Specific adsorption capability (mg of metal adsorbed/kg of AFDW biomass). Specific adsorption was computed in two ways. The amount of metal adsorbed was calculated from the difference in metal concentration before and after contacting. It was also determined from the metal content measured in the separated biomass.
A linearized form of the Langmuir adsorption isotherm equation was plotted:
Eq.
where,
Ym = the maximum specific adsorption;
k = equilibrium constant; and
Y = specific adsorption at residual metal concentration C.
From a plot of C/Y vs. C, the slope (S=1/Ym) gives Ym and the intercept (I=1/kYm) gives k.
Scale up Experiments
A large quantity (kg) of biomass was used in metal contacting experiments (10 L metal solution). Specific adsorption capability obtained from both small and large scale experiments were compared to determine the value of the 50 mL small scale

experiments in predicting bioremoval when both biomass and metal densities are much higher.

Results

A total of 187 small scale (<100 mg biomass) and 6 kg quantities of contacting experiments were performed to screen 16 biomass for their metal adsorption capability. These sixteen biomass species were screened with an initial metal concentration of 1 mg/l. Most of the experiments were carried out with Zn and Cd. The time course experiments performed with Gracillaria and Hydrilla plants show no significant increase in metal adsorption by the biomass after the first 30 minutes of contact. Biomass density experiments for most plants/algal screen in this study exhibited the expected increase in metal removed as biomass density increased. The screening results show that some plants had significantly higher metal uptake than others. Phormidium exhibited specific adsorption of 1092 mg/kg Cd, and 823 mg/kg Zn, and water milfoil 532 mg/kg Cd and 823 mg/kg Zn. Both plants had higher adsorption capability than other plants when exposed to 1 mg/l of metal ion solution.

Further contacting experiments were conducted with certain plant species examined in this study. Water milfoil exhibited a low residual Cd metal concentration (<0.1 mg/l) while still adsorbing at 600 mg/kg. Specific adsorption of Cd onto Spirulina was 600 mg/kg at a residual concentration of 0.7 mg/l. Gracillaria green and red exhibited a 650 mg/kg and 300 mg/kg of adsorption at 0.5 mg/l of residual concentration. The initial screening results showed that Hydrilla, Hygrophyllum and Alligator weed had reached adsorption saturation (<300 mg/kg) at this residual concentration (0.5 mg/l). With residual concentration of 1.0 mg/l, the specific adsorption of Zn onto Spirulina was 1000 mg/kg. The highest adsorption of Zn onto Alligator weed, Hydrilla and Gracillaria red and brown was <200 mg/kg. These plants reached saturation adsorption at 0.5 mg/l. Phormidium, another filamentous blue-green algae, showed a high adsorption capability. Experimental results demonstrated the specific adsorption of Pb ranged between 1657 to 4580 mg/kg as biomass density varied from 0.001 to 0.005 kg/l. At biomass density of 0.0025 kg/l, the specific adsorption of Pb was 4580 mg/kg with residual concentration of 0.21 mg/l. These initial screening experiments demonstrate Water milfoil, Phormidium, and Spirulina were the most promising plants, having high adsorption capability at low residual concentration.

The scale-up (kg quantity) and small (<100 mg) contacting experiments were also performed. Gracillaria brown, Ulva, Hydrilla were exposed to metal solutions containing 0.25 ppm Cd, and 1.00 ppm Zn, and Spirulina was exposed to 1.25 ppm Cd and 5.00 ppm Zn solution. The results show that both kg and small scale experiments exhibited little variation on specific adsorption. The results further demonstrate that data for scale-up of the bioremoval process can be initially obtained from a small scale laboratory experiment using low biomass densities.

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DEVELOPMENT OF CHEMICAL PROFILES FOR U.S. DEPARTMENT OF ENERGY LOW-LEVEL MIXED WASTES

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ABSTRACT

Chemical and radiological profiles of waste streams from U.S. Department of Energy (DOE) low-level mixed wastes (LLMWs) have been developed by Argonne National Laboratory (ANL) to provide technical support information for evaluating waste management alternatives in the Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS). The chemical profiles were developed for LLMW generated from both Waste Management (WM) operations and from Environmental Restoration (ER) activities at DOE facilities. Information summarized in the 1994 DOE Mixed Waste Inventory Report (MWIR-2), the Pacific Northwest Laboratory (PNL) Automated Remedial Assessment Methodology (ARAM), and associated PNL supporting data on ER secondary waste streams that will be treated in WM treatment facilities were used as the sources for developing chemical profiles. The methodology for developing the LLMW chemical profiles is discussed, and the chemical profiles developed from data for contact-handled (CH) non-alpha LLMW are presented in this paper. The hazardous chemical composition of remote-handled (RH) LLMW and alpha LLMW follow the chemical profiles developed for CH non-alpha LLMW.

INTRODUCTION

Chemical and radiological profiles of waste streams from U.S. Department of Energy (DOE) low-level mixed wastes (LLMWs) have been developed by Argonne National Laboratory (ANL) to provide technical support information for evaluating waste management alternatives in the Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS). LLMW is material that is both a Resource Conservation and Recovery Act (RCRA)-hazardous waste and a low-level radioactive waste. DOE LLMW contains RCRA-regulated chemicals or special waste types in a form or concentration sufficient to render the waste hazardous under the guidelines of Title 40, Part 261, of the Code of Federal Regulations (CFR). Two major sources of LLMW in the DOE system are (1) inventory and operations-generated wastes (i.e., Waste Management [WM] LLMWs) and (2) wastes generated from site restoration or decontamination and decommissioning (D&D) of DOE facilities (i.e., Environmental Restoration [ER] LLMWs). The WM LLMWs include the LLMW currently (1994 inventory) in storage and the LLMW projected to be generated through 2013 at 43 sites. The ER LLMWs include the secondary LLMW streams expected to be generated from the treatment of contaminated soils and from the D&D of facilities at 14 DOE sites through approximately 2030.

Handling of LLMW is classified as either contact-handled (CH) for waste with a dose-at-waste surface <200 mrem/h or remote handled (RH) for waste with a dose-at-waste surface >200 mrem/h. The handling category determines the level of protective shielding required to safely store and process the material. LLMW is also classified as either alpha LLMW or non-alpha LLMW. The alpha LLMWs have combined activities from transuranic (TRU) radionuclides with half-lives greater than 20 years (between 10-100 nCi/g). The non-alpha LLMWs have TRU activities <10 nCi/g. Radiological profiles for DOE LLMW have been developed and are described in a

separate paper prepared by Wilkins et al. (1). Currently, the four radioactivity classifications CH non-alpha LLMW, CH alpha LLMW, RH non-alpha LLMW, and RH alpha LLMW determine the handling and routing for treatment and/or disposal of all LLMW.

WASTE STREAM CATEGORIES FOR LLMW

The most nearly complete information available on LLMW is contained in the recent update of the Mixed Waste Inventory Report database (2). This database is referred to as MWIR-2 in the EM PEIS and identifies the types of chemicals in over 2,000 waste streams of LLMW from major DOE installations. The LLMWs are grouped into nine major waste types: aqueous waste, organic liquids, solid process residues, soils, debris waste, special waste, inherently hazardous waste, unknown, and treated waste. In MWIR-2, these 2000+ waste streams within the nine major waste types are further condensed into 109 waste stream treatment codes on the basis of their physical-chemical characteristics. Many of these 109 waste stream treatment codes have similar physical-chemical compositions so that the waste streams can be treated with the same technologies to reduce or stabilize the toxic materials within the waste. For the EM PEIS, the 109 waste treatment codes have been further condensed into 32 treatment categories; the first 23 treatment categories constitute more than 90% of all LLMW. A baseline treatment flowchart for LLMW management has been developed by the Mixed Waste Treatment Project (3) to discuss the treatment technologies required for reducing or stabilizing toxic materials in these first 23 treatment categories. Descriptions of hazardous components in these 23 treatment categories are summarized in Table I.

TABLE Ia

TABLE Ib

HAZARDOUS CHEMICAL CONSTITUENTS FOR LLMW

Detailed chemical analyses of the RCRA-hazardous constituents in LLMW are available for only a limited number of LLMW streams. In MWIR-2, only about 5-10% of the 2,000+ waste streams contain quantitative data on chemical composition. Much of the information on the hazardous chemical composition of LLMW is derived from site-specific (process) operational knowledge. The composition and concentration of RCRA-hazardous chemical constituents for the first 23 waste treatment categories of CH non-alpha WM LLMW were estimated from 1) compilation of the chemical data presented in MWIR-2 and 2) an engineering assessment of the industrial processes that generated the respective LLMW streams. In the EM PEIS, chemical profiles for alpha LLMW were not developed separately because of the relatively small volume compared with non-alpha LLMW.

Hazardous chemical constituents for ER-derived LLMW have been developed on the basis of chemical concentrations for specific secondary waste streams outlined in the Automated Remedial Assessment Methodology (ARAM) database developed by Pacific Northwest Laboratory (4), as well as engineering estimates of the sources of LLMW generation. The ARAM was developed for the purpose of estimating the effects of different ER cleanup strategies on waste volumes, cost, labor, and reduction of public risk (4). In the ARAM database, the ER waste volumes are estimated on the basis of semirestricted ER alternatives. The chemical concentrations of secondary wastes are estimated from the measured concentrations of primary wastes; these secondary waste streams will be shipped to waste management facilities for treatment. It is assumed that ER wastes will be shipped to WM treatment facilities in two forms: pretreated, awaiting final treatment at WM facilities, and fully treated, awaiting final disposal. Two additional assumptions have been included in the ARAM LLMW estimates. First, radioactive, friable-asbestos-contaminated waste is included as an ER LLMW. Substantial amounts of ER friable-asbestos-contaminated radioactive waste are expected from ARAM results for the D&D of DOE facilities. Second, inadequate information exists for ARAM to distinguish between low-level waste (LLW) and LLMW during the D&D of DOE facilities. For the EM PEIS, it is estimated that 9% of the D&D waste listed as LLW in the ARAM results will be LLMW. The ARAM estimates are available for 14 major DOE sites for only 4 waste stream categories. The chemical profiles for ER-derived alpha LLMW and RH LLMW follow the profiles developed for the equivalent ER-derived CH non-alpha LLMW. Information about historical DOE site operations and industrial processes, as well as MWIR information, suggest that the waste streams of a given treatment category from different sites in the DOE complex are similar in chemical composition.

Currently, over 100 individual chemical species are identified in the MWIR-2 and ARAM databases; many of these species are either present in minute quantities,

limited to only a few waste streams, or not regulated as RCRA-hazardous chemicals. To provide a consistent assessment of chemicals that is comparable across waste treatment categories, the chemical profiles developed are condensed into 16 distinct chemicals and/or grouped into classes of chemicals that are common to most waste streams. These include six toxic metals, three inorganic chemicals, and seven classes of organic chemicals that are grouped according to important treatment parameters such as density, solubility, volatility, and degree of chlorination. Table II presents chemical profiles for WM LLMW. Chemical profiles for ER LLMW are provided in Table III.

TABLE II

TABLE III

DISCUSSION

The WM LLMW database in MWIR-2 and the ER LLMW database in the ARAM do not provide sufficient quantitative chemical characterization of LLMW streams. Several assumptions have been made to utilize the information provided in these databases, and these assumptions introduce uncertainties into the LLMW chemical profiles. A 10% reduction factor was used to convert gross waste volume to net LLMW volumes based on estimates from typical industrial waste filling practice. The estimated LLMW volumes and the ratio of specific waste stream volume to total waste volume estimated for each waste category were used to determine major waste streams and major contaminants under each waste category. The accuracy of the reduction factor used might result in uncertainty in the presence of contaminant in a particular waste category. The estimations of future waste generation volumes in these databases lead to uncertainties in developing the chemical profiles. At some DOE sites, the future waste generation rates are only projected to a single year, or to 5 or 10 years. The 20-year projection is estimated by assuming that the most recent generation rate will be continued for the remainder of the 20 years. Additional uncertainties might be introduced regarding the possible changes of future waste-generating operations. The assumption that chemical profiles are waste-stream-specific is made because insufficient data are available to develop site-specific chemical profiles. There is no justification for this assumption, and the true chemical profiles may vary from site to site. Limiting the chemical profiles to 16 chemicals may overlook an important chemical contaminant that appears only in minute quantities in the database but may be present in large quantities in a poorly characterized waste stream. Such an oversight can lead to significant errors when the developed chemical profiles are applied to estimate emissions generated from various treatment processes.

The chemical profiles developed for DOE LLMW are useful for the following tasks: 1) proposing appropriate treatment technologies for waste management and 2) estimating secondary wastes and emissions generated from various waste treatment processes. Currently, the lack of quantitative waste stream information makes the development of chemical profiles very difficult. Existing information is made useful on the basis of several assumptions; however, the uncertainties introduced by these assumptions cannot be accurately assessed with currently available data.

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Session 38 -- Weapons Grade Plutonium and HEU Disposition - Worldwide
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38-1
ELECTRICITY FROM SURPLUS MILITARY PLUTONIUM THE BEST TECHNICAL AND COMMERCIAL OPTION
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ABSTRACT
Against the background of the 1995 review conference of the Nuclear Non-Proliferation Treaty and the surplus stocks of fissile material from the Former Soviet Union and US weapons programs, there is mounting interest in the available options for longer-term management of these materials.
The long-term storage of weapons components does not meet disarmament and non-proliferation objectives of all responsible bodies involved in these programs. The two most readily realizable options (1) for effective plutonium disposition are:
- fabrication into mixed plutonium and uranium oxide (MOX) fuel.
- vitrification or equivalent immobilization prior to storage and disposal as high-level waste.

The nuclear industry, for its part, fully supports the NPT (2) and is ready to bring its experience to bear. The required facilities can be implemented with confidence, drawing heavily on experience gained from US and overseas nuclear programs, although there are a number of uncertainties relating to the immobilization option which require further development. This paper describes the current facilities at Sellafield, and summarizes wider European experience in manufacture, transport and use of plutonium fuels in existing commercial LWRs together with associated waste management and transportation. The paper also briefly addresses the implications for the reactor operator loading plutonium fuels and the implications of incorporating significant quantities of plutonium in vitrified waste.
MOX fuel in LWRs is proven technology with a lifetime cost which can be predicted with confidence. In Europe, the civil nuclear industry has an established program of plutonium recycle and of vitrification of high-level wastes arising from commercial reprocessing facilities. These vitrified wastes include only trace quantities of plutonium, although the engineering skills for design of plant with higher plutonium incorporation exist.
MOX FUEL EXPERIENCE
MOX fuel loadings have been demonstrated in Europe since 1963 with the first loadings in Belgium (LWR) and the UK (gas-cooled reactor). Over 300 tons of MOX fuel have been fabricated to date, recycling 15 tons of reactor-grade plutonium. Programs to load, or investigate MOX fuel have been undertaken in the US, Canada, Euratom countries, Switzerland and Japan. It is perhaps worth restating that, from the earliest times of the commercial nuclear program, the expectation was that plutonium from reprocessing (and depleted uranium arising from enrichment plants) would feed fast reactor programs. Although fast reactor development programs survive in Europe and the Far East, dates for commercial realization are in the distant future and there has been a refocussing of plans for use of separated plutonium in existing reactors.
BNFL's MOX Demonstration Facility (8thm/y) is now operational and draws on experience of 30 years Pu fuel production in the UK. This plant has a unique "short binderless route" for blending UO2 and PuO2 powders, using a high energy attritor mill to achieve intimate missing and a spheroidiser to condition the powder before it is used for the pelleting process. BNFL is currently constructing the Sellafield MOX Plant (SMP) with a capacity of 120 thm/yr using this "short binderless route". This is the third MOX fuel plant on the Sellafield site. The fuel manufacturing process and plant design meet the highest current standards in terms of safety (including occupational dose control appropriate for plutonium from high burnup LWR fuels) and safeguards.
Great attention has been focused on criticality safety, with "safe by shape" vessels and minimal potential for hidden process hold-up. The Company requirement for occupational dose is that the average exposure of the group of works associated with the plant should not exceed 5 mSv/yr (whole body external plus internal) in addition to "As Low As Reasonably Practicable" (ALARP). To meet this stringent criteria for a plant of the scale of SMP requires significant automation and shielding.
Rapid residue recycling and the "short binderless" pellet production route reduce hold-up problems and total inventory. Passive Non-Destructive Analysis (NDA) determination of inventory coupled with the use of sophisticated statistical accounting software achieve Near Real Time Accountancy. The BNFL Plutonium Inventory Monitoring System (PIMS, derived from a Los Alamos technique) employs an array of neutron detectors distributed throughout the plutonium facility and achieves a high

degree of accuracy for measurement of the distribution of Plutonium across an extended area, delivering accurate Near Real Time inventory information. In addition to information for BNFL's management of the facilities, a separate parallel system provides inventory information direct to the international Safeguards inspectors (3).
The SMP project was approved by the BNFL Board mid-1993 and the project will be complete within fiver years of approval.
For the reactor operator, handling and loading of MOX fuel from MDF has proved straightforward and, for fuel manufactured from surplus weapons plutonium, occupational dose implications, even during fuel inspection, will be well within acceptable limits without modification to reactor facilities.
TABLE I
TABLE II
VITRIFIED/IMMOBILIZED PRODUCT
Specifications for the vitrified HLW product from reprocessing plants (and similar material from the US Defence program) have been subject to significant national or international scrutiny and commercial facilities are in operation in France and the UK. Japan is also committed to recycle and a new (second) reprocessing plant and vitrification facilities are under construction. The agreed limiting specification for plutonium in vitrified product from the European plants is more than an order of magnitude below the levels under discussion for weapons Pu disposition.
Although there would not be any fundamental engineering problems in designing a vitrification facility for 1-4% (or even 7%) Pu in glass, there would be concerns at the long-term safety/environmental implications in terms of solubility, criticality, and proliferation resistance. Costs for a large vitrification facility would be of the same order of magnitude as a MOX facility but the product would have no "value" and disposal costs would be uncertain.
PLUTONIUM STORAGE
In the UK, the International Standards for Physical Protection* have been implemented. Plutonium has been stored at the Sellafield site for over 40 years (civil material for 30 years). BNFL's pro-active policy on plutonium management demands that plutonium stocks are held in custom designed and built stores, security, material control, safeguards and accountancy requirements. PuO_2 separated in THORP is stored in strong, leak-tight, triple-walled stainless steel cans with a thermal capacity of 125 W (nominally 7 kg Pu). The current store capacity is 7020 containers (45 mt Pu). Plutonium entering the store is independently verified for international safeguards purposes and monitored thereafter by safeguards inspectors through a Containment/Surveillance system. All operations within the storage area are controlled remotely to reduce operator access to the absolute minimum. In view of the chemical reactivity of plutonium metal, storage as oxide (or a similar ceramic) is preferred to storage of metal.
PLUTONIUM TRANSPORT
BNFL and its associate companies have been transporting plutonium safely in various forms (nitrate, power, fuel) by all modes for over 30 years. In the past 16 years, there have been a number of international shipments comprising 80 deliveries of which at least 90% were as PuO_2, with the balance being as fabricated plutonium in various forms. PuO_2 powder is transported in SAFKEG 2812C packages, the design of which has been tested beyond requirements, including surviving a drop of over 500 meters. There have been no security or safety incidents during this transportation program (4) BNFL has developed a larger package for plutonium transport (1680) to carry four THORP product containers.
A SYSTEMS APPROACH?
The essential stages for implementing a satisfactory plutonium management regime are:
dismantling and short-term storage which will reduce the materials to a form with no classified features.
conversion of the "declassified" plutonium into a form suitable for interim storage and capable of being put under an independent safeguards regime. BNFL has operated such a process from 1975 to 1985.
fabrication of nuclear fuel or vitrification/immobilization of the plutonium.
irradiation of the fuel or storage of the vitrified product until disposal.
For the vitrification/immobilization option:
development of specification for glass/alternative and assessment of technical

(physical properties, short and long-term safety), safeguards and (lifetime) cost implications of storage and eventual disposal of this additional glass waste-form at a Federal facility.

technical demonstration and establishing database to support safety assessments if the proposed specification is in any respect outside the currently validated parameters.

construction/modification of vitrification/immobilization and interim storage facilities.

It is unclear to what extent the cots would displace costs for waste disposal but more than likely that the bulk of the cost would be additional. In addition to the cost of design and licensing, construction, operation and decommissioning a vitrification and storage facility, the cost of glass disposal at Yucca Mountain is also a topic of debate.

For both options:

Any handling of plutonium material on this scale will lead to plutonium contaminated wastes. These waste will vary in chemical and physical characteristics as well as contamination levels. In the US, wastes arising from Defence programs are destined for WIPP. In the UK, a new facility is being built at Sellafield to handle TRU wastes arising from past and future plutonium facility operations. The properties of encapsulated TRU waste have been extensively studied by BNFL. A systematic evaluation of encapsulation alternatives (5) including glass, ceramics, cements, polymers and bitumen has shown the best process encapsulation of TRU waste to be cementation.

This work has demonstrated a long term stable product will be achieved with high Pu loading. The limit on Pu content is assessed as being defined by criticality requirements, not by product quality. Disposal studies have also shown the use of cement to have advantages. The high pH minimizes solubility and the matrix adsorbs Pu effectively further reducing solubility by orders of magnitude.

Consideration of these waste opens up, at least in principle, an alternative approach to the optimization of the plutonium disposition process, in that it is possible that the only waste form for direct disposal could be TRU waste, with the other product being MOX fuel.

Determination of the optimal solution requires a systems approach taking into account:

current form (and location) of the surplus plutonium.

material feed specification for MOX or immobilization.

waste arisings from processing (and waste acceptance criteria).

waste storage and disposal options/viability including any facilities already in operation or development.

time to implement each option.

consideration of political and public acceptance factors.

BNFL has competed an optimization process similar in many respects to the above for new facilities currently under construction or commissioning at Sellafield for wastes (including TRU wastes) arising from reprocessing metal oxide fuels (THORP), MOX fuel fabrication and decommissioning (including TRU waste). In the UK, wastes which exceed 4 Gbq/t (108 nCi/g)a or 12 gbq/t (325 Ci/g)bg and do not generate significant heat are classified as "Intermediate Level Waste" (ILW).

BNFL has established a comprehensive strategy for managing all such wastes arising from past, present and anticipated future operations at the Sellafield complex; cement (with differing formulations according to the wastes to be incorporated) is the chosen matrix for ILW. Four major new waste treatment facilities are newly operational or under construction. These waste will be disposed of in an underground repository.

A further factor which BNFL has considered in the context of product from the commercial reprocessing plant is the extent of possible contaminants in the feed materials for MOX fuel fabrication. For the commercial reprocessors and their customers, plutonium is treated as a resource to be used efficiently, whereas the US priority is expected to be "disposition" of the surplus plutonium as quickly as possible and a number of options for increasing Pu content in MOX fuel have been proposed. What is needed to determine the optimum strategy for MOX is a clear determination of the "spent fuel standard" and, since this standard is qualitative, possibly an equivalent DHLW standard for immobilized waste.

The surplus plutonium from the Russian and US weapons programs differs in some

respects from plutonium arising from commercial reprocessing of higher burnup civil fuels from both gas-graphite and water reactors and particular consideration needs to be given to acceptance criteria both for wastes and for product material (whether glass or MOX). Although there must be no question of compromising reactor safety margins, and the reactor operator must be concerned at any reduction in availability through, for example, increased outage frequency, the need for stringent criteria for neutron poisons in the fuel or clad, for example, may be relaxed if the objective is to consume plutonium rather than to generate electricity in a fuel-efficient way. The effect on a candidate reactor operator's fuel cycle costs will clearly be key to assessing his/her view of fuel specification but this is a contractual matter between the reactor operator and DOE as the (MOX) fuel supplier. There is an opportunity for a risk-based assessment of the current condition of the surplus weapons plutonium to identify options for medium term disposition and urgent steps to take to stabilize the material at least in a manner which will not prejudice future use or immobilization. This is necessarily an adjunct to the review of some of the facilities themselves and, ideally(!) a review of material and facility needs in terms of waste arising and optimum waste forms/treatment process should be undertaken.

IN FAVOR OF MOX

The final "product" from the process of MOX use in LWRs is irradiated fuel similar in all significant respects to irradiated uranium fuel, in terms of safety, safeguards and requirements for final disposal and can be treated in the same way as other routine irradiated fuel arisings.

The principle benefits of the MOX disposition option are:

- minimization of high-level waste volume.
- earliest disposition of the surplus fissile plutonium inventory.
- avoidance of significant long-term plutonium storage costs.
- increased confidence that the principal waste form (irradiated MOX fuel) can be disposed of in the repository under evaluation at Yucca Mountain, Nevada.

IN CONCLUSION

conversion of surplus military grade plutonium metal to PuO_2 has already been demonstrated.

it is safer to store plutonium as PuO_2 than as metal.

the highest levels of safeguards, security and safety in handling, storage and recycle of plutonium in commercial reactors are already proven and available in Europe.

consumption of surplus plutonium as MOX fuel in LWRs offers the most credible and economic option for reducing the plutonium inventory and the least technical risk is terms of final disposal and hence the best means to meet disarmament and non-proliferation objectives.

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38-2

UTILIZATION OF Cs137 TO GENERATE A RADIATION BARRIER FOR WEAPONS GRADE PLUTONIUM IMMOBILIZED IN BOROSILICATE GLASS CANISTERS

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ABSTRACT
One of the ways recommended by a recent National Academy of Sciences study to dispose of excess weapons-grade plutonium is to encapsulate the plutonium in a glass in combination with high-level radioactive wastes (HLW) to generate an intense radiation dose rate field. The objective is to render the plutonium as difficult to access as the plutonium contained in existing U.S. commercial spent light-water reactor (LWR) fuel until it can be disposed of in a permanent geological repository. A radiation dose rate from a sealed canister of 1000 rem/h (10 Sv/h) at 1 meter for at least 30 years after fabrication was assumed in this paper to be a radiation dose comparable to spent LWR fuel. This can be achieved by encapsulating the plutonium in a borosilicate glass with an adequate amount of a single fission product in the HLWs, namely radioactive Cs137. One hundred thousand curies of Cs137 will generate a dose rate of 1000 rem/h (10 Sv/h) at 1 meter for at least 30 years when imbedded into canisters of the size proposed for the Savannah River Site's vitrified high-level wastes . The United States has a current inventory of 54 MCi of Cs137 that has been separated from defense HLWs and is in sealed capsules. This single curie inventory is sufficient to spike 50 metric tons of excess weapons-grade plutonium if plutonium can be loaded at 5.5 wt% in glass, or 540 canisters. Additional Cs137 inventories exist in the United States' HLWs from past reprocessing operations, should additional curies be required. Using only one fission product, Cs137, rather than the multiple chemical elements and compounds in HLWs to generate a high radiation dose rate from a glass canister greatly simplifies the processing engineering requirements for encapsulating plutonium in a borosilicate glass.

INTRODUCTION AND BACKGROUND

The Clinton Administration has identified the disposition of excess plutonium and highly enriched uranium from the dismantlement of nuclear weapons as one of the most challenging problems for the U.S. Government to solve. An Interagency Working Group is to provide coordination among the multiple U.S. Government bureaus and departments involved. The U.S. Department of Energy (DOE) has a Special Fissile Materials Disposition Program Office to address the problem. Near-term activities for this office include identifying and characterizing alternative approaches to disposition of special fissile materials and completing documentation required by the National Environmental Policy Act for implementation of disposition alternatives. A recent National Academy of Sciences (NAS) study examined ways to dispose of excess plutonium and highly-enriched uranium. (1)

The three preferred disposal options in the NAS study are 1) transforming the plutonium into intensely radioactive spent fuel similar to that already produced by existing commercial reactors, 2) vitrification of the plutonium with radioactive high-level wastes (HLW) in the form of glass logs, and/or 3) direct disposal in deep boreholes.

The Lawrence Livermore National Laboratory (LLNL) has the lead responsibility for assessing immobilization of plutonium in glass or ceramic disposition forms, for assessing geological disposition, and in assisting the DOE in evaluating other disposition alternatives. This is being done using systems engineering to systematically define and evaluate the options and alternatives.

The NAS study introduced the term "spent fuel standard." Meeting a spent fuel standard means to render plutonium as inaccessible as the plutonium found in spent fuel from existing commercial reactors. The magnitude of the radiation field barrier from decay of radioactive fission products would be one crucial metric of a spent fuel standard. An important requirement for this radiation field barrier is a long life. Of the over 100 radionuclides in HLW, the most persistent gamma-ray emitter is Cs137, which has a 30-year half-life. Cesium-137 is also the dominant gamma emitter in aged spent-fuel from existing commercial light water reactors. The Cs137 gamma ray is penetrating and requires 7 cm of steel to reduce its intensity by an order of magnitude.

Further, cesium can be incorporated in a borosilicate glass matrix. Using a single chemical element as the radioactive source to generate a radiation field barrier significantly simplifies the process flowsheets for vitrifying plutonium in a glass matrix, when compared to using multiple fission products of varying concentrations as would be required for incorporation of HLW. A substantial inventory of relatively pure Cs137 was separated from the Hanford HLW and is currently stored in sealed capsules at various sites as cesium chloride. If sufficient inventories of these Cs137 capsules exist, then further extraction is not required from existing HLWs.

Cesium capsules can be readily transported to any site selected for the vitrification facility.

ASSESSMENT, ASSUMPTIONS, CHARACTERISTICS, INVENTORY, AND CALCULATIONS

This paper describes the feasibility assessment and calculational approach in the following subsections. These subsections cover 1) assessment objectives, 2) key assumptions, constraints and limitations, 3) radiological characteristics of Cs137, 4) current U.S. inventories of separated Cs137, and 5) a set of parametric calculations and sensitivity analyses making up the feasibility assessment.

Assessment Objectives

Technical feasibility is assessed for using Cs137 as the radiation barrier in canisters of plutonium encapsulated in glass. This assessment is based on the radioactive decay properties of Cs137, the current inventory of Cs137 in U.S. defense wastes, the compatibility of Cs137 with the envisioned glass vitrification process, and the expected time frame for completing disposition.

Assumptions, Constraints, and Limitations

The inventory of excess weapons-grade plutonium requiring encapsulation in glass is assumed to be 50 metric tons. Canister dimensions used for encapsulating high-level wastes at the Savannah River Site (SRS) are assumed. (2) The SRS canisters are constructed of 304L stainless steel having a 0.375-cm (3/8-in.) wall thickness, a 61-cm (24-in.) o.d., and an overall height of 3 m. Canisters are partially filled to contain some 1680 kg of glass. Glass density is taken as 2.73 g/cm3. Using these numbers, the glass radius would be 29.5 cm, and it would have an effective height of 2.2 m.

Radiological Characteristics of Cs137

Cesium-137 has a 30-year half-life. It beta-decays to Ba137 (5%) and Ba137m (95%). (3) The Ba137m then decays with a half-life of 153 seconds through an isometric transition to the stable form, Ba137. About 90% of the Ba137m decays produce the penetrating 662-keV gamma-ray. The gamma yield from metastable barium equals 85% (90% of the 95% yield) of the Cs137 activity, which is 87 Ci/g. The 30-year half-life should be adequate for the interim storage period that will be needed before permanent geologic disposal of the glass canisters can be implemented.

Current Cs137 Inventories

Most of the current U.S. inventory (some 1109 curies of all radionuclides in 397,000 m3 of HLW, as of December 1990) is the result of past DOE defense activities and is stored at the SRS, Hanford (HAN) or at the Idaho Nuclear Engineering Laboratory (INEL). A small amount (2,200 m3 containing of 3107 Ci) of commercial HLW was generated during 1966-72 commercial reprocessing operations at the West Valley Demonstation Plant (WVDP). Details of these inventories are maintained in a U.S. integrated database. (4,5) Except where noted, the inventory data and descriptions in this section are taken directly from those references.

Approximately 131,700 m3 of alkaline HLW has accumulated at the SRS during the past three to four decades. This HLW is being stored in underground, high-integrity, double-walled, carbon-steel tanks. The current inventories include alkaline liquids (61,300 m3), sludges (14,800 m3), salt cakes (55,500 m3), and precipitates (125 m3) that were generated primarily by the PUREX reprocessing of nuclear fuels and targets from production reactors. All the SRS wastes totaled 5.6108 curies (December 1990).

The 12,000 m3 of HLW stored at INEL consists of 8,500 m3 of liquid waste and 3,500 m3 of solid calcine. Liquid HLW was generated at the Idaho Chemical Processing Plant primarily by the reprocessing of spent fuel from naval propulsion reactors and reactor testing programs. This acidic liquid waste is stored in underground stainless-steel tanks that are housed in concrete vaults. Most waste was then converted to a calcine solid and store space retrievable stainless-steel bins that are housed in underground reinforced-concrete vaults. The INEL wastes total 6.3107 curies (December 1990).

Commercial reprocessing at the Nuclear Fuel Services plant was terminated in 1972, and no additional HLW has been generated since that time. As of December 1990, the 1,231 m3 of HLW stored at WVDP consists of 1,136 m3 of alkaline waste (1,090 m3 of liquid plus 46 m3 of sludge), 50 m3 of acidic waste, and 45 m3 of an inorganic ion-exchange material (a zeolite) loaded with radioactive cesium (Cs134, Cs135, and Cs137). The alkaline waste was generated by reprocessing commercial and Hanford N-Reactor spent fuels. As generated, the waste was acidic; treatment with excess sodium hydroxide resulted in the formation of an alkaline sludge. The small amount of acidic waste now in storage was generated by reprocessing a batch of

thorium-uranium fuel from the Indian Point-1 Reactor. Storage of the alkaline waste is in an underground carbon-steel tank, while the acidic waste is stored in an underground stainless-steel tank. The total curies at the West Valley site is 2.7107 (December 1990).

The 253,600 m3 of alkaline HLW stored at HAN are categorized as liquids (26,400 m3), sludges (46,000 m3), and salt cakes (93,000 m3) that are stored in underground single-shell tanks and as slurries (88,200 m3) that are stored in underground double-shell tanks. This waste, which has been accumulating since 1944, was generated by the reprocessing of production reactor fuel for the recovery of plutonium, uranium, and neptunium for defense and other national programs. In all, these HAN wastes contain a total of 3.9108 curies (December 1990).

Most of the high-heat-emitting isotopes (Sr90 and Cs137, plus their separable daughters) were removed from the old waste at HAN, converted to solids (strontium fluoride and cesium chloride), placed in double-walled capsules, and stored in a water basin. (4,6,7)

According to David Peeler of the Pacific Northwest Laboratories, some 1577 cesium-containing capsules were fabricated at the Hanford Reservation (7). Of those 1577 capsules, some 961 are stored at HAN, 371 are stored at other locations, and 245 have been cut or destroyed. The 961 at HAN and 371 at offsite locations contained an average of 4104 curies per capsule, as of April 1994. The 961 HAN capsules contain a total of 39 MCi (megacuries), and the 371 offsite capsules contain 15 MCi. In this paper, we assume that the full 54 MCi (39+15) are available for use in plutonium disposition.

Casks have been routinely used to ship the capsules to various sites for various radiologic applications. The encapsulated cesium is in the form of cesium chloride, and it contains some 2-6 wt% of impurities, which are primarily sodium, aluminum, magnesium, potassium, nickel, iron, chromium, and silicon (6).

Parametric Calculations and Sensitivity Analyses

A series of radiation shielding and other parametric studies for Cs137 in borosilicate glass canisters was performed using the commercial MicroShield point-kernel code. (8) The radioactive cesium was assumed to be uniformly distributed throughout the borosilicate glass. In turn, the glass was assumed to be inside SRS HLW canisters (0.3-m-radius by 2.2-m-high glass cylinder with a specific gravity of 2.7, surrounded by a 0.95-cm-thick steel wall). The dose point was taken to be midway down the cylinder length and 1 meter out from the steel shell. These calculations show that 1105 curies of Cs137 are required per canister to generate a radiation field of 1000 rem/h (10 Sv/h) at 1 meter some 30 years after the glass canister is filled. The 1105 curies are generated by 1.2 kg of Cs137. Within the point-kernel approximation, external gamma radiation is linear in Cs137 content, and this is illustrated in Fig. 1. Linearity means that 0.6 kg of Cs137 produces half the radiation dose of that for 1.2 kg. Figure 2 shows how the dose rate decreases with increasing distance from the canister surface. Figure 3 shows how the radiation field varies with position along the length of the glass cylinder.

The number of glass canisters must be known, if the 54 MCi of separated and encapsulated Cs137 are to be sufficient to generate a radiation barrier for 50 metric tons of plutonium. The maximum plutonium content in borosilicate glass has not been determined. Figure 3 shows a graph of the number of containers required to encapsulate 50 metric tons of plutonium for various plutonium loadings. For this study, plutonium loadings in glass logs were assumed to range between 0.1 and 10 wt%. From Fig. 4, a 0.1 wt% loading requires 30,000 canisters and a 10 wt% loading requires 300 canisters. Canister minimization is highly desirable to reduce wastes and costs associated with handling operations, material consumption, transportation, and geologic disposal. The highest tolerable loading will likely be established by nuclear criticality rather than by chemical or mechanical requirements on the glass. From the MicroShield calculations described above, 1105 curies of Cs137 are required for a single canister to generate a 1,000 rem/h dose rate at 1 meter after 30 years. Since the total inventory is 54 MCi, 540 glass canisters can be fabricated with 1105 Ci each, to meet this radiation specification. Each of the 540 glass canisters will contain about 93 kg of the 50 metric tons of excess plutonium stock. Since the glass in a canister weighs 1680 kg, this implies a plutonium-in-glass loading of 5.5 wt%. This works out to a cesium-to-plutonium weight percentage of 1.3% for the required 1.2 kg of Cs137.

A recent paper by Westinghouse-Hanford personnel proposes combining cesium-137 with

plutonium oxide in containers several centimeters in diameter and several centimeters high. (9) Their calculations show that a 10 wt% cesium-to-plutonium content is required to achieve 1,210 rem/h at 20 years. Exponentially decaying this value to 30 years yields 960 rem/h.

We modeled the hypothetical addition of cesium to plutonium oxide inside of typical DOE plutonium shipping containers. These containers have an outside diameter of 15.2 cm (6 in.) and a steel wall thickness of 0.95 cm (3/8 in.) Assuming that the 5 kg PuO_2 mass has a unity specific-gravity leads to an effective powder height of 37 cm. Modeling with MicroShield shows that 2.3104 Ci of Cs137 would be required for a 1,000 rem/h dose rate at 1 m for the DOE plutonium shipping containers after 30 years. Based on 4.5 kg of plutonium, this calculation shows a cesium-to-plutonium weight percentage of 6%, which is in approximate accord with Ref. 9. The important point here is that the ratio of required cesium to plutonium is higher for small shipping containers as compared to large glass-filled canisters (6 wt% versus 1.3 wt%). This means that the existing stock of separated Cs137 will go further in the larger glass-filled canisters. In addition, heavy glass-filled canisters are more desirable from a safeguards and security perspective.

CONCLUSIONS

This study establishes the technical feasibility of using the existing U. S. stock of separated Cs137 in cesium chloride capsules as the gamma source to provide a radiation barrier for excess weapons grade plutonium encapsulated in glass. The alternative to using a single fission product, Cs137, is to feed a highly variable HLW composition stream containing a multitude of fission products to a melter. Direct use of HLW could introduce some uncertainties in the final glass composition that could effect geologic performance. Further, the high-level wastes at HAN, SRS, and WVDP do not contain concentrations of Cs137 large enough to directly generate a 1000 rem/h radiation barrier. To increase the radiation barrier level, complex chemical recovery and cesium concentration process operations on the HLWs would have to be designed and performed prior to any glass encapsulation operations.

The use of the single fission product Cs137 for generating a gamma-radiation field offers significant advantages in developing the engineering solutions for vitrification of excess plutonium. For example since only two chemical compounds of plutonium and cesium need to be added to a borosilicate glass, the glass frit composition can be more readily tailored to make a vitrified product with optimum geologic performance. These process feed stream simplifications allows the engineering of simplified vitrification process equipment due to only three streams (Pu, Cs and glass frit) feeding a melter. The fact that the cesium is already separated also means that: (1) the Cs137 can be transported to any site capable of hosting a vitrification facility, including one that is not constrained to be located at SRS or Hanford or INEL and, (2) a sufficient inventory is available so that cesium recovery from its current dilute forms at the HLW defense wastes is not required for any vitrification process.

In addition, the cesium capsules are regarded as an unacceptable waste form issue for direct geologic disposal since cesium chloride is water soluble and the cesium isotope is fairly long lived. (10) The utilization and incorporation of Cs137 into a borosilicate glass matrix generates an acceptable waste form for geologic disposal and resolves a current DOE environmental waste management issue, should encapsulation of excess weapons-grade plutonium in glass be selected by DOE as a disposition alternative.

ACKNOWLEDGMENTS

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38-3
VITRIFICATION FOR DISPOSITION OF EXCESS PLUTONIUM

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ABSTRACT

As a result of nuclear disarmament activities, many thousands of nuclear weapons are being retired in the U.S. and Russia, producing a surplus of about 50 MT of weapons grade plutonium (Pu) in each country. In addition, the Department of Energy (DOE) has more than 20 MT of Pu scrap, residue, etc., and Russia is also believed to have at least as much of this type of material. The entire surplus Pu inventories in the U.S. and Russia present a clear and immediate danger to national and international security. It is important that a solution be found to secure and manage this material effectively and that such an effort be implemented as quickly as possible. One option under consideration is vitrification of Pu into a relatively safe, durable, accountable, proliferation-resistant form. As a result of decades of experience within the DOE community involving vitrification of a variety of hazardous and radioactive wastes, this existing technology can now be expanded to include immobilization of large amounts of Pu. This technology can then be implemented rapidly using the many existing resources currently available. A strategy to vitrify many different types of Pu will be discussed. In this strategy, the arsenal of vitrification tools, procedures and techniques already developed throughout the waste management community can be used in a staged Pu vitrification effort. This approach uses the flexible vitrification technology already available and can even be made portable so that it may be brought to the source and ultimately, used to produce a common, borosilicate glass form for the vitrified Pu. The final composition of this product can be made similar to nationally and internationally accepted HLW glasses.

BACKGROUND

The Committee on International Security and Arms Control (CISAC) of the National Academy of Sciences (NAS) was commissioned by General Scowcroft, National Security Advisor to President Bush, to conduct a study on disposition alternatives for management of excess plutonium resulting from disarmament activities. This charter was later confirmed by the Clinton Administration. After receiving input from many sources and evaluating the options, NAS recently issued the results of this study in a report entitled "Management and Disposition of Excess Pu" (1) in January of 1994. The treatise covered many important aspects of the Pu disposition question and with respect to long-term management of the excess Pu stated the following:

"Then two most promising alternatives for achieving these aims (long-term Pu disposition) are:

  fabrication and use as fuel, without reprocessing, in existing or modified nuclear reactors; or

  vitrification in combination with high-level radioactive waste."

A third option, which has not been studied in as much detail, burial of Pu in deep boreholes, was also mentioned as a possible consideration.

As a result of the NAS study, the vitrification option was elevated to the same level of importance as a reactor option and listed as one of two leading preferences

for ultimate disposition of weapons grade Pu.
The Savannah River Site (SRS) has been involved in vitrification of high level waste for several decades and also associated with vitrification of a variety of other types of radioactive and non-radioactive materials (2,3). As a result of over 20 years of vitrification experience and about 40 years of Pu handling and processing experience, the site has been requested to provide input into this subject by a variety of groups and agencies, including the U.S. Department of Energy. These efforts are directed at addressing a variety of Pu disposition alternatives (4-7), including management of weapons grade Pu and also Pu scrap, residues, etc. that currently exist throughout the DOE complex. An early summary of vitrification options is documented elsewhere (8).

WHY VITRIFICATION

There are many potential advantages associated with the vitrification option for long-term management of plutonium. These include the following:

Immediacy of Implementation

Many experts in the field believe it is important to act quickly to immobilize Pu for security, safeguards, safety and environmental reasons. Because of the advanced state of the art of vitrification and as a result of the existing capabilities and experience in this area, there is no other Pu option that can be implemented as rapidly.

Flexibility

The vitrification option provides a common technology for treatment of almost all forms of Pu. This includes not only weapons grade Pu, but also significant quantities of more complex Pu scrap and residue compositions, currently existing within the DOE community and posing additional problems.

Technology Availability

As a result of the High Level Waste program and associated waste management efforts, there have been decades of research on development of techniques and procedures for vitrifying radioactive and hazardous components, equipment and facilities for these tasks, and in specifying waste form qualifications to assure product quality. These efforts are directly applicable to Pu vitrification and could be piggy-backed upon. A very extensive and capable vitrification infrastructure exists containing experienced and dedicated experts throughout federal and national laboratories, academia, and industry throughout the United States as well as in other countries.

Waste Glass Performance

HLW glasses have been demonstrated to have excellent chemical durability, mechanical integrity, radiation and thermal stability. Chemical durability is considered to be the most important technical performance property of a waste glass form. It is important to note that actinide bearing glasses exhibit excellent chemical durability. The leaching of actinides is generally 10-100x better (lower leach rates) than modifiers or alkali cations contained in HLW waste glass systems.

Processing Considerations

The ability to vitrify radioactive materials is not only well developed, but also well demonstrated. In the case of HLW, actual production facilities are in operation world-wide. These include the French process in Marcoule and also La Hague, the German vitrification operation in Mol Belgium, the Sellafield facility in England and others. Construction of the first HLW vitrification facility in the United States, the Defense Waste Processing Facility (DWPF) at SRS, has recently been completed and is scheduled to be in production in about one year. Vitrification of Pu represents an extension of this already available vitrification technology and its many components.

Waste Minimization

Due to the type of equipment and buildings needed to process radioactive materials, and as a result of the vitrification facilities and equipment used to treat other types of radioactive wastes, existing contaminated and non-contaminated facilities could be modified to perform Pu vitrification. These facilities include buildings and equipment designed to receive Pu, store the material, process the Pu, and vitrify it into acceptable products. These facilities currently exist within the DOE complex and although they would require various degrees of modification, their existence would eliminate the need to build and later D&D additional buildings.

Ability to Immobilize Pu in Glass

Immobilization of Pu into borosilicate glass has already been demonstrated in HLW programs. This early work involved immobilization of 7 wt. % of plutonium oxide into

a glass matrix. Higher Pu loading is probable. Hence, there is no question whether Pu can be vitrified- the maximum amount immobilized remains to be defined and will most likely not be determined by solubility limits but by criticality considerations.

Inherent Criticality Control

Due to the composition of borosilicate glasses, boron as well as lithium, which are normally present in HLW forms, can act as poisons to assist in criticality control. This is especially important during handling and processing operations. Other poisons such as gadolinium or erbium and rare-earths could be added to further this effect, which would be especially relevant for long-term repository storage scenarios due to their insolubility. Additional work would be necessary to better assess the effects of potential poisons for criticality control in some of the final disposal options under consideration and for increasing the difficulty in reclaiming Pu from the glass matrix. Criticality control represents the most important consideration in all stages of any disposal option.

Acceptability/ Waste Form Qualifications

The only waste form which has achieved a degree of national and international acceptance for immobilizing HLW is borosilicate glass. HLW glasses already contain Pu, although in very small amounts. It took approximately ten years and thirty million dollars to qualify the SRS HLW waste glass composition. This important and necessary effort could be piggy-backed upon for the Pu vitrification option.

Proliferation Resistance

There are many ways in which Pu can be immobilized into glass to produce durable, safe, proliferation-resistant forms. These options depend on the degree of proliferation resistance required and are directly proportional to the cost and complexity of the operation. For example, simply immobilizing Pu into glass can be achieved rapidly and most easily and provides the highest degree of flexibility. The Pu-glass product produced would be more proliferation resistant that Pu in its weapons form, but could be reclaimed fairly easily by those reasonably familiar with this field. The degree of proliferation resistance could be increased significantly, however, by either initially mixing the Pu directly or by re-melting Pu-only glass, with fission products or existing HLW. The radiation field associated with the radioactive additives would considerably increase the difficulty in obtaining or handling this material and in subsequent transportation and reprocessing operations to reclaim Pu. Proliferation resistance can be further enhanced by the potential size and weight of the product and most importantly, by the safeguards that would be necessary for any undertaking of this type.

OVERVIEW OF A POTENTIAL Pu STRATEGY

Vitrification provides an important option to immobilize and dispose of not only weapons grade Pu, but also many other forms of Pu of concern within weapons producing countries. The major steps involved in a Pu vitrification strategy include the following

Pu Handling and Glass Preparation

I. Preparation (receipt, pre-storage, and pre-treatment)
II. Conversion to melter feed
III. Vitrification into intermediate or final Pu-glass products

Interim and Final Product Storage

IV. Interim product storage
V. Final product disposition

A simplified flowsheet summarizing this overall vitrification strategy is depicted in Fig. 1 followed by a brief description of each of the important phases.

Fig. 1.

Preparation

The vitrification strategy applies to Pu in metal form currently contained in weapons as well as Pu in other forms such as oxides, buttons, scrap, solutions, ash, salts, residues, etc. The other forms of Pu are contained at DOE sites including Rocky Flats, Los Alamos National Laboratory, Hanford, Argonne National Laboratory, Idaho National Engineering Laboratory, and the Savannah River Site. Preparation of these materials will involve receipt, disassembly/ separations in some cases, assaying, interim storage and pre-treatment prior to immobilization.

Conversion

There are two major options that can be used for conversion of Pu from a metallic form to a suitable melter feed. These include a) oxidation- burning it to produce

plutonium oxide powder and b) dissolution- dissolving it to yield a plutonium acid solution. Additional preparation involving Pu scrap and residues could be performed, depending on composition and subsequent melting characteristics. Vitrification can handle either liquid or oxide feeds.

Vitrification

A vast array of electric melting techniques have been developed for vitrification of radioactive wastes over the years and include indirect heating, joule heating, plasma and microwave vitrification. Many of these techniques and their tested glass melters could be used to vitrify Pu. The proposed strategy considers two main options for vitrification:

In Option A, shown in Fig. 1, the treated Pu is melted directly with fission products such as Cs-137 or HLW to produce a highly radioactive Pu glass product. This could be accomplished in several ways and provides the highest degree of proliferation resistance possible. Because of the highly radioactive wastes to be mixed with the Pu, specially contained and shielded facilities and equipment would be necessary.

In Option B, an interim Pu-only glass is first produced as an interim product. This option has the advantages of being able to use a wide range of site-specific vitrification technologies already located at sites containing the waste, can be performed by the use of relatively simple portable equipment and gloveboxes, and can be accomplished most rapidly and easily due to the relatively low levels of radioactivity involved. While the resulting Pu-only glass does not produce the most proliferation resistant glass product, it does provide the most flexibility because it can be transported to other locations that have more highly radioactive wastes, such as HLW or selected fission products, and later be re-melted with this waste to produce more proliferation resistant forms. Since specially contained facilities and equipment are necessary for this part of the operation, current facilities with other missions could be used by meshing into existing programs and schedules to produce duel missions and optimum use of existing resources.

An important concept in this strategy is that a common final waste glass form can be produced with a similar composition to HLW glass (9), which has undergone a very time consuming and expensive process to be certified and made acceptable. A resulting Pu- HLW glass compositions would be anticipated to be as good, if not better, than the already acceptable HLW glass compositions.

Interim Product Storage

A need to store Pu bearing glasses temporarily is important to the vitrification option. The design of the interim storage facility would depend on factors such as the composition and radioactive content of the products and the intended duration of storage before ultimate disposal. Among the most important considerations for this facility are worker radiation exposure, public and environmental protection from radiologic hazards, and material safeguards and accountability.

Final Product Disposition

The reference concept for ultimate disposition of HLW glass and spent fuel is to dispose of this material by deep burial in carefully selected geologic repositories. This is also being considered for Pu bearing glasses. A very significant challenge for any repository scenario is to demonstrate safe and effective performance of products out to very long time periods (1000 to 10,000 years, and longer). This challenge would be expected to be even more formidable for any immobilization alternative containing large amounts of Pu, due to criticality considerations. The waste form and waste package will be designed to prevent criticality from occurring but this must be demonstrated to a very high degree in a very complex environment. While this would not be expected to be as significant an undertaking for a "Retrievable Surface Storage Facility or RSSF", it would be expected to take much more effort to demonstrate in a geologic repository.

The final form and composition of Pu glass would be tailored for technical and political considerations, involving possible reuse or non-reuse by the weapons producer and for optimizing proliferation resistance, especially towards potential non-friendly nations.

VITRIFICATION OF Pu USING EXISTING RESOURCES

As discussed earlier, there exists an infrastructure knowledgeable on vitrification of radioactive and hazardous wastes that could be used for vitrification of Pu. Along with the developed technology, experience and experts, are existing buildings, equipment and supporting hardware and software for such an effort. While buildings

and equipment exist throughout the DOE complex, the following discussion will emphasize vitrification options and facilities associated with the Savannah River Site (SRS). These facilities are shown along with the accompanying main vitrification steps discussed earlier in Fig. 2.
Fig. 2.
At least six separate vitrification options have been identified using SRS facilities. The processing options involve three potential Pu products; a Pu-only glass (interim product), a Pu/Cs-137 glass (final product) or a Pu/HLW glass (final product). Among the main existing facilities which could be modified and are critical to these vitrification options are the following:
Plutonium Storage Facility (PSF)
The mission of the PSF was to receive Pu materials from offsite that would later be processed. The building contains a fully safequarded vault for automated transport and stacking, gloveboxes for opening drums and removing and inspecting contents, and instruments for non-destructive assay and computer accountability. The facility also contains a delivery systems for later processing of materials in an adjacent facility, the New Special Recovery (NSR).
New Special Recovery (NSR)
The mission of the NSR was to process plutonium from throughout the DOE complex. This includes dissolving Pu scrap, oxide or metal from various sources to produce purified Pu metal buttons or oxide powder. The facility contains state of the art glovebox trains for feed preparation, waste handling, dissolution processes, samples and analysis, along with a supporting remote control room. This facility also contains extra room which could be outfitted with glass melters to pursue one of the vitrification options under consideration.
F Canyon (Including the Multi-Purpose Processing Facility or MPPF)
221-F Canyon was the world's first PUREX production plant used to dissolve natural and depleted uranium targets and to recover the uranium and plutonium. It is a large, heavily shielded facility operated by remote means. The building has the potential of being refitted with a melter and used for vitrification in support of another vitrification option of immobilizing Pu directly with HLW or selected fission products. The MPPF is located within the facility and contains eight modules that are now being used for vitrification of Cm and Am using a bushing melter, and will also be used for vitrification and subsequent clean up of Pu scrap on site.
Defense Waste Processing Facility (DWPF)
The mission of the DWPF is to immobilize the 34 million gallons of HLW currently being stored at SRS into borosilicate glass. This represents the first waste vitrification facility constructed in the United States and is scheduled to be in production in about one year. All of the operations necessary to process, vitrify, containerize, seal and decontaminate HLW glass units are present, including supporting capabilities to control the process and verify product quality. This facility and its equipment could be modified to vitrify larger quantities of Pu than presently contained within HLW.
Important for DWPF options is not to adversely effect the current, important mission of the facility. This could most easily be achieved by the staged approach mentioned earlier. There are two primary ways that Pu vitrification could be conducted using the DWPF. First, during the scheduled melter change out of the DWPF, the facility and equipment could be modified and Pu bearing feed introduced either directly as a Pu feed or as a Pu bearing glass frit, to produce Pu-HLW glass forms in 304L stainless steel containers. These units would be similar to HLW glasses in canisters, 2-ft. in diameter and almost 10-ft. high. An important potential advantage of this option is that waste loading to the glass could be increased by 1-2%, which would still result in a highly durable form which could immobilize 50 MT of Pu as part of the HLW program, and without increasing the number of waste canisters produced.
The second main option involves placing the interim product, Pu-bearing glasses, into HLW canisters and pouring HLW over them in the current HLW vitrification campaign. While this option does not produce as integral a product as the Pu-HLW glass, it does produce a form with similar radiation characteristics (proliferation resistance) and is considerably easier and less expensive to implement into the current facility and schedule.
SUMMARY
Vitrification is a technically viable option to immobilize and manage Pu resulting

from disarmament activities as well as a wide range of existing Pu scrap and residue compositions. A vitrification infrastructure exists from waste management programs. This includes expertise and experience, personnel, buildings, equipment and supporting capabilities, which could be used to implement vitrification of Pu in a time expedient and cost effective manner.

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COMPOSITIONS AND DURABILITIES OF GLASSES FOR IMMOBILIZATION OF PLUTONIUM AND URANIUM

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ABSTRACT

Investigations have been performed to determine the suitability of glass as a host for surplus fissile material removed from nuclear weapons. The U.S. Department of Energy - Office of Technology Development has sponsored research at the Savannah River Site to develop durable glass compositions that are compatible with high concentrations of plutonium and uranium. These investigations are also being performed to provide baseline actinide glass durability and processing data.
Two glass forming systems are being evaluated. One of the systems is a commercial borosilicate glass and the other an iron phosphate glass. Both glass systems have a very high degree of compatibility with actinide oxides and are considerably more durable than conventional high-level waste glasses. The iron phosphate glass has a melting temperature in the 1100C range and has the higher uranium and plutonium solubility. The borosilicate has a melting temperature in the 1425C range and is the more durable (on the order of fused silica) glass.

INTRODUCTION

As the nuclear weapon arsenals of the United States and the Former Soviet Union are reduced, metric tonnage quantities of fissile material must be dispositioned (1). One of the potential disposition options for U.S. weapons material is vitrification into a plutonium or uranium glass product (2). The U.S. Department of Energy - Office of Technology Development has sponsored a program at the Savanna River Technology Center (SRTC) to develop suitable glass formations for the long term safe

storage of uranium and plutonium as well as americium, neptunium and curium. These "actinide glasses" are the focus of this paper (3).
Three of the most important questions which must be answered for any actinide glass product are 1) what is the maximum actinide oxide loading, 2) what is the chemical durability, and 3) how can it be produced. The experiments described below were performed to provide a preliminary technical baseline on actinide solubility in glass and glass durability. Glass processing characteristics and limits will be discussed in subsequent publications.
This paper discusses the chemical composition and durability of two types of actinide glasses under development. One of the glasses is a commercial borosilicate composition developed in the 1930's for use as an optical glass (4). This glass, referred to as the Lffler glass, was selected for study due to the very high (55 weight percent) lanthanide oxide content (5). Lanthanides are commonly used as actinide surrogates (6). There was, therefore, a high degree of confidence that this glass would be chemically compatible with high concentrations of actinides. The other glass is an iron phosphate. This glass was selected for study due to the combination of low melting point and high durability (7,8). In the initial studies thorium and uranium were used as the actinides. Because of the low radioactivity of these elements, the glasses could be prepared and tested on the bench top.
The specific objects of the initial study were to determine:
1. maximum weight percent loading of thorium and uranium oxide in the borosilicate and phosphate glass,
2. chemical durability of the glasses as a function of actinide loading.
Data obtained were used to determine initial frit compositions and processing parameters for the plutonium melts. The first plutonium glasses have been prepared and are being tested in the shielded cell and glovebox facilities at SRTC.
EXPERIMENTAL
Glass Fabrication
The initial glass samples fabricated were made from reagent chemicals. The batches were made in high-form alumina crucibles. Water was added to all reagent chemical batches to ensure proper mixing. The batch was allowed to dry overnight at 90C prior to firing. The crucibles were then placed in a furnace and ramped to temperature no faster than 8C per minute. The appropriate melt temperature was held for a period sufficient to ensure complete melting. The glass was then cast into a graphite mold and annealed. Glass frit was produced in the same manner except the melt was cast into water and then crushed into powder.
Glass Durability Testing
Relative durability of the glasses discussed above was measured using the ASTM C-1285 standard nuclear waste glass durability test method, commonly referred to as the Product Consistency Test, or PCT (9). The PCT protocol calls for crushed glass powder to be reacted with ASTM-I deionized water in a closed vessel (10). The test conditions included:
  10 milliliters ASTM-1 H2O per gram 100-200 mesh glass powder,
  90C test temperature,
  7 day test duration.
PCT leachates were analyzed for soluble glass constituents by Inductively Coupled Plasma - Emission Spectroscopy (ICP-ES). U and Th concentrations in the leachates were determined by Inductively Coupled Plasma - Mass Spectroscopy (ICP-MS). Leachate solution pH was measured by a glass bulb electrode.
Relative glass durability was calculated from the PCT data. Normalized Loss (NL[1]), a common expression of relative glass durability was calculated for each glass tested. NL[1] is a function of the concentration of a soluble glass constituent cation in the PCT leachate solution and the concentration of the cation in the glass. NL[1] is expressed in the units grams of glass dissolved per liter of leaching solution. The following relationship is used to calculate NL[1]:
Eq.
where
c[i] = concentration of "i" in solution (mg/L).
f[i] = weight fraction of "i" in glass.
X-Ray Diffraction
X-Ray Diffraction was performed on all samples to identify any crystalline phases. Scanning Electron Microscopy was also performed as a complementary technique if crystalline phases were detected. None of the glass samples discussed above had any

discernible crystalline character.
RESULTS
Glass Production
Minor adjustments to the base Lffler glass composition were made in order to lower the melting point and liquidus temperature and raise the melt viscosity (11,12). This was accomplished by reducing the total lanthanide oxide content and increasing the alumina, lead oxide, and silica fractions of the glass composition. The compositions of one series of glasses, which bracket the expected process parameters for actinide glass production, are given in Table I. All glasses shown in Table I, with the exception of Lan-22, were melted at 1425C. Lan-22 was melted at 1460C.
TABLE I
The next series of glasses consisted of uranium and thorium oxide added to the target glass composition shown in Table I. Uranium and thorium oxide were substituted on a 1:1 molar basis for cerium and neodymium oxide. Melts containing 1, 5 and 9 mole percent (9 mole percent @ weight percent) uranium oxide (calculation basis UO2) and thoria (ThO2) were successfully processed at 1425C.
The base composition iron phosphate glass, as shown in Table II has a melting point @ 1100C. No composition adjustment was necessary to the base glass prior to adding uranium or thorium oxide. Thorium and uranium oxide were substituted on a 1:1 molar basis for Fe2O3. Melts containing 10, 14 and 17 mole percent (17 moles percent @ 30 weight percent) actinide oxide were successfully processed at 1150C.
Iron phosphate glass frits were made for plutonium processing. Frit and PuO2 were mixed together and then heated to 1100C (5 hours at temperature). Melts containing 10 and 17 mole percent PuO2 were successfully processed. A frit composition (denoted Frit + Ba) developed for the plutonium glass melts is shown in Table II.
TABLE II
Relative Glass Durability
The Lffler glasses are quite resistant to aqueous attack. The PCT experiments on the non-radioactive glasses (Table I) show very uniform results. All six Lffler glasses tested were more durable than two common durable glasses - Vycor and fused silica. This is graphically represented in Fig. 1. Table III lists the PCT leachate Si concentration for the Lffler glasses, fused silica and Vycor. Normalized Release, NR[Si] and the Glass ID's used in Fig. 1 are also shown in Table III.
Thorium and uranium bearing Lffler glasses are also quite resistant to aqueous attack. The PCT results show little or no difference between the durability of actinide bearing and non-radioactive compositions. Also, uranium and thorium oxide appear to have identical effects on glass durability. There is no difference between the durability of glasses with 1 percent and 5 percent actinide oxide content. Glasses with 9 mole percent actinide oxide were slightly less durable. These results are displayed in Fig 2. It should be noted the relative durability of the Lffler glasses is approximately 3 orders of magnitude better than the standard for high-level waste glass (13,14).
The iron phosphate glasses are also quite resistant to aqueous attack. PCT experimental results indicate the iron phosphate glass to be approximately 2 order of magnitude more durable than the standard for high-level waste glass. Thorium and uranium bearing glasses have similar PCT response up to the 20 weight percent actinide oxide concentration. Both thorium and uranium have a negative effect on the durability of the iron phosphate glass beyond 20 weight percent oxide loading. Beyond twenty weight percent, uranium glasses are appreciably less durable than thorium glasses. These data are graphically represented in Fig. 3. The leachate solutions are also significantly more acidic as the uranium oxide content increases beyond 20 weight percent. These data are graphically represented in Fig. 4.
TABLE III
Fig. 1.
Fig. 2.
Fig. 3.
Fig. 4.
DISCUSSION
The Lffler and iron-phosphate are glass compositions with high concentrations of actinides. Also, both glasses are extremely resistant to aqueous corrosion. Interestingly, the Lffler and iron phosphate glass are completely different chemically. The base compositions share no common oxides. Lffler glasses are processed at temperatures consistent with commercial borosilicate glasses. Iron

phosphates are processed at temperatures 300C lower, in the range of high-level waste glasses. It is felt, therefore, that these glasses may be used for different actinide vitrification missions - in the manner of complimentary products. The Lffler glass has been selected as the optimum glass formulation for vitrification of SRS americium and curium (11). This glass is compatible with existing commercial melters and appears well suited for mass production. It is completely compatible with lanthanide neutron poisons (Gd, Sm, Eu), and certainly, boron. The use of dual neutron poisons in the frit and glass compositions has obvious safety benefits. The iron phosphate does not have the same degree of compatibility with lanthanide neutron poisons as the Lffler glass, but is easily processed in low temperature furnaces. Moreover, the glass appears to be fairly tolerant of fluoride and reducing agents. For this reason, the iron phosphate is being proposed for disposition of heterogeneous, sub-critical masses of plutonium by direct vitrification. This should significantly lessen the number of pretreatment processes which must be developed to handle U.S. plutonium inventories.

CONCLUSIONS

There are four principle conclusions.

1. The solubility of thorium and uranium oxide in a representative Lffler borosilicate glass is approximately 9 mole percent (20 weight percent).
2. The durability of the Lffler borosilicate glass is extremely high, equivalent or better than fused silica. Thorium and uranium oxides have identical effects on Lffler glass durability.
3. The solubility of plutonium, uranium, and thorium oxide in the iron phosphate glass is approximately 17 mole percent (30 weight percent).
4. The durability of the iron-phosphate glass is considerably better than the standard for high-level waste glass. Glasses with high concentrations of uranium are slightly less durable than corresponding thorium bearing glasses.

The Lffler composition is the recommended glass composition for vitrification of SRS Am and Cm. The original composition has been tailored to be more compatible with the tetravalent actinide oxides and existing commercial melter systems. Based on the processability and extremely high chemical durability, it is concluded that this glass should also be considered for vitrification of metric tonnage quantities of weapons plutonium and uranium.

The iron phosphate glass has extremely high chemical compatibility with actinide oxides. It also has a high chemical durability and can be easily processed on the crucible scale. Based on these data, this glass should be considered suitable for disposition of small, heterogeneous sources of plutonium and uranium.

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Session 39 -- Stakeholder Involvement in Nuclear Issues
Co-chairs: Tony Villegas, Battelle-PNL;
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39-2
INTERNSHIPS: FORGING STRATEGIC PARTNERSHIPS
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ABSTRACT
Tribal governments are affected by a variety of environmental challenges. The legal doctrine of federal trust responsibility provides an avenue for tribal governments to build their infrastructure and human resources to handle these environmental challenges. Federal support for the development of tribal expertise, however, is limited by the increasingly tight federal budget. The Nez Perce Tribe proposes that building the necessary internal expertise is vital to strengthening its sovereignty and protecting the health and welfare of its members and neighboring jurisdictions. The Nez Perce Tribe in an effort to accelerate the development of its expertise and promote a clear, more comprehensive national environmental protection regime seeks to expand its long-term strategic education partnerships to include industry at work in the environmental restoration and waste management fields. The legal and political nature of environmental protection issues requires full consideration and stronger partnerships between all parties involved.
INTRODUCTION TO NEZ PERCE TRIBE AND WASTE MANAGEMENT
In a set of promises between two sovereign nations, the Nez Perce Tribe exchanged vast territorial holdings for protection, material aid, a relationship with the United States. This set of promises is contained in the Treaty of 1855. The Treaty of 1855 established a federal trust responsibility that requires federal agencies to act in the best interests of the Nez Perce Tribe when the Nez Perce Tribe does not have the capability to act in its own best interests.
For example, in the area of environmental protection, the Nez Perce Tribe has an understanding of what a healthy environment is and how to maintain that environment. This knowledge is based on traditional knowledge and cultural practices. While there is a lot to be gained by the United States and other non-Indian societies from an equitable and fair exchange of ideas and practices, the larger non-Indian society has been busy in developing more complex industrial processes along with their associated externalities (pollution). Accompanying the technical difficulty of addressing the industrial externalities is the politically and legally tortuous task of understanding and operating within the environmental arena. The Nez Perce Tribe is aware of the complexity of the environmental problems, however, within the national context it is next to impossible to negotiate these waters without assistance. Obtaining this necessary assistance while maintaining and nourishing the traditional Nez Perce knowledge and practices is the goal of Nez Perce ERWM Department's internship program.

The Nez Perce Tribe has environmental concerns, jurisdiction, and activities in three geographic areas: on reservation, in ceded territory and in possessory rights areas. "On reservation" is an area seen on most maps as the Nez Perce Reservation. "Ceded territory" is land beyond the exterior boundaries of the Nez Perce reservation that once encompassed the land claimed exclusively by the Nez Perce Tribe. "Possessory rights areas" are locations beyond ceded territory traditionally used the Nez Perce Tribe in common with other tribes. These possessory rights areas are in three states and in the ceded territories of other Indian nations. The possessory rights areas are guaranteed by the US government and are acknowledged by all other jurisdictions.

The US Department of Energy's Hanford Reservation and its related environmental management challenges affect the Nez Perce possessory rights areas most immediately along the Columbia River. Since the focus of this paper is on building strategic partnerships through internships, it is not necessary to focus on the well-documented problems at Hanford.

The Nez Perce Tribe has entered into a cooperative agreement with the Department of Energy as the mechanism for Nez Perce involvement at Hanford. The Nez Perce/DOE cooperative agreement covers the following main areas: policy and planning; clean-up and technology development; education and human resources development; public involvement; cultural resources; data management; transportation and emergency response; and health studies. Each of these areas has deliverables assigned to both the Department of Energy and the Nez Perce Tribe. The spirit of the cooperative agreement is based on a government-to-government relationship of mutual respect and as the name implies, cooperation. The Nez Perce Tribe has an on-going effort to educate the Department of Energy to its federal trust responsibility and the roles and levels of interaction that can be appropriate vis--vis the Nez Perce Tribe. Building strategic partnerships through internships can expand this education effort to other environmental management organizations.

Nez Perce Education: Historical Background

Traditionally, Nez Perce Tribal members went from being children to adults. There was no "adolescence" as we recognize the term today, with all of the implications that go with that stage of life. Children were traditionally taught by three main methods--by example, storytelling and experience. The adults of the Tribe were responsible for being role models and teachers, setting examples for the young to emulate. Although all members of the Tribe had this responsibility, the grandparents were normally the primary teachers and mentors of the children. By virtue of their age and experience, grandparents were equipped with the knowledge and the wisdom to provide role modeling, values, and behavior that would be learned by the children. Grandparents and certain other tribal elders were also "storytellers." The oral tradition of the Nez Perce people includes many stories and legends that on the face, appear to be humorous and many times even "off color." Coyote, "Its-iyaya" is the main character of many Nez Perce legends and plays many roles: the trickster, the wise one, the clown and the leader depending on the legend. The stories and legends always have a moral and this is the reason for their telling. Storytelling also passed down from grandparents and elders to children the history of the Tribe. By the time children were 12 or 13 they had been taught the necessary values and history to be considered adult. About the same time they were expected to get their "Wyakin," their guardian spirit who would remain with them for their lifetime as a guide and helper. The Wyakin was always a living thing. Usually an animal or bird and was received after praying and fasting in a remote mountain area.

Education for Nez Perce children was traditionally a process that was so interwoven into the culture that it was "painless" compared to the educational process today. There may have been tests, in that young men were expected to demonstrate what they had learned by showing their horsemanship, hunting prowess, fishing skills, tool making, and living the values they had learned. Similarly, young women were expected to demonstrate weaving, cooking, sewing and other skills they had been taught.

The education of Nez Perce children on nature; and environment, was an integral part of the culture and of their spiritual development. Since life depended on water, fish, meat, plants, herbs and medicines, there was spiritual recognition of these things and was extended to such objects as rocks. Anything related to nature was a part of the spirituality and culture of the Nez Perce people and was treated with respect. An example of this is when a deer was killed by a hunter, the hunter knocked out the deer's eyes so that thanks was given to the deer for the meat and

hide he would provide for the people.
Contemporary Indian Education
With the onslaught of European contact, the Nez Perce people were conquered physically, but never spiritually or culturally. Missionary schools were established on the reservation in the mid 1800's. The schools endeavored to teach Anglo ways and meted out harsh physical punishment to children who spoke the Nez Perce language, participated in dances or ceremonies or refused to comply with rules of English standards and protocol. Presbyterian missionaries were particularly harsh with children who attempted to practice Nez Perce customs or speak the language and were physically whipped as punishment. Anything Nez Perce was considered by the missionaries to be "heathen". The English language was picked up quickly by children and the forced imposition of European ways was eventually adopted by the majority of children in these schools as well. During this time period, the conversion of the Nez Perce to Christianity was the overriding purpose of the schools and although other subjects were taught, the missionaries were successful in accomplishing this conversion, mainly to the Presbyterian and Catholic faiths, causing factions in the Tribe that played a major part in the Nez Perce War of 1977 and continue to brew today.
European teaching methods have never been completely successful with Nez Perce students. Teaching methods differed so drastically from the traditional ways of learning that the adaptation was extremely difficult for most students and continues to be difficult today. In 1992, the public school on the Nez Perce Reservation with the highest Nez Perce enrollment aggregated test scores to see how groups were accomplishing in the system. The test scores were aggregated by race and sex and the results showed that the highest scores were earned by white females, followed by white males, Indian females, then Indian males. It became apparent that the educational system for the most part is failing Nez Perce students. The school located in Lapwai, Idaho, has a 70% Indian enrollment and has some Nez Perce teachers and administrators within the system. The school district is struggling to find ways to teach Nez Perce students so that they can successfully accomplish in school and be prepared to go on to college. Of course, there are exceptions among the Indian students with some earning high grades and not experiencing problems with learning. An interesting facet of the study, showed that Indian students accomplished on an equal level with white students until about grades 6 and 7. One theory is that once Indian students must deal with abstract concepts, learning takes a drastic dip on the scale. Math, science, and English are the areas of biggest impact and are of concern to the Nez Perce Tribe since Fisheries, Forestry, Water Resources, wildlife and Environmental Restoration, particularly in respect to the Hanford Nuclear Reservation cleanup are all in need of Nez Perce scientists and professionals. This trend appears to be true nationwide and educators currently struggle with the answers to this problematic situation.
Introduction to the Nez Perce ERWM Internship Program
The Nez Perce Tribe is implementing an internship program based on a model developed by the Council Of Energy Resource Tribes (CERT). This program identified interns as young as recent high school graduates and provides them career counseling and internship opportunities throughout their college years and beyond. The Nez Perce ERWM Department currently has seven interns, and is currently in the process of hiring a full-time education programs specialist. The ERWM Department is intent on providing opportunities to interns that best meet student, ERWM, and intern-host needs.
The Nez Perce ERWM
Department plans to develop qualified professionals with a wide variety of work experience based in a sound understanding of traditional Nez Perce culture and values. The students begin with some introductory level opportunities and as the student's interests experience and professional contacts increase, they can begin to take more control over the direction of their placements. Some students see the internships as a possibility to gain a broad-based view of their fields. Students with engineering internships have the option to experience policy and planning internships as well.
Inception of the Nez Perce ERWM Internship Program
In 1992, when the Nez Perce Tribe received funding from the US Department of Energy to participate in the Hanford cleanup, it became painfully apparent that the Tribal membership did not include scientists qualified to work in this area. Nez Perce

students who had been able to get through the public school system, were just beginning to graduate from universities, but had taken up fisheries biology or forestry, knowing that those Tribal departments were also desperately in need of scientific expertise from Tribal membership. The ERWM Department and the Nez Perce Trial Executive Committee (Tribal Council) agreed that the best approach to develop in-house expertise for this Department was to immediately start a strong on-going internship program to develop Nez Perce scientists and professionals.

An attractive aspect of the Tribal internship program is its innovative and pragmatic approach to education and training. If a student, for example, began in his or her first year of college and went through a graduate program, he or she could spend at least six summers rotating through employers such as the Nez Perce ERWM Department, the US Department of Energy (including headquarters, sites, labs, and contractors), universities and other companies dealing with environmental management and related activities. In some cases if a student is willing to take a year off from school, an annual internship can be arranged, in certain instances with companies and/or agencies providing technology development internships. The possible sites are endless and interest has even been expressed by international companies.

Nez Perce ERWM Internship Program Mechanics

The ERWM Department began its internship program by seeking out college students who were in science areas and had experienced some degree of success in these programs. Students were asked to submit letters of reference, college transcripts and a letter outlining their interest in the ERWM internship program. Five students were selected initially who were at various levels in college and had a variety of majors or goals all in the science areas, pre-medicine, civil engineering, geology, botany, and environmental engineering. The program allows for students to work during the summer, Christmas and spring breaks and in special instances we look forward to accessing year long internships with especially attractive companies or agencies. Universities and colleges attended by the interns were also varied geographically and otherwise and include the University of Idaho, Lewis Clark State College, Boston University, Brigham Young University and Colorado State University. All interns quickly developed an interest in Hanford and the particular technological and health problems presented there. Interns the first and second years were placed with the Nez Perce Tribal ERWM Department, the Council of Energy Resource Tribes (CERT) in Denver, and the US Department of Energy and its contractors, Battelle, Westinghouse, Bechtel, Kaiser, with DOE labs and with other companies and agencies dealing with technology development and health issues.

Each intern is assigned a specific project. Projects have included risk assessment; GIS mapping; botanical identification of traditional plants and herbs; and research on a wild and scenic river designation for the Hanford Reach stretch of the Columbia River, the last free-flowing section of the River. Each project concluded with recommendations for the Nez Perce Tribal Executive Committee to consider in taking an official position. The projects are not make-work type projects but have all produced quality research and/or work which have been used by the ERWM Department and the agencies providing the internship sites.

In the summer of 1994 two high school seniors were added to the program and interned at US DOE headquarters in Washington, DC, one working in the Office of Science Education and Technical Information and one in the Office of Environmental Management's Office of Public Accountability. One of these students has dropped out of the program but the other continues to participate. It has been determined that selecting students who have already begun college and have demonstrated some success is more efficient and effective for the Tribe and for the students.

Success of the Nez Perce ERWM Internship Program

We believe the primary reason for the success of the ERWM internship program is the holistic approach in dealing with the interns. The Nez Perce ERWM Department has endeavored to combine science and culture; a circle of learning that provides the interns contemporary scientific facts and knowledge, and concurrently, provides confirmation of the Nez Perce value system and culture. The fact that the Hanford Nuclear Reservation is situated on Indian lands and is rich in cultural and natural resources has driven home the importance of combining the scientific and cultural knowledge. The finding of burial grounds during the construction of the Environmental and Molecular Sciences Laboratory in 1994 made it abundantly clear that American Indians with a tribal value system need access to decision-making

roles in order to deal with instances that present problems to tribes that are both scientific and cultural. The appreciation and respect which is applied to all living things especially cultural and natural resources by Tribal peoples, unfortunately was not, and still is not, applied decision-makers in the field of environmental management. Accompanying the career counseling and educational support, the ERWM staff subtly encourage interns to investigate the cultural and spiritual aspects of their being such as traditional dancing, drumming, crafts and the language. Most interns have embraced one or more cultural areas. Not only do these activities assist them in exploring their past and present culture, but it enhances self-esteem and well-being. Identity is important to any student and these students are fortunate to have a rich and living culture with which to identify.

Another aspect of the program is a strong staff monitoring effort of all interns. There is a tribal saying that it "takes a whole village to raise a child". this is very similar to the Nez Perce Tribe's philosophy. The ERWM Department staff have endeavored not only to provide the employment and training opportunities to interns but to also provide as a whole, staff mentorship which includes friendship, personal support, counseling, gentle discipline and financial support. There is intense involvement between the individual ERWM staff and the ERWM interns. The staff is there in many of the same aspects as parents, grandparents and teachers. A mutual respect between the staff and interns has developed during the time the program has been in operation. To understand completely the type of mentorship which exists, it is important to understand that the Nez Perce Tribe numbers about 4,000 members with about 1900 living on the Nez Perce Reservation. This provides a very small cohesive group which shares many interests, sorrows, successes and failures. Everyone knows everyone else and so between the Nez Perce staff members and the Nez Perce interns, relationships already existed. In all cases, the relationships,, however, have become much closer and much more trusting.

Summary and Plans for the Future

The internship program in the ERWM Department of the Nez Perce Tribe is replicable, however at this time it is being aggressively pursued by only one Tribe. It is a program which we hope to develop and share with other tribes interested in development of their own technical expertise. It is a program which may also prove to be of great value to mainstream America.

The Nez Perce Tribe also proposes to developed education support programs with the public schools on the Nez Perce reservation. These support programs are being developed with the schools principals and student identified needs. Beyond the in-school support the Nez Perce ERWM is identifying after-school, hands-on educational opportunities aimed at enrolled Nez Perce Tribal members students.

The Nez Perce ERWM places a strong emphasis on educating Nez Perce youth because for the Nez Perce Tribe to remain a sovereign people, it must be Nez Perce people who make decisions for the Nez Perce Tribe. The more abundant and powerful tools that the Nez Perce people can learn and use to promote Nez Perce values and traditional knowledge within the national context the more equitable future exchanges, and more fruitful future partnerships will become.

OPPORTUNITIES FOR STRATEGIC PARTNERSHIPS

Opportunities to build strategic partnerships with the Nez Perce Tribe, through its ERWM Department education programs exist for industry, government and other organizations that focus on building tribal capacity, address environment justice concerns, emphasize science and engineering education, or wish to establish relations with a tribal government located in the Northwest United States with a variety of concerns affected by US DOE activities related to Hanford. The opportunities are limited by ERWM staff and resource constraints.

The ERWM Department has begun strategic partnerships with: the US Department of Energy Office of Environmental Management in Washington, D.C. and at its Richland Operations Office; and the Council of Energy Resource Tribes in Denver. Other partners in the Nez Perce education efforts are being approached with the hope of fulfilling the stated goals of the Nez Perce Tribe.

CONCLUSIONS

The Nez Perce Tribe has historically supported education for its members. Gaining the necessary knowledge of scientific and engineering disciplines is crucial to protection of Nez Perce treaty rights along the Columbia River and elsewhere. The Nez Perce ERWM internship program supports the acquisition of this knowledge and practice of these skills within the context of traditional Nez Perce culture and

values. In an effort to maintain the high quality of its program, the Nez Perce ERWM Department has identified the need to encourage strategic partnerships with government, industry, and other institutions involved in the environmental management and related fields. The technical nature of radioactive and hazardous waste management and the national political climate of greater accountability and cost-effectiveness should encourage all participants in environmental management related activities to focus on long-term, strategic partnerships. The Nez Perce Department of Environmental Restoration and Waste Management recognizes this situation and has begun to forge these new partnerships through its internship program.

39-3
PARTNERING WITH PUEBLOS: INVOLVING AMERICAN INDIANS IN ENVIRONMENTAL RESTORATION ACTIVITIES AT LOS ALAMOS NATIONAL LABORATORY,
LOS ALAMOS, NEW MEXICO
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ABSTRACT
In this paper, we review the history of the relationship between Los Alamos National Laboratory and neighboring pueblo communities and identify recent successes in improving that relationship. In particular, we discuss an initiative undertaken in connection with the Laboratory's Environmental Restoration Project. During the course of this initiative, Laboratory staff and San Ildefonso Pueblo officials have established channels of communications that have yielded significant dividends, both in terms of the quality of government-to-government interactions between the two entities and in terms of cooperative efforts that promote environmental investigation and characterization activities, both on Laboratory and Pueblo lands. Among the specific accomplishments this initiative has fostered is representation by tribal members on Laboratory technical teams, to sensitize team members to the cultural and religious significance of sites where samples are collected.
HISTORY
Los Alamos National Laboratory, located in the mountains of northern New Mexico approximately 32 miles northwest of Santa Fe, occupies a unique position in the US Department of Energy's nuclear complex. It was in Los Alamos, during the "Manhattan Project" years that scientists from Europe and the United Stated worked together to develop and test the first atomic bomb, thus heralding the beginning of the nuclear age.
Development of nuclear weapons has left a legacy of radioactive and hazardous waste. Although the dangers of acute radiation exposure were known at the time of the earliest work at Los Alamos, no one knew with any certainty what problems would arise if waste products were buried for long periods or released to the environment. Nevertheless, waste was buried in material disposal areas situated around the Laboratory site. Liquid waste also was disposed in a few of the canyons that drain the Pajarito Plateau, where the town of Los Alamos is located.
Los Alamos National Laboratory and the town of Los Alamos are surrounded by the communities of Santa Fe, Espaola, and several smaller, predominantly Hispanic, villages. A number of Indian pueblos, the closest of which are San Ildefonso, Santa Clara, San Juan, and Cochiti, are also the Laboratory's neighbors.
The area surrounding the Laboratory is steeped in pueblo history. According to oral tradition, some of the Anasazi, or "old ones," who inhabited Mesa Verde (in the four corners area of Colorado) migrated to Chaco Canyon in New Mexico. In the early 1200s, some of these Chacoan Anasazi established themselves on the Pajarito Plateau. During the 1500s, many of the inhabitants of the plateau moved to the pueblo villages that now are located along, or near, the Rio Grande. Among their descendants are the residents of San Ildefonso, Santa Clara, San Juan, and Cochiti pueblos.
Many archeological sites are located on the Pajarito Plateau and in the canyons that dissect the plateau. For instance, Bandelier National Monument contains several

beautiful ruins that have been excavated, stabilized, and opened to the public. Ceremonial rooms ("kivas"), great houses, and cliff dwellingsexcellent examples of passive solar engineeringat Longhouse, Tyuonyi, and Tsankawi ruins afford the visitor a glimpse into Anasazi village life of the 1300s.

The entire Pajarito Plateau is rich in prehistoric cultural materials. The area of the plateau occupied by the Laboratory contains many archeological sites, and over the years most of these sites have been surveyed, documented and numbered. These sites range from simple field houses to large pueblo villages.

Pueblo residents claim the Pajarito Plateau as their ancestral lands. Although they value the Laboratory as a provider of well-paid jobs, the institution is still viewed by pueblo peoples as an foreign presence. Indeed, Laboratory operations have had a major impact on the plateauecologically, culturally, and economically. Of all the modern pueblos, San Ildefonso, whose western boundary adjoins the Laboratory's, has felt the greatest impact.

The Environmental Restoration (ER) Project at the Laboratory, therefore, considered it important to establish a way for the pueblos to become involved in environmental restoration activities, especially those targeting the canyons that drain the Pajarito Plateau.

ER PROJECT GOALS

The ER Project was established in its present form at the Laboratory in 1989. The project has three goals:

Protect human health and the environment from exposure to releases of hazardous, radioactive, and mixed wastes from historical treatment, storage, and disposal practices at the Laboratory.

Meet the environmental cleanup requirements of the Laboratory's permit to operate under the Resource Conservation and Recovery Act (RCRA), specifically, under Module VIII (known as the Hazardous and Solid Waste Amendment Module, which governs ER Project activities). (1) The Environmental Protection Agency issued Module VIII in May 1990 and modified it in May 1994.

Perform these activities faster, better, and cheaper.

REGULATORY PROCESS

The ER Project follows a complex regulatory process for investigation and characterization activities:

1. A potential release site, or group of sites, is the subject of a work plan, which sets forth the sampling methodology necessary to implement the RCRA facility investigation (see step 2). The Environmental Protection Agency must approve the work plan.
2. The RCRA facility investigation identifies the nature and extent of contamination that could lead to exposure among human and environmental receptors. When the need is identified, the Laboratory may initiate an expedited cleanup, i.e., a voluntary corrective action that reduces time to completion. The Laboratory may also adopt a more formal approach involving the next two steps in the process.
3. The corrective measures study evaluates cleanup alternatives to reduce risks to human and environmental health and safety in a cost-effective manner. A corrective measures study is performed only if the RCRA facility investigation indicates that corrective measures are needed.
4. The corrective measures implementation applies the chosen remedy, verifies its effectiveness, and establishes ongoing control and monitoring measures.

LABORATORY IMPACT ON THE PUEBLOS

Figure 1 shows a map of Los Alamos National Laboratory and the surrounding communities. Many communities in the area surrounding Los Alamos are very concerned about the environmental impact that past and present Laboratory operations have on their communities. Their main concerns are the effects of releases of contaminants from the Laboratory on water, soil, and air, as well as the effects of waste stored or disposed on Laboratory property.

Environmental surveillance results (2) show that contamination may have migrated offsite through the canyons of the Pajarito Plateau to the Rio Grande. San Ildefonso and Cochiti pueblos are located downstream from the canyons that drain the Los Alamos town site and Laboratory lands. Several other pueblos are also located downstream from the Laboratory. Pueblo dwellers located downstream of the Laboratory are primarily concerned about contamination in surface water and ground water. In the pueblos located upstream of the Laboratory, residents have indicated that contamination of the air and of the meat in animals they hunt for food is their

primary concern.
Fig. 1.
THE BUREAU OF INDIAN AFFAIRS, U.S. DEPARTMENT OF ENERGY, AND SAN ILDEFONSO PUEBLO MEMORANDUM OF AGREEMENT
The Bureau of Indian Affairs, U.S. Department of Energy, and the Pueblo of San Ildefonso signed a memorandum of understanding (MOU) in the early part of 1987 (3). This MOU addressed some of the concerns of the tribal government and the pueblo community regarding contamination of air, water, soil, and ceremonially significant plants. Additional tribal concerns related to the possible effects of contamination on deer and elk populations.
The MOU established a formal protocol for sampling soil, foodstuffs, air, and water at the pueblo. The Laboratory agreed to collect and analyze material taken from vegetable gardens and the meat of cattle raised on the reservation. The Laboratory agreed to share the results of the sampling activities with the tribal council.
RECENT CONTACTS WITH SAN ILDEFONSO PUEBLO
ER Project investigations for Field Unit 1 which includes the canyons, are trying to identify movement of contaminants through the canyon systems to offsite. These canyons are the subject of a single, unified set of ER work plans. Some of these canyons run through San Ildefonso lands. It was appropriate, therefore, to involve tribal officials from this pueblo to participate in the formulation of the work plans, especially with respect to sampling activities that might directly affect cultural and religious sites.
The ER Project staff was invited to make a presentation to a meeting at San Ildefonso Pueblo in March 1994. The governor of San Ildefonso and members of the tribal council were favorably impressed with the prospect of active participation in ER activities, as was an observer from Santa Clara Pueblo. In June, the ER Project received a recommendation that the ER technical team working on the canyons be joined by a knowledgeable member of the tribe. This representative's function would be to inform team members about San Ildefonso residents' concern for their environment and about religious and cultural issues that might be raised during the preparation of the workplan and characterization activities. The tribal representative joined the team in July, and the process of familiarizing the technical team with the Pueblos' concerns have already been incorporated in the draft work plan's first four chapters. Our tribal representative received all the appropriate radiological and waste operations training he would need to work in the field, as well as training in site-specific safety procedures.
At the beginning of August 1994, ER Project staff met with the pueblo governor and received permission to do an ecological survey of lower Los Alamos Canyon, which runs through San Ildefonso lands. This was a major breakthrough for the ER Project. ER staff now had direct access to areas within pueblo boundaries if accompanied by a tribal representative. Without such access, sampling work in Los Alamos Canyon could only have been performed up to, but not beyond, the Laboratory boundary. Now, technical teams would be able to characterize the entire drainage up to its confluence with the Rio Grande, thereby allowing ER Project staff to investigate relationships between contamination on Laboratory property and that found in the most important New Mexico river.
In September and the early part of October 1994, the San Ildefonso tribal government gave permission to the Environmental Assessment/Resource Evaluation Group at the Laboratory to perform the first phase of a biological characterization survey on San Ildefonso lands in Los Alamos and Pueblo canyons. The purpose of the survey was to identify habitat types and their potential for harboring threatened and endangered species. The area to be surveyed borders the Rio Grande and extends west to Basalt Springs on the eastern boundary of the Laboratory.
Our tribal representative accompanied the team members conducting the survey. The first phase of the biological survey consisted mainly of identifying different types of vegetation in the canyon area. Members of the biological survey team laid out seven vegetation transects consisting of a series of 20-x-50-cm quadrates. The canopy cover and diameter at breast height of trees over 3 ft. high were measured in circular plots with a radius of 30 ft. The team sampled plant communities on the banks of the Rio Grande, on canyon slopes, and in the bosques (wooded areas). Percentages of grass, forb, and litter cover were determined for each quadrate. The team also identified different plant species and prepared herbarium specimens.
When the survey was completed in the beginning of October, 1994, the team

established a total of 10 to 15 transects, which should yield a clear picture of the plant communities inhabiting the canyon. These transects will be monitored every five years. Both Laboratory researchers and San Ildefonso officials will have solid baseline data with which to gauge how the canyon environment is affected by Laboratory operationsinformation that will prove useful to the tribe as it formulates its own environmental policy and program.

MILESTONES IN THE LABORATORY-SAN ILDEFONSO RELATIONSHIP

So far, collaboration with San Ildefonso has greatly benefitted the ER Project. We have been able to

establish a positive working relationship with the Pueblo of San Ildefonso;

begin developing work plans for characterizing stretches of Los Alamos and Pueblo canyons running through San Ildefonso lands to the Rio Grande;

raise ER Project and subcontractor staff's consciousness concerning the pueblo's cultural and religious concerns;

provide opportunities for training so that tribal representatives can function as part of field teams.

San Ildefonso tribal leaders acknowledge the benefits of a more open and positive relationship with the ER Project and the Laboratory. The governor has now designated an additional tribal environmental representative who regularly attends meetings of the Canyons technical team. In February, a Santa Clara tribal member will be joining the ER Project.

WIDENING THE SCOPE OF LABORATORY-PUEBLO INTERACTIONS

In November 1994, the Laboratory signed cooperative agreements with three pueblosSan Ildefonso, Jemez, and Cochiti (4). These agreements, witnessed by Charles Curtis, Undersecretary for the Department of Energy, complement a 1992 accord between the pueblos and the Department of Energy and promise to build confidence and a positive working relationship between the pueblos and the Laboratory. It also establishes a Los Alamos Pueblos Project that will carry out the objectives of the agreement.

The cooperative agreement formalizes the government-to-government relationship between the pueblos and the Laboratory. Accordingly, all interactions between the ER Project and the pueblos will henceforth follow clearly defined channels. The agreement establishes the tribal government liaison in the Laboratory's Stakeholder Involvement Office as the facilitator for any activities involving the ER Project and the pueblos.

DEVELOPING THE ENVIRONMENTAL RESTORATION PROJECT KNOWLEDGE BASE OF THE TRIBAL REPRESENTATIVE

As a member of the technical team for the Canyons, our tribal representative has learned and continues to learn more about the unique geology of the Pajarito Plateau, the distribution of the perched aquifers across the plateau, and the possible pathways along which contaminants could spread through the environment. Because his cultural background has proven to be so valuable to the ER Project, our tribal representative has also been involved in working with a second field unit. This field unit includes Bayo Canyon, which adjoins San Ildefonso lands. This canyon was the location of past Laboratory experiments which involved detonating high explosives to determine the effects on small quantities of radioactive materials. He has joined the Bayo Canyon field team to help team members take into account the unique cultural and religious significance of Bayo Canyon to San Ildefonso Pueblo. He also communicates relevant information to the Pueblos.

THE FUTURE OF LABORATORY-TRIBAL RELATIONS

A near-term goal of the ER Project is to train students from the different Pueblos in sampling techniques, and general field work during summer vacations. This can be accomplished in cooperation with the existing Undergraduate Student Program at the Laboratory. Instruction would be accomplished through course work and on-the-job training. If sampling needs to be done on pueblo lands, the ER Project could have a ready pool of field technicians available either to accompany ER staff or for trained students to take samples if the land is not accessible to non-tribal members. The students are likely to acquire job skills that make them very valuable to the Pueblos' environmental programs which are now being established. In addition, they would be employable by the Laboratory or its subcontractors.

Other pueblos that have heard about the ER Project's initiative have indicated an interest in the ER Project's work on the canyons. Hopefully, the Laboratory will be able to accommodate their participation in the future.

The ultimate success of this initiative will be measured by whether the pueblos are

confident that the chosen cleanup remedy for the canyons is the most beneficial for their people. For the Laboratory's ER Project, the ultimate success will be that the cleanup has been accomplished faster, better, and cheaper and that we have developed a mutually trusting, respectful relationship with the pueblos.

ACKNOWLEDGMENTS

This partnership with San Ildefonso Pueblo would not have been possible without the assistance of the tribal government liaison in the Stakeholder Involvement Office at the Laboratory. Gilbert Suazo was instrumental in providing access to the tribal leadership and made us aware of the unique government-to-government relationship the Laboratory has with our pueblo neighbors.

Without the vision and support of Laboratory EES-15 Deputy Group Leader Everett Springer and Principal Investigator Pat Longmire, this partnership effort would not have come to fruition. Both of them, as well as Laboratory Field Project Leader Allyn Pratt, continue to provide their support and technical knowledge.

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39-4

RISK-BASED SYSTEMS ANALYSIS FOR EMERGING TECHNOLOGIES: APPLICATION OF A TECHNOLOGY RISK ASSESSMENT MODEL TO PUBLIC DECISION MAKING

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ABSTRACT

The Risk-Based Systems Analysis Model was designed to establish funding priorities among competing technologies for tank waste remediation. The model addresses a gap in the Department of Energy's (DOE's)* "toolkit" for establishing funding priorities among emerging technologies by providing disciplined risk and cost assessments of candidate technologies within the context of a complete remediation system. The model is comprised of a risk and cost assessment and a decision interface. The former assesses the potential reductions in risk and cost offered by new technology relative to the baseline risk and cost of an entire system. The latter places this critical information in context of other values articulated by decision-makers and stakeholders in the DOE system. The risk assessment portion of the model is demonstrated for two candidate technologies for tank waste retrieval (arm-based mechanical retrieval - the "long reach arm") and subsurface barriers (close-coupled chemical barriers). Relative changes from the base case in cost and risk are presented for these two technologies to illustrate how the model works. The model and associated software build on previous work performed for DOE's Office of Technology Development and the former Underground Storage Tank Integrated Demonstration, and complement a decision-making tool presented at Waste Management 1994 for integrating technical judgments and non-technical (stakeholder) values when making technology funding decisions.

INTRODUCTION

Because of the extraordinary uncertainty inherent in research and development (R&D) activities, decision-makers must choose among still undeveloped or partially developed products for roles in large and complex technical systems, such as for high-level waste tank remediations for the U.S. Department of Energy (DOE). They must attempt to make funding decisions about appropriate technologies within tight timeframes. Decision-makers who have fiscal responsibility for program budgets have difficulty in determining whether the technology provides sufficient benefit to the overall system to warrant its development. Unfortunately, the tools to accomplish these tasks have proven cumbersome. Current systems engineering or environmental impact statement analyses that address entire systems are larger and more costly than is practical for the multiple, highly detailed decisions required to guide a technology development program. Also, these more complex models are not typically amenable to online sensitivity analyses, which are required when there are expert disagreements in predictions of a new technology's performance under varying conditions. In addition, a number of good activities are underway to improve stakeholder involvement in the decision-making process for choosing technologies, which adds the dimension of increased inputs into the process.

Because of this complexity and uncertainty, the current decision-making processes often suffer from a lack of disciplined assessment of the factors, particularly of the potentials for risk and for cost-reduction, which are usually considered by all parties to be critical. The current processes can lead to biased judgements because the technologies are rarely evaluated in a systems context.

The Risk-based Systems Analysis Model, a computer-based model for assessing emerging technologies, addresses these gaps in the decision-making tools available for selecting what technologies to fund when those technologies must function as part of a large and complex system. It provides a simplified systems perspective that is more practical for technology decisions than are the more complicated systems engineering models. It combats biases in judgment by summarizing system-level information that is difficult for experts to consider when assessing individual technologies. It accommodates differences of opinion in assumptions or predicted performance through sensitivity analysis. Finally, the interface is easy to display and understand, which makes this model useable online in forums involving multiple decision-makers, including public stakeholders and regulators who want to see (or have the opportunity to assess and differentially weight) how a technology performs on a broad range of variables.

RISK-BASED SYSTEMS ANALYSIS MODEL DESCRIPTION

The model uses a structured systems methodology for estimating the risks and the costs of new technologies for remediating DOE underground storage tanks (UST) relative to a base case. Figure 1 describes the envisioned decision-making process underlying the model. The base case establishes a flowsheet and material balance for all stages of a baseline UST remediation (the Hanford technical strategy is currently modeled). The model incorporates new technologies (test cases) into this complete remediation system to allow a disciplined assessment of changes in the risk and cost of specific functions as well as overall system risk and cost. The values for each module are summed to provide total system values. Potential risks to workers and to the public associated with both routine and failure conditions are currently represented. These data are combined with additional decision variables (e.g., program schedule) and stakeholder values (e.g., economic impacts, final land use) to ensure that risk factors are placed in a broader decision context. Where data are not available, risk and cost analyses are performed for each module using available data, expert judgment, and simplifying assumptions.

Fig. 1.

The assessment methods selected had to be:

- comprehensive and "cradle to grave" in scope;
- easy to apply early in the technology evaluation process (when limited information is available); and,
- amenable to changes in assumptions or estimates to facilitate discussion among users.

System Characterization

System characterization requires that all of the major processes and process streams be identified and that an overall material balance be established. First, the major processes are labeled as unit operations. Each unit operation is described in terms

of the information categories listed in Table I.
It is particularly important that all process streams discharging to the environment (vapor or liquid transport) be characterized, as these are the primary contributors to the public risk values. This information is used to estimate routine worker radiation exposure and potential for and severity of accidents associated with each unit operation, under both routine and nonroutine conditions.
TABLE I
The unit operations are grouped into functional groups labeled as modules. The model's baseline risk assessment consists of 13 modules that encompass the complete set of tank remediation functions, including waste retrieval, treatment, disposal, and tank and facility decontamination and decommissioning. Each module is a group of similar or related process steps linked to a specific technology and is comprised of one or more unit operations or major components. The modular format allows users to add or replace modules to define the test cases and provides a practical breakdown of overall system risk and cost. Figure 2 shows the process flow defined by the modules for the base case. The model's modular format provides a mechanism by which new modules can be added or modules can be modified to form test cases. This facilitates the incorporation and comparison of the impacts of new technologies with the baseline.
Following organization of the modules into an overall flow diagram, stream flows between modules are identified and labeled. Then, a material balance is completed for all critical streams entering and leaving each module in terms of the total liquid mass, total solids mass, total water mass, and mass of contaminants of concern.
Risk Evaluations
The objective of the model's risk calculation is to provide a consistent basis for qualitatively evaluating the potential influence that alternative technical modules may have on overall health risks associated with the remediation system. The specific steps involved in the risk assessment portion of the model are shown in Fig. 3. The risk assessment spans both operational and post-operational phases of remediation and addresses all potentially significant sources of contamination and all potential contaminant receptors. Potential risks associated with both routine and failure conditions affecting both workers and the public are addressed.*
Fig. 2.
Fig. 3.
Calculational methods were selected that are consistent with the fact that they will be applied early in the technology evaluation process, when limited or only qualitative information will be available regarding technology characteristics. The specific methods selected for demonstrating the model were chosen for their ease of application and familiarity; they may (and, in some cases, should) be further reviewed and modified. See Peters et al. (3) for the detailed calculations.
Health risks are calculated for six variables:
  workers and the general public,
  routine and failure events, and
  two timeframes:
- near-term exposures associated with the active installation and operational phases, and
- future public exposures associated with residual contamination and disposed wastes following completion of the active operations.
In the case of workers, health-related impact is evaluated in terms of radiation dose to individual members of the work force. In the case of the public, health-related impact is addressed in the context of a maximally exposed individual's (MEI) (1) incremental increase in risk (probability) of death from cancer and (2) hazard index (HI), which is the ratio of calculated toxic material exposure to the toxic material reference doses. Public health risks associated with past and future UST leaks and routine facility environmental releases are currently estimated using the Multimedia Environmental Pollution Assessment System (MEPAS) computer code (Droppo et al., 1989). For identifying and evaluating potential failure events, standard safety analyses exist, but they require a minimum level of facility and process information that is not available for these technology evaluations. Thus, a subjective approach was selected that relies on expert judgement: risks to workers and failure risks are determined using a Priority Planning Grid (PPG), developed under DOE funding and commonly used at the Hanford

Site. This process uses qualitative frequency consequence values (unitless) that have been derived from selected elements of the relative ranking system implemented in the PPG. The PPG has established eight levels of consequence (ranks), within which nine impact attributes are compared. The attributes of interest for the model risk evaluation are public safety, worker safety, environmental contamination, and cost.

Results generated by the risk calculations, while given as discrete numerical quantities, should not be viewed as estimates of actual or potential human health risk associated with the tank waste remediation technologies. The calculation results are intended to support consistent comparative evaluations of the influence that alternative technologies may have on overall system risk, and should be viewed as qualitative estimates only.

Cost Calculations

Cost information is required to provide a basis for direct comparisons of alternatives in terms of overall cost and cost-benefit. The model measures benefit in terms of risk reduction. Given the high uncertainties in risk and cost data available for emerging technologies, costs were appropriately estimated using relatively simple cost estimating software and best engineering judgement. All costs are converted into Total Net Present Worth (TNPW) in 1994 dollars. The complete waste retrieval, treatment, and decontamination and decommissioning (D&D) process is assumed to take 13 years (10 years processing and 3 years D&D).

The following costs are calculated:

capital (facility design, purchase and construction, and equipment purchase and installation),

labor (exempt, nonexempt, and bargaining unit personnel),

utilities (steam and electricity),

CENRTC and GPP (capital equipment not related to construction and general plant process),

disposal (for low-level and high-level waste only), and

D&D (decontamination and decommissioning of all tanks and facilities when remediation is complete).

System Analysis Model Software

The System Analysis Model Software is a linked system of Microsoft Excel 4.0 spreadsheets that provide the user with a computerized directory of the flow processes being evaluated by the model. By using the mouse to click on a series of buttons or menu items, the user can call up a particular case (i.e., the Base Case, Test Case I, or Test Case II), display a complete flow diagram of any module within the given case, and then find out more information about the module and any unit operation within the module. Available information also includes the set of assumptions used to characterize the module, the details of the mass balance at that point in the model, and the results of the risk analysis and the cost analysis. This layered information system, which displays only as much information as required by the user, allows the user to easily compare modules, unit operations, and risk/cost analyses results for different system configurations. It also provides access to any of the critical assumptions or parameters, making the process transparent. This serves both educational and negotiation purposes: non-experts can view the critical parameters throughout the model and experts can test alternative assumptions or judgements of performance.

EXAMPLE RISK ASSESSMENT RESULTS

The model has been demonstrated using the Hanford Tank Waste Remediation System as a base case, with two retrieval technologies for test cases: an arm-based retrieval system and close-coupled chemical sub-surface barriers. Complete descriptions of these test technologies are provided in Peters et al. (3).

Table II shows the relative contributions of each module to the base case systems and costs. Comparisons of the base case to each of the test cases is presented in relative terms in Table III. Table III shows, for example, that Test Case I (a close-coupled barrier) promises to reduce by 41% the chance that the MEI will get cancer, reduce by 47% the chance that the MEI will be harmed by toxicants, increase by <1% the accident risk, and increase by 2% the collective routine worker radiation dose. These reductions are achieved at a 24% increase in cost, or an addition of $3,300,000,000. In contrast, at slightly less cost increase (19%, or $2,500,000,000), Test Case II (robotic sluicing) may provide relatively greater risk reductions to the public. We cannot confidently conclude from the level of analysis

in the model that a test case will reduce the incidence of cancer by a specific amount, only that it will do so to a greater or lesser extent than will the baseline.

APPLICATION

The ultimate value of a technology decision tool is to provide decision-makers with a way to identify development efforts that, if successful, would provide significant overall benefit as related to cost. Health risk and cost are two critical components of overall benefit, but they are not the only relevant factors. McCabe et al. (2) have argued that decision-makers and stakeholders wish to consider environmental impacts, programmatic risks (e.g., schedule), socio-political issues (e.g., positive economic impact), other aspects of technical performance (e.g., ease of implementation), and regulatory compliance when assessing new technology. Stakeholders at the Hanford Site, which has the biggest UST remediation problem, have identified a core set of 14 values that they want to guide remediation decisions. Other stakeholder involvement exercises have uncovered additional values that may be discriminators for technology development decisions (1).

The decision process that is envisioned for application with the Risk-Based Systems Analysis Model involves the following basic steps:

Identify criteria. Decision-makers and stakeholders would agree on the criteria against which technologies should be evaluated; if this includes risks other than those currently modeled here, the data and calculations used to measure risk may be adapted to make the model more relevant (e.g., to address such things as environmental risk, schedule, and regulatory compliance).

TABLE II

TABLE III

Evaluate criteria. The Risk-Based Systems Analysis Model provides direct input to relevant risk and cost criteria; experts may use the output from the model to make holistic judgments or the output can be used to directly measure system performance for different technologies; other criteria may be evaluated using expert or stakeholder judgment.*

Apply weights. In typical decision environments, there is not single decision-maker or set of value functions to, for example, weigh cost against amount of risk reduction. We envision using a number of alternate weighting schemes to analyze how sensitive overall benefit scores are to each criterion. Decision- makers would look for those technologies that perform well regardless of weighting scheme.

Feedback. Decision-makers, stakeholders, experts, and modelers would engage in a dialogue to address outstanding questions or issues; this may result in some redefinition of the risk assessment model or its parameters.

Recommend. The final step is a recommendation to higher-level decision makers that a technology be funded or not, either as is or reconfigured to minimize outstanding risks or to maximize other performance criteria.

The model generates disciplined assessments of two of the most complex and heavily weighted variables in most funding decisions: risk and cost. However, the decision interface software also allows decision-makers to consider additional criteria, have those evaluated by multiple stakeholders, weight them using alternate value sets, and produce rankings of candidate tank technologies under each weighting scheme. Additional criteria that stakeholders may hold in evaluating technologies include System Performance, Socio-Political Interests such as protecting land use and tribal rights, and Regulatory Compliance.

SUMMARY AND CONCLUSIONS

The Risk-Based System Analysis Model provides a framework for evaluating the risk and cost of new technologies relative to a baseline remediation plan. The general approach incorporates new technologies into complete remediation systems to allow a disciplined assessment of overall system risk and cost. Risk and cost data can then be combined with additional decision variables, and evaluated from multiple perspectives (stakeholders, technologists, etc.) to produce a disciplined assessment of emerging technologies.

Application of this model will ensure that funding priorities reflect the risk/cost tradeoffs, stakeholders can understand the basis for risk/cost assessments, the final recommendation will consider risk and cost in the context of additional stakeholder/user concerns, and the decision process itself will enhance the dialogue and understanding of the different parties (technical experts, stakeholders, users, etc.). In that way, it is a practical tool for building consensus among users,

technologists, stakeholders, and final decision-makers.

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DEVELOPING PROTOTYPE INDICATORS OF VALUE AND COSTS ADDED THROUGH PUBLIC INVOLVEMENT PROGRAMS

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ABSTRACT

Although the use of public involvement continues to grow at the Department of Energy (DOE) and other federal agencies, our understanding of public involvement remains relatively rudimentary. Information about public involvement for managers, practitioners, and stakeholders is typically restricted to descriptions of the types of public involvement strategies or activities available for use or case histories of single efforts. The documented attempts at evaluating public involvement activities are limited to attitude surveys of participants or self-referential assessments of whether objectives have been met. To date, no methods or indicators of public involvement effectiveness have been developed. This research project moves beyond this descriptive stage by identifying dimensions that public involvement participants believe add or subtract value to their projects. Face-to-face interviews were conducted with public involvement participants--managers, practitioners, regulators, and other stakeholders--to ascertain dimensions of value. These dimensions were operationalized as a series of social indicators that make tangible the abstract values expressed by respondents. Indicator metrics were also identified and are being tested for reliability and validity across a wide range of public involvement efforts.

INTRODUCTION

There is a pressing need to document the value and costs of public involvement for the participants in these processes--the stakeholders--and to present this information to decision makers in a way that helps them assess the value and costs of managing public input through a public involvement program.

This research effort was designed to develop prototype indicators of the value and costs added to projects by public involvement efforts. The first step towards this goal was to identify the various dimensions of value and costs and to assess how those dimensions are distributed among different participants. Of critical importance during this phase of the research was the documentation of how participants characterize the value and costs of public involvement. Next, we used the identified dimensions of value and cost to draft a set of indicators. Indicators of the value added or subtracted by public involvement fall into three categories--process, outcome, and cost. Finally, we developed potential metrics for the indicators to be used to validate the prototype indicators.

Our review of the literature suggests that empirical analysis of public involvement processes is in its formative stages. The current literature fits generally into one

of two categories: 1) case studies or 2) normative prescriptions of how and why to do public involvement. Case studies generally provide assessments that examine the effectiveness of a public involvement process in achieving program goals and objectives. This type of research is primarily self-referential and as a result does not facilitate comparison across public involvement processes. The second type of public involvement research focuses on the reasons why and how public involvement processes should be implemented. Consensus building and the fulfillment of democratic principles are often cited as support for public involvement processes. As expected, little documentation of the value or costs actually added through public involvement exists. Nevertheless, the review of existing literature did highlight some of the dimensions of value that appear to be important to stakeholders. Issues of public acceptability, accessibility, good decision making, education and learning, time commitments, and trust were commonly discussed themes in the extant literature. This information was used to assess the validity of the indicators that were developed through the analysis of data collected in the interviews.

METHODS

The first phase of this research was to identify the various dimensions of value or cost added through public involvement processes. The data used to identify the dimensions were obtained through interviews with the participants of three different DOE-sponsored public involvement processes. These projects were selected on the basis of three criterion:

1. A wide range of stakeholders participated in the public involvement processes. Stakeholders included managers, public involvement practitioners, regulatory agency representatives, technology developers, and public interest group representatives.
2. The public involvement processes were in different stages of completion when this research was conducted. This variation facilitated the thorough examination of process versus outcome value responses.
3. The public involvement processes are perceived as successful. Past research indicates that participants of successful processes are more likely to articulate benefits as well as disadvantages of the process in which they were involved than are participants of less successful programs.

Several caveats must be noted about the generalizability of the project results to other circumstances. All three DOE projects involved the same or similar types of stakeholders. In fact, many of the respondents interviewed for this project have participated on more than one of the three projects. Most of the respondents were experienced public involvement participants and very familiar with the issues surrounding the individual projects. All three DOE projects involved the same site, although the problems addressed in the projects were slightly different. In addition, the problems addressed in the projects have not typically elicited passionate responses or involvement from stakeholders, although respondents consider the problems addressed by the projects to be serious and long-standing. All three projects were viewed by participants and observers as successful, yet none of the projects have implemented any decisions informed by the public involvement process as of the writing of this report.

While project similarity facilitated the analysis of the data collected for this report, it must be remembered that many types of public involvement exist. The dimensions of value identified by participants in these three projects may be limited by the circumstances described above. Care must be taken in generalizing the results from this limited sample to the larger population of public involvement efforts.

The sampling strategy for this project was not randomized or standardized. The key sampling criteria were diversity and representativeness. Our objective was to represent the full range of stakeholders--managers, practitioners, regulatory agency representatives, technology developers, and public interest groups--in our sample. Informed participants helped identify potential respondents.

An interview protocol was designed to elicit information from respondents about three main topics: 1) expectations of what participants wanted to accomplish during the process; 2) benefits and costs of the involvement process; and 3) the effect of the public involvement process on decision making.

In an attempt to identify the multiple dimensions of value, the respondents were asked to answer each question in reference to themselves personally, to their organizations, to the project with which they were involved, and to the community.

The essence of several questions was quite similar, although the wording of the questions was different. This method of multiple questioning was used in order to ensure that all facets of the different issues were explored.

Interviews were semi-structured, a format which allowed the interviewer to probe for both process responses (articulation of values or costs experienced as a result of participating in the public involvement process) and outcome responses (articulation of values or costs experienced as a result of the specific outcome(s) of the public involvement process). The flexibility of the semi-structured format also helped avoid undue repetition of questions. Interviews were conducted with teams of two interviewers. Audio equipment was not used, however notes were taken during the interviews.

Once the data were collected, the dimensions valued by respondents were identified through analytic coding--the process of categorizing and sorting the data. The coding was a two-phase process. In the initial coding phase, the data were sorted into discrete categories for which we developed conceptual labels that described the values and costs reported by respondents. For example, in this first round of sorting, many respondents talked about "improving the project." We used this as a conceptual label for all those values related to making the project better. The responses to each question were analyzed individually using these conceptual labels to identify the values reported by each respondent. After the individual questions were analyzed, the values were then compared across questions.

The conceptual labels developed in the first phase constitute the dimensions of value from which the indicators were developed in the second phase of focused coding. During this phase, the value dimensions (the conceptual labels) were examined for similarities and differences, which often resulted in a re-sorting of the components of the value dimensions. Following this second coding, individual indicators were developed that capture the value dimension in a way that can be measured. For example, the values included in the dimension of "improving the project" were incorporated into indicators such as "project efficiency," "cost avoidance," and "decision acceptability," all of which were components of the original value dimension. The final set of indicators are not necessarily mutually exclusive in that similar value dimensions may fit into more than one conceptual category.

At this point we re-visited the public involvement literature to see if the major dimensions of public involvement were adequately represented in the selected indicators. Stakeholder accessibility to decision making was not revealed through the interviews although it is emphasized in the literature. It may be that this was not an issue for respondents because they had secured access through their successful participation. Accessibility is included as a draft indicator but should be tested through further interviews with public involvement participants.

Finally, potential metrics for each indicator were identified in order to further validate the indicators in future work. Three sources informed the construction of the metrics: 1) the public involvement literature; 2) interview data; and 3) expert judgment and experience. The list of metrics provided for each indicator is not meant to be exhaustive but to provide examples of possible ways that the indicators can be measured.

THE PROTOTYPE INDICATORS

The prototype indicators were developed to reflect the literature and the interview responses. When respondents talk about what is valuable, they use either specific and unique examples--"then manager X and regulator Y stopped talking to each other until we brought in a facilitator for each meeting"--or conceptual ideas--"having people with lots of different ideas is important." These specific and abstract dimensions were operationalized into "indicators" which can be measured through observation (and counting) of specific behaviors. For convenience, these indicators were divided into three different categories:

- process indicators that focus on the dimensions of value found in the procedures and processes used in the public involvement effort;
- outcome indicators that focus on the dimensions of value found in the outcomes of the public involvement effort; and
- cost indicators which describe the direct and indirect costs of the public involvement effort.

Each of the prototype indicators is described briefly below. Tables I, II, and III describe examples of potential metrics for each prototype indicator.

Process Indicators

Five process-related indicators are discussed below. Potential indicators of process-related indicators are arrayed in Table I.

1. Accessibility to decision making process: The ability to participate in any given public involvement process is dependent on several variables which can be grouped together as accessibility issues. These variables emphasize decision makers willingness to meaningfully involve stakeholders in decision processes. This issue was not commonly mentioned by respondents although the literature on public involvement considers it fundamental.
2. Diversity of views represented in public involvement process: The organization and maintenance of any public involvement program depends on the identification and selection of participants who represent the range of public interests, positions, and values. Almost all respondents reported that the diversity of views represented not only the community and its interests, but enhanced and improved discussion.
3. Opportunities for participation: While public involvement participants bring their own needs and desires to the process, they all expect to participate in the process in some meaningful way. Respondents reported that participation provided the opportunity to voice their concerns and views not only to the decision makers, but to other participants as well.
4. Integration of concerns: Public involvement is enhanced for most participants by the identification and integration of stakeholder concerns. "Getting all the issues on the table," "showing respect for our point of view," and "improving and guiding" the project were all common refrains from respondents about the value of identifying stakeholder concerns. "Affecting the test plans," "prioritizing project activities," and "providing vision and direction" were common responses of how stakeholder concerns were integrated into project activities. Interviewees from all three programs were particularly satisfied with the agency's willingness to hear and respond to their concerns.
5. Information exchange: The two-way exchange of information is at the heart of public involvement programs. Every respondent reported that the process(es) they were involved in allowed information to flow and understanding to develop on both sides of the table. Information of value to participants included technical, regulatory, and public acceptability.

Outcome Indicators

Five outcome-related indicators are discussed below. Potential metrics for outcome-related indicators are arrayed in Table II.

1. Project efficiency: Public involvement is frequently claimed to reduce the amount of time required to complete a project. Public involvement activities are likely to add time "up front," especially in identifying stakeholders, educating participants about public involvement, and developing procedures that identify and integrate stakeholder concerns. Respondents claimed that regardless of these additional time requirements, time was saved on their projects in terms of faster decisions, fewer work stoppages, fewer lawsuits, and quicker implementation of decisions.
2. Cost avoidance: Many of the project "savings" reported in public involvement efforts are actually an avoidance of potential costs. These avoided costs are notoriously difficult to measure and document. However, respondents perceive these "savings" to be substantial in some cases and important in all projects. Respondents reported that typical "savings" include avoided litigation and additional studies or investigations, regulatory streamlining, and acceleration of the decision process.
3. Project/decision acceptability: Underlying all of public involvement is the desire to improve chances that decisions will be acceptable to the widest range of stakeholders, including agency staff. The majority of respondents reported that the public involvement process they were involved in enhanced project acceptability. From a manager's perspective this can mean avoiding unanticipated conflict in the future, accelerating regulatory processes, and justifying continued funding. Other stakeholders are likely to view acceptability as the point or goal of all public involvement efforts.
4. Mutual learning: Learning is the supposed outcome of education as a process and mutual learning occurs when all stakeholders learn from their participation in public involvement efforts. Mutual learning suggests that all stakeholders see themselves as responsible for finding a solution, rather than defending a particular cause or agency. Respondents told us that when stakeholders put their personal agendas aside, focused on the problem(s) at hand, and worked together to develop

solutions, mutual learning was occurring.
5. Mutual respect: Participation in public involvement processes allows stakeholders to work, talk, and solve problems with individuals who are often perceived as the "enemy." If successful, this opportunity results in an increase in respect for other people, their perspectives, and positions they take on issues. Respondents report that other participants are usually "de-demonized;" they are no longer seen as the enemy. For mangers, this often translates into viewing the input of stakeholders as valuable to resolving problems. Other stakeholders begin to see agency staff as individuals they can talk with and who listen to concerns.

Cost Indicators

Seven direct cost indicators and four indirect cost indicators are discussed. Potential metrics for direct and indirect costs are arrayed in Table III.

1. Direct costs: Direct Costs can usually be measured through traditional accounting practices once the indicators have been selected. The following direct cost indicators have been identified:
  staff labor or reimbursement for participation
  time
  facilitation services
  facilities
  materials
  travel
  specialists/experts
2. Indirect costs: Indirect Costs cannot usually be measured through traditional accounting practices, although participants are likely to recognize that they are "paying" these expenses. The following indirect cost indicators have been identified:
  Time: 1) cumulative: many stakeholders participate in more than one public involvement process. Their sense of time requirements may reflect the number of projects they participate in or a ratio of the current project to total projects time commitment. 2) perceptions: many stakeholders perceive their public involvement activities as painful or distasteful. This may be reflected in a distortion of the memory of how much time is required for public involvement participation. 3) attention: decision making efforts are frequently stretched out over many months, with actual public involvement required only sporadically. However, stakeholders are required to attend to the progress of the decision making which may be reflected in a perception that time requirements are heavy.
  Opportunity: Participation in, funding of, and commitment to public involvement efforts often requires foregoing other resource-dependent activities. These "lost opportunity" costs are notoriously difficult to measure, but the consequences are apparent to most stakeholders and agencies.
  Authority and influence: Participation in public involvement efforts is perceived by some stakeholders as a diminution of their decision making authority or influence. Respondents told us that they not only feared a loss of their decision making authority, but also a loss of credibility within their organization, a fear that information they shared would be used against them in the future, and peer pressure in the public involvement effort that was difficult to justify outside the effort.
  Emotional: Many respondents reported that they paid a heavy emotional cost for participation in public involvement activities. They report personal verbal attacks, demands that cannot be met, lack of institutional support, and general frustration as common side effects of public involvement.

Discussion of the Findings

While this research project was designed to develop a series of prototype indicators of value and cost added through public involvement, several findings from the research methods, interviews, and analysis are worth noting.

Public involvement participants were able to articulate specific dimensions of value, both positive and negative, brought to a project by public involvement activities. However, we only spoke with participants in effective programs and it may be possible that respondents from less effective efforts may have difficulty in identifying dimensions of positive value or may reveal somewhat different dimensions of value.

No participants discussed the value of accessibility to decision making although this issue is considered fundamental in the existing literature. This may be due to

the successful nature of the participants' access to programs which they considered effective. Or, people who have been actively involved for a longer time and with more projects may articulate different values. We may find that participants with less effective access or in less successful programs value and articulate concerns about accessibility (or lack thereof) to public involvement.

Almost without exception, respondents identified having a wide diversity of views represented in the public involvement process as valuable and important to them. Almost all respondents also reported that accommodating the different values and needs of this diverse group of stakeholders was difficult, if not impossible, to accomplish within the public involvement effort. Integrating input from stakeholders was attempted through allowing stakeholders to change the agenda as required, flexibility in defining problems, providing feedback on the use of input, and having stakeholders evaluate the public involvement effort.

Several respondents discussed project "savings" resulting from public involvement programs. These "savings" typically take the form of costs avoided such as avoided litigation, additional investigations, and program components. When asked to assess the monetary value of these avoided costs, respondents typically provided large, rounded-off figures such as $100 million or $45 million, which suggests that the "savings" are estimates. While avoiding litigation, new research, and schedule project components can save a large amount of money, these "savings" are hypothetical in nature. While the perception of stakeholders "saving" large amounts of money should not be denied, relying on these numbers for indicators of value added through public involvement poses both ethical and methodological questions. For example, how can one assess the amount of money saved by avoiding a hypothetical lawsuit on a hypothetical issue brought by hypothetical stakeholders? These types of "savings" can only be evaluated through a rigorous and comparative evaluation of completed projects.

Measuring the direct costs of public involvement should be relatively easy using traditional accounting practices. Indicators of direct costs also serve as metrics. One indicator of cost that will be interesting to track is change in costs over time. As public involvement becomes more efficient, costs should decrease or, at least, not increase for projects of similar magnitude. These changes will need to be measured against a "baseline" of current costs. Measuring indirect costs will present problems similar to measuring process and outcome dimensions. This does not mean that indirect costs should not be tracked. Respondents' perceptions that they are "paying" the price for participating in public involvement efforts are often indicated in how they feel about indirect costs such as lost opportunity, loss of authority and influence, and emotional wear and tear.

When asked how much time is required to participate in public involvement efforts, most respondents believe that their time commitment is quite large. When pressed to document the time spent on preparation, participation, and follow-up, however, the tallies turned out to be quite modest. We try to capture this discrepancy between actual amount of time and perceived amount of time in the indicators of indirect costs of cumulative time, perceptions of time, and length of attention required (as described in Section 5.4). Actual time commitment to public involvement needs to be tracked carefully and completely while exploration of these indirect time costs proceeds to determine what contributes to the idea that public involvement "takes a lot of time."

CONCLUSIONS

This first step of the effort to develop prototype indicators of the value added through public involvement identified those dimensions of public involvement that were valued, either positively or negatively, by participants in three different public involvement efforts sponsored by the Department of Energy. The dimensions were operationalized into indicators which could be measured and potential metrics for such measurement. However, this is only the first step in a series of steps that are necessary to develop useful indicators.

The next step is to take these prototype indicators back to a sample of the initial respondents to determine whether these indicators truly capture the dimensions of value they identified in their interviews. Validation of the indicators will also benefit from talking with public involvement participants beyond these three projects, outside of DOE, in different stages, and with more varied outcomes. Participants from different types of public involvement efforts are likely to accept the proposed indicators but will probably suggest others that reflect their own

programs and experiences. The indicators can be refined using all validation sources.
After this validation and refinement process, the indicators will need to be tested in ongoing and completed public involvement programs. Procedures for obtaining data on the metrics will need to be developed. And then, baseline data collection will be necessary. The indicators should be tested on programs that are considered successful as well as unsuccessful. If possible, the indicators should also be tested on public involvement efforts sponsored by DOE and other federal agencies such as the Department of Defense as well as state-level efforts such as those sponsored by the Washington State Department of Ecology. It is only through testing the indicators on a wide array of programs that their general usefulness will be assessed.
In the short term, the indicators can be used as a discussion point with Hanford and PNL managers currently sponsoring public involvement activities as well as those managers who have yet to do so. Hanford and PNL managers who sponsor or participate in public involvement activities can help in the validation of the indicators. They can be asked to review this report and to provide input based on their own positive and negative experiences. These managers may also have pragmatic suggestions for metrics for the indicators. Their participation in any future testing phases can also be solicited at this point.
For those Hanford and PNL managers who have yet to sponsor or participate in public involvement activities, the prototype indicators can be used to discuss common costs and benefits of using public involvement to manage public input to benefit projects. The indicators are likely to identify many of their fears of participation, particularly the indirect costs and time commitments. The discussions can also be used to validate the prototype indicators and to ensure that no major dimensions have been overlooked in this process. The indicators (and report) can begin an iterative discussion process with these managers as they begin to think about public involvement.

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A NON-LINEAR PROCESS FOR MULTI-STAKEHOLDER INVOLVEMENT IN ENVIRONMENTAL MANAGEMENT ACTIVITIES
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ABSTRACT
Governmental and industrial institutions involved in radioactive waste management are now entering a new era in which they must significantly expand public involvement. Thus, the decision making process formerly utilized to direct and guide these institutions must now be shifted to take into consideration the needs of many more stakeholders than ever before. To meet this challenge, these institutions now have the job of establishing an entirely new set of accurate, sufficient and continuous information pathways between themselves and many divergent stakeholder groups in order to create lasting relationships built on trust and mutual respect. In this paper we outline the necessary considerations for establishing the pathways for successful multi-stakeholder decision making, based upon a non-linear, cognitive-linguistic grid that has demonstrated a high degree of effectiveness in human interaction. Such effectiveness occurs because human beings are non-linear open systems, and require decision making strategies that reflect our very nature.
   The world that we have made as a result of the level of thinking that we have done so far, has created problems we cannot solve at the level of thinking at which we created them.
   -Albert Einstein
INTRODUCTION
The international industrial and governmental community involved in radioactive waste management and environmental remediation are now entering a new era of greater public involvement. Thus the decision making processes formerly utilized to direct and guide these institutions must now be shifted to take into consideration the needs of many more stakeholders than ever before. To meet this seemingly difficult challenge, these institutions will likely need to adopt new strategies for interaction that provide accurate, sufficient and continuous information pathways

between themselves and the many divergent stakeholder groups. These strategies would provide the means to establish trust and mutual respect, as well as serve as a vehicle for staying open and responsive to the needs of all internal and external stakeholders: industrial partners, governments, including the various federal, state and local regulatory agencies, and all related community groups. In this paper we present a process for creating such a strategy.

BACKGROUND

Throughout the history of Western civilization, the methods humankind has devised for problem solving have tended to be very linear in nature. As noted by Davies (1), the greater part of Western science has been founded on the method of "reductionism", whereby the properties of a complicated system are understood by studying the behavior of its component parts. To give a simple example of such linear thinking, Davies states, "there is probably nobody who understands all the systems of a Boeing 747 airliner, but every part of it is understood by somebody. We are happy to say that the airliner's behavior as a whole is understood, because we believe that an airliner is just the sum of its parts."

The first Western scientists were preoccupied with linear physical systems which are especially amenable to a reductionist approach. Indeed, Isaac Newton, whose laws of motion and gravitation gave birth to the age of science, also gave rise to the philosophical and social underpinnings of reason. If simple cause and effect relationships could describe the gravitational force on a falling apple, they could describe everything else in nature as well.

In recent years scientists have come to recognize more and more systems that must be understood holistically or not at all. These systems are described mathematically by equations known as "non-linear." Poincar, the 19th century mathematician and physicist, was among the first to recognize the problems associated with reductionism in his examination of "closed" systems, i.e. a system made up of just a few interacting bodies sealed off from the outside world.

Briggs and Peat (2) describe the problem of Poincar as follows. For a system containing only two bodies, such as the sun and earth, or earth and moon, Newton's equations can be solved exactly, and the orbits obtained are stable. Add a third body and the equations cannot be solved, the orbits only approximated by a technique called perturbation theory. To the ideal two-body system, Poincar added a term that increased the non-linear complexity (feedback) that corresponded to the small effect of the movement of the third body. Although most of the possible orbits for the two bodies are only slightly altered by the motion of the third body, Poincar discovered that with even the very smallest perturbation, some orbits behaved in an erratic, even chaotic way. "His calculations showed that a minute gravitational pull from a third body might cause a planet to wobble and weave drunkenly in its orbit and even fly out of the solar system altogether."

As noted above, modern science has focused more and more on systems whose description is considered holistic. Chaos theory, fractals, and the advent of the hologram are examples of this way thinking. David Bohm (3) perhaps summarizes it best when he theorizes that the universe must be fundamentally indivisible, in which the observer cannot be separated from the observed.

As a result of Newtonian thinking, and other conventions of dualistic thought, many human decision making strategies have become/are linear in nature. They tend to impose arbitrary limitations as well as positing simple cause and effect relationships rather than considering the myriad of constantly changing factors that come into play in each and every moment, and their mutual effect on one another. In organizations or systems where the articulated needs are relatively uniform, or homogeneous, there may be fewer disparate variables to consider. As a result, the levels of complexity they face seem, for the moment, to be manageable within the limits of what linear decision making strategies can solve.

In order for decision makers to keep a system manageable when faced with increasing numbers of variables, they tend, over time, to fix external boundaries and limit external feedbackcreating a closed system. For example, since its inception, the nuclear enterprise has a history of operating in this closed systems manner. In it formative years, the Joint Committee on Atomic Energy, a small elect group of United States Senators and Representatives, with little or no external feedback, made decisions regarding the development and deployment of both civilian and military applications of nuclear energy. In the name of national security, both the decision making, and the resultant programs, were not only kept from public view, they were

kept from other governmental review, except on a "need to know" basis. Hence a homogeneous viewpoint, coupled with fixed boundaries and little or no feedback, characterized the dawn of the nuclear age.

Such a reductionist stance gives rise to strategies which create an atmosphere susceptible to intractable conflicts with other "outside" systems that may not necessarily agree that they are outside the boundary that has been internally established. Moreover, allowing such an outside system to merely perturb the closed system, can, as observed in numerous public forums regarding nuclear issues, render the closed system chaotic and unstable, as is the case with natural phenomena.

Now as industrial and governmental institutions are challenged to open their formerly fixed boundaries to include increasingly varying viewpoints from the outside, the levels of complexity are increasing many fold. The difficulties in using linear, closed systems decision making strategies are likely to become more evident at the interface between internal and external stakeholder groups.

THE NEED FOR A NON-LINEAR MODEL

Our attempts, as decision makers, to recognize, sort, and track seemingly disparate views around the represented multiple categories of human need in what are labeled "intractable" circumstances can often seem like an impossible task. In fact, and in experience, this task often is impossible when decision makers use linear strategies for decision making. Linear problem solving methods, such as compromise, trade-offs, settlements and win/lose scenarios are likely to ignore significant relevant needs. Even win/win scenarios, which may take into account certain relevant needs, differs from our approach in that "win/win" seems to imply that there is something to winstill one-half of a polarized concept. We strive to eliminate any concepts like "winning" or "losing" that could lead to possible fragmentation of the whole. Rather we continually pay attention to the interaction between needs and the context out of which those needs arise. Without this level of attention we will likely delay or deny a final suitable resolution in which all sides recognize and agree that the best solutionbased upon present evidence and current resourceshas been attained.

The central problem we must address, as we search for resolution, is that human communication has been predicated on a set of linear and closed system assumptions, partially deriving from the fact that language itself is sequential and linear. Yet, as human beings, we are open living system organisms, continually adapting to the feedback from the environment in which we live. This constant reorganizing and transforming ourselves is the essence of living or staying alive. Using a closed systems paradigm for languaging our experience is at best inappositea category errorand, at worst, creates a dissonant distortion in experience which often fuels individual discomfort and potential conflict.

Our open systems model can be thought of as a "map" or "guidance system" which facilitates the translation of closed systems conventions in language into an open systems language which allows a more seamless movement through the ideation and decision making process. When skillfully utilized, this model helps to break the cycle of this closed systems taint in our experience. It provides a means to establish a new level of congruence in thought formation and interaction which can significantly increase understanding and cooperation in decision making processeseven between people who previously viewed their positions as intractable.

TOWARDS NON-LINEARITY

Essentially, life is non-linear, teeming with variables, all interacting, all interinfluencing and thereby yielding what we refer to as change. This seems to be an accurate view of living transactions, whether at the molecular level, cellular level, level of the organism, groups of organisms, and the subset of organisms called "human," as well as the collections of activities, organizations and institutions we create.

As we shift our decision making strategies to be more congruent with an open living systems paradigm, our ability as decision makers to recognize the consequences of the pattern of perception we are applying to the definition of our need, or the situation, is critical in the non-linear decision making realm. Accurately identifying if there are any closed systems elements in any transactions that may lead to conflict, mismanagement, or insufficiency of need gratification is essential, as is accurately and completely tracking ourselves and with others at any given moment. Further, we must be able to recognize and understand the consequences of the organization of the pattern of information that we are holding as we participate in a decision making process.

To help us do this, it might be useful to describe the landscape or backdrop on which this phenomena called human experience is being played out. From David Bohm's point of view, "physical reality is not a collection of separate objects" as we might perceive it, "but more an undivided whole that is in perpetual dynamic flux." In his view, "mind and matter are not separate substances. Rather they are different aspects of one whole and unbroken movement." He refers to this flow as "holomovement""holo" indicating that the manner in which perceived reality is structured can be likened to holography, and "movement" referring to the flux and constant state of change. This perspective is correlated with our approach borne out in this non-linear decision making modela language of holomovement.

ACCURATE AND COMPLETE INFORMATION

We as human beings are attempting to strategize sufficient means for meeting our needs. Yet, most of us do not have accurate or complete information about the essential functional elements of how to do so. Therefore, when we run into the limits of our knowledge we often create some scenario that fits for the limits of our knowledge and try to act on it to the best of our ability.

Both the nuclear industrial/governmental community and the public have been prone to finding "solutions" without having complete and accurate information. In addition to a lack of technical information regarding the magnitude and the extent of environmental degradation and its effects, both sides have failed to find the means to provide the other with the sufficient information necessary for informed decision making. In the absence of sufficient information we suffer the consequences of the limits of the information we possess. Any living system deprived of or depriving itself of information will, in all probability, function in a diminished capacity. The variations of these diminished capacities lead to a critical level of chaos and confusion, with many observers believing that the situation is impossible to remedy. Yet, when given sufficient and accurate information, the open system organism begins to course correct and bring itself back into balance.

As the environmental legacy of the weapons complex continues to come to light, greater and greater public awareness has placed even greater pressures to open the system. The public no longer wishes to become informed after the fact; rather the public wishes to become an active participant in the decision making regarding the environmental, economic and health impacts on their communities and on themselves.

FROM COMPLEX TO SIMPLE

Oftentimes, professionals in leadership positions have access to information and choose to withhold or partially disclose it. This is a behavior calculated to exclude, borne out of fear that some important need is threatened.

The continued development and production of nuclear weapons has been tightly controlled by the Department of Energy, which has been solely responsible for the various decisions regarding the development of new weapons systems, the construction of new facilities, and particularly the allocation of resources regarding production versus safety and the environment. In the name of "national security," the DOE was able to make "unilateral" decisions, a closed system strategy that ignores the interconnected, undivided wholeness by excluding from the decision making process most members of the set of what we call our "nation"with jarring consequences. This process of exclusion is a form of fragmentationthe major culprit in creating the phenomena referred to as complexity.

Complexity usually exists when we do not readily have a means to unravel and understand some new phenomena we face which do not resemble something we have handled before, something we might refer to as a puzzle. Yet, if we possess the informational grid to deal with the puzzle phenomena, we may, all of a sudden, be able to refer to the puzzling experience as simple rather than complex.

Experiencing complexity is an indicator of insufficient and/or inaccurate information in a system at the individual or group level. "Complexity" is relative to the organization of the organism labeling or describing their experience as such. Our non-linear pattern of languaging serves as a grid of manageable scale to identify the differences that make a difference in reducing complexity and intractability in human interaction and decision making. It simultaneously identifies the markers that are essential ingredients to hold in mind in order to course correct when you perceive that you may be moving toward an unstable, closed systems position. Reporting experience from this defined grid of interactive focal pointsrepresenting open system transactionsindividuals or groups of decision makers are able to begin to experience openness, wholeness, movement (emerging

cooperation), and simplicity (clarity) in each moment.
When we have such a precise reflecting surface in which to see a moment to moment view of our complete position with ourselves internally, as well as the interactive interface between our position and the position of those with whom we participate in the decision making process, something important happens. Information is now being managed in an open systems frame as it is being formed by all of the participants in the decision making process. When individuals understand this open systems non-linear frame, and apply it, the arena of complex and intractable conflict can move into the domain of simpleprovided everyone involved in the decision making process is willing to play the game by participating with the self-organizing principles of this open systems strategy.

NON-LINEAR OUTCOMES

As we align ourselves with these principles, we are likely to begin to experience many levels of internal conflict and confusion evaporating, replaced by an evolving calm, order, and balance that facilitates fuller understanding and acceptance of our own needs and position. This can be evidenced in fuller attention and enjoyment of the process we are engaged in, and more effective and satisfying relations with the other participants in the "conflict."
Further, in the realm of decision making, many of the so-called impediments related to differences seem to blend, while boundaries that once separated are now seen as boundaries that connect. The sometimes stubborn and demanding threats of loss of power, influence, money, position, and overall well-being inherent in closed systems approaches can finally become pass_ as part of the human condition. In using our non-linear grid to restore the perceptual capability for experiencing essential wholeness and interconnection, the strategies for resolving differences result in cooperative outcomes.
In circumstances that have been defined as complexintractable, multi-stake holder group conflictswe refer to the team of individuals working with the multiplicity of positions as open systems translators. As translators, we are dealing with phenomena from the position of reminding that-which-is-whole that it is whole. This translating encourages us to stop treating ourselves and the world around us as if it is fractured, fragmented and closed off rather than ultimately interconnected with every other element of alivenessthus restoring us as decision makers to our perception of wholeness within our own process of ourselves. As we participate in decision making with others (who are also being reminded through translation of their wholeness) we are able to fractally replicate the pattern of wholeness in all levels of complexity and intractability. Creating and acting as a self-healing system eliminates the so-called stubbornness of difference, and brings the possibility of unity back into direct individual and collective experience.

OUR MAPPING PROCESS

In discussing this model we cannot simply intellectually transfer it as a fixed entity to each of you, so that you can map it onto your existing definitions and beliefs about yourself and the on-going transactions around you. In one sense, this is not a "model" at all, nor is it simply a piece of knowledge to be understood and added to your existing data base. Instead, it is a mapping process to constantly order and rearrange your data base in each moment as new bits of data, or signals, come into your systemyour awareness. This essential constant rearranging is necessary in order to understand the meaning you are creating and are part of in the relative unfolding of all processes in which you find yourself. In fact, this moment to moment observing and shifting and reporting the shifting is the core of decision making.
How we identify and report this shifting is central in solving the problem of intractable conflict in human decision making. If we use a linear construct to report our experience to ourselves, we will likely perceive and report ourselves as separate from the phenomena we are observing and interacting with, thus seeing the outside world as colliding with us, as if it were doing something to us. This is the recipe for victimization, and potential conflict arising from the perception of victimizationthe creation of an "enemy."
In using this mapping process we are able to skillfully focus upon and report the fact that we are moment to moment creating meaning as it emerges from the sea of mutual arising contingenciesthe multiplicity of events we call signals and which we arrange into what we call information. In this non-linear living system which is us, there can be no victims and no enemies. In fact, the arbitrary boundaries we draw to

create meaning are simply the distinctions we provide to signal where we are focusing in the flux of variables and where we would like to influence someone else to focus with us. We are, in fact, asking for connection and cooperationnot divisiveness and separation.
To the degree that any of us have relied upon linear descriptors and descriptions to understand our experience of ourselves, we are misguiding ourselves about the nature of the transactions we are in with ourselves, particularly in the realm of decision makingwhether with ourselves or someone else. Of course, the more "someone elses" there are, the more trouble we will be in if we are using linear descriptors for a non-linear event. It is a matter of scale, in that we are in some trouble when we are in conflict with ourselves individually, yet we are in great trouble when we are in conflict with ourselves collectively, be it interpersonal hatred of difference, gang or family violence, ethnic cleansing, or degradation and pollution of the planet. Unfriendly competition and fear of limited resources, which can lead to the hoarding of information or other things essential for the common benefit, are all consequences of using linear organizing principles for non-linear eventsour lives!
What we have to offer here is a perceptual course correction for translating our linear closed systems strategies for human interaction into non-linear open systems strategies for ultimate restoration of the possibility of connection and cooperation. A means to collectively experience confluence of need and the marshaling of resources to met those needs in ways that do not unilaterally exclude any one. The illusion that we could exclude anyone is fueled by the delusion that we could possibly transact business together as non-linear organisms through the means of linear strategies and languaging.
The mislabeling of events, whether it be ourselves or elements in the universe around us, is the crux of our problem in intractable conflicts. Mistaking our definition/label of events for the experience of the event itself is our category error. To treat that which is moving (holomovement) as if it were static is our basic mistake. Therefore, the more we get into describing events between us as if they are "real," rather than just a momentary description of the pattern of information we are in with ourselves, and projecting that "realness" onto the rest of the events around us, the more we can get into trouble. These types of strategies will inevitably lead to some type of break down in our process of relating to one another, regardless of the task or function in which we are involved.
In fact, when using linear descriptors for non-linear events we are setting an inaccurate pattern in motion that will be borne out in all levels of relating: intrapersonally, interpersonally, intra-group, inter-group, intra-organizationally, inter-organizationally, intra-regionally , inter-regionally, intra-globally, inter-globally, on into infinity. The essence of our fractal universe is set in motion from the pattern we create, be it accurate or inaccurate. The consequences of inaccuracy are inordinately grave. Making the choice to shift the habit of identifying and patterning ourselves as closed linear systems, closed off from the rest of the world and, therefore, vulnerable to it, is essential for sustaining life.

CONCLUDING COMMENTS

In every corporate or institutional setting where I have applied the model in the past 20 years, it has been at the request of the "people in charge" who recognize that they are not achieving what they hoped or thought possible to achieve and, much to their dismay, things are breaking downalarm and crisis are at the doorcredibility, effectiveness, and profitability are threatened.
Whether in the corporate boardroom, the laboratory, or the production line, the problem is always the samemislabeling events leads to the breakdown of the system which eventually effects something or someone adversely, be it profits or the satisfaction of the work force. The solution process is ultimately always the same as wellto translate the configuration of events and the language describing the events, and the policies and regulations governing the events, from a closed systems linear motif into an open systems non-linear pattern of understanding and operation. This usually involves helping the players involved find the thread of how they lost and are losing themselves, defining the differences that truly make a difference in sustaining the non-linear pattern, and thereby, through apposite (negative) feedback restore the clarity and balance necessary to keep the system functioning in a salubrious fashion where each person involved can have their personal and professional needs met. I introduce them to the living exercise of decision making

by noticing the change they are in constantly and beginning to feel accord with this change and skillfully track themselves in the change, rather than straining to resist the change.
This non-linear model can be a standard of possibility for human decision making. Decision makers choosing to use this methodology will likely need in vivo coaching to unlearn the linear assumptive conventions in language and thought as they simultaneously learn the skill of applying non-linear strategies. The model provides a means for an entire individual, cultural and cross cultural course correction from adversarial to cooperative strategies in any setting provided the participants are willing to choose to use it.
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Session 40 -- New Approaches and Successes in Expediting Cleanup
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40-1
PROGRESS TOWARD IMPROVING FRONT-END PLANNING FOR DOE ENVIRONMENTAL MANAGEMENT (EM) PROJECTS--STEPS TO IMPLEMENT THE DATA QUALITY OBJECTIVES (DQO) PROCESS ACROSS ALL EM DATA OPERATIONS
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ABSTRACT
Challenged with doing more with less, the U.S. Department of Energy (DOE) Environmental Management (EM) program is placing strong emphasis on achieving significant productivity improvements in environmental data operations (i.e., sampling and analysis) through better front-end project planning. An important tool for helping achieve this objective is the Data Quality Objectives (DQO) process. Initially defined by the U.S. Environmental Protection Agency (EPA) and adopted as EM policy, DQO planning of environmental data collection is beginning to take a strong foothold at a number of DOE field sites. To accelerate the spread of DQO planning throughout the EM program, the EM Office of Compliance and Program Coordination Analytical Services Division is well within the second phase of a three-phase approach in working with DOE field sites to introduce and integrate the DQO process into field environmental data operations. This phased approach combines the strategies of technical assistance and training, creating DQO case study "success stories," field assessments, and formation of an EM Headquarters and Field Focus Group on Data Quality Objectives and Data Management. Highlighted in this paper are eight DQO case studies initiated in FY 1995 that demonstrate a broad range of applications for the DQO process, including environmental restoration, decontamination and decommissioning, waste characterization, and innovative technology demonstration.
INTRODUCTION
The U.S. Department of Energy Environmental Management program is challenged with accomplishing more in a time of shrinking budgets. As EM drives toward "early action on the ground," (1) new emphasis is focused on better planning and management of activities. One area in which EM can demonstrate significant productivity improvement is environmental data operations (i.e., sampling and analysis). Recognizing this potential, Assistant Secretary of EM, Thomas P. Grumbly took an

important step in this direction by issuing his September 7, 1995 EM policy statement that the Data Quality Objectives process be used in all environmental projects where there may be a need to collect significant environmental data.
The DQO process is a series of logical planning steps to identify and design more efficient and timely data collection efforts. This process is defined by the U.S. Environmental Protection Agency in its September 1994 final Guidance for the Data Quality Objectives Process (EPA QA/G-4), and it is a requirement for EPA Superfund projects. The DQO process has several notable strengths. It brings together the right players (managers, technical staff, regulators, and other stakeholders) early in the planning process to gain consensus and commitment about the drivers and scope of the project. This interaction leads to a clear understanding of the problem and issues, the actions needed to address the problem, and the acceptable level of uncertainty in the data for making decisions. Through this planning process, data collection and analysis are optimized so only those data needed to address the appropriate questions are collected. This saves time in planning and decision acceptance and resources by eliminating unnecessary data collection.
The DQO process has most often been applied to site remediation projects and to some extent, to waste characterization (e.g., Hanford tanks HLW waste characterization). However, DQO planning has a much wider application for designing sampling and analysis activities. The types of questions answered by environmental data cut across all of the EM program. These questions and their applications include:

What, where, and how much contamination? When is it clean or safe?

- Site remediation
- Facility deactivation
- Decontamination and decommissioning (D&D)
- Environmental monitoring (groundwater, surface water, air)

What process knowledge is appropriate and how much sampling and analysis is needed?

- Waste characterization
- Site remediation
- D&D

What existing information can be used and what new data are needed to verify facility acceptance into the EM program?

- Facility transition

What determines technology effectiveness and how is it demonstrated?

- Technology development

The DQO process also forms the basis for a related approach created by DOE. In FY 1992, DOE enhanced the DQO process as applied to environmental restoration sites by merging it with remedy selection and implementation to address all facets of an environmental restoration project. This enhanced process, called the Streamlined Approach for Environmental Restoration (SAFER) combines DQO planning and the Observational Approach operational framework, balancing the reduction of uncertainty for decision making by collecting new data with managing uncertainty through contingency actions. Descriptions of SAFER and DOE's applications are the subject of another paper in this session of the WM95 Proceedings (2).

STEPS TOWARD IMPLEMENTING DQO PLANNING ACROSS THE EM PROGRAM

Over the last three years, the Analytical Services Division (ASD), now within the DOE Office of Compliance and Program Coordination (EM-20), has been working in a phased approach with DOE field sites to introduce and integrate DQO planning into EM environmental data operations. ASD began its initial phase with a Complex-wide training program and early technical assistance to a few high profile sites (i.e., Hanford, Oak Ridge, and Savannah River). These sites have achieved resource savings and improved relationships with regulators and other stakeholders on several projects. Some specific examples are provided in another paper in this session of the WM95 Proceedings (3). Indicative of the recognized benefits of the DQO process, the initial sites are now expanding DQO planning sitewide to many of their projects and supporting it with line management funding.
After this early demonstration phase, ASD moved into phase two, stepping up its activities to enhance wider use of DQO planning in EM. In August of 1994, ASD sponsored a widely attended workshop on Applications of Data Quality Objectives Planning to EM-Wide Environmental Data Operations, where candidates for DQO case studies were solicited from around the DOE Complex. The objectives of these case studies are to:

provide hands-on training of site personnel while helping them achieve savings on their own projects;

help new sites get started in using DQO planning on their projects; and

provide documented templates on a range of DQO applications for use by the Complex.

Eight DQO case studies have been initiated in FY 1995 so far. These case study projects include waste characterization, evaluation of innovative waste characterization technology for regulatory approval, environmental remediation, and reactor and building D&D. Each of these case studies and the progress to date are described in the next section.

Phase three of implementing DQO planning across all EM environmental data operations is expected to begin in late FY 1995 to early FY 1996. The DQO/SAFER training program will continue, as it has through all phases, providing classroom instruction to sites where requested. ASD also plans to continue its technical assistance through DQO facilitation and statistical support to the field for four to six new case studies, targeting sites where DQO planning has not yet been used and new types of applications for the DQO process. Finally in phase three, ASD will initiate a program of field assessments. The intent of these assessments is to: (1) help sites evaluate their environmental data collection processes and procedures, including the planning aspects, for effectiveness in generating credible and cost effective data for environmental decision making, and (2) give them advice on how to make improvements.

Since the time DQO planning was first introduced into DOE, a number of field sites have made strides in adopting this process, both with and without ASD's technical assistance. However, there is significant room for more experience and broader use around the Complex. With the objective of facilitating and enhancing wider field implementation of DQO planning to achieve productivity savings, a new Quality Improvement Team Focus Group on Data Quality Objectives and Data Management, led by ASD, has been formed in EM. This group is comprised of representatives from all of the EM Headquarters line management organizations and several of the field offices to provide a cross section of perspectives and lessons learned on integrating DQO planning into EM field operations. The Focus Group will champion improving communication to increase awareness and understanding about the benefits of DQO planning throughout the program and promote ways to help field sites identify projects and activities where DQO planning could effectively improve operations.

DQO CASE STUDY PROJECTS

Following is a brief discussion of the eight DQO case study projects and their status as of January 30, 1995. The DQO planning phase on these projects is in varying stages of completion at this time, ranging from early discussion to near completion. In all cases where planning is underway, the DQO process has significantly helped the project teams in focusing the questions to be addressed and has often resulted in critical scope changes in response to identification of new issues not considered initially.

Waste Characterization: Hanford Ion Exchange Modules

Ion exchange modules (IXM) used to remove radioactive ions from Hanford storage basins have been stored at the K-Basins (105-KW, 105-KE, and 183-KW chlorine vault storage area) for a number of years. During this time, varying amounts of hydrogen and other potentially explosive gases have accumulated in the module chambers. The IXMs need to be stabilized and moved to more permanent storage locations, but there is a potential environmental and safety concern in moving them without testing their potential for explosion first. An action level for hydrogen of greater than 1 percent by volume within the IXM chambers is considered to be protective of human health and the environment and will minimize worker exposure during the stabilization process. Testing will be used to categorize the IXMs in one of three categories: 1) safe to move as is; 2) requires purging before it can be safely moved; or 3) requires purging and reconfiguration before it can be safely moved. DQO planning, which is complete, has been used to design a sampling program to make these determinations. Because there are no historical hydrogen data from IXMs at K-Basin, preliminary sampling and analysis of one or two IXMs is necessary to complete a statistically defensible design for making decisions with the appropriate level of certainty. The DQO process was used to 1) design this preliminary sampling and 2) produce alternative options for the number of samples required from subsequent IXMs in the complete sampling design based on probable results from the

preliminary sampling.
Through the DQO process, another issue was identified by the DOE and site contractor project team. The concern is how quickly the concentrations of potentially explosive gases in the IXM chambers are changing so that subsequent samples would be required to adequately represent the gas concentrations. This aspect of the sampling design will be addressed using a hydrogen generation and breathing model. The DQO process also helped refocus some of the initial project decisions. For example, it was realized that IXMs do not have to be monitored after their receipt for storage or disposal at the Hanford Central Waste Complex, as previously thought. This enabled the team to narrow their focus to the safety concerns surrounding the hydrogen gas concentrations in the IXM before moving or purging them. DQO planning also resulted in avoiding duplication of efforts. The IXMs in the trench at the 200-W Burial Ground became a priority and through the DQO process, it was decided that they could be moved from the trench directly to their final destination, rather than to an intermediate storage location after sampling/purging, as originally planned. By including the IXMs in the trench as a related issue in the modified scope of the initial DQO case study project, considerable planning time was saved overall.
Innovative Technology Evaluation: Fourier Transform Infrared (FTIR) Spectroscopy
Waste drums containing plutonium contaminated material that are designated to be sent to the Waste Isolation Pilot Plant (WIPP) in New Mexico for disposal must undergo head space gas testing and analysis for volatile organic compounds as part of the WIPP waste acceptance criteria and Land Disposal Restriction requirements imposed by the EPA. Analysis of head space gas has been performed using a method approved by the EPA, consisting of taking a sample of gas and analyzing it by conventional gas chromatography/mass spectrometry (GC/MS). Considering the large estimated volume of waste to be sent to the WIPP, this sample collection and analysis method is anticipated to be prohibitively expensive and time consuming (approximately $1,000+ per drum with a two day turnaround). An alternative technology, FTIR Spectroscopy has been proposed to test the head space gas. FTIR spectroscopy analysis can be performed in minutes rather than hours. This method is anticipated to be far less expensive than conventional testing because it will allow on-line analysis of waste streams.
The requirements set by the EPA for the GC/MS method in the No Migration Determination for WIPP are 25 percent precision and 30 percent accuracy. The decisions to be addressed by this case study are: 1) Does FTIR spectroscopy perform as effectively as GC/MS, and 2) can FTIR spectroscopy support decisions about whether waste drums meet WIPP waste acceptance criteria or other applicable criteria for head space gas. Two aspects of this case study make the application of the DQO process somewhat unique. First, the DQO process is usually applied to identify error tolerance for the uncertainties associated with making remedial decisions. In this case, the process is being used to make decisions about the performance of a new technology. Secondly, DQOs are being used to evaluate and compare experimental test GC/MS and FTIR spectroscopy data sets, rather than design a sampling and analysis program for collecting new data. Because most of the FTIR data are already available, DQOs will be used to assess whether further tests are required and to perform data quality assessment on the already existing test data.
Many of the steps of the DQO process have been worked through with the project team and specifications for some new data are being determined. To help with the final data quality assessment, DOE plans to assemble a panel of experts in the technology and type of measurements made to evaluate how comparable the FTIR spectroscopy results are to GC/MS and how well they meet the DQOs. This data quality assessment will form the defensible basis for presenting a case to the EPA for their approval to use FTIR spectroscopy.
Site Characterization: Paducah Waste Area Grouping (WAG) 22 Burial Grounds
The Paducah WAG 22 Burial Grounds began as a RCRA site but became a CERCLA site after it was placed on the EPA National Priorities List (NPL). Two solid waste management units in the WAG were identified as targets for field sampling to support a risk assessment and possible remediation. It appears that groundwater has been contaminated with radionuclides, metals, and chlorinated hydrocarbons. Some offsite contamination has also occurred. Although the two waste management units may be a source, all of the sources of offsite contamination have not been identified. The purpose of the sampling and analysis is to provide additional data for identifying the nature and extent of releases of hazardous waste constituents from the two

areas. The data will be used to formulate a Feasibility Study or to support the "no action" alternative.
Originally, the purpose of the investigation was to fill "data gaps" identified in the existing database. The DQO process emphasizes that data should be collected only to answer specific questions identified during planning, rather than to fill apparent data gaps. After applying the DQO process, it became clear that "data gaps" and "data needs" are not the same thing. This focused the project team better and allowed them to formulate a conceptual model of the site, identifying sources, pathways, and receptors. At that point, the site contractor was independently able to complete the process and design the sampling and analysis plan. Recognizing the value of DQO planning, Paducah will be applying the process to additional projects at the site.
Environmental Restoration: Nevada Test Site (NTS) Underground Testing Area (UGTA)
The NTS UGTA was used for nuclear testing over a wide area, with one-third of the 825 tests performed below the water table. Based on the physical characteristics of the site, vast quantities of soil and groundwater could have been adversely affected by this testing. The problem is compounded because groundwater, with the exception of some perched zones, is an average of 500 meters below the land surface. There are no clear external regulatory drivers to investigate or remediate this site and there appears to be no immediate endangerment to human health or the environment posed by the UGTA. However, DOE is committed to address the UGTA location as part of their effort to restore the environment from radioactive contamination. The State of Nevada follows a "non-degradation" of groundwater policy but has not defined what "degradation" standards might be imposed. Therefore, the objective of this DQO case study is to identify what further site characterization is required before a decision on future actions can be made. This will be based on reaching consensus with the regulators about decision drivers, such as groundwater non-degradation. If remedial action is required, decisions will include what remedies are feasible, practical, cost effective and capable of reducing risk to acceptable levels now and in the future. Discussions with the State on alternative future actions are in the early stages.
Environmental Restoration: Nevada Test Site (NTS) Double Tracks (Tonopah Test Range)
The NTS Double Tracks was a nuclear detonation area where soils have been contaminated by radionuclides and fission products, particularly plutonium. In the short term, the Air Force would like to continue using the Double Tracks area as a conventional bombing training range. This creates a risk from resuspension of plutonium contaminated soil. Long term future land use of the area could range from continued bombing to agricultural. Although no RCRA constituents have been identified at the site and thus, no clear external regulatory driver for an investigation/remediation, DOE has undertaken this investigation because of its commitment to state and federal regulators and to the public to address the radioactive contamination.
The scale of the decision making is unclear at this point; is the contamination at Double Tracks confined to soil hot spots or is it widespread across the area? The DQO process will be used to focus planning on what needs to be characterized, how remedial actions will be evaluated, and what are the ultimate disposition of the waste and waste acceptance criteria. Specific concerns stem from a question of whether field testing or screening techniques can be used to characterize the soils rather than conventional laboratory analysis. An additional aspect to this project is that the Nevada Test Site has many locations with problems similar to the Double Tracks. The process used to evaluate and remediate this location may be used as a template to apply to other NTS areas under consideration.
Environmental Restoration: Savannah River Site (SRS) R-Reactor Seepage Basins/108-R Overflow Basin
Six seepage basins associated with the R reactor operated at this location between 1957 and 1964. They have been capped with clay, and subsequently sprayed with asphalt. The R reactor area has been designated a CERCLA site, and remedial options will be evaluated once a site investigation has been completed. Monitoring wells installed in the 1960s have detected some groundwater contamination in the area, and there is evidence that radionuclides continue to be present in the soils. The main concern identified to date appears to be ecologically related rather than human

health risk. However, the site decisions defining contaminant thresholds for considering remedial action are now being developed to determine whether ecological impacts or human health will drive risk. Several project issues need to be resolved: 1) whether groundwater contamination is a result of releases from the seepage basins and should be addressed in the seepage basin remedial investigation/feasibility study (RI/FS) or whether it is part of a more complex, multi-plume contamination problem from R reactor, itself, requiring an independent study; and 2) whether the RI/FS can or should be integrated into the R Reactor D&D operation.

A characterization work plan written in 1992 was found unacceptable by some of the stakeholders, and will be redone using DQO planning. The regulators are familiar with the DQO process, having successfully applied it with SRS staff on other site projects, and were anxious to use it for this study. To date, the DQO process has been used to help design preliminary data collection to get a better idea of whether the plume is isolated or attributed to additional sources. Based on the preliminary information, broader characterization will be designed.

Integrated D&D and Site Remediation: Oak Ridge K-1410 Area Complex Demolition Project

The Oak Ridge K-1410 Area Complex Demolition Project consists of two contaminated buildings and concrete pads, a neutralization pit, associated underground power lines, and the grounds surrounding the buildings. The objectives of the project are to perform applicable CERCLA actions to protect human health and the environment and D&D actions for the most cost effective way to deal with the facilities. This project is being planned as an integration of D&D decisions (such as possible decontamination, removal, and disposal of the buildings, concrete pads, and power lines) and remediation decisions (such as removal of hot spots and remediation of soil, groundwater, and surface water). Project activities to be accomplished that are defined so far include: 1) detailed characterization of the area, 2) building demolition feasibility study, 3) removal of transferable radiological contamination, 4) removal of hazardous waste materials, 5) either demolition or stabilization and isolation of the facilities (depending on D&D decisions), and 6) closeout of the project. The DQO process will be used to identify data requirements within each project activity.

EPA and state concurrence is required on all CERCLA decisions and actions but concurrence is not required for D&D decisions. However, it is necessary for regulators to see both sets of decisions because the project is integrated. Recognizing that the project team did not have a clear picture of the process relating the logic and actions of CERCLA decisions to D&D decisions, the DQO process is being used initially with the regulators to define a model linking the CERCLA and D&D decisions so it is understood where regulator decision making authority rests and where it does not.

Reactor D&D: Argonne National Laboratory East CP5 Reactor

The Argonne CP5 was a heavy water graphite research reactor built in 1952 and abandoned in 1979. The fuel and a substantial portion of the ancillary research equipment have already been removed from the reactor building. However, tritium contamination is still a major concern. Characterization, removal, and disposal of the reactor core structure are the primary technical activities to be addressed. An environmental assessment and initial D&D work scope were prepared in 1982. This work scope needs to be reviewed and revised as appropriate based on current standards and information. Argonne would also like to use the CP5 D&D project to help develop a general model approach to management of facility D&D that can be applied to other projects.

During the initial steps of the DQO process, the DOE, site contractor, and independent experts planning team identified a number of higher level management issues to be resolved before D&D could commence. Central to these are which D&D activities should be performed by the site contractor and which should be subcontracted to outside contractors. Deciding on options for the organizational and reporting structure of the D&D project team was another key issue. Also discussed was whether to use the CP5 D&D project to demonstrate innovate technologies. Based on these discussions, the project team was able to prepare: 1) a proposed organization and reporting responsibility matrix for the CP5 project team; 2) a scheme for the planning and implementation phases of the project and content lists for the associated planning documents; and 3) recommendations to DOE for subcontracting. While working through these issues, the next step will be to

assemble a technical planning team that will apply the DQO process to developing a current day approach to D&D of the reactor core structure.

CONCLUSION

The DQO process has gained a strong foothold at a number of DOE field sites. Early successes and demonstrated benefits have stimulated these sites to expand its use to their other EM projects. Lessons learned have also filtered to other DOE sites where DQO planning is being initiated. While progress to date is favorable, DOE is still a long way from institutionalizing this front-end planning tool as a routine part of its environmental data operations. In the current world of ever increasing fiscal constraints, generating credible decision making data in a more cost effective manner is the only option. DOE can no longer afford old practices of repeating data collection over and over until we get it right. It must be the right data the first time. The DQO process is an effective tool for helping achieve this objective. With the combination of this significant fiscal driver, the recognition of DQO "success stories," and continued facilitation and technical assistance from ASD and the Quality Improvement Team Focus Group on Data Quality Objectives and Data Management, we anticipate an accelerated spread of DQO planning across the Complex over the next few years.

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DATA QUALITY OBJECTIVES PROCESS IMPLEMENTATION IN THE DOE OAK RIDGE ENVIRONMENTAL RESTORATION PROGRAM: IMPROVEMENTS IN COMMUNICATION AND EFFICIENCY

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ABSTRACT

The cost of collecting and analyzing data to address U.S. Department of Energy (DOE) remediation problems comprises a major portion of the total DOE annual budget. Lessons Learned have shown that many past sampling efforts were not essential for reaching interim or final solutions. Sampling design approaches, such as the data quality objectives (DQO) process and the streamlined approach for environmental restoration (SAFER), are now available and provide proven methods to "get to the bottom line" on the essential elements of making a remediation decision. Given the impending major cuts in DOE spending, we cannot afford not to use the DQO and SAFER processes for all future remediation efforts.

This paper outlines the history of the DQO process implementation for the Oak Ridge Environmental Restoration (ER) Program, describes problems associated with past planning efforts that resulted in less than adequate sampling program designs, provides examples of program improvements that have occurred during 1993 and 1994 because of DQO process implementation, and discusses the future direction for improved integration of data collection and decision making.

INTRODUCTION

A major portion of the total DOE annual budget is earmarked for the collection and analysis of environmental data to address DOE remediation problems. Our "Lessons Learned" records indicate that many of the past sampling efforts have not contributed significantly to decisions about interim or final solutions. Sampling

design approaches such as the data quality objectives (DQO) process and the streamlined approach for environmental restoration (SAFER) provide proven methods to "get to the bottom line" on the essential data necessary for remediation decision making. Given the impending major DOE spending cuts, we cannot afford not to use the DQO and SAFER processes for all future remediation efforts.

The goal of an efficient environmental restoration (ER) program should be to locate areas of unacceptable contamination and remediate them to levels defined as acceptable by the principal decision makers (or stakeholders). The sooner and more cost-effectively this is accomplished, the better it is for both DOE and the public. Data collection programs serve as effective tools for gathering the information we need to make sound decisions but cannot, by themselves, lead us to the best answers for remediation questions.

Selection and implementation of the best remedial decision requires the development of a conceptual model that can be used to make decisions that will address the problem at hand with an acceptable level of certainty. A conceptual model is generally a series of interconnected mathematical models and statistical descriptions of the site that are built out of data. There is no clearly defined "level" of quality that data must meet before it can be used to build a conceptual model. The only requirement is that the model have sufficient integrity, on the whole, to yield reliable results that are within the tolerance for error as defined by the stakeholders. In order to understand the uncertainty of the conceptual model, we must understand (if not measure) and prioritize the major sources of sampling error.

The following sections describe the design weaknesses of past sampling programs in the Oak Ridge ER Program, discusses improvements currently being made, and indicates the direction in which DOE Oak Ridge Operations would like to see the program move.

BACKGROUND OF DQO PROCESS IMPLEMENTATION IN OAK RIDGE

Over the last two years, the design of the ER Program's data collection processes has changed from focusing on complete characterization to focusing on the collection of data sufficient for decision making. The motivations for this change were the need for improved cost effectiveness, the desire to accelerate schedules, and information from "Lessons Learned" that showed us when too much, too little, or inappropriate data had been collected to be useful in remediation decision making. Quality Assurance staff in both DOE Headquarters and the regulatory agencies have served as catalysts for this shift by providing improved policies and direction to field elements and by backing up this direction with sound technical leadership and on-site assistance.

The ER Program was created in 1984, when Resource Conservation and Recovery Act (RCRA) corrective measures and closure regulations were the principal drivers for the mitigation of contaminant releases from disposal areas that had received hazardous and mixed wastes. While planning for these early efforts was conducted with the assistance of technical experts and involved mutual agreement between DOE and the regulators [the U.S. Environmental Protection Agency (EPA) and the State of Tennessee], the main emphasis was on schedules and reduction of regulatory compliance issues; the technical rationale for data collection often took a secondary role.

These conditions sometimes created sampling program designs that lacked a technical rationale and were inefficient in terms of cost control and scheduling. Some of the major weaknesses in past programs include the following.

Lack of adequate involvement by DOE and regulators in the design of sampling plans. Before 1993, sampling programs were often designed by DOE management and operations contractors and their subcontractors with minimal initial involvement by DOE and the regulators. DOE technical staff often did not conduct focused review of the sampling strategy until time for draft work plans to be reviewed. Because schedules were often tight, these reviews were performed to quickly to ensure data adequacy, sound technical rationale, or proper linkage to actual feasible decisions. This method of planning resulted in an unnecessarily high risk of collecting too much, too little, or inappropriate data.

Inadequate statistical design. Sampling program designs often lacked a sound statistical basis. In most cases, the statistical certainty that the sampling program provided was not based on an allowable tolerance for decision error established by the decision makers. Thus, demonstrating that the conceptual model

based on the data would meet the needs of the decision maker needs was difficult. Also, the numbers of samples to be collected were often driven by the schedule and the budget rather than by the quantities needed to make statistically reliable decisions.

A disproportionate emphasis was placed on analytical laboratory quality control. Until recently, there has been a misconception that tight lab quality control and lab data validation will ensure that the quality of the data collected will be acceptable for decision making. Unfortunately, decisions are not based on the actual data. Rather, they are based on the conceptual model that is developed from the data collected. Perfectly good analytical data, unless it is sufficiently representative of the remediation problem, may not yield a quality conceptual model. A representative conceptual model requires:

a sufficient number of samples to ensure that the statistical sampling error is within decision tolerance,

a quality control program emphasis on all major sources of sampling error and not just the analytical lab, and

appropriate data validation and assessment methods that check to ensure that all sources of sampling error are considered in proportion to their importance.

Inadequate use of historical data in building final conceptual models. For many contaminated sites on the Oak Ridge Reservation, there is a base level of historical and current knowledge regarding the nature and extent of contamination that is present and the potential exposure routes that present human health and ecological risks. Past remedial investigation planning efforts have sometimes placed improper emphasis on this invaluable source of information for decision making. There was a tendency to collect new data that repeated existing data from past studies and current data from sources not related to the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). This was done because the sampling program planners did not consider the data collected in their survey to be of sufficient quality for reliable decision making. The yardstick that the sampling program designers were using to measure data quality, however, was not performance based. They were too focused on the quality of individual data points (e.g., whether or not the data were associated with a Level IV analytical data package) rather than on whether the data, on the whole, yielded a reasonable conceptual model of the site.

Lack of adequate public input in sampling design planning and decision making processes. In some cases, public participation in the selection of feasible remedial decisions was not given proper emphasis until the time of the CERCLA-mandated proposed plan. In one case, a large data collection program was built around obtaining data to support an interim decision about the isolation of hydrologic waste (i.e., waste capping) for a large land-waste disposal area. The regulators and DOE thought the capping action was the best course to minimize releases and achieve regulatory compliance. However, when the capping decision was proposed to the public, they rejected it because they considered the proposed expenditures too large and not warranted, given the relatively low risk that the site posed to the off-site environment (the site contributed less that 2% of the total radiological releases from the plant site). A minimum level of sampling may have been sufficient to ascertain that the contaminant loading from this waste site was very small, and CERCLA-driven priorities could have been assigned to sites where risk-based decisions were more appropriate.

As is true of most problems, elaborating on cause and effect from a retrospective position is easy. The true test of the value of the conclusions reached, however, is whether the "Lessons Learned" can be applied within a program and produce quality improvement. As discussed in the next section, we in the ER Program feel that we are in the process of significant quality improvement.

IMPROVEMENTS IN PLANNING

Despite past weaknesses in sample program planning, several recent improvements have resulted in cost and schedule savings and have provided for a sounder technical rationale for remedial investigations. These improvements include development of policies and procedures that institutionalize the DQO and SAFER processes into everyday business; provision of program-wide training to educate DOE, the regulators, and contractor staff about the DQO and SAFER processes; modification of the Oak Ridge Federal Facility Agreement to require the use of DQO-based scoping meetings during CERCLA sampling program planning, and the use of headquarters-assisted pilot projects to demonstrate technical successes and train

staff.
Two pilot projects where extensive improvements were realized include the Clinch River Remedial Investigation Program and the Oak Ridge National Laboratory (ORNL) Waste Area Grouping 1 Impoundments Remedial Investigation. The first of these was specifically designated to use the DQO process, while the latter used SAFER. The logistics of each of these pilot programs and the associated improvements are described briefly below.

Clinch River Remedial Investigation

A DQO pilot project was conducted for Phase II of the Clinch River Remedial Investigation during 1993 and 1994. This pilot provided a means for substantially reducing the number of samples needed for decision making and accelerated the schedule for a Record of Decision.

The Clinch River is the main surface water receiving stream for the Oak Ridge Reservation (see Fig. 1). The history of DOE-associated contamination of the Clinch River extends back to 1943, when the Atomic Energy Commission began operations in Oak Ridge. The most significant contaminant releases occurred during the 1950s and early 1960s, and the main contaminants of current relevance were mercury from the Y-12 Plant and Cesium-137 from ORNL (other major releases of short-lived nuclides occurred, but these have long since decayed). Because of these releases, there is now widespread sediment contamination in Watts Bar Reservoir downstream of the Oak Ridge Reservation.

The principal area of concern includes mid-channel sediments for approximately 23 river miles of the Clinch River (Upper Watts Bar Reservoir) and 38 river miles on the Tennessee River downstream of the Clinch River confluence (Lower Watts Bar Reservoir). Substantial reservoir sedimentation has occurred since the period of major DOE releases, so the main inventory of contamination is now covered by two or more feet of cleaner sediment, as well as by the reservoir water column. Numerous other nonradiological contaminants have also been detected in the deep and surface sediments and waters of the Clinch River, including other metals and organics. Most of these, however, are common to industrialized watersheds, and their presence is probably only partially attributable to DOE Operations. Of these other contaminants, the PCBs are of greatest public concern; fishing advisories have been issued to limit fish consumption.

To address the current contamination concerns, a two-phase CERCLA remedial investigation was initiated in 1989. Phase I, completed in 1993, was focused on obtaining data for confirmation of past contamination studies and for conducting preliminary site characterization and screening-level risk analyses (human health and ecological). The Phase II sampling program, for which the pilot DQO project was conducted, was focused on completing characterization of the nature and extent of contamination and on providing data for final risk analyses. Although the Phase II work plan had been approved by DOE and EPA before the DQO scoping meetings began, DOE technical and management staff felt that there was sufficient opportunity for scope reductions through application of the DQO process. Thus, they coordinated with the regulators to revisit the planning stage for the Phase II work.

DQO scoping meetings for the Phase II Remedial Investigation were held from fall 1993 through winter 1994 and culminated in the issuance of a revised Phase II Sampling and Analysis plan that substantially reduced the number of samples required for decision making. Most of the sampling program reductions were achieved by improving regulator, DOE, and contractor understanding of the nature of the problems in the Clinch River, the feasible options for addressing these problems, and the general level of error tolerance allowable in these decisions.

Some of the major agreements reached, along with their associated impacts to the sampling and analysis plan, are listed in Table I. An additional benefit of the DQO process for the Clinch River Phase II Remedial Investigation was that it provided a route for communicating with the regulators about accelerating the schedule for remedial decisions. The original scheduled date for a Record of Decision was 1999. During the DQO Process, it was decided that separate remedial decisions could be made for Lower and Upper Watts Bar Reservoir, with the decision for the Lower Reservoir to be made in 1995 (a four-year acceleration), and for the Upper Reservoir in 1998 (a one-year acceleration). These schedule accelerations result in both cost savings and public relations improvements. One valuable Lesson Learned for the Clinch River DQO Process is that decision makers are uncomfortable with establishing quantitative limits on uncertainty (i.e., Step 6 of the DQO Process). For the Clinch

River, these limits were established only in a qualitative manner. Although this prevented realization of the full power of the DQO process, the use of qualitative limits provided sufficient comfort to the decision makers and still resulted in substantial improvements.
Fig. 1.
TABLE I
In summary, the DQO process for the Clinch River allowed DOE and the regulators:
to better understand the specific risk issues associated with Clinch River contamination,
to focus on attainable remedial options, and
to qualitatively evaluate how much data actually is needed to build a conceptual model for making comfortable decisions about these feasible options. The end result was substantial savings in cost and time. Table I lists documented cost savings in sample collection logistics; these, however, are only a portion of the cost savings realized. Schedule reductions and associated program size reductions will also lead to large reductions in program management costs.
Oak Ridge National Laboratory Surface Impoundments
During 1994, in cooperation with DOE Headquarters, the ER Program began performing a pilot project using SAFER to address remediation of two contaminated unlined surface impoundments at ORNL. The implementation of the SAFER process which combines the observational approach and the DQO process, resulted in a focused remedial investigation and feasibility study (RI/FS) work plan that explicitly linked data collection needs to stakeholder-defined remediation decisions. As a result, substantial data collection savings were realized by gathering the minimum data sufficient to evaluate possible remedial actions and to implement the selected action. The ORNL surface impoundments (Fig. 2) were constructed in 1945, and served through the mid 1960s as settling or equalization basins for low-level radioactive and process waste water as part of ORNL waste treatment systems. The impoundments were identified for rapid remedial action under CERCLA because their sediments were releasing contaminants to the groundwater and to adjacent surface water. Administrative controls protect current ORNL workers from exposure to contaminants; however, there is a potential risk for future site occupants. And, should an impoundment berm fail, large quantities of contaminant would be released to an adjacent surface stream (White Oak Creek), hence to the off-site environment.
At the onset of the SAFER process, much data collected as a result of various sampling efforts throughout the years existed for the impoundments. These data were gathered and assessed to develop a conceptual model that focused on hydrologic fluxes and contaminant movement. This model was presented and discussed in project scoping meetings with stakeholders that included DOE, EPA Region IV, and the State of Tennessee. During these meetings, the model was used to obtain stakeholder agreement on both the contamination problem and future exposure scenarios. Project scoping meetings resulted in several stakeholder agreements that provided for a well-defined project focus and a streamlined plan for remediation. Two key agreements were:
Remediation should be focused on risk reduction for an industrial land use. Because the site is currently industrial and is likely to be an industrial site for the foreseeable future, it was agreed that any acceptable remedial action must be sufficiently protective for industrial use. If, however, a partial cleanup protected an industrial use of a site, additional cleanup activities may be required if the site use changed in the future.
Feasible remedial decisions should be evaluated for implementation. Several possible remedial solutions were identified that would address potential health and environmental concerns in an industrial use scenario. These alternatives included simply filling and capping the impoundments, hydrologic isolation or immobilization of contaminants, consolidation of sediment into containment structures, and sediment removal for off-site disposal.
With these agreements, the project team was able to develop a series of questions that needed to be addressed to:
confirm the applicability of the conceptual model to selecting feasible remedies,
evaluate possible remedial actions, and
implement the selected action.
The questions were laid out to focus on the most likely alternatives. Specific data requirements to address the questions (i.e., to address risk assessment, ARARs,

effectiveness, cost, etc.) were identified, along with the uncertainty acceptable in those answers. Representatives of contractors responsible for the data collection, alternatives selection, remedial design, and remedial construction participated with the stakeholders in this process to be sure that data collected during field investigations would support their specific tasks.

This planning approach resulted in a relatively simple set of data requirements that considered acceptable error in characterization. Little additional data were required to support the RI/FS, and data collection was accomplished within a few weeks. This focused data collection resulted in substantial cost and time savings for sample collection and analytical work.

In summary, the SAFER process focused data collection on simply confirming an existing site conceptual model needed for evaluating and selecting among a defined set of remedial actions for future industrial land use. Without benefit of the DQO approach, extensive data might have been collected that were neither needed nor required to support remedial decision making. The linkage of data collection needs to actual decisions during this project also greatly accelerated the schedule and efficiency of producing the feasibility study because the data needed for alternative evaluation were available and much of the combined RI/FS document could be written while data were being collected and evaluated. The end result was that the project schedule was shortened from 24 months (the original baseline estimate) to 15 months for the period between the scoping workshop and delivery of an RI/FS report. This saved several hundred thousand dollars. Collecting only the data needed for decision-making also provided additional, although undocumented, cost savings. The substantial economies gained for this relatively simple project indicate that much more substantial cost savings could be gained for more complex projects.

Fig. 2.

FUTURE DIRECTIONS

Despite the progress that has been made, much remains to be done. When conscientiously applied, the DQO and SAFER processes provide substantial savings in time and dollars. Since the present culture has not, however, shifted completely to decision-based data collection planning, it is imperative that program managers (DOE, regulator, and contractor) continue to strengthen the role of the DQO/SAFER processes in routine planning.

In response to the need for improvement, DOE Oak Ridge will continue to focus on the DQO/SAFER processes as important management tools. Some of the specific areas that DOE Oak Ridge intends to focus on in coming months include:

Extending DQO/SAFER processes to remedial design and remedial construction. The pre-Record of Decision component of the cleanup process is small fraction of total cleanup costs; however, it has been given the most emphasis in the DQO process. The challenge, therefore, is to not only to streamline the process for arriving at remedial decisions, but also to more rapidly and cost-effectively design and implement remedial actions. The DQO process can be effectively employed to specify data needed for safe and effective design and construction of feasible remedial alternatives. These data can, in turn, be collected during the remedial investigation planning phase, saving time and effort. Obtaining stakeholder agreements on allowable uncertainties in remedial action performance may also prevent over-designing and, thus, provide substantial cost savings.

Using DQO/SAFER processes for decontamination and decommissioning planning. The decontamination and decommissioning process closely mirrors the CERCLA process, although the level of regulatory involvement has not been as intense. Several current decontamination and decommissioning planning efforts are being conducted using the DQO process for project scoping. Preliminary results are positive.

Improving quantitative specification of decision-error tolerance for improved sampling plans. Decision makers are uncomfortable discussing acceptable decision errors in quantitative terms. Because the quantity of data collected depends directly on the statistical uncertainty that can be accepted in the data-related decisions, it is critical that this issue be addressed. It should be made clear from the start at all DQO scoping meetings that establishment of quantitative error tolerances will be given high priority.

Increasing the use of field analytical laboratory methods and performance-based data validation/data assessment. Requested analytical methods and data validation and assessment processes are often not closely aligned with the actual data quality objectives of the project. This is due in part to an improper understanding of the

uncertainties associated with the analytical measurement process in relation to overall sampling program uncertainty and the tolerable level of decision error for the project. In some cases, a better conceptual model can be developed by using available funding to perform a larger number of field lab analyses in lieu of costly offsite lab services. Data validation can also be streamlined by concentrating efforts on those analyses where the consequences of decision error are greater and by eliminating reviews of redundant quality indicators. Data assessment should also become a routine practice to ensure that the overall dataset is suitable for use in the conceptual model.
In conclusion, the funding for DOE cleanup programs is shrinking and, unfortunately, remediation of many of our major sources of contamination has only recently been initiated. Based on previous spending patterns, it will be difficult to sustain a viable and effective remediation program if we follow the rigid and exhaustive CERCLA path of the recent pastit is simply too costly. Fortunately, the DQO and SAFER processes offer a proven method of cost and schedule reduction. The success in realizing their benefits, however, is up to us.

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THE STREAMLINED APPROACH FOR ENVIRONMENTAL RESTORATION (SAFER): STATUS REPORT ON ONGOING PILOT PROJECTS

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ABSTRACT
The Department of Energy (DOE) developed the Streamlined Approach for Environmental Restoration (SAFER) as a tool to effectively recognize and manage the uncertainties inherent in the environmental restoration process. The SAFER approach integrates the Data Quality Objective (DQO) process with the Observational Approach. DOE and the Environmental Protection Agency (EPA) agreed to jointly test SAFER on a pilot project scale. In January, 1994, DOE selected four facilities to host SAFER Pilot Projects: Hanford in Washington, Mound Plant in Ohio, Oak Ridge National Laboratory (ORNL) in Tennessee, and Savannah River Site (SRS) in South Carolina. The Hanford Pilot focuses on remedial design and remedial action at a reactor area. At the Mound Plant, two of the three SAFER projects involve removal actions. The third Mound project involves property transfer to the City of Miamisburg. The ORNL SAFER pilot project addresses the remedial investigation (RI) / feasibility study (FS) for a series of surface impoundments. SRS hosted two pilot projects, both of which were in the RI planning stage. The success of the pilot projects is being measured both quantitatively and qualitatively by an independent performance measurement team. Although the SAFER Pilot Projects are only 75 percent through the allotted time, significant accomplishments can already be credited to applying SAFER.

INTRODUCTION
The Department of Energy (DOE) developed the Streamlined Approach for Environmental Restoration (SAFER) as a tool to facilitate focused identification and management of key uncertainties inherent in the environmental restoration process (1,2). SAFER embodies the strengths of the Data Quality Objectives (DQO) process and the Observational Approach, which are integrated to form a comprehensive methodology, applicable to all stages in the life of an environmental restoration project. The primary features of the SAFER process include: 1) emphasis on up-front planning to ensure that only data required to make a decision about the site are collected; 2) management of uncertainties whenever possible by contingency plans, allowing remediation to progress; and 3) stakeholder involvement throughout the process. Planning using the SAFER approach incorporates and enhances the DQO process, which

was developed by the Environmental Protection Agency (EPA) (3,4,5,6). This process begins with the participation of stakeholders to develop a conceptual model that summarizes the existing site knowledge and to explicitly state the problems at the site that need to be resolved. Decisions that must be made in order to resolve the problems are the basis for formulating data needs. With stakeholders involved, the balance between the acceptable uncertainty in the resultant data and the cost of the data can be established early to optimize the sampling plan.

The Observational Approach, a basic geotechnical engineering technique (7,8,9), is integrated with the DQO process effectively using the conceptual model to identify probable conditions and reasonable deviations. Reasonable deviations are addressed in a contingency matrix that specifies a contingency plan. Monitoring protocols trigger implementation if the deviation is detected. If realistic contingency plans cannot be developed for a deviation, then additional data collection may be warranted to reduce or eliminate the uncertainty about that condition.

Formulation and implementation of SAFER has been a collaborative effort by four different offices within DOE: Office of Program Integration (EM-43), Office of Transportation, Emergency Management, and Analytical Services (EM-26), Office of Environmental Activities (EM-22) and Office of Environmental Guidance (EH-23). The DOE SAFER team has worked for several years to develop and refine the SAFER process. This includes providing workshops to disseminate the concepts and helping to implement the process.

In 1992, DOE approached the EPA with the idea of testing SAFER on a pilot project scale. The primary objectives of the pilot study were to: (1) measure the effectiveness of the SAFER process as implemented at several sites, and (2) identify reactions by participants, both positive and negative, to implementing SAFER broadly throughout the DOE complex. EPA supported the idea of piloting the SAFER process at sites within the DOE complex, but requested that DOE stipulate the nominated sites for the pilot study satisfy three criteria: 1) projects must be part of National Priority List (NPL) sites; 2) DOE, EPA, and cognizant State authorities must jointly nominate the candidate projects; and 3) the local DOE Project Manager must have signature authority for the primary deliverables on the project. In January 1994, the DOE Deputy Assistant Secretary for Environmental Restoration and Waste Management selected four DOE facilities to host SAFER Pilot Projects: Hanford in Washington, Mound Plant in Ohio, Oak Ridge National Laboratory (ORNL) in Tennessee, and Savannah River Site (SRS) in South Carolina. SAFER team support to the pilot projects will continue through the summer of 1995.

Of particular importance during the pilots is how the effectiveness of SAFER in these projects can be measured. An independent performance measures team is developing qualitative and quantitative measures that will be applied to the pilots. A final report will be released in the Fall of 1995 that describes the sites, how SAFER was applied, and the measures of success. This paper serves as an interim report on the progress and success of the SAFER pilot projects.

PILOT PROJECT PROGRESS

When the pilots started, the subject projects were at different stages in the remedial process: Hanford's pilot project is in Remedial Design/Remedial Action (RD/RA); at Mound, two of the pilot project sites are interim removal actions and one site is a land transfer associated with an ongoing RI; at Savannah River the 2 projects are in the Remedial Investigation (RI) planning stage; and the ORNL project is planning the Feasibility Study (FS). The following is a brief description of each site and the progress through December 1994; measures of success are discussed in the next section.

Hanford Pilot Project

The Hanford Pilot focuses on Operable Unit (OU) 100-BC-1 contained within the 100-BC Reactor Area. OU 100-BC-1 is a source overlying a groundwater OU, which flows directly into the Columbia River. The 100-BC-1 OU contains 44 waste sites, all resulting from managing 183 million liters of liquid and 40 cubic meters of solid effluents from B Reactor operations. The waste sites include cribs, trenches, retention basins, burial grounds, pipelines, river outfalls, and septic tanks. Contaminants of concern include fission products (primarily Cs-137, Co-60, Sr-90, Eu-152, and Eu-154) and chromium.

DOE, EPA, and State regulators, through a tri-party agreement, have determined that a single Interim Action Record of Decision (ROD) will be applied to the entire 100-BC Reactor Area. The Proposed Plan is scheduled to be signed in February 1995;

the ROD and Design Report are scheduled for June 1995. Following signature of the ROD, remediation of at least one waste site must begin prior to October 1, 1995, with other waste sites following thereafter. The SAFER pilot is focusing on assisting DOE/Richland Operations in meeting the goal of completing the design and beginning remediation of one waste site, following ROD signature in June.
The SAFER process is being used in the following aspects of the pilot project:

To identify a process that will be used to develop remediation levels (i.e. cleanup goals) for incorporation in the 100-BC ROD and to translate final remediation levels established in the ROD into stopping rules for the waste site remediation.

To establish priorities for waste sites within the 100-BC-1 OU. Criteria are being developed to determine the order in which different waste sites within the OU will be selected for remediation, and to identify the waste site that will be the focus of the first interim action at 100-BC-1.

To develop conceptual design criteria for the selected waste site, including identification of data needs for the remedial design and development of monitoring plans and contingency plans to manage uncertainties.

Mound Plant Pilot Project

Two of the three SAFER projects at the Mound Plant are removal actions: 1) B Building Solvent Storage Shed and 2) Area 7 soil remediation. The third project involves determining that property is not contaminated and can be transferred to the City of Miamisburg.
The B Building solvent storage shed pilot addressed a 33- by 15-meter area of soil contamination beneath a solvent storage shed. Historically, the B Building storage shed was used to store waste and product-grade solvents from B Building. Before the shed was constructed, drums of solvents were stored in an outdoor storage area adjacent to the present shed location. Prior to SAFER team involvement, the removal action called for complete site characterization, removal of the shed, and use of soil vapor extraction (SVE) to remove contaminants from the soil. The objective of the site characterization was to determine depth to bedrock in order to evaluate the feasibility of using SVE.
For the B Building Storage Shed, SAFER team assistance and recommendations were applied to the following:

A conceptual model and matrix of probable conditions, possible deviations, and contingencies were used to show the project team that there was no need for a complete site characterization prior to beginning the SVE. A simple and cost-effective contingency plan was identified that could be implemented quickly in the field if it was determined that the depth to bedrock was insufficient for the SVE system to operate effectively at certain wellheads.

The conceptual model also helped the project team to separate demolition of the shed from remediation of the soil. This separation allowed on-site staff to demolish the shed, eliminating the need for a separate contract. Thus the removal action could focus on remediation of the soil.

The second SAFER pilot site at Mound was the Area 7 Soil Removal. Area 7 was historically used to deposit construction debris, thorium drums, and other wastes. In 1959, 3 dump truck loads of soil contaminated with Ac-227 and Ra-226 were deposited in or near an abandoned septic tank in the northeast section of Area 7. The septic tank historically had received only sanitary waste from the administrative buildings at the Mound Plant between 1946 and 1952, when it was abandoned. The exact location of the septic tank is unknown. Results from previous sampling efforts have shown elevated levels of Ac-227 and Ra-226 in the soil near the suspected location of the septic tank and elevated levels of Ra-226 in the soil downslope. Prior to the commencement of the SAFER pilot, the response action was focused on removing the septic tank, including a full site characterization to find the tank.
SAFER involvement in the project had the following effects:

The conceptual model helped focus the team on the contaminated soil rather than the tank. This re-focusing made a significant difference in the nature and scope of the removal action and formed the basis of the revised work plan.

SAFER processes were used extensively to develop the basic design approach, removal goals, and preliminary decision rules.

The third SAFER project at Mound was a property transfer action. The property involved is land purchased in 1981 as a buffer area around the Mound Plant. No DOE

operations were conducted in the area. Prior to DOE ownership, the land was used primarily for agriculture. DOE wants to release the property to the City of Miamisburg for industrial development. Although no DOE operations took place in the immediate vicinity, thorium re-drumming (including handling and disposal of thorium) was done in an adjacent area. Previous sampling efforts indicated a potential for migration of thorium onto the area considered for transfer. No other potential releases of hazardous substances have been identified.

SAFER was introduced in the project in the following ways:

Developed a land transfer strategy for the New Property and identified a formal method of approval (i.e., an "intent to transfer" letter) that would ultimately lead to formal EPA approval of the transfer. The strategy used a phased approach wherein the clean areas (based on distance from operations) would be transferred first, followed by sampling of potentially contaminated areas to determine whether they could be transferred with no remedial action.

Applied decision rule logic to the analytical results from the "clean" area to document the absence of contamination, thereby supporting the transfer. The same decision rule logic was used to develop the sampling plan for the potentially contaminated area.

Oak Ridge National Laboratory (ORNL)

The ORNL SAFER pilot project began in the RI report and FS work plan stage of a CERCLA remedial action. The focus of the project is the four Waste Area Group (WAG) 1 surface impoundments. There are two lined and two unlined impoundments located adjacent to a stream in the highly industrialized main plant area of ORNL. The surface impoundments were used to store low-level mixed liquid wastes that were generated by ORNL operations from 1945 through 1976.

Since the remedial investigation had been completed, substantial historical and sampling data was available. These data included historical operational activities and analytical results from samples of groundwater, soil, and sediments in the vicinity of the impoundments. Based on these data, the impoundments are cut into or are very close to bedrock, which is fractured limestone aquifer. Groundwater beneath the impoundments is known to be contaminated and continued release of contaminants from the unlined impoundments to groundwater and surface water is a virtual certainty.

Sludges in the impoundments contain radionuclides (Sr-90, tritium, Pu-239, Pu-240, Cs-137, and possibly U-233), metals, pesticides, and polychlorinated biphenyls (PCBs). The Sr-90 and tritium are of particular concern because of their mobility in groundwater.

SAFER influenced the direction, schedule, and cost of the pilot project in the following ways:

Encouraged the Tennessee state regulators to become fully integrated members of the project team. A State regulator now attends all pilot project meetings, and provides valuable insight into the specific types of information the State wants to see in an RI/FS in order to evaluate the remedial alternatives under consideration.

The conceptual model helped focus the RI/FS document on key issues and decisions, resulting in a smaller, clearer, and more concise document. The conceptual model also was instrumental in establishing a common understanding of the goal through early development of the problem statement and remedial objectives, and continual reevaluation of the model in response to changing information.

Use of decision rules enabled the site project team to identify a limited set of data needed to analyze the feasibility of remedial alternatives, and to eliminate collection of all other data or to postpone data collection to the remedial design stage of the remediation.

Savannah River Site (SRS)

SRS is hosting two pilot projects, the F- and H-Area Retention Basins and the D-Area Oil Seepage Basin, both were started at the RI planning stage. Although both sites were used for liquid waste disposal, the similarity ends there. SAFER has been applied to both sites with success.

The F- and H-Area Retention Basins were unlined surface impoundments designed to receive contaminated cooling water discharged when leaks in the cooling system caused operational upsets. The contaminated liquid was delivered to the basins via large diameter (24- to 36-inch) process pipelines. However, only the F-Area Retention Basin had inactive pipelines that were included in the pilot project. The two retention basins were operational for more than three decades, up to the

late 1970s. Releases to the environment occurred via infiltration from the basins themselves and potentially from leaks along the pipeline. Additionally, both basins had historically overflowed into adjacent streams, contaminating the overflow areas. Based on RESRAD modeling outputs, the primary contaminants of concern are Cs-137 and Sr-90, because these are predicted to be the most mobile.
When the retention basins were taken out of service, the F-Area basin was remediated to the standards of the 1970s, but the H-Area basin was fenced and left as is. At the F-Area basin, the contaminated soils and sludge were excavated down to the original bottom of the basin and disposed of. Analyses of samples collected at the bottom of the excavation indicated that Sr-90 and Cs-137 had migrated into the soil underlying the basin. Sr-90, the more mobile of the two, was present 2 meters below the basin bottom. However, the threat to groundwater appears to be minimal because groundwater is 16-20 meters below ground surface. The excavation was filled with clean soil and revegetated.
Access to the H-Area Retention Basin and the overflow area is restricted to minimize exposure to high radionuclide concentration, estimated to be greater than 10 mrem/hr at several locations. The fenced area, which includes the basin and surrounding berm, is covered with vegetation, including numerous trees. The vegetation is part of the problem at this site, because it is too radioactive to dispose of in a solid waste landfill. At times, the basin contains standing water, reflecting the high groundwater level (i.e., 3 to 4 meters below the surface).
The SAFER process has been applied to the F- and H-Area Retention Basins project in the following ways:

Facilitated the use, and stakeholder acceptance, of the RESRAD model to estimate the migration of radioactive contaminants through soil. Through use of RESRAD, DOE was able to significantly reduce the number to contaminants of concern to the two most mobile.

SAFER consensus building and integration of the stakeholders as members of the project team facilitated the use of innovative technology, including a robotic pipecrawler and hydropunch. The pipecrawler remotely videotaped the process pipeline, and the videotape was used to identify potentially contaminated areas (i.e. cracks or joints). The hydropunch facilitated collection of soil and water samples. The regulators were willing to consider use of these innovative approaches because they were integrated into the project team, they were excited about SAFER, and SRS agreed to confirm the information derived from the pipecrawler.

Both EPA and State regulators were active participants in the early decision-making stages of this pilot. They often provided timely and valuable insights into their issues and concerns, allowing the project team to re-focus their activities and eliminate false starts.

The second SAFER pilot project hosted by SRS is the D-Area Oil Seepage Basin. Prior to acceptance of the D-Area Oil Seepage Basin as a SAFER Project, there was an interim action planned that will remove the source material (drums and sludge) and replace the disturbed soil back into the trenches. The RI will focus on the soil remaining after the interim action and the groundwater. This site was constructed in 1952 as a series of 2-meter deep unlined trenches for disposal of waste oils and other fluids not suitable for burning in powerhouse boilers, general office, and cafeteria waste. Until 1973 the waste in the trenches was periodically burned. After 1973 open burning ceased, but disposal in the trenches continued until 1975 when the trenches were backfilled with soil, burying an unknown number of drums.
The trenches are in recent fluvial sediments of the Savannah River floodplain which consist of sand, silt, and clay. Alluvial groundwater level fluctuates seasonally between 1 and 5 meters below the ground surface. Thus, during periods of high water level, groundwater is within the trenches and in periods of low water, the trenches are in the vadose zone. Groundwater flow is to the southwest towards a wetland area located about 60 meters from the basin. The alluvial aquifer is separated from the underlying aquifer unit by a clay layer.
Soil in the trenches was sampled and analyzed in 1984 and 1993. Based on comparisons between maximum observed values and Preliminary Remediation Goals (PRGs), the contaminants of concern are a dioxin congener, arsenic, and dieldrin. In groundwater, trichloroethylene and its degradation products have been detected and floating free product was bailed from a borehole.
SAFER has influenced the RI portion of remedial action at the D-Area Oil Seepage Basin in the following ways:

Regulatory stakeholders had early input into the development of the conceptual model, problem statements and decision rules that form the basis of the RI work plan.

The problem statements and decisions needed to resolve the problems focused the scope of the RI. In particular, only the top 1 meter of soil will be considered for its potential to expose a human receptor by direct contact. Soil deeper than 1 meter will only be considered for its potential to continue to impact groundwater.

For the RI work plan, a phased sampling and analysis plan was developed for saturated zone soils with the objective of assessing the potential effects on groundwater. The first phase will focus on determining retardation factors for the contaminants present in the saturated zone soils. If contaminants are not highly retarded, the objective of the second phase will be to assess whether the saturated zone soils will continue to be a source of contamination that poses an unacceptable risk to human health or the environment.

MEASURES OF SUCCESS

The success of the pilot projects will be measured in two ways -- quantitatively and qualitatively. The quantitative measurements will be savings in time and money that can be attributed to SAFER. Comparisons will be made between previous baselines, regulatory milestones, and comparable projects and the actual expenditures on the projects. These kinds of measurements cannot be made until action that SAFER was applied to is complete.

Qualitative measurements are how the participants view the SAFER process and their satisfaction with how it was applied. These measures come from interviewing the participants of each pilot project, including representatives from DOE, the DOE contractor, any subcontractors, State regulators, and EPA. A site-specific survey instrument is developed for each hosting facility, and all interviews are conducted by the same two interviewers to maintain consistency. Through December, only the interviews at the Mound Plant had been completed. Interviews at ORNL and SRS have started and these will be completed in January 1995. Hanford interviews will be conducted in the spring of 1995. Notable measures of success through December 1994 are:

Mound: B Building Solvent Storage Shed - the uncertainty analysis eliminated the planned site characterization, resulting in a 40 percent cost savings and completion of the project 6 months earlier than planned.

Mound: Area 7 Soil Removal - the uncertainty analysis eliminated the planned site characterization, saving close to $300,000 and implementing the field response earlier than planned. Mound personnel felt that the technical quality of the work plan was significantly improved with the SAFER input.

Mound: New Property Transfer - developed decision rule logic that demonstrated the sufficiency of available data to make the decision that the property was "clean" and could be transferred; no additional data were required.

ORNL: WAG 1 Surface Impoundments - (1) developed a focused work plan, (2) facilitated early field work, (3) RI field work and RI/FS report will be completed within 7 quarters, half of the time originally estimated, resulting in significant savings, and (4) State regulator attendance in project meetings has been extended formally to all restoration projects at ORNL.

SRS: F- and H-Area Retention Basins - (1) facilitated the use of innovative technologies, resulting in reduced sampling locations and avoided costs of $750,000; (2) State regulator participation increased State confidence in using RESRAD modeling to limit the contaminants of concern and avoided $450,000 in additional analytical costs; and (3) State and EPA regulators attending monthly scoping meetings on all SRS restoration projects.

SRS: D-Area Oil Seepage Basin - (1) the conceptual model effectively focused the RI on the impact on groundwater and recognized the minimal potential for deeply buried soils to directly contact human receptors, thereby limiting human health risk assessment to the top four feet of soil and (2) decision rule logic produced a bifurcated soil sampling plan that considers the risk posed by the vadose zone (top 4 feet of soil) differently than the risk posed by the saturated zone; this is expected to result in substantial cost and time savings in the remedial action phase of the project.

CONCLUSIONS

Although the SAFER Pilot Projects are only 75 percent through the allotted time, significant accomplishments can be credited to applying SAFER. At each site, there

has been nearly universal acceptance and enthusiasm for the SAFER process by DOE, contractors, and regulatory stakeholders. SAFER has changed basic ways of doing business at these sites; for example, at ORNL and SRS regulator involvement has been formalized in scoping meeting attendance much earlier in the process than in the past. Gone are the days of submitting a work plan that is completely new material to the regulators and waiting for comments. Now, before the work plan is drafted, the site team and the regulators have jointly looked at what is known about the site and made some preliminary decisions about how the site should be approached. Another example of change in practice is at Mound where removal actions will not automatically be scheduled and costed for a site characterization; instead an uncertainty analysis will be done to determine the need for further characterization.

What DOE hopes to demonstrate with the SAFER pilot projects is that the SAFER process should be the standard way of doing environmental restoration business. When it does, the SAFER acronym can disappear from use, because there will be no difference between environmental restoration and SAFER. When that comes to pass, the ultimate goal of the SAFER Pilot Projects will have been realized.

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DEVELOPMENT OF A PHASED APPROACH STRATEGY FOR REMEDIATION

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ABSTRACT

The comprehensive Remedial Investigation (RI) Feasibility Study (FS) Remedial Design (RD) and Remedial Action (RA) is not the only route through which site remediation can be achieved. A phased approach to remediation integrates early actions and final actions to achieve early cleanup of selected site problems while at the same time contributing to the most logical and efficient solution for achieving remediation at an Operable Unit (OU). By aggressively planning and implementing early cleanup opportunities offered by CERCLA removal and remedial authorities, a phased approach can expedite remediation, reduce risks, reduce costs, demonstrate progress, and respond to stakeholder and other priorities while moving the OU most quickly to final remediation. However, early actions have not been widespread because their

usefulness and advantages are not fully understood and methods for combining these actions into a phased approach have not been articulated.
Since a comprehensive RI/FS culminating in a final Record of Decision (ROD) is essentially the only process by which a site can be removed from the National Priorities List (NPL), phased approach actions must be integrated with final cleanup. This is accomplished through development of a phased approach strategy and a consensus memorandum. A phased approach strategy guides application of a phased approach. Basic elements of the phased approach strategy include identification of the site problems and specific candidates for early actions. A consensus memorandum is based on the phased approach strategy and actually initiates an early action. A consensus memorandum is a brief statement of intent that describes specific scope and approach for each early action. Both a phased approach strategy and consensus memorandum are developed jointly by the Department of Energy (DOE), Environmental Protection Agency (EPA), and State agencies at the OU level.
DOE has developed a Phased Approach/Early Action Guidance (in draft) to support use of a phased approach for remediation. DOE based the guidance on current EPA and DOE guidance and initiatives. The purpose of this paper is to provide the basis for DOE's Phased Approach, identify advantages of using a phased approach, describe the relationship of DOE's phased approach to other EPA/DOE streamlining initiatives, and describe development and implementation of a phased approach strategy
PHASED APPROACH BASIS
The use of early actions during environmental restoration is not a new concept. Statutory authority is provided in the Comprehensive Environmental Response Compensation and Liability Act (CERCLA) that gives considerable discretion is determining appropriate actions for quickly reducing risks. In addition, EPA and DOE have established policies that encourage the use of early actions. The National Oil and Hazardous Substances Pollution Contingency Plan's (NCP's) bias for action and the discussions found in EPA guidance (EPA 1993) contain extensive direction for expediting cleanups and maximizing reduction in risks to human health and the environment. The DOE's RI/FS guidance (DOE 1994) also describes how early actions can be taken during and RI/FS.
CERCLA Authority
Authorization for early actions comes from two sections of CERCLA: removal authority from Section 104 and remedial authority from Section 106. The distinction between these two authorities is not greatly significant in the DOE Environmental Restoration (ER) program since DOE is the lead agency the same DOE funds are used regardless of which authority is selected and the same EPA and State regulatory staff are generally involved in the oversight function. However, a necessary step under CERCLA is to declare which authority is being used. Many site problems can be addressed by either authority, although some Federal Facilities Agreements (FFAs) may restrict this flexibility. Procedural and documentation differences exist depending on which authority is used. Where flexibility among authorities exists, site managers need to consider the advantages of each authority in making decisions.
NCP Support
The regulations promulgated by EPA in the NCP state that sites should be remediated using early actions as necessary or appropriate to achieve significant risk reduction quickly, streamline analysis commensurate with the scope and complexity of the problem being addressed, or expedite the completion of total site cleanup. Using this bias for action and streamlining principles help effectively manage site problems and expedite the reduction of risk posed by the site. EPA links the phasing of actions as part of the ongoing strategic planning at a given site with the ultimate goal of implementing final remedies. Phasing actions support the strategic planning process by helping determine the types of actions and analyses necessary or appropriate and the optimal timing of those actions.
EPA Guidance
A cornerstone of the EPA's streamlining philosophy is an approach described in Guidance for Evaluating Technical Impracticability of Ground Water Remediation (OSWER Directive 9234.2-24) and Considerations in Ground-Water Remediation at Superfund Sites and RCRA Facilities (OSWER Directive 9283.1-06). These documents encourage "early actions to control plume migration and remove contaminant sources, reducing risks and providing information useful in identifying the restoration potential of the site." EPA also notes that "phasing of activities does not lengthen or deter the remediation process; rather if approached properly, phasing of

activities should expedite the process by reducing risk and by bringing final cleanup levels closer to completion of the RI/FS." DOE Guidance
The DOE's RI/FS Guidance (1993) extended the "bias for action" encouraged in the NCP to RI/FS activities. Specifically, the guidance addresses the using limited field investigations (LFIs) to support early actions. The Phased Approach/Early Action (draft) guidance now extends the use early actions from the discussion in the DOE RI/FS Guidance.
PHASED APPROACH ADVANTAGES
Several advantages of using a phases approach have already been mentioned such as expediting the process, reducing risks promptly, and moving final cleanup levels closer to completion of the RI/FS (i.e., demonstrate progress). Other advantages include reducing costs and responding to stakeholder and other priorities. This section further describes how these benefits are realized by using the phased approach and may be used a basis for measuring success of the approach.
Expedite Action
A phased approach can result in actions that overall are quicker and more efficient (thereby expediting the process in two ways). Implementing actions quickly reduces the time to affect risk reduction and lower overall restoration costs. More efficient use of resources (e.g., less data collection, less alternatives development, and better tailoring of the action to the site problems) also allows the final RI/FS to be focused on the more complex problems that remain after the early actions are completed.
A phased approach also emphasizes opportunities for parallel or concurrent conduct of several activities that are historically carried out sequentially (e.g., investigation, decision, design). A particular emphasis of the phased approach is to complete preliminary remedial design documents during planning. By using data and documents to serve multiple purposes, a phased approach can reduce the overall time needed to move through an investigation and begin actual remediation.
Reduce Risks
Early actions can limit exposure and halt migration of contamination quicker than comprehensive RI/FS/RD/RA approaches. This directly supports the main intent of CERCLA and the NCP. Early protectiveness is usually the strongest justification for developing or implementing a phased approach strategy. A phased approach should always identify opportunities for early risk reduction.
An additional advantage of a phased approach exists where final technologies for a site problem are not yet available but where wastes pose a current (or near-term future) risk to the environment, workers, or other receptors. DOE faces many such situations (e.g., Weldon Spring Quarry). Where such situations exist (e.g., surficial radiation hot spots), a final ROD would likely require delay of a solution until the technology was developed or constructed or until a final waste management decision was made. The phased approach facilitates interim risk reduction through such activities as removal and storage. Although a phased approach may result in implementability issues such as the need for interim storage capacity, and may result in additional Operations and Maintenance (O&M) costs to maintain compliance, careful definition of the problems to be addressed with an early action (e.g., a priori agreements about the definitions of contaminant concentrations that constitute hot spots or unacceptable risks and the amount of wastes that will be excavated) can offset these problems and improve the overall effectiveness of the entire environmental restoration program for the OU.
Reduce Costs
A phased approach leads to cost reduction similar to the ways in which it contributes to expediting an action. Three primary methods exist. First, a phased approach leads to better focused studies of reduced scope (i.e., not final) that generally require fewer data to support decisions. Second, by selecting the most appropriate authority, actions are commensurate with the complexity of the problem. That is, a comprehensive RI/FS is not needed to select a remedy for a problem with a relatively obvious solution. Third, by allowing for the concurrent preparation of remedial design documents, the overall amount of time for preparing documents and conducting actions can be reduced. This results in lower overall program costs.
Demonstrate Progress
Phased approaches show earlier progress (e.g., implementing actions) to the stakeholders than the comprehensive CERCLA approach because they result in the early cleanup of actual problems. Demonstrating early progress alone may be enough reason

to implement a phased approach. If progress can be coupled with the other advantages (e.g., early risk reduction), a clear benefit of a phased approach is the ability to accomplish needed cleanup earlier than planned.

Another advantage of a phased approach is building momentum leading to an improved overall process for conducting environmental restoration. Even small accomplishments achieved under a phased approach can build momentum for additional progress, in many instances leading to a new or even more logical approach to addressing whole OU or facility-wide environmental remediation challenges.

A phased approach also can show progress by providing an avenue for testing new techniques, management approaches, or even technologies. CERCLA already encourages the use of treatability studies for trying new technologies. In some instances, a treatability study can be incorporated as part of a phased approach. If a technology proves useful and effective, the phased approach also provides a forum for continuing the technology (as an early action) before a final remediation decision is made.

Respond to stakeholder and Other Priorities

Integrating stakeholder concerns and incorporating new information learned during a phased approach may lead to changes in the priorities for addressing site problems. A phased approach process provides a forum for responding to stakeholder concerns. For example, on the basis of stakeholder concerns, several DOE sites have made significant changes in the priority given under original plans to remediate certain site problems most amenable to early transfer for public or other non-DOE uses. A phased approach provides the flexibility to address these changing priorities quickly and efficiently.

RELATIONSHIP TO EPA/DOE STREAMLINING INITIATIVES

Several streamlining initiatives have been developed that demonstrate EPA and DOE's commitment to achieve cleanup with the most practical, efficient, and quickest means possible. These initiatives provide structured approaches to achieve early risk reduction and streamline the process. Two of these initiatives, Superfund Accelerated Cleanup Model (SACM) and Resource Conservation and Recovery Act (RCRA) Stabilization Initiative were developed by EPA. A third, the Streamlined Approach for Environmental Restoration was developed and is being used by DOE.

SACM

EPA has initiated a CERCLA streamlining initiative, the Superfund Accelerated Cleanup Model (SACM) program, which utilizes removal authorities at remedial sites to achieve earlier risk reduction and to increase the efficiency of actions. Principles of the SACM program are as follows:

- Provide an ongoing process for evaluating site-specific conditions and need for action.
- Allow for cross-program coordination of response planning.
- Facilitate prompt risk reduction through early action.
- Ensure appropriate cleanup of long-tern environmental problems.
- Ensure early public notification and participation.
- Define conditions where removal actions are appropriate.

SACM principles are met through the phased approach developed by DOE. For example, DOE's phased approach emphasizes prompt risk reduction through early action, public notification and participation, and defining conditions where removal actions are appropriate.

RCRA Stabilization Initiative

EPA has developed a stabilization initiative that is similar to SACM but relies on different statutory and regulatory preferences for action. Stabilization initiatives generally rely on well-understood technologies to limit the migration of contaminants, to reduce immediate threats, and to contribute to understanding the range of existing problems. DOE is preparing a separate guidance on developing a phased approach for RCRA actions, however, principles discussed in this paper are generally applicable for phasing RCRA actions as well.

SAFER

DOE has developed the Streamlined Approach for Environmental Restoration (SAFER) as its own streamlining initiative. SAFER specifically addresses management of uncertainties during remediation at DOE sites. SAFER uses the concepts of probable conditions, deviations, decision, and contingency plans to achieve enhanced planning, focused investigations, and aggressive alternative assessment and remedy selections. SAFER streamlines the CERCLA process by providing a framework to

optimize the management of uncertainty through the use of a conceptual model, identification of site problems, incorporating new information as it is learned, and integrating the extended project team in decisions about characterization and remediation. SAFER's tenets are fully integrated into DOE's phased approach.

PHASED APPROACH STRATEGY DEVELOPMENT AND IMPLEMENTATION

Once the commitment to using early actions and developing streamlining opportunities is made, a plan for identifying and implementing those opportunities must be developed. Ideally, this would be a prospective process that begins once a site is listed on the NPL and continues through the RI/FS, the remedy selection process, remedial design, remedial action, to deletion from the NPL. Even if opportunities have been missed, advantages can still be realized by integrating phased approach principles into what has already been accomplished.

A strategy is necessary to ensure a coordinated approach is taken by DOE, EPA, and the State. This common vision of what and how problems will be addressed ensure proper phased analysis and response is taken that doesn't preclude implementation of the expected final remedy. Specifically, a phased approach strategy (see strategy outline in Table I) should contain the following elements:

A statement identifying site problems included in the OU and briefly summarizing what is known about each (i.e. site problems).

A description of which site problems will be addressed using early actions and which will be addressed through an RI/FS and final record of decision (ROD) (i.e. candidates for early action).

A list of the type of early action (e.g. time-critical removal, early remedial action) that will be used to address each site problem amenable to early action and authority (i.e. removal or remedial).

The primary objectives that each early action will achieve (i.e. strategic objectives).

Brief text explaining the rationale for each assignment (i.e. consensus of the phased approach strategy).

A preliminary schedule, through the final remedial action(s) (i.e. implementing an action).

Each of these elements is discussed in this section.

Table I

Site Problems

Understanding of site problems will vary depending on how much information is currently available. Regardless of current understanding, definition of site problems is necessary to develop a phased approach around an agreed upon list of problems being addressed within the OU. Description of the problems must be technically based commensurate with available information. Then as increased understanding of the OU is achieved, new information can be technically linked to refine problem definitions.

In general, site problems are discrete aspects of an OU that may require remediation. Problems should be definable in terms of an environmental medium, geographic features, the types of waste present or suspected, or the types of waste units that exist. Identification of problems should begin by analyzing existing site data, including results of any removal actions, preliminary assessment and site investigations, and the NPL listing process. Consistent with the phased approach principles advocated by EPA and DOE, additional data collection and analysis should be focused and limited to narrow site problems, accelerate response actions, and identify additional information needed to scope complex problems.

The most logical place to begin identifying phasing opportunities is with the conceptual site model (see Fig. 1). This model is developed as soon as initial site understanding is achieved and revised as further understanding is gained. The conceptual site model is the primary tool for presenting the known or suspected source-pathway-receptor connections. Since site problems are most often developed in terms of sources and pathways and these are the most appropriate for taking early actions (e.g., by removing or containing a source or by eliminating a pathway).

Fig. 1.

Candidates for Early Action

Working from the agreed upon list of problems for an OU, several factors must be applied to help determine whether or not early actions can be successfully implemented. These factors include the early actions appropriateness, feasibility, and potential for achieving benefits. Laying out potential early actions identified

in the conceptual site model and balancing them against all these factors will help prioritize where to begin taking early action.
Basic criteria for determining when early actions are appropriate include complexity, urgency, and potential responses. Site problems that are less complex and straightforward do not justify development of a full range of alternatives. These obvious solutions may create opportunities to take actions quickly affording significant risk reduction. When the need to protect human health and the environment outweighs the need to examine all alternatives, prompt action can usually be justified. Finally, when the choice of alternatives is limited by requirements, program precedence, guidance, or practicality, the remedies can be implemented without studying the few remaining alternatives in detail.
Sufficient site understanding is required to understand how site conditions and configurations influence selection of early actions. Certain situations exist when the ability to perform early actions is impossible. For example, an early action to remove drums may be infeasible because adequate storage, treatment, or disposal capacity is not readily available. This emphasizes the need for thorough planning to ensure time and resources are not wasted pursuing infeasible early actions.
Ultimately, the ability to demonstrate the success of early actions by achieving ER benefits will provide the strongest most quantifiable rationale for taking early action (e.g., expediting actions, reducing risks, achieving remediation progress quicker, reducing costs, and responding to stakeholder priorities). By balancing all these benefits among potential early actions, a means for prioritizing where to begin taking early action can be developed based on greatest chances for success.
Type of Action/Authority
As discussed in the phased approach basis section, a necessary step under CERCLA is to declare which authority is being used to justify an early action. The actions that can be taken under CERCLA removal (104) authority include emergency, time-critical, and non-time-critical removal actions. The actions that can be taken under CERCLA remedial (106) authority include early, interim, and final remedial actions. Other variations exist among EPA regions and Federal Facility Agreements (FFA) (e.g., expedited response action, removal) in how these actions are termed or can be used. However, differences should not prohibit a broad use of the phased approach.
These various authorities can be placed into two groups with the exception of final remedial actions which must be addressed through the complete RI/FS process. The first group, emergency and time-critical removal actions, generally focus on the critical interventions need to address imminent threats. Actions within this group can be taken with minimal time for investigation, public involvement, evaluation of alternatives, and design. The second group, non-time critical removal actions and early/interim remedial actions, require near term action due to risks that warrant response prior to a detailed investigation. These actions should be most amenable to remediation, unhampered by lack of site information, least encumbered by requirements, and avoid controversial remediation decisions. This group requires a more complex sequence of planning and decision activities and can allow for more investigation, planning, documentation, and stakeholder involvement.
From a practical standpoint, the applicable authority will not significantly affect the opportunities to streamline, reduce risk, save time, and save money. The differences are related to administrative requirement (e.g., documentation required) and/or procedural (e.g., requirement to comply with applicable or relevant and appropriate requirements (ARARs)). Decision on what authority to use should be based on the advantages of each authority. Again, this relies on sound strategic planning to understand the influences of these decision on all site problems, candidates for early actions, and overall benefits that can be achieved.
Strategic Objectives
Early actions are appropriate where the need for action is obvious and the nature of the required action is relatively clear. Because of this, objectives for early actions can be established more clearly that is generally possible at the beginning of a comprehensive RI/FS. These objectives should specify contaminants and media of concern, potential exposure pathways and receptors, remediation goals, and contribution to the overall remedy.
Early actions are not required to achieve final remediation levels, achieve all ARARs, or even provide final protectiveness of human health and the environment. Therefore, early action strategic objections should detail site problems to be

addressed and the actions to be taken as well as the problems that will not be addressed. This will provide focus for planning early actions and evaluating the remaining actions required for final remediation.

Consensus on the Phased Approach Strategy

Due to differing interests in remediating a site among the various stakeholders, agreement on the tentative actions and objectives much be reached. Any unresolved issues can weaken the strategy and eventually result in delays or abandonment of early action efforts. This consensus should be worked out in three phases:

- Confirming that an action is required, deciding on the specific scope, and committing to begin early stages of the remedy design process.
- Agreeing on how to ensure early actions are consistent with future, full-scale remedial actions.
- Reaching common understanding and resolution of regulatory and other issues that could hinder or prevent early action.

By design early actions focus on specific activities that can be taken immediately to mitigate obvious and dangerous threats. In order for coordinated work to begin among all involved parties, they must commit to begin acting on a specific scope of work. In the DOE Phased Approach Guidance, this consensus is documented in a consensus memorandum (see outline in Table II) . After this consensus is reached, agreement on design and implementation issues must be worked out as follows:

- Appropriateness of a phased approach for the OU.
- Identification of some site problems that are candidates for early actions.
- Identification of site problems deferred to a comprehensive RI/FS/RD/RA.
- Identification of the steps and schedule for implementing a phased approach.

Table II

An early action must contribute to the overall objectives of the comprehensive RI/FS. The NCP also requires early actions to be consistent with the final remedial action. Therefore, consensus should also be obtained on:

- Whether the early action will interfere with any future, full-scale remedial actions.
- Where potential interferences might occur, the risks, and how they can be avoided or mitigated.
- The follow-up actions needed as part of the comprehensive RI/FS to prepare a final ROD.

Understanding and agreement between all involved parties must be reached on issues fundamental to implementing an early action. Resolution of these issues will drive what needs to be accomplished, operating conditions, and approaches. Consensus must be reached on at least the following issues:

- Interim cleanup levels.
- ARARs and waivers.
- Risk assessment methodology.
- Use of innovative technologies.
- Treatment/storage/disposal of remediation-derived wastes
- Land use
- Use of institutional controls

SUMMARY

Use of early and final actions in an integrated, phased approach offers several advantages to DOE facilities implementing environmental restoration activities under CERCLA. These advantages include expediting progress, reducing costs, reducing risks, demonstrating progress, and responding to stakeholder priorities. Development of a phased approach strategy with regulators is critical to effective implementation of a phased approach. DOE is finalizing guidance on developing a phased approach strategy and early actions that will be available in mid-1995. Similar approaches are appropriate for RCRA corrective actions.

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ACCELERATING RCRA CORRECTIVE ACTION: THE PRINCIPLES OF THE DOE APPROACH*

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ABSTRACT

The U.S. Department of Energy (DOE) is involved in the remediation of environmental contamination at many of its facilities under the Resource Conservation and Recovery Act (RCRA). RCRA's corrective action provisions were established by the Hazardous and Solid Waste Amendments of 1984 (HSWA). These provisions provided a broad mandate for the U.S. Environmental Protection Agency (EPA) and the States to require corrective action at hazardous waste treatment, storage, and disposal facilities (TSDFs). The goal of RCRA corrective action is the cleanup of releases of hazardous waste or hazardous constituents from solid waste management units (SWMUs) at TSDFs. In response to the HSWA mandate, EPA established a program for the conduct of RCRA corrective action that was similar to that established under the Comprehensive Environmental Response Compensation and Liability Act (CERCLA). In addition, EPA developed and implemented its "stabilization" initiative as a means of quickly addressing immediate risks posed by releases until long-term solutions can be applied. Progress toward environmental restoration under the RCRA corrective action program has improved since EPA implemented its stabilization initiative, but remains slow.

To improve the efficiency of environmental restoration at its facilities, DOE is developing guidance and training programs on accelerated environmental restoration under RCRA. A RCRA guidance document, entitled "Accelerating RCRA Corrective Action at DOE Facilities," is currently being developed by DOE's Office of Environmental Policy and Assistance. The new guidance document will outline a decision-making process for determining if acceleration is appropriate for individual facilities, for identifying, evaluating, and selecting options for program acceleration, and for implementing selected acceleration options. The document will also discuss management and planning strategies that provide a firm foundation for accelerating RCRA corrective action. These strategies include a number of very basic principles that have proven effective at DOE and other federal facilities, as well as some new approaches.

The purpose of this paper is to introduce DOE's new guidance document, discuss the general approach presented in the guidance for accelerating RCRA corrective action, and to emphasize some of the more important principles of effective management and planning.

INTRODUCTION

The U.S. Department of Energy (DOE or Department) is currently involved in the remediation of environmental contamination at many of its facilities under the Resource Conservation and Recovery Act (RCRA). RCRA's corrective action provisions were established by the Hazardous and Solid Waste Amendments of 1984, commonly referred to as HSWA. HSWA provided a broad mandate for the U.S. Environmental Protection Agency (EPA) and the States to require corrective action at hazardous waste treatment, storage, and disposal facilities (TSDFs). The goal of RCRA corrective action is the cleanup of releases of hazardous waste or hazardous constituents from solid waste management units (SWMUs) at TSDFs. The term SWMU includes both units used to manage hazardous waste and units used to manage nonhazardous waste.

In response to the HSWA mandate, EPA established a program for the conduct of RCRA corrective action that was similar to that established under the Comprehensive Environmental Response Compensation and Liability Act (CERCLA). The RCRA program was initially implemented through a series of guidance documents. Then, on July 27, 1990, EPA issued a proposed rule that outlined the RCRA corrective action program (55 FR 30798). Although a portion of the proposed rule has been finalized

[Corrective Action Management Units (CAMUs) and Temporary Units (TUs) - 58 FR 8658, February 16, 1993], the bulk of the rule remains proposed and EPA continues to use the proposed rule as guidance. In addition, in 1991, EPA implemented its "stabilization" initiative, as a means of taking interim measures at RCRA sites to address immediate risks posed by releases until long-term solutions can be applied (U.S. EPA. Managing the Corrective Action Program for Environmental Results: The RCRA Facility Stabilization Effort. October 25, 1991).

Overall, EPA estimates that about 100,000 SWMUs at 5,800 facilities are potentially subject to RCRA corrective action requirements, and that 15,000 SWMUs at 2,600 of the 5,800 facilities may actually require remediation (Draft Regulatory Impact Analysis. March, 1993). Current sources conflict regarding the number of SWMUs that have actually been remediated to date, partly because much of the progress that has been accomplished is due to stabilization activities. It is clear, however, that progress toward environmental restoration under RCRA corrective action is slow.

To improve the efficiency of environmental restoration at its facilities, DOE has placed environmental restoration initiatives on a "fast-track." DOE continues to emphasize taking remedial action based on sound scientific and engineering data, but is placing a new focus on accelerated, cost-effective remediation, in partnership between the Department and stakeholders. In achievement of its mandate to provide relevant and effective environmental guidance to Departmental program and field elements, the DOE Office of Environmental Policy and Assistance (EH-23), is developing guidance and training programs on accelerated environmental restoration under RCRA and CERCLA. A RCRA guidance document, entitled "Accelerating RCRA Corrective Action at DOE Facilities," is under development.

One key element of DOE's approach to accelerating RCRA corrective action that is being incorporated into the new guidance document is effective management and planning. Effective management and planning is a prerequisite for compliance with any complex regulatory program, but is especially important for environmental restoration. The new guidance document will identify management and planning strategies that will help to provide a firm foundation for accelerating RCRA corrective action. These strategies include a number of very basic principles that have proven effective at DOE and other facilities, as well as some new approaches. The purpose of this paper is to introduce DOE's new guidance document, discuss the general approach presented in the guidance for accelerating RCRA corrective action, and to emphasize some of the more important principles of effective management and planning.

WHAT IS ACCELERATED RCRA CORRECTIVE ACTION

In order to define what accelerated RCRA corrective action is, it is first necessary to examine the conventional RCRA corrective action process. The conventional RCRA corrective action process is depicted in Fig. 1 (adapted from EPA's May 31, 1994 RCRA Corrective Action Plan - Directive No. 9902.3-2A). The process consists of four basic phases; the RCRA Facility Assessment (RFA), the RCRA Facility Investigation (RFI), the Corrective Measures Study (CMS), and Corrective Measures Implementation (CMI). During the RFA, SWMUs are identified and the potential for release determined. RFAs at most TSDFs were completed during the late 1980's and early 1990's. RFA's are typically conducted by the regulator; the subsequent phases are conducted by the facility with regulator oversight. If SWMUs are determined to have the potential for a release, the requirement to conduct an RFI is incorporated into the facility's RCRA permit or into an enforceable order, and the facility conducts the RFI. A CMS is then conducted to evaluate remedial technologies for those releases determined to be significant. The requirement to conduct remediation is then incorporated into the facility's RCRA permit (or enforceable order) and implemented during the CMI. Corrective action ends at a facility when remediation has been completed at all SWMUs, and is also implemented through the RCRA permit or order. Finally, interim measures, which include primarily short-term actions intended to stabilize releases, may be implemented at any time during the process, but are most likely to be implemented prior to or during the RFI phase.

For a typical small or medium sized facility, it may take up to five years or more to reach the CMI phase. For larger facilities, including many DOE facilities, the time required to reach the CMI phase can be in excess of ten years or more. While much of this time is spent meeting requirements as spelled out in the statute and regulations, and coordinating activities with other laws, primarily CERCLA, there exists a number of opportunities for accelerating the process. The main objective of

the new DOE guidance being developed is to outline a decision-making process for determining if acceleration is appropriate for individual facilities, for identifying, evaluating, and selecting options for program acceleration, and for implementing selected acceleration options.

Acceleration for DOE facilities is being examined with two major objectives. DOE's first objective is to reduce risks and prevent further release migration in a faster time frame, pending long-term solutions. This objective is comparable to EPA's stabilization initiative. The second objective is to complete remediation at all SWMUs on a facility-wide basis, also in a faster time frame. Once remediation is completed at the facility, the regulator would proceed with a permit or order modification, resulting in a release from RCRA corrective action requirements.

APPROACH TO ACCELERATING RCRA CORRECTIVE ACTION

An overview of the process being developed by DOE to achieve acceleration is depicted in Fig. 2. Most facilities are in the RFI or CMS phase and many have implemented interim measures at individual SWMUs. Therefore, the new guidance document will present a decision-making process that considers where the facility is in the corrective action process at the time accelerated alternatives are being considered.

Fig. 1.

Fig. 2.

The decision-making process under development within the new guidance will consist of four basic steps. In the first step, the facility will determine if acceleration is feasible at the facility at that particular point in time. Factors to be considered here include the status of permits, orders or agreements, relationship with the regulator and stakeholders, and funding and resources. For example, if a permit requiring corrective action was recently issued, it may be advisable to refrain from suggesting changes in requirements so soon.

If acceleration is determined to be an option, the facility will examine its operations and makes changes as necessary to facilitate identification, evaluation, selection, and implementation of acceleration alternatives. The facility also examines facility-wide considerations, such as waste management capacity and need for new units. This is the second step of the process. It is at this point where the principles of effective management and planning, which are the focus of this paper, are examined. More information on these principles is provided below.

During the third step, specific options for acceleration are examined and selected for individual SWMUs or SWMU groupings. These options will be referred to within the new guidance as "tools" for accelerating RCRA corrective action. Tools that may be applicable are dependent on the phase of RCRA corrective action that the SWMU or SWMU grouping is currently in. Referring back to Fig. 1, for example, many SWMUs at DOE facilities are currently in the RFI stage. Tools that may be considered during the RFI stage to help accelerate the process include streamlining data collection activities, conducting a preliminary or focused CMS, and conducting voluntary corrective action, among others. The new guidance will provide facilities with a systematic process for identifying, evaluating, and selecting options for program acceleration at SWMUs or SWMU groupings.

During the fourth step of the process, selected acceleration alternatives are implemented, monitored, and completed. The new guidance also discusses interaction with other laws and includes many examples illustrating application of the principles, as well as the tools. The guidance is currently under development and assuming that resources remain available, is expected to be released by the end of the calendar year.

PRINCIPLES OF EFFECTIVE MANAGEMENT AND PLANNING FOR RCRA CORRECTIVE ACTION

Effective management and planning is a prerequisite for compliance with any complex regulatory program. While there are certain principles of effective management and planning that are recommended components of an adequate environmental restoration program, these principles become essential if the facility is to be successful in accelerating RCRA corrective action. These include a number of very basic principles that have proven effective at DOE and other federal facilities. These principles will be discussed within the new DOE guidance, and as indicated above, are examined as part of the second step of the acceleration process. The most important of these principles are summarized below, with specific reference to how they can be used to promote acceleration.

Prevent Releases/Minimize Waste - If future releases can be prevented, facilities

can devote resources to addressing past releases. The most important principle of accelerating RCRA corrective action is to prevent future releases. The prevention principle pretty much goes without saying with respect to new releases. What is not so obvious is preventing further releases at existing SWMUs. Release prevention will be especially important as facilities proceed under RCRA corrective action, from investigation to remediation. Active remediation provides many opportunities for additional releases. Examples include overflow of drums used to collect contaminated groundwater, leaking pipes from ground-water extraction and treatment systems, and transfer of contaminated soil from one area to another from improperly cleaned remediation equipment. Facilities should develop proactive programs to: 1) Prevent future releases; 2) Detect releases quickly if they do occur; 3) Take corrective action quickly to stop the release and prevent (further) release migration and conduct cleanup; and 4) Continually monitor the effectiveness of ongoing prevention programs and modify these programs when they are not working.

An important facet of release prevention is waste minimization. Simply put, the less waste created, the fewer opportunities for releases and environmental contamination. During RCRA corrective action activities, waste minimization can be practiced by: 1) Carefully delineating areas requiring active remediation; 2) Segregating hazardous waste from non-hazardous (solid) waste; 3) Treating or otherwise managing contaminated media in-situ, where possible; and 4) Judiciously using water or detergents to rinse equipment, such as drilling rigs.

Determine Regulator and Facility Priorities - Because most RCRA corrective action activities must be approved or imposed at TSDFs through the RCRA permit or an enforceable order, the priorities of the regulator can strongly influence the feasibility of accelerating corrective action at a particular facility. Therefore, DOE should assess regulatory agency and facility priority as they examine their prospects for acceleration. By targeting scarce resources to address the most pressing problems, overall risk reduction at facilities can be achieved in a faster time frame.

Regulatory agency resources are not available to take immediate corrective action at all of the facilities that warrant corrective action. In recognition of this fact, in 1992 EPA established the National Corrective Action Prioritization System (NCAPS). NCAPS is a computer program that EPA uses to rank facilities for corrective action purposes. Using NCAPS, EPA evaluates SWMUs at a facility for actual and potential releases. A facility ranking of high, medium, or low is then assigned by EPA based on an estimation of the environmental benefit that would be realized from accelerated cleanup. NCAPS is intended to be a life-cycle prioritization system, where facilities are continually re-assessed to determine priority. As of the end of calendar year 1993, EPA had determined the priority for corrective action for most major RCRA facilities. Nearly 40 percent of all facilities were ranked by EPA as high priority; approximately 30 percent were ranked as medium priority, and 30 percent were ranked as low priority.

The new guidance document will encourage DOE facilities to learn their NCAPS ranking and EPA's justification for the ranking, including the contribution of individual SWMUs to the ranking. In most cases, the NCAPS ranking of a facility can be learned by contacting the regulator. This information is important for determining if acceleration of corrective action is feasible on a facility-wide basis, and for prioritizing individual units for action.

Generally, facilities with high NCAPS rankings will be the first to receive attention from the regulator. In this case, it is likely that the regulator will have initiated development of permit conditions or an enforceable order, and in some cases, permit conditions or an enforceable order may already be in-place. There may be less of an incentive for the regulator to consider other alternatives in this situation because of limited resources. Nevertheless, viable acceleration alternatives can be proposed to the regulator, especially if the alternative would be expected to achieve risk reduction in a shorter time frame. High-ranked facilities are therefore encouraged to identify the options available for accelerating RCRA corrective action, and to work with regulators and stakeholders to implement feasible options, even if permit conditions or orders are under development or already in place. More important, if a permit or order has not been established, facilities can suggest accelerated alternatives during the negotiation period. In addition, facilities can work with the regulator to structure the permit or order to maintain flexibility, such that better ideas do not require a formal

permit or order modification prior to implementation.
Facilities ranked as medium or low priority are not as likely to receive immediate regulator attention. If medium or low priority facilities are already subject to permit conditions or orders, these facilities are likely to experience a reduction in regulatory oversight. At the least, the schedule for completion of milestones, such as RFI work plans, may be drawn-out over a longer period of time. At these facilities, and especially at facilities that do not have permit conditions or orders in place, there may be more opportunities for undertaking specific types of accelerated actions, such as voluntary corrective actions. In fact, EPA is encouraging lower-ranked facilities to consider voluntary actions.
The DOE facility should also consider developing a facility-specific ranking for individual SWMUs or for SWMU groupings. Developing a facility-specific ranking is important, because, like EPA, DOE also lacks the resources to do everything at once. Equally important, facilities may have reasons that regulators would be reluctant to consider for wanting to take action at certain SWMUs earlier than others. For example, some facilities may wish to remediate certain portions of their facility earlier than others considering future activities (e.g., new construction). Based on the facility-specific priority ranking, the facility can identify acceleration options for implementation at SWMUs or SWMU groupings and discuss feasibility with the regulator and stakeholders.
Understand the Regulatory Program and the Regulator - Understanding the regulatory program and the regulator is one of the keys to successful acceleration of the RCRA corrective action process. Following is an example that illustrates the importance of understanding the regulatory program and the regulator.
In the early days of RCRA corrective action, many facilities were directed by regulators to conduct comprehensive investigations, including analyses for all RCRA 40 CFR Part 261 Appendix VIII constituents. Some facilities resisted conducting unnecessarily broad and costly investigations and negotiated a more reasonable scope for their facility investigations. At the same time, other facilities blindly accepted these requirements, not realizing that there were opportunities for negotiation. It didn't take these facilities long to realize the costs associated with overly broad investigations, however. These facilities then went through a painful process of working with their regulator to modify their permit or order to focus on a more reasonable investigation. Meanwhile, time and resources were wasted. The point is to underscore the importance of understanding the regulatory program. It is important for facilities to understand all the options available to them for conducting and accelerating RCRA corrective action. Most important, facilities must understand that RCRA, and corrective action in particular, is for the most part a negotiated program. While ultimately, the regulator is responsible for issuing permits, orders, or agreements that compel the facility to conduct RCRA corrective action, the facility can influence requirements and schedules through negotiation. Equally important to knowing the regulatory program, is knowing the regulator(s), both organizationally and individually. Each EPA Region and State has its own goals, objectives, policies, and procedures. Hence, the program is inconsistently implemented by the Regions and States; what "works" in one location, will not necessarily work in another. Also, the preference of the individual corrective action official may influence how corrective action is imposed at a particular facility. It is, therefore, important for facilities to understand the regulatory program as well as the individual regulator. Facilities should: 1) Employ or contract with personnel that have experience with the regulator; 2) Learn as much as possible about the program and the regulator through training courses, guidance documents, symposia, conferences, trade journals, peers, etc.; 3) Keep open the lines of communication through status and similar meetings; and 4) Research decisions made regarding other facilities (public and private) located within the Region or State.
Remain Current With New Developments - RCRA in general, and the RCRA corrective action program in particular, are evolving programs. Programs evolve as a result of public opinion, political pressure, new technological developments, court decisions, fiscal limitations, and many other factors. EPA's stabilization initiative, which put new focus on implementing interim measures as a means of early risk reduction, is a prime example of program evolution. The CAMU and TU rule is another good example of program evolution. Knowing that the CAMU and TU rule was in the process of being finalized, many facilities urged their regulator to delay decisions on

corrective measures pending promulgation of this final rule, because of the new waste management options it was expected to provide. More recently, EPA published for comment a draft list of screening levels for 107 of the most common contaminants found in soil (59 FR 67706, December 30, 1994). These levels are concentrations below which cleanup would not be required. Concentrations above these levels mean that further study would be required prior to determining the need for cleanup. This is a particularly important development in the context of corrective action acceleration because many SWMUs may now qualify for determinations of no further action.

RCRA corrective action is particularly vulnerable to evolution because the bulk of the regulations that would formally implement the program have yet to be promulgated. New developments will also occur in related programs that will directly affect RCRA corrective action. For example, the pending CERCLA reauthorization is expected to define risk goals, remedy cleanup standards, consideration of land use, and other issues that will affect RCRA corrective action. RCRA is also up for reauthorization, which provides more opportunity for program evolution.

It is important to remain current with these new developments. DOE facilities should: 1) Designate one or more individuals with the responsibility of maintaining currency with new developments and upcoming events, and communicating with the rest of the organization; 2) Maintain subscriptions with major and local environmental newsletters; 3) Establish contacts with individuals in DOE's Office of Environmental Policy and Assistance; 4) Obtain relevant DOE and EPA guidance materials and attend training programs, conferences and symposia and other information transfer events; 5) Communicate frequently with the regulator; and 6) Communicate frequently with peers, and especially other DOE facilities.

Establish a Good Rapport With the Regulator - Early relations between DOE facilities and their RCRA regulators can be characterized by a mutual distrust and lack of cooperation. DOE facilities have more recently begun to establish better relationships with their regulators. The advantages of establishing a good relationship with regulators are significant. Most important, regulators are more receptive to new ideas and new approaches, such as those put forth in the new DOE guidance being prepared. In addition, a good rapport facilitates the establishment of a team approach to problem-solving, which can be critical when innovative solutions are being examined.

It takes time and a concerted effort to establish a good relationship with the regulator. DOE facilities should: 1) Be open to new ideas and approaches put forth by the regulator; 2) Be cooperative rather than antagonistic; 3) Invite the regulator on frequent site tours to show progress or accomplishments 4) Establish an onsite location for the regulator (larger facilities only); and 5) Hold regular meetings with the regulator where progress and problems can be discussed in an team setting.

Establish a Strong Public Participation Program - DOE management has long recognized the importance of public participation in decision-making associated with environmental restoration activities. In 1991, guidance on this subject was published by DOE's Office of Environmental Guidance, entitled "Public Participation in Environmental Restoration Activities" (DOE/EH-0221, November 1991). Then in October 1992, DOE's Assistant Secretary for Environmental Restoration and Waste Management (EM) issued a statement of policy on public participation, declaring that the overall goal is "to create an open and accessible decision-making process that results in decisions that are technically and economically feasible, environmentally sound, health and safety conscious, address public values and concerns, and can be implemented." This policy statement was followed by DOE's "Public Participation Guidance for Environmental Restoration and Waste Management (March, 1993), which advocated public participation planning.

Due in part to these concerted efforts, many DOE facilities have made significant progress in improving relations with their local communities. DOE facilities are making a genuine effort to educate the public on site remediation activities and to involve the public in actual decision-making. Public participation for accelerated actions needs to be considered, therefore, in light of the facility's existing structure for public participation and community relations. Most DOE facilities will have facility-wide community relations plans for RCRA and CERCLA activities. These plans specify, in general terms, the community relations activities that will be appropriate for environmental restoration activities. However, each specific

accelerated undertaking may have additional needs and/or create additional demands for public involvement. Another activity that the DOE facility should consider, therefore, is conducting a strategic analysis of the need for, and implications of, developing a specific and more detailed public involvement plan as part of the initial strategy for acceleration.
Regulators, the public, and other stakeholders must be provided with the opportunity to take an active role in the decision-making process. However, all parties need to realize first that compromise is not always easily achieved, second that decisions are subject to funding limitations, and most important, that decision-making regarding RCRA corrective action is not a democratic process. While DOE has substantial opportunities for suggesting courses of action, and stakeholders have significant opportunities for input into the decision-making process, ultimately, the decision regarding available options under RCRA lies with the regulatory authority. All parties to the decision need to be flexible to the extent possible and compromise where necessary.
Consolidate SWMUs Into Groupings - Another important principle is to establish (or re-establish) SWMU groupings. SWMU groupings can be established based on a number of factors, including unit type, unit age, unit design, physical (3-dimensional) location, waste type and constituent content, release characteristics, regulator priority, facility priority, relative risk to human health and the environment, and many other factors.
Individual SWMUs should be organized into major categories, subcategories, and even smaller categories if appropriate. In essence, the smallest SWMU grouping should be established based on the least common denominator concept. Only those SWMUs with the same type of problem, the same release characteristics, and that may potentially be subject to the same corrective action (accelerated or not) at the same time, should be in the lowest subcategory.
SWMU consolidation into SWMU groupings is especially useful for large facilities with multifaceted problems at different physical locations, as is the case with many of DOE's larger facilities. DOE facilities under the CERCLA program have been practicing consolidation of contaminated sites into what are called operable units (OUs) for years. Most often, major OU categories are established based on physical location. The contaminated sites within the OUs are then further organized into subcategories based on many of the factors listed above. DOE has taken a site-specific approach to organization of sites into OUs. A site-specific approach to similar organization of SWMUs under the RCRA program should also be taken.
During the process of SWMU consolidation into groupings, facilities are encouraged to critically examine the manner in which SWMUs should be organized. Moreover, if SWMUs are already organized into groupings, which is the more likely case, changes should be suggested if deemed appropriate. Further, SWMU groupings should be established such that future changes that may be appropriate can be implemented without having to modify permits, orders or other agreements.
The advantages of SWMU consolidation into groupings relate to ease of management and facilitated decision-making. In particular, if SWMUs are properly categorized based on the least common denominator concept, accelerated options will be easier to evaluate, easier to implement, and cost less from an administrative standpoint.
Delegate Authority - One of the less obvious but nevertheless important principles of accelerating RCRA corrective action is delegation of authority. The DOE field organization should delegate decision-making to the lowest level possible, preferably to the SWMU or SWMU grouping (operable unit) manager. While decisions that have significant funding or policy implications need to be approved at a fairly high level within the DOE organization (often with headquarters approval and oversight), there are many decisions that can be approved at a much lower level. The less approvals required for any one decision, the less time it will take to implement that decision. Delegation of authority to the lowest level possible can have a significant effect on accelerating the process. In accordance with this principle, DOE facilities should carefully evaluate all the decisions that need to be made during the corrective action process, and categorize these decisions. Each category of decision can then be delegated to the lowest level possible within the DOE organization.
Obtain Expert Assistance - Remediation programs like RCRA corrective action and CERCLA are among the most complex regulatory programs. Successful implementation of these programs requires many different technical and non-technical fields of

expertise. In addition, many facets of RCRA corrective action require interaction with other regulatory programs, primarily CERCLA, but including many other federal laws and state equivalents. It is virtually impossible for any one person, or for even a group of several people, to possess all the knowledge to make RCRA corrective action work. Another important principle of accelerating RCRA corrective action is to obtain expert assistance where necessary.

Perhaps the greatest potential for delay that can be introduced by the facility is the need for re-work. A good example is moving stored remediation waste from one location to another because the original location did not meet regulatory standards. These types of errors, which can also result in substantial fines and other actions, are often caused by the failure to recognize the need for expert assistance. Expert assistance can save time as well as resources. DOE facilities should maintain a standing group or panel of technical and regulatory experts for input into corrective action decisions, work plans, reports, and similar documents. DOE facilities should also seek DOE headquarters input regarding particularly innovative actions and actions that may set precedents.

REVIEW AND CONCLUSION

As the nation progresses more and more towards the goal of doing more with less, federal facilities in particular need to find innovative ways to meet their obligations under the various environmental laws. The new guidance document that DOE is developing for "Accelerating RCRA Corrective Action at DOE Facilities," including both the principles of effective management and planning, and the tools for accelerating RCRA corrective action, will provide some new ideas toward meeting this goal.

Acceleration of corrective action, however, will not always mean lower costs. On the contrary, in many cases where corrective action is accelerated, the costs will increase, at least in the short term. On the other hand, in most cases risk reduction would occur sooner. As accelerated alternatives are examined, each facility will need to evaluate cost and risk reduction associated with specific actions, considering overall available funding. It is clear that tough decisions are in our future.

40-6

EXPEDITING PROGRESS IN THE HANFORD ENVIRONMENTAL RESTORATION PROGRAM

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ABSTRACT

The U.S. Department of Energy, Richland Operations Office (DOE-RL) has adopted a "project" philosophy for planning and executing the Environmental Restoration (ER) Program at Hanford. This approach centers around task specific project teams supported by highly specialized functional groups.

The newly formed Environmental Restoration Contract (ERC) Team has assembled an organization of transitioned employees from existing site contractors and personnel with private sector environmental cleanup experience. Representatives from DOE-RL have also assumed active roles within the ERC Team. Working together, the ERC Team has improved project efficiency, resulting in significant cost savings and shortened schedules.

INTRODUCTION

With the unique challenges and near term implementation of remedial action projects associated with the Hanford ER Program, DOE-RL recognized the need to pursue different approaches for managing and executing the work. Due to the magnitude of the cleanup effort, the uniqueness of the contaminants, and the associated costs, there was a need to ensure that state-of-the-art methods and technologies from environmental cleanup efforts worldwide were considered and employed, as appropriate. Conversely, because of the site's complex waste management history, the institutional memory of the existing site work force was vital for characterizing the site and planning the remedial activities. Coupled with these factors was the desire by the DOE, Congress, and local stakeholders to have a cost efficient process whereby cleanup could be expedited without an abundance of unwarranted studies, permits, and paperwork.

To proceed within this framework, DOE-RL issued an $800 million Environmental

Restoration Contract to a team consisting of four companies recognized for their expertise in environmental cleanup and overall project management: Bechtel Hanford, Inc., CH2M Hill Hanford, Inc., IT Hanford, Inc., and TMA Hanford, Inc. The scope of work included:

Remedial action in the 100, 200, and 300 Areas
Decontamination and decommissioning in the 100, 200, and 300 Areas
Design and construction of a waste disposal facility
Demonstration and implementation of innovative technologies
Deactivation of the N-Reactor Complex
Sitewide support for all ER activities

The contract was awarded in March 1994, and transition (including transfer of the work force from the existing site management and operations contractor) was complete on July 1, 1994. To stress the importance of cost efficiency, the fee for the contract was established as "performance-based" with incentives for cost/schedule improvements.

NEAR TERM CHALLENGES

To effectively execute the scope of work and meet DOE-RL's expectations, the ERC Team faces three major challenges:

Increase Efficiency - Move from operational mode to cleanup (project) practices - Streamline and focus high impact support functions, such as procurement and regulatory analysis
Incorporate Private Sector Knowledge/Experience - Are we doing the right things? - Are we doing things right?
Promote DOE/Contractor Team Concept - Develop common strategies - Encourage mutual "buy-in" to project commitments

PROJECT PHILOSOPHY

To address these challenges, the ERC Team evaluated the life cycle of a typical manufacturing complex. During the initial construction phase, activities at most facilities involve clearly defined, non-repetitive tasks, executed by specialty contractors. During the operations phase, work is usually focused on more repetitive functions, with an in-house work force, production-line procedures, and a multi-layered management structure. As a facility's operating life ends, a deactivation and restoration phase is initiated, where once again the work becomes non-repetitive and task oriented. However, the scope of work during the restoration phase is often not as clearly defined as it was in the construction phase, particularly when environmental cleanup is involved. This build/operate/restore cycle is illustrated in Fig. 1. The Hanford facility is currently transitioning from the operations phase to the restoration phase.

Fig. 1.

To effectively manage the restoration work, the ERC Team implemented a "project" approach, where specialized resources are assembled into project teams. Work is planned and executed through the teams and, after completing the work scope, these teams are disbanded and the resources reassigned. For this approach to work, costs and schedules must be based on a well-defined baseline scope so that resource loading can be predicted and balanced within the overall ERC Team. Additionally, effective project control systems are needed to track the work closely, so that progress and costs can be monitored and reported quickly.

Division of the ERC work scope among individual project managers is based on the type of work involved. These project managers are responsible for defining resource needs, obtaining the appropriate resources from functional (discipline) departments, and guiding the teams as necessary to execute their assigned scope of work. They are accountable for cost, scope, and schedule. The individual teams are responsible for completing their assigned tasks according to an overall project plan, and are accountable to their project manager. A key element to the success of this approach is having representatives from DOE-RL assume active roles within the project teams.

Several functional departments exist to support the projects. Some of these groups provide general sitewide support such as auditing, human resources, contract administration, and community relations. Other functional groups such as engineering, project controls, procurement, and field support provide specific support services, technical oversight, and resources to individual projects. These groups use skills and experience from numerous other projects to improve efficiency in high impact administrative processes such as procurement and regulatory analysis. Functional department managers are accountable for maintaining a well-trained and

experienced staff to support the projects, addressing sitewide issues in a consistent and cost-effective manner, deploying resources to the project teams on an as-needed basis, and providing technical oversight.
An organization chart indicating the relationship between the projects and the functional groups is shown in Fig. 2.
Fig. 2.
Such an approach was quite foreign to much of the transitioned work force, and orientation was needed on the new project philosophy. Also, to infuse private sector experience, many people with specialized skills in environmental remediation needed to be relocated to Hanford from the ERC Team's offsite offices. Once everyone was on board, special team building sessions were held to establish working relationships and develop specific commitments for each project area and the functional groups. Both the ERC Team and DOE-RL participated in these sessions. The result was a cohesive team with a common focus.

INCREASED EFFICIENCY

Initiating a new work process usually involves a steep learning curve. During the first few months, a decline in productivity is typical prior to seeing the gains that are ultimately anticipated. However, for the Hanford ER Program, productivity gains needed to be immediate. DOE-Headquarters was closely scrutinizing the initial efforts to assess the viability of having specialized ER contractors at other sites within the DOE Complex. Simultaneously, budgets were being tightened, as the administration looked for ways to make DOE more cost effective. The ERC Team was asked to complete the remaining fiscal year 1994 work scope with $19 million less than previously budgeted. For fiscal year 1995, $201 million was allocated for $233 million worth of work scope. Coupled with these challenges were significant regulatory and contractual milestones that had to be met and the activities associated with relocating personnel, establishing work procedures, and setting up an office for over 800 employees.
To meet this challenge, the ERC Team implemented the project approach described above. Each of the project managers were provided resources from the functional organizations consisting of transitioned and transferred staff to support the effort. The resulting teams were empowered by management to complete the work scope. Due to the project approach, the ERC Team is seeing positive results including:

Completed the fiscal year 1994 work within the reduced funding target

Reduced applicable procedures from 2400 to about 1700

Reduced turnaround time issuing documents to outside entities by as much as 75 percent

Streamlined the procurement process - 45 day bid/award cycle for multimillion dollar remedial action subcontract
- 35 day bid/award cycle for a site preparation subcontract

Met all Federal Facility Agreement and Consent Order milestones -38 on schedule
- 24 ahead of schedule

Met several project team efficiency challenges that resulted in $23 million of documented cost savings

PRIVATE SECTOR KNOWLEDGE AND EXPERIENCE

A key to the success of the ERC Team is to incorporate into the project the knowledge and experience gained from private sector construction and environmental projects. This involves questioning the way things are being done for the restoration at Hanford. After five decades of highly specialized manufacturing activities, it can be a challenge to fully align the work force with a new mission (see Fig. 1).
Many work practices used at Hanford have been developed through years of experience in handling radioactive materials and hazardous wastes. The wealth of this knowledge needs to be integrated into the cleanup process. However, there are many work practices that may no longer be applicable to the type of work being performed by the ERC Team. In addition, innovative ideas developing outside the DOE Complex may have direct applicability at Hanford. This is an area where the teaming of experienced Hanford personnel with people from the environmental cleanup industry has distinct advantages. Some things done to integrate this knowledge and experience include:

Combined Bechtel Hanford, Inc. construction/project management experience with TMA Hanford, Inc., CH2M Hill Hanford, Inc., and IT Hanford, Inc. environmental expertise on individual project teams

Integrated key individuals from ERC management team into the transitioned work force

Made "home office" resources available to support the project teams on an as needed basis - consulted on fabrication of groundwater pump and treat systems - provided input on monitoring requirements for the prototype Hanford Barrier

Evaluated and redefined waste disposal facility needs/construction from a commercial operations perspective - $92 million estimated savings in cost from the baseline estimate for construction of an Environmental Restoration Disposal Facility for waste resulting from site cleanup (see Fig. 3)

Established an independent technical advisory group to counsel the ERC Team on approaches and methodologies to implement the cleanup

Fig. 3.

DOE/ERC TEAM CONCEPT

Full support from DOE-RL was vital for the ERC Team to successfully instill the "project" philosophy. Cooperation and participation at all levels within both organizations was needed to develop an effective project team. Therefore, the DOE project manager has become a key member of the ERC project management team, and individuals from DOE-RL staff are actively participating in all of the ERC project teams.

Early during the transition, senior DOE-RL and ERC managers attended an offsite planning session where goals/objectives were discussed. The result was a common and specific commitment statement, signed by all that were present. The group also formed a "Results Management Team" to monitor progress on the commitments, make quick and informed decisions, and continually guide the project toward its mission. Several months later, another session convened with over 100 representatives from all aspects of the ER Program. At this session, the commitment statement of the DOE and ERC senior management team was explained and discussed. Project teams including both DOE and ERC members worked together to analyze their tasks, align themselves with the mission, define roles and responsibilities, identify/resolve key issues, and establish project specific commitment statements. The participants realized that the success or failure of the project would be dependent on everyone working as a team toward a common goal. The session resulted in additional alignment and focus on the ER cleanup mission at Hanford.

SUMMARY

Establishing a "project approach" to the ER Program is already paying dividends. Overall project improvement has been directly reflected through reduced costs and shortened schedules. Private sector knowledge and skills are being integrated with the expertise of the existing work force to improve products, focus on specific problem areas, and question the scope/purpose of things that are being done. A strong partnership between DOE-RL and the ERC Team is being developed and is proving successful in directing the work towards a common mission. This partnership, along with the project approach to work planning and execution, will help retain focus and improve efficiency, both of which are essential during a period of declining budgets.

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LESSONS LEARNED FROM EXPEDITING REMEDIATION PROJECTS THROUGH INTEGRATING REGULATORY AGENCIES INTO REMEDIATION DESIGN AND PERMITTING TEAMS

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ABSTRACT

The Department of Energy, Colorado Department of Public Health and the Environment, and the Environmental Protection Agency established a joint working group which worked together closely during the planning, permitting, and design phases of a project to close/remediate five Solar Evaporation Ponds at the Rocky Flats Environmental Technology Site. The joint working group was a new strategy established to expedite the schedule on this technically complex project. Results from implementing the joint working group indicate that the integration of the regulatory agencies as an up-front stakeholder can be a successful method of expediting a regulatory-based remedial action project. This paper presents the

successes and lessons learned through integrating the regulatory agencies into the design and permitting team. In addition, the paper recommends methods to enhance the success of this strategy. Conditions will be presented to help identify the types of projects that could benefit from this new approach.

INTRODUCTION

The Department of Energy (DOE) operated five lined Solar Evaporation Ponds (SEPs) at the Rocky Flats Environmental Technology Site (RFETS) in Jefferson County, Colorado, from 1953 until 1986. Figure 1 presents a location map of the RFETS with respect to the surrounding region.

Fig. 1.

The SEPs which are identified as individual hazardous substance site (IHSS) Number 101 are included in the RFETS Operable Unit 4 (OU4). The SEPs were used primarily to store and evaporate low-level radioactive process wastes and neutralized acidic wastes containing aluminum hydroxide with high levels of nitrate. Operation of the SEPs has resulted in contamination of the surrounding soils and may be a potential source of ground water contamination.

Under an Interagency Agreement (IAG) with the Colorado Department of Public Health and the Environment (CDPHE) and the U.S. Environmental Protection Agency (EPA), DOE is required to close and remediate OU4 in accordance with an approved Interim Measures/Interim Remedial Action (IM/IRA) program. The IAG integrates the Resource Conservation and Recovery Act (40 CFR 265) and Comprehensive Environmental Response, Compensation and Liability Act requirements for surface impoundment closure and environmental restoration.

As a measure to expedite the closure/remediation of the SEPs to meet aggressive IAG milestone dates, the DOE, CDPHE, and EPA agreed to work together closely throughout the planning, permitting, and design phases of the project. The goal of this new strategy was to involve the regulatory agencies throughout the design and permitting processes so that the DOE, CDPHE, and EPA could jointly present and defend the IM/IRA to the public.

A joint working group was formed with a member from each regulatory agency to provide input throughout the project. The joint working group met on a weekly basis to identify and resolve issues. In this capacity, the regulatory agencies were cognizant with respect to the development of technical details and provided concurrent review as the project progressed. The RFETS operator (EG&G) and their permitting/design subcontractor were also members of the joint working group.

This paper presents the successes and lessons learned through integrating the regulatory agencies into the design and permitting teams on the OU4 IM/IRA. In addition, the paper recommends methods to enhance the success of this strategy. Conditions will be presented to help identify the types of projects that could benefit from this new approach.

JOINT WORKING GROUP COMPOSITION AND MEETING CONDUCT

A joint working group was formed with a member from the CDPHE, EPA, and members from the DOE, EG&G, and the permitting/design subcontractor. This joint working group met on a weekly basis to identify and resolve issues. The meetings were conducted informally with participation by all group members. However, the primary participants were the DOE, CDPHE, and EPA. The DOE representative was typically the meeting chairperson. Detailed meeting minutes were kept for the project administrative record.

BENEFITS FROM THE JOINT WORKING GROUP INTEGRATION

The establishment of an integrated joint working group provides numerous potential benefits to a project. One of the most significant benefits is that issues which could impede permit/design approval are identified early so that they can be addressed and resolved prior to the formal submittal of the permit application and design documentation for regulatory review. Early identification and resolution of issues reduces the potential to have to re-work aspects of the permit/design due to a denial from the regulatory agencies. The joint working group concept therefore has the potential to save the cost of permit/design re-work. In addition, the project schedule may not be impacted by re-work.

A second major benefit is the ability to reach agreement upon methodologies that will be used to screen and manipulate data. For example, the joint working group debated the methodologies with respect to:

1. Statistical manipulation of data,
2. Calculation of preliminary remediation goals,

3. Risk assessment, and
4. Computer modeling to demonstrate protectiveness to human health and the environment.

By working together, the DOE, EPA, and CDPHE established methodologies to process data in a manner acceptable to all three organizations. Early agreement with respect to methodologies can save budget and schedule since a reworking of data should not be required upon formal submittal of the permit document. The potential cost and schedule savings are significant because data manipulation methodologies are early project activities that provide the basis for important project decisions. Re-working data manipulation activities has significant cost and schedule impacts and jeopardizes all project decisions.

A third benefit of the joint working group interactions is the early identification and agreement on Applicable, Relevant, and Appropriate Regulations (ARARs). Early identification of ARARs allows the project to design and strategize appropriate measures to meet the regulatory requirements.

A fourth benefit of the joint working group is the ability to identify and resolve technical issues and design details prior to formal agency review and approval. The interactions of the joint working group allow DOE to design a project that has a high probability of acceptance by the regulatory agencies. The regulatory agencies benefit by being cognizant of many details of the project prior to the formal review process.

LESSONS LEARNED THROUGH JOINT WORKING GROUP INTERACTIONS

Several lessons have been learned through the interactions of the joint working group. Cognizance of these issues may help enhance the success of future joint working groups. The first lesson learned is that the opinions and positions that a team member brings to the joint working group may not reflect the position and policies that the individual's organization endorses. Different subgroups and departments of the various organizations do not always agree among themselves. It is therefore important that the members of the joint working group routinely brief their management. Meeting minutes should be widely distributed to the joint working group members as well as their respective colleagues and managers.

The second lesson learned is that it is often difficult to reach final closure/agreement on issues. In general, the more controversial an issue is, the more difficult it is to reach closure. Difficult issues typically involve disputes with respect to regulatory interpretation. Resolution of these issues takes time because the members of the joint working group often need to consult with regulatory/legal experts within their own organizations. It is important to identify issues as early as possible and schedule adequate time for the various organizations to research and develop a position/solution on the issue. Many issues require re-evaluation as a project evolves and becomes increasingly complex. Therefore, not all closures/agreements are final.

A third lesson learned is that the joint working group meetings can be long and tangential issues may interfere with accomplishing the meeting agenda. This is a potential problem with any meeting situation. The best way to keep a meeting on track is to:

1. Establish the agenda prior to the meeting by soliciting input from the different representatives.
2. Publish the agenda prior to the meeting.
3. Have a strong meeting facilitator to ensure that the agenda is followed.

It is best to use the meetings to establish the project criteria rather than design the project details by committee. This typically reduces the number of time consuming tangential issues. An effective process is to use the joint working group to establish and agree on performance criteria. The DOE and their subcontractors then design a solution or method to meet the criteria. A presentation of the findings/results is made to the joint working group at a subsequent meeting. This strategy is effective because the regulatory agencies are involved in establishing the performance requirements that must be accomplished. The DOE and their subcontractors are able to investigate alternatives to meet or exceed the requirements in the most cost effective and technically feasible manner.

The joint working group should maintain a list of open issues which is reviewed at each meeting. This will ensure that an issue does not get neglected. A list of previous agreements should be maintained. This list captures the significant decisions and provides an easy reference document to identify which decisions may

change as the project evolves and/or becomes increasingly complex.

METHODS TO MAXIMIZE THE EFFECTIVENESS OF JOINT WORKING GROUPS

This section is intended to provide advice on how to maximize the benefits that can result from close interaction between the DOE and the regulatory agencies throughout the planning, design, and permitting phases of a project.

Joint working group members should be empowered by their organizations to make decisions representative of the organization's policies. Therefore, senior members of an organization should be involved in the joint working group. The individual members of the joint working group should have primary responsibility for permit approval and/or implementation of the project.

The joint working group should keep a list of the open issues so that they will not be neglected. The issues list should be reviewed at each team meeting to see if any group member has further information that may help resolve any open issue. In addition, an action items list should be maintained so that each member of the joint working group is cognizant of their responsibilities to support future meetings and decisions. The actions item list should be reviewed and updated at each joint working group meeting.

It is important that detailed meeting minutes are taken at each meeting. These minutes are likely to become part of the project's administrative record. The minutes should record the decisions and agreements made at each meeting, and capture the viewpoints of the various organizations on issues that are discussed, but not resolved. The members of the joint working group should be allowed to review and ratify the meeting minutes prior to their formal issuance. The meeting minutes provide an excellent means for others who are involved in the project (but not members of the joint working group) to remain cognizant with respect to the decisions that are being made.

TYPES OF PROJECTS WHERE A JOINT WORKING GROUP MAY BE SUCCESSFUL

The establishment of a joint working group is not necessary for all projects. Projects with the following characteristics are candidates for the implementation of a joint working group.

1. Projects with a regulatory driven expedited schedule
2. Large projects involving numerous constraints and complex technical/regulatory issues.
3. Projects involving the use of proposed or newly promulgated regulations
4. Projects that may involve public controversy.

The establishment of a joint working group is important for projects that are expedited due to an agreement with regulatory agencies because it requires the agencies to participate in the expedited schedule. The agencies will be cognizant and partly responsible for schedule adherence. The agencies will therefore understand and may contribute to a situation where a schedule extension is requested.

Projects involving numerous constraints and/or complex technical and regulatory issues are candidates for establishing a joint working group because regulatory agency involvement will be useful in solving the complex issues within the project constraints. Interactions of a joint working group will be beneficial for early identification and resolution of the issues which could impact schedule and budget when the permit and/or design package is submitted for formal review and approval.

If a project will require the implementation of proposed or newly promulgated regulation, then it is appropriate to create a joint working group so that the regulators can provide direction and interpretation on how to best implement a new regulation.

If the implementation of a project may spark public controversy, then the establishment of a joint working group should be considered. The potential for the facility owner and the regulatory agencies to jointly propose and defend a project to the public can ease the public review period, and increase the potential to implement the project with few modifications.

CONCLUSION

The conclusion drawn through the establishment of a joint DOE, CDPHE, and EPA working group for the RFETS OU4 Solar Evaporation Pond Closure project is that the strategy was effective at identifying and resolving technical/regulatory issues to allow the project to proceed on an agency driven expedited schedule. The integration of the regulatory agencies as an up-front stakeholder can be a successful method of expediting a regulatory-based remedial action project.

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Frank Coffman, ITC
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LAND DISPOSAL RESTRICTIONS PHASE II-UNIVERSAL TREATMENT STANDARDS FINAL RULE: IMPACTS ON MIXED WASTE TREATMENT

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ABSTRACT

The Environmental Protection Agency (EPA) published the Land Disposal Restrictions (LDR) Phase II Rulemaking in the Federal Register on 19 September 1994 (59 FR 47982). This rule establishes the universal treatment standards, promulgates treatment standards for newly identified toxicity characteristic organic wastes, and requires that underlying hazardous constituents be addressed in certain characteristic wastes. Phase II also clarifies the regulation of debris treated by immobilization technologies, the applicability of treatability variances, and the treatment standards for lab packs. Most of the impacts of the Phase II rule are not unique to mixed waste; they are applicable to hazardous wastes regulated only by the Resource Conservation and Recovery Act. The Department of Energy (DOE) waste managers who have recently issued Site Treatment Plans as a requirement of the Federal Facility Compliance Act will benefit from an examination of changes brought about by the Phase II rule that includes applicability of universal treatment standards, newly identified waste treatment standards, underlying hazardous constituents, and other mixed waste management issues. This paper provides an overview of LDR Phase II rule changes, and gives examples that illustrate how Phase II rule changes may affect how mixed wastes are managed at DOE facilities. Also, a summary of the expected changes in Phase III, Phase IV, and Phase V proposed rules is given.

INTRODUCTION

The Environmental Protection Agency (EPA) published the Land Disposal Restrictions (LDR) Phase II Rulemaking in the Federal Register on 19 September 1994 (59 FR 47982).* In the Phase II rule, EPA establishes the universal treatment standards, promulgates treatment standards for newly identified toxicity characteristic (TC) organic wastes, and requires that underlying hazardous constituents be addressed in certain characteristic wastes. EPA also clarifies other issues related to LDR in the Phase II rule, such as the regulation of debris treated by immobilization technologies, the applicability of treatability variances, and the treatment standards for lab packs.

Most of the changes and clarifications to LDR provided in the Phase II rule affect the management of mixed waste. This paper will provide an overview of LDR, elaborate on the Phase II rule, discuss the ramifications of the rule on mixed waste management, and then provide examples.

LAND DISPOSAL RESTRICTIONS OVERVIEW

In the Hazardous and Solid Waste Amendments (HSWA) to the Resource Conservation and Recovery Act (RCRA), Congress specifies that hazardous waste must be treated prior to land disposal in a manner that substantially diminishes the toxicity of the waste or substantially reduces the likelihood of migration of hazardous constituents from the waste. To accomplish this, Congress requires EPA to promulgate regulations specifying levels or methods of treatment of hazardous waste. In response, EPA promulgated the Land Disposal Restrictions for Title 40 of the Code of Federal Regulations Part 268 (40 CFR 268).

As required by HSWA, the regulations in 40 CFR 268 contain "treatment standards" that identify levels or methods of treatment for hazardous waste. The standards apply to all hazardous waste for which a treatment standard has been promulgated unless a variance, extension, or exemption applies. The standards must be met prior to land disposal of the waste. Land disposal includes, but is not limited to, placement of hazardous waste in any of the following:

Landfill
Surface impoundment
Waste pile
Injection well
Land treatment facility
Salt dome or salt bed formation
Underground mine or cave
Concrete vault or bunker intended for disposal

Many requirements contained in LDR apply prior to disposal, such as the dilution prohibition, storage prohibition, testing requirement, and recordkeeping requirement. To further understand LDR, this section will provide information on LDR treatment standards, the applicability of LDR, prohibitions contained in LDR, and requirements for testing and recordkeeping identified in LDR.

Treatment Standards

In 40 CFR 268, EPA identifies treatment standards for all hazardous waste codes that existed at the time HSWA was enacted. EPA identifies treatment standards for a number of waste codes identified as hazardous after HSWA was enacted (newly-identified wastes). Treatment standards are established by evaluating the best demonstrated available technology (BDAT) for the treatment of a waste. The standards are normally identified by waste code, divided into wastewater and nonwastewater categories, and identified as either a specified technology or a constituent concentration level. If a specified technology is identified as a treatment standard, that technology must be used to treat the waste or a treatability variance must be obtained. If constituent levels are specified, the waste must be treated by an appropriate technology until all identified constituents reach or fall below the designated level. The BDAT is used to determine the constituent level, but another treatment technology may be used to treat the waste to that level. Each waste code may have a number of constituents to treat to a specified level. In the past, the treatment level for a particular constituent could vary from one waste code listing to another.

Prior to the Phase II rule, the treatment standards were listed in 40 CFR 268.42 for specified technologies, and in 40 CFR 268.41 and 268.43 for constituent concentration levels. The levels listed in 40 CFR 268.41 were based on constituent concentration in an extract of the waste after using the Toxicity Characteristic Leaching Procedure (TCLP). The levels listed in 40 CFR 268.43 were based on total concentrations of the hazardous constituents.

For hazardous waste debris, the treatment standards can also be based on the matrix of the waste. These standards are identified in 40 CFR 268.45 as a specified technology and include extraction, destruction, and immobilization treatment processes. The appropriate technology to treat the debris depends on the type of debris and type of contaminant(s) present. If debris is treated by an extraction or destruction technology, it is not regulated as a hazardous waste after it is treated.

Applicability

The LDR requirements of 40 CFR 268 apply to all generators and transporters of hazardous waste with a treatment standard, as well as all owners and operators of treatment, storage, and disposal facilities (TSDFs) that handle the waste. A newly identified waste is not subject to 40 CFR 268 until a treatment standard has been promulgated. In addition, the effective date of LDR may be delayed, a variance may be obtained, or an exemption may apply. These delays, variances, and exemptions are in the form of a national capacity variance, case-by-case exemption, treatability variance, equivalent method variance, no-migration petition, or surface impoundment exemption.

The effective date of a treatment standard for a particular waste code may be delayed by a national capacity variance or a case-by-case extension. A national capacity variance is provided when EPA determines that sufficient treatment capacity for certain hazardous waste codes is not available nationwide. The effective date for a treatment standard is extended until treatment capacity for the waste is available. (Other requirements such as analysis and recordkeeping may still apply.) A case-by-case extension may be obtained from EPA for 1 year and extended for 1 more year if treatment capacity for a specific waste cannot be made reasonably available. To obtain the extension, the petitioner must demonstrate certain information identified in 40 CFR 268.5.

Variance from an established treatment standard can be obtained in the form of a treatability variance or an equivalent method variance. A treatability variance may be granted by EPA for a unique waste stream where the existing treatment standard, expressed as a concentration or specified technology, is not applicable. EPA will establish a new treatability group for that waste stream if the variance is granted. If a specific method is identified as the treatment standard, an equivalent method variance enables EPA to approve another process as the treatment standard if performance of the alternative process is equivalent.

Exemptions from meeting treatment standards may be granted if a no-migration petition is approved or the surface impoundment exemption applies. A no-migration petition can be granted by EPA for specific waste disposed at a specific site if it can be demonstrated that no migration of hazardous constituents from the disposal unit will occur for as long as the waste remains hazardous. Hazardous waste that does not meet treatment standards can be treated in a surface impoundment pursuant to the surface impoundment exemption if certain conditions are met.

Prohibitions

In addition to the prohibition on land disposal of waste that does not meet treatment standards, LDR contains two other significant prohibitions. These are the dilution prohibition and the storage prohibition. The dilution prohibition states that hazardous waste subject to a treatment standard cannot be diluted as a substitute for legitimate treatment unless it is a necessary part of the treatment process. The storage prohibition states that in most situations hazardous waste subject to a treatment standard cannot be stored unless it is to accumulate sufficient quantities of waste to facilitate proper recovery, treatment, or disposal.

Testing and Recordkeeping

All generators and TSDFs managing hazardous waste with a treatment standard are subject to waste analysis, notification, and recordkeeping requirements contained in the LDR. Hazardous waste generators must determine if the waste meets applicable treatment standards and present their findings to hazardous waste facilities that will receive the waste. A generator may determine if the waste meets applicable treatment standards based on knowledge of the waste or by conducting analysis. Treatment and storage facilities must test hazardous waste according to a waste analysis plan to determine if the waste meets applicable treatment standards and present their findings to hazardous waste facilities that will receive the waste. Disposal facilities must also test hazardous waste prior to disposal according to a waste analysis plan to ensure that it meets applicable treatment standards. All hazardous waste facilities must maintain on-site records of these activities.

Changes to the Land Disposal Restrictions Program

Since the first statutory deadline (i.e., hard hammer) addressing LDR (Solvents and Dioxins Rule) in 1986, there have been changes regarding the technologies available to treat hazardous wastes, as well as the availability of data regarding the treatability of certain constituents. As a result, EPA has taken the initiative to take actions to reform the LDR program, to update, clarify, and facilitate implementation of already existing requirements. EPA's first RCRA Reform Initiative (Phase I) was the Hazardous Debris Rule promulgated on 18 August 1992 (57 FR 37194), which modified treatment option for debris contaminated with hazardous waste. Additional modifications to the LDR program are a result of the 25 September 1992 U.S. Court of Appeals decision on Chemical Waste Management v. EPA, 976 F.2d 2 (D.C. Cir. 1992) to vacate the treatment standards for ignitable and corrosive hazardous wastes. EPA has taken action to comply with this decision by promulgating an interim final rule (24 May 1993, Federal Register, 58 FR 29860) establishing revised treatment standards for ignitable and corrosive wastes. This interim final rule retains the requirement to remove the hazardous characteristic (by deactivation), and also requires that the waste be treated for any underlying hazardous constituents (approximately 216 constituents) reasonably expected to be present in the waste.

The latest action of the RCRA Reform Initiative is the promulgation of the Phase II rule.

PHASE II CHANGES TO THE LAND DISPOSAL RESTRICTIONS

Under the Phase II rule, EPA provides a number of changes and clarifications to LDR. EPA establishes universal treatment standards, promulgates treatment standards for newly identified toxicity organic wastes, and requires that underlying hazardous

constituents be addressed in certain characteristic wastes. The Phase II rule also clarifies other issues related to LDR, such as the regulation of debris treated by immobilization technologies, the applicability of treatability variances, and the treatment standards for lab packs.

Universal Treatment Standards

Under the Phase II rule, EPA promulgates a single universal treatment standard (UTS) for each organic, metal, and cyanide constituent in wastewater and nonwastewater form. The constituent concentration treatment level is the same regardless of the hazardous waste code that identifies the constituent. This eliminates the variation in concentration levels that existed from one waste code listing to another. The UTSs are established for all hazardous wastes for which treatment standards have been promulgated, except for the following:

- TC metal wastes (Waste Codes D004-D011)
- Most hazardous wastes with a specified technology identified as the treatment standard

Implementation of the UTSs will change some of the standards identified as constituent concentration levels. More specifically, 59 percent of the previous constituent levels will remain the same, 33 percent will change within a factor of 10, and 8 percent will change more substantially. (Three percent will become significantly more stringent.) However, the adjustment in constituent treatment levels should not require a change in the technologies used to treat the waste. The technology basis for establishing the new constituent levels is primarily the same as the basis for the old levels.

For organic nonwastewaters, the BDAT for determining UTS constituent levels is combustion. The BDAT for determining UTS constituent levels in organic wastewaters is the same as that identified for the hazardous waste code: F039 multisource leachate. The BDAT for the F039 wastewaters is biological treatment followed by chemical precipitation, or wet-air oxidation followed by carbon adsorption followed by chemical precipitation. The BDAT for determining 12 of the 14 nonwastewater metal UTS constituent levels (which excludes TC metals) is high-temperature metal recovery or stabilization. The two other BDATs are vitrification for arsenic, and recovery by roasting or retorting for high concentrations of mercury. The BDAT for determining metal wastewater UTS constituent levels is chemical precipitation. For metal wastewater treatment, operating conditions such as flocculating agents, reagent concentrations, retention time, and mixing may need to be adjusted to comply with the new standards. The UTS levels for cyanide wastewaters and nonwastewaters are based on alkaline chlorination as the BDAT.

The universal treatment standards became effective on 19 December 1994. In states that have authority to implement RCRA but not LDR, EPA will enforce the new standards until the states incorporate the modifications. States authorized to implement LDR for some or all waste streams will continue to implement these treatment standards. The new UTSs will not apply to those waste streams until the state has incorporated them into state law.

Treatment Standards for Newly Identified Wastes

Under the Phase II rule, EPA promulgates treatment standards for newly identified TC organic wastes, TC pesticide wastes, coke by-product wastes, and chlorotoluene wastes. Because these wastes were identified as hazardous after the enactment date of HSWA in 1984, they are "newly identified wastes" for the purposes of the LDR program. (On 29 March 1990, EPA promulgated a rule that identified organic constituents [in addition to existing EP metals and pesticide constituents] and levels at which a waste is considered hazardous based on the characteristic of toxicity.)

The TC organic wastes (hazardous waste codes D018-D043) and the TC pesticide wastes (hazardous waste codes D012-D017) are determined to be hazardous through application of the TCLP. The coke by-product and chlorotoluene wastes are generated by specific industrial processes.

The new treatment standards for TC organic wastes are based on incineration for nonwastewaters. For wastewaters, the standards are based on existing data and technologies identified for treatment of these constituents in other wastes. The treatment standards for TC pesticide wastes already existed based on Extraction Procedure (EP) constituent levels, not TCLP levels. The Phase II rule establishes treatment standards based on constituent levels from the TCLP. The new standards are based on the same technologies used to establish the EP standards. The treatment

standards for the TC organic and the TC pesticide wastes are equivalent to the UTSs. The Phase II rule applies to those TC organic wastes that are managed in systems other than those regulated under the Clean Water Act (CWA), those engaged in CWA-equivalent treatment prior to land disposal, and those injecting into Class I deep injection wells. In addition, TC wastes rendered noncharacteristic are still within the scope of the Phase II rule.

Underlying Hazardous Constituents

The UTSs also apply to all underlying hazardous constituents in TC pesticide and organic wastes (D012-D043), as well as the constituent responsible for the TC designation. The Phase II rule also modifies the applicability of underlying hazardous constituents to D001 and D002 characteristic hazardous wastes. An underlying hazardous constituent is defined as any constituent listed in the universal treatment table, except zinc and vanadium, that can reasonably be expected to be present at the point of generation of the hazardous waste at a concentration above the constituent-specific treatment standard.

The compliance monitoring and notification requirements contained in LDR were modified to minimize the amount of information that must be placed on the LDR notification in certain circumstances. In addition, the certification and notification requirements were modified where necessary to address underlying hazardous constituents present in TC organic or pesticide wastes. A generator must identify all underlying hazardous constituents in TC organic or pesticide wastes. Generators may base this determination on their knowledge of the raw materials used, the process they operate, and the potential reaction products of the process, or upon the results of analysis. The generator also must notify TSDFs receiving the waste of the constituents, when required.

Other Issues Addressed in the Phase II Rule

Additional issues related to LDR that the Phase II rule clarifies include the possibility of removing immobilized hazardous debris from under Subtitle C regulation, the applicability of treatability variances, and the treatment standards for lab packs.

EPA decided in the Phase II rule to revisit the issue of whether immobilized hazardous debris, if treated in certain ways or treated to meet certain limits, would be excluded from Subtitle C regulation. Several issues were discussed including the inappropriateness of the TCLP for debris, the use of a 50 psi (pounds per square inch) structural test as a performance standard, and the use of a leach test performed on intact debris. Although industry supported the idea of removing immobilized hazardous debris from regulation under Subtitle C, there was no supporting data or other information submitted to EPA to support these claims. Therefore, the Phase II rule did not modify the debris rule and EPA will reassess this determination in the future. This reassessment will include evaluation of exclusions as part of the Hazardous Waste Identification Rule (HWIR) process.

Existing LDR regulations (40 CFR 268.44(a)) allow for variances from treatment standards if a waste cannot be treated to specified treatment standards or if the treatment technology on which the standard is based is inappropriate for the waste. Guidance on obtaining LDR treatability variances is provided in the Phase II rule as is guidance on treatability variances for soil, including site-specific, non-rulemaking variances. The preamble to the Phase II rule provides clarification on the applicability of the variance, the necessary data required, and the proper procedure for obtaining a variance.

Appendices IV and V of Part 268, which promulgated alternative treatment standards for hazardous wastes placed in lab packs, are replaced with a new Appendix IV. Appendix IV now identifies which hazardous wastes cannot be placed into lab packs. Prior to Phase II, Appendices IV and V listed the hazardous wastes that could be placed into lab packs. The specified method of treatment is still incineration, combined with the requirements of 268.42 (c)(4) (ash residues from lab packs containing D004-D008, D010 and D011 are treated to meet the characteristic metals treatment standards), will ensure that all underlying hazardous constituents present in characteristic wastes will be treated.

IMPACTS ON MIXED WASTE MANAGEMENT

Most of the effects of the Phase II rule on mixed waste management are not unique to mixed waste. The same effects exist for hazardous waste that is regulated only by RCRA. However, this section identifies the specific changes to mixed waste management brought about by the Phase II rule.

Applicability of Universal Treatment Standards
In several previous cases associated with mixed wastes (as well as hazardous wastes), a constituent regulated under a treatment standard for one mixed waste was also regulated in another waste at different concentration levels. Under the Phase II rule, EPA eliminates these differences in concentration limits for the same constituent to provide a better assessment of treatability and to reduce confusion. Although the Phase II rule does not require treatment for additional hazardous constituent concentrations, treatment levels are for some constituents lower. In most cases, these changes should not require technologies identified in site-specific treatment plans to be modified, since the new treatment standards are based on existing technologies. The site-specific treatment plans are also based on existing technologies.
In situations where constituent concentration treatment levels are lowered and the applicability of a treatment technology must be demonstrated, treatability studies will become an important step. Treatability studies will be used to determine the optimal process conditions needed to achieve the desired treatment standard. If, based upon the results of the treatability study, a technology cannot achieve the required constituent concentration treatment level, the technology will have to be redeveloped or redesigned.
Newly Identified Waste Treatment Standards
Under the Phase II rule, EPA promulgated treatment standards for newly identified TC organic wastes and modified treatment standards for TC pesticide waste. For hazardous waste, these treatment standards became effective on 19 December 1994. For mixed wastes, these treatment standards will become effective on 19 September 1996. EPA determined that sufficient alternative treatment capacity for mixed radioactive wastes contaminated with newly listed and identified wastes is not available and has granted a 2-year national capacity variance for these wastes.
As a result of the national capacity variance, any site that generates these newly identified mixed wastes may choose to treat the mixed waste before 19 September 1996, if treatment capacity is available or to dispose of the mixed waste without treatment if disposal capacity is available to accept these mixed wastes.
Ultimately, treatment capacity is needed for any future-generated mixed waste that exhibits only the characteristic for TC organics or TC pesticides. In addition, LDR treatment standards may already exist for other applicable listings or characteristics.
Underlying Hazardous Constituents
The requirement to treat characteristic wastes for underlying hazardous constituents does apply to mixed waste. Specifically, a mixed waste that exhibits the ignitable or corrosive characteristics must also be treated for any underlying hazardous constituents that are reasonably expected to be present in the waste. Newly identified mixed waste with a treatment standard promulgated under the Phase II rule must be treated for underlying hazardous constituents when the treatment standards become effective. For pesticide and organic treatment standards based on the TCLP, the requirement will become effective at the end of the national capacity variance.
The requirement to address underlying hazardous constituents significantly impacts the characterization of mixed waste. In addition, new treatment steps may be required for specific mixed waste streams. All ignitable, corrosive, and organic mixed waste will need to be treated for metals, thus requiring a two-step treatment process. In addition, certain corrosive wastes may require treatment for organics or metals using additional treatment technologies, where before neutralization would have been the only required treatment process to deactivate the corrosive component of the mixed waste.
The requirement for underlying hazardous constituents may also impact certain mixed wastes that are listed under 40 CFR part 261, subpart D, since any mixed waste that carries a listed hazardous waste code must also identify any applicable characteristic waste codes. This requirement was promulgated in a previous LDR final rule (Third Thirds), in which EPA established special requirements for purposes of part 268, that a waste will carry the waste code for any applicable listing under subpart D, as well as one or more waste codes under 40 CFR part 261, subpart C, where the waste exhibits a characteristic. As a result of the Phase II rule, additional characteristic hazardous wastes require treatment for underlying hazardous constituents and would therefore apply to more listed hazardous wastes.
Other Mixed Waste Management Issues

Additional issues that may impact mixed waste management from the Phase II rule include removing immobilized mixed waste debris from Subtitle C regulation, the applicability of treatability variances, and the treatment standard for lab packs. Several large volume mixed waste streams (i.e., surface contaminated or activated lead solids, and inorganic debris) could be affected by a change in the requirement for immobilized mixed waste debris to no longer be under Subtitle C control. This change would allow mixed waste debris to be disposed of in a radioactive waste management unit as opposed to a disposal unit for mixed wastes. Available disposal capacity is much greater at radioactive waste management units than at mixed waste management units. In addition, several requirements and performance standards currently exist to which immobilized mixed debris would be subject as a result of the radioactive component of the mixed waste. The treated/immobilized mixed waste would not allow toxic contaminants to remain at levels that could pose a hazard to human health and the environment absent Subtitle C regulation. Therefore, it is anticipated that existing performance standards applied to mixed waste debris treated using an immobilization technology would not have to be modified.
If mixed waste treatment technologies are unable to meet LDR treatment standards, additional guidance is provided in the Phase II rule regarding treatability variances.
For any mixed waste lab packs that contain characteristic metal wastes, the incinerator ash will require stabilization prior to land disposal.
EXAMPLE OF THE EFFECTS OF THE PHASE II RULE EFFECTS ON MIXED WASTE MANAGEMENT
Universal Treatment Standards
Before Phase II Final Rule
For mixed waste containing F001 spent solvents (trichloroethane and freon), the LDR treatment standard for the nonwastewater form was a concentration-based treatment standard. This treatment standard required compliance at or below the constituent concentration of 5.6 mg/kg in the waste for trichloroethane and 33 mg/kg in the waste for freon.
After Phase II Final Rule
For mixed waste containing F001 spent solvents (trichloroethane and freon), the LDR treatment standard for the nonwastewater form is 6.0 mg/kg for trichloroethane and 30 mg/kg for freon.
Newly Identified Treatment Standards
Before Phase II Final Rule
Mixed wastes only exhibiting the characteristic of TC organic or TC pesticide wastes were not subject to LDR and could be land-disposed with treatment.
After Phase II Final Rule
Mixed wastes only exhibiting the characteristic of TC organic or TC pesticide wastes can be land-disposed without treatment until 19 September 1996, or the mixed waste can be treated to meet the waste code specific treatment standards including underlying hazardous constituents.
Underlying Hazardous Constituents
Before Phase II Final Rule
For a treatability group that consists of aqueous liquids that exhibit only the characteristic of corrosivity and contained methanol (that was not a listed hazardous waste and did not exhibit the characteristic for TC organics) the required LDR treatment would be neutralization of the liquids. The waste would no longer be a mixed waste and could be disposed of as a low-level waste once it is solidified.
After Phase II Final Rule
Aqueous liquids (corrosive) treatability group that only exhibitS the characteristic of corrosive and contained methanol as an underlying constituent requireS an alternative treatment for the organic component of the waste. The preferred treatment option is to use evaporative oxidation for the methanol component followed by neutralization of the treated aqueous liquid and condensate. The neutralized liquids would then be stabilized. The waste would no longer be a mixed waste and could be disposed of as low-level waste.
WHAT TO EXPECT IN THE FUTURE
As a result of the U.S. Court of Appeals decision in 1992, EPA has a schedule according to which it will address the decisions in this court ruling. For example, there is an upcoming Phase III proposed rule that is expected in 1995 under which EPA will be examining treatment standards for characteristic hazardous wastes that will be managed in CWA systems, Safe Drinking Water Act (SDWA) Underground Injection

Control (UIC) class I wells, and for CWA zero-discharge facilities. The final rule of Phase III is scheduled for release in early 1996. The Phase IV proposed rule, expected in June 1995, will address treatment standards for the TC metal wastes (D004-D011). In addition, the mineral processing wastes, which were formerly excluded from RCRA Subtitle C regulation under the Bevill Amendment, are considered to be newly identified and will also be addressed in the Phase IV rule.

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PROGRESS ON INTEGRATING ENVIRONMENTAL RESTORATION DATA INTO MIXED WASTE UPDATES

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ABSTRACT

The Federal Facility Compliance Act required the Department of Energy to compile specific information on the volumes, characteristics, and treatment plans for mixed wastes, and to provide this information to the Environmental Protection Agency and affected states. The initial format for reporting this information was developed largely on the basis of operations wastes. Although this format is acceptable for stored environmental restoration wastes, it was found to be poorly suited for reporting wastes that could be generated by future cleanup activities, many of which have not yet been well defined. Modifications to the process initially used to report restoration wastes were made to allow them to be reported in a manner that more accurately reflects uncertainties inherent in such estimates. This revised approach allowed for the smooth integration of data for restoration wastes with those compiled by the waste management team for mixed waste inventory updates.

BACKGROUND

For the past two years, the Department of Energy (DOE) has been involved in an intensive effort to collect and consolidate information on mixed wastes from both operating facilities and inactive sites, to satisfy reporting requirements for the Federal Facility Compliance Act (FFCA). This information is being provided to the Environmental Protection Agency (EPA) and affected states, and it includes descriptive data on both stored and anticipated wastes, as well as treatment plans for those wastes. The approach and structure of the reporting mechanisms for this mixed waste information have evolved during this time, with the goal being a comprehensive framework that can be used for future updates.

Although environmental restoration (ER) wastes account for a major portion of the total waste volume anticipated from DOE activities over the next 30 years, the initial structure for reporting mixed waste information was developed primarily on the basis of operations waste. During preparation of the mixed waste inventory report and accompanying data bases, several issues were raised with regard to incorporating ER information into that format. Most problems arise from the fact that specific ER activities that will be conducted in the future are not yet known for most sites, so waste projections (let alone mixed waste projections) are preliminary at best. Thus, considerable uncertainty is associated with the plans and costs being developed for treatment facilities based on these early estimates many of which involve conservative assignment to the mixed waste category, absent characterization data that may not be available for several years.

It is important to recognize that these wastes may not be generated at all, depending on site-specific cleanup decisions. In-place responses such as capping or access restrictions are being evaluated for certain areas at a number of ER sites, so some contaminated environmental media such as soil and groundwater at these sites may never become waste. The staged nature of the decision-making process for these sites makes it difficult to provide full information on waste volumes and characteristics, and subsequent treatment plans, prior to the establishment of formal decisions that are developed from the input of many stakeholders including EPA, the states, and affected communities.

Problems also arise in projecting information for secondary waste streams, which include personal protective equipment that will be worn by remedial action workers, as well as filters, sludges, and spent resins from the anticipated treatment of

multiple waste streams. Additional difficulties are associated with data on wastes that may be generated by decontamination and decommissioning activities, which are currently being tracked by various programs and represent a considerable source of uncertainty in the volume projections for DOE mixed wastes. A similar issue is involved in tracking the transition of wastes from environmental restoration to waste management responsibilities. Further administrative issues include the staggered timing of updates for the mixed waste data base versus the site treatment plans (conceptual, draft, and proposed), and the unavailability of specific ER contacts in the field to help coordinate with waste management (WM) counterparts. Several of these issues and their resolutions are discussed further below.

ISSUES AND RESOLUTIONS

Level of Detail. Information was collected for the mixed waste inventory report (MWIR) data base at the "waste stream" level, and completion of a 30-page questionnaire was requested by the WM team for each waste stream. This level of detail was not available for ER wastes, especially for the materials that are projected to be generated, because characterization and response planning activities are ongoing at the restoration sites.

For stored ER wastes, this issue was addressed by completing the questionnaire to the extent information was available, using standard responses in many cases (for example, with regard to treatment plans where decisions had not yet been finalized and the options were therefore still being assessed). The waste streams were defined as single containers or a group of containers with wastes of similar physical properties and contaminant conditions. Although many ER wastes consist of heterogenous mixtures of contaminated media that do not fit easily into standard categories, they were assigned to the most appropriate treatability group code that had been developed by the WM team to facilitate summary roll-ups and integrated planning. The original questionnaire has also been revised and shortened considerably, to focus on key data needs for FFCA reporting.

For projected ER wastes, a simplified profile sheet was developed to capture on a single page the information requested by the FFCA, to the extent it was available. Standard responses were provided for many entries (such as "good management practices" for the waste minimization activities). In some cases, no volume estimates could be provided yet, but a profile sheet was prepared to present other information (such as the projected source and type of waste) and also to serve as a placeholder in the event that a given waste stream is generated by a future ER activity that is currently under consideration at the site.

Stage of Site Decision-Making Process. A considerable amount of information on treatment plans and waste projections was requested by the WM team for each waste stream. To provide these data, final decisions on the remedies that affect waste generation and treatment would have to be available for each site. Comprehensive decisions have not yet been made on the remedies to be implemented at most ER sites. Thus, the eventual fate of the variety of contaminated materials typically found at a site which may or may not involve the generation of mixed waste cannot reasonably be projected prior to the evaluation of response options and the receipt of input on proposed actions from the communities, states and EPA regions involved. To do otherwise, i.e., to provide details that presume a certain course of action, might imply that the decision-making process was being circumvented at these sites, and that is not the case. This issue was addressed by caveating the information provided for ER mixed wastes as preliminary. In addition, some responses were simply noted as "unknown" or "not available" at this time, and others included standard language (see examples above) that reflected the predecisional stage of the process.

Consistency with the Site Treatment Plans. Concurrent with the collection and presentation of information for the MWIR and its updates, sites have also prepared several evolving versions of site treatment plans (STPs) for issuance to EPA and the states according to an established schedule. Issuance of the MWIR and its updates has not followed this same schedule, so data differences have occurred and the inconsistencies have been questioned.

The development of characterization and planning information at ER sites is an ongoing process, and data will change over time. To resolve the issue of apparent inconsistencies associated with different reporting periods, the schedules of the MWIR data base updates and STP revisions are being synchronized. Closer coordination between the two reporting activities is also reducing duplicative data entry

efforts. For example, information from the 1994 MWIR data base is being preloaded into the format for the 1995 update, as supplemented by information from the current STP, so field personnel need only change those entries for which new information has been developed since the most recent reporting of mixed waste data.
Integration of ER and WM Treatment Plans. Information is being collected for the FFCA on treatment plans for mixed wastes from both operating (WM) facilities and restoration (ER) sites. Using a single reporting structure developed from an operational perspective did not accommodate some of the ER considerations, such as the evaluation of bench-scale and pilot-scale facilities and the possibility of multiple feeds that have not yet been well characterized, or in some cases even identified.
To address this issue, additional clarification and flexibility have been incorporated into the treatment facility questionnaire developed by the WM team for FFCA reporting purposes. Treatment plans for mixed waste originate at the site level, whether it is an ER or WM waste. Coordination between the respective elements at sites where both are present has helped integrate the treatment planning activities for the FFCA updates.
SUMMARY
Resolutions continue to be developed for these issues, and progress is being made toward establishing a workable structure for mixed waste reporting. Successful integration of ER mixed waste data into the overall FFCA reporting framework is critical, especially given DOE's plan to provide further updates of mixed waste information to EPA and the states. Applying an approach that acknowledges the preliminary nature of ER waste projections is helping to solve some of the problems associated with earlier mixed waste reporting for the FFCA. Also, close coordination between the ER and WM teams has helped accommodate the flexibility needed to reflect differences in the two programs and has increased the consistency of information being presented for DOE mixed wastes from operations and restoration activities.

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DEVELOPMENT AND STATUS OF THE AL MIXED WASTE TREATMENT PLAN
OR
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ABSTRACT
Nine Department of Energy (DOE) sites reporting to the Albuquerque Office (AL) have mixed waste, waste that is chemically hazardous and radioactive. The hazardous waste regulations require the chemical portion of mixed waste to be treated to certain standards. The total volume of low-level mixed waste at the nine sites is equivalent to 7000 drums, with individual site volumes ranging from 1 gallon of waste at the Pinellas Plant to 4500 drums at Los Alamos National Laboratory. Nearly all the sites have a diversity of wastes requiring a diversity of treatment processes. Treatment capacity does not exist for much of this waste, and it would be expensive for each site to build the diversity of treatment processes needed to treat its own wastes. DOE-AL assembled a team that developed the AL Mixed Waste Treatment Plan that uses the resources of the nine sites to treat the waste at the sites. Work on the plan started in October 1993, and the plan was finalized in March 1994. The plan uses commercial treatment, treatability studies, and mobile treatment units. The plan specifies treatment technologies that will be built as mobile treatment units to be moved from site to site. Mobile units include bench-top units for very small volumes and treatability studies, drum-size units that treat one drum per day, and skid-size units that handle multiple drum volumes. After the tools needed to treat the wastes were determined, the sites were assigned to provide part of the treatment capacity using their own resources and expertise. The sites are making progress on treatability studies, commercial treatment, and mobile treatment design and fabrication.
To date, this is the only plan for treating waste that brings the resources of several DOE sites together to treat mixed waste. It is the only program actively

planning to use mobile treatment coordinated between DOE sites.

THE PROBLEM OF MIXED WASTE AT DOE-AL SITES

In response to frustration with a lack of progress in the Department of Energy (DOE) in treating and disposing of mixed waste in accordance with the Resource Conservation and Recovery Act (RCRA), Congress passed the Federal Facilitates Compliance Act (FFCAct) in 1992. Generally, mixed wastes are wastes that have a hazardous component, as defined in the RCRA regulations, and a radioactive component. The FFCAct requires each DOE facility to negotiate a site treatment plan (STP) with the state in which the facility is located. The STP must specify how and when mixed waste will be treated.

Nine sites that have mixed waste report to the DOE Albuquerque Office (DOE-AL):

- Grand Junction Project Office; Grand Junction, Colorado;
- Inhalation Toxicology Research Institute; Albuquerque, New Mexico;
- Kansas City Plant; Kansas City, Missouri;
- Los Alamos National Laboratory; Los Alamos, New Mexico
- Mound Facility; Miamisburg, Ohio;
- Pantex Plant; Amarillo, Texas;
- Pinellas Plant; Pinellas, Florida
- Sandia National Laboratories, California; Livermore, California; and
- Sandia National Laboratories, NM; Albuquerque, New Mexico.

The level and type of radioactive contamination group the waste into different categories based on DOE definitions. The nine sites have low-level mixed waste, and a few sites have transuranic mixed waste. Transuranic mixed waste will be handled following a national program developed by the DOE. Treatment is therefore needed for only low-level mixed waste.

Waste Volumes

Approximately 7000 drums of low-level mixed wastes are at the nine sites. Volumes at individual sites range from 1 gallon at Pinellas to 4500 drums at Los Alamos. Five of the nine sites have less than 50 drums of waste, and three of those have less than 10 drums. Few waste streams are greater than 50 drums. The wastes are diversified. Even sites with small volumes have waste that requires a diversity of treatment approaches. For example, the Grand Junction Project Office has less than 10 drums of waste made up of 19 waste streams.

The Problem

The problem is that various treatment processes are needed at most of the nine sites. The treatment must be implemented quickly to meet the intent of the FFCAct. Because the activity is funded by taxpayers and there are serious competing needs for tax dollars, the treatment must be implemented as inexpensively and efficiently as possible.

THE AL MIXED WASTE TREATMENT PLAN

The purpose of the AL Mixed Waste Treatment Plan is to use the resources of the nine sites to create real treatment capacity for mixed waste that minimizes the time and cost. Each site is responsible for negotiating a site treatment plan with its state agencies. The plan offers resources outside those of the individual sites that can be used in planning the site treatment plan.

Methodology for Developing the Plan

The plan was developed by the Treatment Selection Team: four representatives of the sites, two representatives of DOE-AL, and one consultant on regulatory affairs.

The overall approach used to develop the plan is that used in the classical solution of any engineering problem:

- define the problem;
- determine what is given to work with;
- determine basis for solution; and
- solve the problem.

In defining the problem, the team took an approach different from past efforts. The team visited each site and discussed the waste, existing and planned treatment, and site capabilities. Instead of reducing the information into computer forms, the information was recorded as text wherever possible. This approach was important in characterizing the waste because it created a visual picture of the waste and allowed the team to maintain the true identity and character of the waste throughout the development of the plan. A data sheet with text describing the waste is included in the plan document for each of the 141 waste streams.

In solving an engineering problem, engineers identify givens, things that affect

decision-making. The team then prepared fact sheets for information affecting decision-making and included those in the plan document.
Each team member reviewed all the waste data sheets and fact sheets so that all team members had a common background for problem-solving.

Developing the Basis

Developing a basis bounds a problem so that an engineer can define the problem to solve it. To bound this problem, the waste streams were divided into treatability groups that were the basis for solving the problem. The 141 individual wastes found at the sites were manually separated into categories, then waste streams, and finally waste substreams that were treatability groups. Each progressive step recognized the characteristics of the waste that affect treatment. The 48 waste substreams or treatability groups were arranged on matrix sheets that include the quantity and site identification number for each waste.
A base treatment was selected for each substream. Base treatment is not the best treatment or a selected treatment, but a treatment approach the team thought could handle all the wastes in substream. Base treatments were treatment approaches that the sites recommended or that the team felt could treat all the waste in the substream.
The validity of the substreams as treatability groups was verified by ensuring that each waste included in a substream could be treated using the base treatment.

Developing the Solution

Several ground rules were established for developing the plan. The treatment options considered were directed toward the volumes and waste types found at the DOE-AL sites. Treatment options evaluated must be implementable within five years. Treatment options considered must have a realistic approach to shipping waste; that is, shipping waste to commercial facilities for treatment or shipping small volumes of waste between DOE sites for treatability studies is reasonable, but shipping between DOE sites for treatment or disposal is not practical in the short term. The ground rule on shipping waste is based on input on the states' attitudes during site visits. Finally, common sense must be used in rating and selecting alternatives. Solutions must fit the problems.
Using criteria that addressed regulatory standards, public acceptance, safety, scalability, and probability of success, alternative treatment approaches were rated against the base treatment for each of the 48 waste substreams. The top three treatment approaches were ranked as first, second, or third. If two or more approaches were rated equally, both were given the same rating. A list was made for each treatment option, which was ranked as first, second, and third choice for each substream. Matrices were prepared showing the waste that could be treated with each treatment option. These matrices are essentially client lists for the treatment approaches that rated the highest.
The matrix sheets were laid out on a table. The team assessed whether any of the treatment approaches were unavoidable, things that had to be done regardless of what other treatment approaches were used. The unavoidable treatment matrices were saved. Next, the team determined whether there were any treatment approaches for which there was no other option or for which a site was well along in design and fabrication of treatment capacity. These were also selected. The team then determined whether there were any obvious winners, treatment approaches that handle appreciable volumes of waste and that were easily implemented. These were selected. The team looked for and rejected obvious failures, treatment approaches that handle limited wastes. The selection process left nine treatment approaches on the table. The team evaluated each of these treatments individually to determine whether they fit into an overall approach.
Using the waste matrices showing the volumes and locations of waste in each treatability group, the team analyzed each selected treatment option and determining how it should be used. The actions needed to implement the options were then assigned to the sites. Distribution of assignments was based on available resources site interest and site expertise. The assignments are summarized in Table I.

Summary of the Plan

The plan makes use of treatability studies, portable treatment units in sizes ranging from bench-scale to skid-sized units, off-site treatment capacity, and the ability to survey some waste out of the radioactive designation.
The plan defines an activity for each selected treatment option and assigns a site to be project manager for that activity. The plan does not give specific direction

about how each site completes its assignments but allows each site to use its own initiative to find the most efficient approach to completing the assignment.
The plan establishes an Overall Program Manager, the Grand Junction Project Office (GJPO), to implement the plan, coordinate overall activities, and maintain a master schedule. Support working groups quickly resolve issues related to implementation of the plan.
Problem Areas
The plan presents some new concepts that create problems.
Interdependency of sites. Each site has tried to be self-sufficient in its waste management activities. The plan requires that sites depend on one another to create treatment capacity. This approach raises questions about how the concept can be incorporated into the site treatment plan negotiations and who is liable if one site fails to meet a schedule affecting other sites.
Permitting portable treatment. Permitting portable treatment units is no worse than each site's permitting multiple treatment units individually. The plan raises a question about whether there is a more efficient way of permitting portable treatment units that recognizes the individual rights of the states.
Transportation of portable treatment units. Using portable treatment units means that the units are moved to the waste, rather than the waste being moved to treatment. What are the states' concerns about moving portable treatment units?
Orphaned waste. The plan is based on the characterization data available at the time of the site visits. The treatment options selected are appropriate for a variety of wastes. Even still, some wastes are expected to be orphaned as characterization improves and the design and implementation of treatment alternative progress. The plan makes the Overall Program Manager responsible for evaluating orphaned waste and for determining whether additional treatment options are needed.
APPLICABILITY
The plan has direct applications in the Department of Energy. Using the plan, five of the nine sites can treat their low-level mixed waste in two years using off-site commercial facilities and treatability studies. Within five years, enough treatment capacity can be available to treat the wastes at the remaining sites.
The plan also has broad applications elsewhere.
Other DOE sites with similar volumes and types of waste will be interested in the possibility of expanding the use of mobile treatment units other sites.
Equipment vendors will be interested in the types of equipment they can supply to support building the mobile treatment units.
Commercial mixed waste generators will be interested in the methodology, which applies to their mixed waste problems.
Commercial waste management firms will be interested because the plan defines the types and volume of waste planned for commercial treatment, and because the plan may offer opportunities for these firms to provide packaged mobile treatment units or to operate the units.
CURRENT STATUS
Each site has prepared a project implementation plan for its assignment, and GJPO has prepared an overall schedule. The status of individual assignments is summarized in Table II. All the sites have used the AL Mixed Waste Treatment Plan as the basis for STPs.
Support working groups have prepared recommendations for approaches to permitting and preparing safety documentation for mobile treatment units. A recommendation is complete for supporting design and fabrication needs at sites needing help. A draft recommendation for using Type A containers for transporting and storing mobile treatment units is under review. Progress is being made on NEPA documentation, commonality of design issues, decontamination of units, and the feasibility of using portable enclosures.
The Portable Treatment Working Group developed a draft recommendation for maintaining and operating the mobile treatment that is being reviewed. The group strongly recommends that a central organization, either a DOE site or a commercial firm, be put in charge of all the mobile units providing treatment services to the sites.
CONCLUSIONS
Using commercial treatment and treatability studies, the AL Mixed Waste Treatment Plan can, in two years, bring five of the nine sites into compliance with the land disposal restrictions for storage under the Resource Conservation and Recovery Act.

Enough capacity will be created within five years to treat all the remaining mixed waste. The plan presents an economic alternative to treating mixed waste that can be expanded to other DOE facilities.

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MEDIATED ELECTROCHEMICAL OXIDATION AS AN ALTERNATIVE TO INCINERATION FOR MIXED WASTES*

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ABSTRACT
Mediated Electrochemical Oxidation (MEO) is an aqueous process which oxidizes organics electrochemically at low temperatures and ambient pressures. The process can be used to treat mixed wastes containing hazardous organics by destroying the organic components of the wastes. The radioactive components of the wastes are dissolved in the electrolyte where they can be recovered if desired, or immobilized for disposal.
The process of destroying organics is accomplished via a mediator, which is in the form of metallic ions in solution. The ions, initially at their lower valence state, are oxidized to a higher valence state at the anode surface. The oxidized ions in turn destroy organics throughout the bulk of the fluid. This results in a very efficient process, since the system can be optimized to oxidize the mediator at the anode, while the organics are dispersed at low concentrations throughout the anolyte. At Lawrence Livermore National Laboratory (LLNL) we have worked with worked with several mediators, including silver, cobalt and cerium. We have tested mediators in nitric as well as sulfuric acids.
We have recently completed extensive experimental studies on cobalt-sulfuric acid and silver-nitric acid systems for destroying the major organic components of Rocky Flats Plant combustible mixed wastes. The study was conducted on small-scale laboratory equipment and on a bench-scale facility that incorporated an industrial-sized electrochemical reactor. Organics tested were: Trimsol (a cutting oil), cellulose (including paper and cloth), rubber (latex), plastics (Tyvek, polyethylene and polyvinyl chloride) and biomass (bacteria). The process was capable of destroying almost all of the organics tested, attaining high destruction efficiencies at reasonable coulombic efficiencies. The only exception was polyvinyl chloride, which was destroyed very slowly resulting in poor coulombic efficiencies.
Besides the process development work mentioned above, we are working on the design of a pilot-plant scale integrated system to be installed in the Mixed Waste Management Facility (MWMF) at LLNL. In addition to the electrochemical process, this system will comprise the requisite secondary processes to replenish the reagents for continuous operation. These include processes for electrolyte regeneration and mediator recovery. The system will also be completely integrated with upstream and downstream processes (for example, feed preparation, off-gas and water treatment, and final forms encapsulation). The conceptual design for the MEO-MWMF system has been completed and preliminary design work has been initiated. Demonstration of the process with low-level mixed wastes is expected to commence in 1998.

INTRODUCTION
Mediated Electrochemical Oxidation (MEO) is an aqueous process which destroys hazardous organics by oxidizing a mediator at the anode of an electrochemical cell; the mediator in turn oxidizes the organics within the bulk of the electrolyte. With this process organics can be nearly completely destroyed, that is, the carbon and hydrogen present in the hydrocarbon are almost entirely mineralized to carbon dioxide and water. The MEO process is also capable of dissolving radioactive materials, including difficult-to-dissolve compounds such as transuranic oxides. Hence, this process can treat mixed wastes, by destroying the hazardous organic components of the waste, and dissolving the radioactive components. The radioactive material can be recovered if desired, or disposed of as non-mixed radioactive waste. The process is inherently safe, since the hazardous and radioactive materials are completely contained in the aqueous phase, and the system operates at low temperatures (below 80C) and at ambient pressures.

Mediated Electrochemical Oxidation was originally developed for dissolution of difficult-to-dissolve forms of transuranic oxides, but later was found to be effective for oxidizing many organic materials. Extensive development work on this technology has been carried out at PNL and at LLNL, in the United Kingdom, and in France (1-4). At LLNL, work in the past was concentrated on understanding the basic science and modeling the dissolution and destruction mechanisms. To this end, the reaction rates of water with Ag(II) were measured using spectrophotometric methods, and the diffusivity of silver ions in nitric acid was estimated using a rotating disk electrode (5). Other mediators studied were cobalt, iron and cerium. Mediators were tested in nitric and sulfuric acids. The breakdown of organics, such as ethylene glycol, was modeled in detail with the formation and eventual destruction of intermediate compounds (6-8). Dissolution of transuranic oxides was also modeled and system studies were conducted to optimize system operating parameters (9,10). Also, a full-scale system was built for transuranic oxides dissolution and tested with surrogate materials (11).

More recently work at LLNL has focused on the destruction of the organic components of major low-level mixed wastes streams at the Rocky Flats Plant (12,13). The work comprised the destruction of Trimsol, a cutting oil; various cellulosic substances, including wipes and cloth; biomass; and rubber (latex) and plastics (Tyvek, polyethylene and polyvinyl chloride). The emphasis of the work was on process development, that is, the determination of optimal process conditions for destruction of each organic. The parameters which can be controlled in the MEO process are system temperature, strength of the electrolyte, type of mediator and electrolyte, and current density at the electrodes. The destruction rates of various organics and the destruction and coulombic efficiencies were measured as a function of the process parameters which were varied in each experiment.

The destruction efficiency in this work refers to the total destruction of the organics, that is, the degree of complete mineralization of the carbon in the organics to carbon dioxide. Hence, it is a more stringent measure of destruction than the commonly used Destruction & Removal Efficiency (DRE). The DRE measures the destruction or removal of the original organics only, regardless of whether they are merely transformed into other organics, or are completely destroyed.

The coulombic efficiency refers to the theoretical amount of electric charge needed to destroy the organics versus the actual amount required. Usually the coulombic efficiency will be less than 100 percent since the total destruction of the organic requires a number of steps, some of which may be limited by homogeneous or heterogeneous (surface) kinetics. In such cases, so-called parasitic reactions of the mediator with the water in the electrolyte will consume some of the current. However, it should be noted that if the decrease in coulombic efficiency is due to operation of a cell above its limiting current, oxygen will be generated at the anode. This will not only waste current but will also require higher cell voltages. The combined effect of both will result in much higher power consumption. At LLNL great care has been taken to operate the cells at or below their limiting current. The limiting current is the maximum amount of useful current at the anode which is utilized to convert the mediating ions from their lower to their higher valence states. It depends upon the anode area and geometry, the mediator concentration, the mechanics of flow, and electrolyte properties that govern diffusion of the mediator to the electrode surface.

PROCESS CHEMISTRY

The chemical reactions given here are for a process based on silver as mediator and nitric acid as electrolyte. In addition to the electrochemical process, the secondary processes to regenerate the reagents and remove secondary wastes are discussed.

Starting with the reactions in the electrochemical reactor, silver in the form of Ag(I) is oxidized to Ag(II) at the anode:

$$Ag^{+} \rightarrow Ag^{++} + e^{-} \quad (1)$$

The redox potential for this reaction is 1.98 V with respect to a standard hydrogen electrode. The high potential allows the Ag(II) to attack organics, eventually converting the organic carbon to carbon dioxide. For example, for cellulose:

$$C_6H_{10}O_5 + 24Ag^{++} + 7H_2O \rightarrow 6CO_2 + 24Ag^{+} + 24H^{+} \quad (2)$$

The Ag(I)/Ag(II) redox potential is high enough to break down water molecules. Although this is a competing reaction which consumes Ag(II) ions, it may not be entirely parasitic since OH radicals may be formed as intermediate products. The complete reaction is written down as:

$$2H_2O + 4Ag^{++} \rightarrow O_2 + 4Ag^{+} + 4H^{+} \quad (3)$$

Ag(II) is very reactive and will dissolve many radioactive compounds which are otherwise very hard to dissolve. For example, many transuranic oxides are notoriously difficult to dissolve completely even in strong solutions of nitric and hydrofluoric acids. However, Ag(II) very efficiently dissolves the oxides converting them to ionic form:

$$MO_2 + 2Ag^{++} \rightarrow MO_2^{++} + 2Ag^{+} \text{ Eq.} \quad (4)$$

For the silver-nitric acid system, the fluids on the anode and cathode sides of the electrochemical cells must be kept separate, since nitrous acid which is produced at the cathode reduces Ag(II). The silver is introduced only at the electrolyte on the anode side (anolyte), and the electrolyte on the cathode side (catholyte) is separated by means of a porous divider or an ion selective membrane. The reactions shown above are those that occur at the anode or in the anolyte.
The hydrogen ions formed in the reactions above migrate through the divider to the cathode. There the nitric acid is reduced to nitrous acid. Hydrogen gas is not formed at the cathode unless the concentration of nitric acid there is low (below 2 M).

$$HNO_3 + 2H^{+} + 2e^{-} \rightarrow HNO_2 + H_2O \quad (5)$$

This reaction indicates the secondary processes required to support the primary electrochemical process. First, the nitrous acid generated must be converted back to nitric acid. Otherwise the nitrous acid will eventually decomposes to form NOx:

$$2HNO_2 \rightarrow H_2O + NO + NO_2 \quad (6)$$

Contacting the nitrous acid with oxygen regenerates the nitric acid, and prevents the formation of NOx:

$$2HNO_2 + O_2 \rightarrow 2HNO_3 \quad (7)$$

However, since oxygen is only very sparingly soluble in the solution, typical contactors in the form of packed bed columns would have to be very tall to attain reasonable conversion efficiencies. A turbo-aerator was developed at LLNL which achieves very high efficiencies in a small volume (14). The turbo-aerator draws the gas and the fluid together, and passes them through a row of stator blades which disperses the gas into very small bubbles. The intimate mixing and the high surface area contribute to measured efficiencies of 95 to 99%. The contactor is installed in the catholyte flow loop.
In addition to the nitrous acid, reaction 5 shows that water is generated on the cathode side. Although water is broken down on the anode side as shown in reactions 2 and 3, there is a net accumulation of water in the system when hydrocarbons are destroyed. Essentially, all the hydrogen present in the organic molecule is converted to water, which must be removed from the system. This is done using an evaporator in conjunction with a fractionation column. The electrolyte is passed through the evaporator and a small amount is evaporated off, enough to remove the extraneous water. The concentrated electrolyte is returned to the cells. The vapor from the evaporator contains both water and nitric acid, which are separated in the fractionation column. The column can be designed to produce water which is directly sewerable, while the acid is reused.
Besides the secondary systems to convert the nitrous acid and remove the water, a system is required for silver recovery. When chlorinated organics are destroyed in the anolyte, chloride ions are formed which immediately combine with silver ions to

form insoluble silver chloride. The silver chloride precipitate is removed from the electrolyte by settling or centrifuging. It is then introduced into a hot solution of sodium hydroxide and hydrogen peroxide. The silver chloride is reduced to silver:

$$AgCl + NaOH + H_2O_2 \rightarrow Ago + NaCl + H_2O + O_2 \quad (8)$$

The silver is filtered or centrifuged and dissolved in nitric acid for reuse. The chloride remaining in solution is in the form of salt (sodium chloride). It is dried and disposed of via polymer encapsulation.
Since silver chloride is almost insoluble, it affords a method for removal of silver from the electrolyte, when desired. For example, when the electrolyte becomes loaded with dissolved radionuclides and metals, they must be removed; but the silver in solution must be removed first for reuse. By adding chlorides into the electrolyte, the silver precipitates out and is separated. Then the solution is boiled off until almost dry by passing it through the thin film evaporator. The evaporator bottoms are carried out, and if desired, the radionuclides can be recovered by ion exchange. If recovery is not desired or feasible, the radioactive material is disposed of via grouting or ceramicization.
Hence, the MEO primary (electrochemical) process requires secondary processes for the conversion of nitrous acid to nitric acid, recovery of silver, and regeneration of nitric acid by removal of water. A functional flowsheet showing the relation of these processes is shown in Fig. 1. The flowsheet also shows the secondary waste generated and suggests further treatments that may be required before they are discharged or disposed. A conceptual design for the integrated MEO system based on this flowsheet has been completed for the Mixed Waste Management Facility (MWMF) at LLNL. Preliminary design for the system is currently underway.
Fig. 1.
Fig. 1. Flowsheet for Integrated Mediated Electrochemical Oxidation System.
EXPERIMENTAL WORK AND RESULTS
The experiments performed for electrochemical destruction of organics were of two types. A large number of small-scale experiments were performed where the process parameters could be easily varied. They were carried out in laboratory-sized H-cells with a capacity of destroying approximately 0.5 g of organics at a time in a batch mode. These experiments gave a preliminary assessment of the important process parameters and their influence on destruction rates. A smaller number of experiments were then carried out in a bench-scale facility containing an industrial-sized commercial electrochemical cell. This system was built with an Imperial Chemical Industries (ICI) FM-21 electrochemical reactor, which is widely employed in chlor-alkali chemical plants. The reactor presently contains a cell stack composed of two anodes and three cathodes. It has a capacity of 3000 A of limiting current at mediator concentrations of 0.5 M, and can destroy up to 0.5 kg/h of organics in either continuous or batch modes of operation. The cell voltage drop at maximum current is below 2 V. The bench-scale experiments were conducted to demonstrate organics destruction on large industrial-scale equipment so results could be confidently extrapolated to plant-sized operations.
The parameters varied in the small-scale experiments were temperature, acid concentration and acid-mediator combination. Two types of acid-mediator combinations were tested: silver in nitric acid, and cobalt in sulfuric acid. The concentration of the nitric acid was varied from 4 to 10 M, and that of sulfuric from 4 to 6 M. The temperature of the electrolyte was varied between 20 and 70 C. The concentration of the mediator in all tests was 0.5 M. In general, the results indicated that higher temperatures, higher acid concentrations and a more aggressive acid-mediator combination (silver with nitric acid) resulted in higher destruction rates. An example of the influence of these parameters is shown in Fig. 2 for destruction of cellulose. Here the initial destruction rate of cellulose is plotted as a function of temperature. Since the destruction rate is directly proportional to the limiting current in the cell it is normalized by the current. It is seen from Fig. 2 that the destruction rate rises steeply with temperature, especially for the cobalt-sulfuric acid system. The silver-nitric acid system outperforms the cobalt-sulfuric acid system, although they tend to converge at higher temperatures and acid concentrations. The effect of acid concentration is not very strong for cellulose, but that is not the case with all organics.
Fig. 2.

Fig. 2. Cellulose Destruction in Small-Scale Experiments.
Various organics of interest to Rocky Flats Plant were tested for destruction using the MEO process. These included different types of cellulosic substances (wipes and cloth), Trimsol (a cutting oil), rubber and plastics (latex, polyethylene, polyvinyl chloride and Tyvek), and biomass. A comparison of the destruction rates of the organics tested are shown in Table I. The results in the table are for the silver-nitric acid system at high acid concentrations (8-12 M) and high temperatures (70 C).It is seen that reasonable destruction rates were achieved for most organics tested except for polyvinyl chloride.
In addition to the destruction rates, the coulombic and destruction efficiencies of the process are of interest.In general, the destruction efficiency increases as more current is passed through the system, that is, with a decrease in coulombic efficiency. The results of the trade-off between destruction and coulombic efficiencies are shown in Fig. 3 for Trimsol and Fig. 4 for cellulose. These results are from experiments conducted in the bench-scale system with silver and nitric acid at acid concentrations of 8 M and temperatures of 70 C. It is seen in both cases that destruction efficiencies of 99% or higher can be achieved at coulombic efficiencies of 70%. To obtain destruction efficiencies of 99.99% requires coulombic efficiencies in the range of 40 to 50%.
Fig. 3.
Fig. 3. Destruction and Coulombic Efficiencies for Trimsol Destruction in Bench-Scale Experiments.
Fig. 4.
Fig. 4. Destruction and Coulombic Efficiencies for Cellulose Destruction in Bench-Scale Experiments.
CONCLUSIONS
The Mediated Electrochemical Oxidation (MEO) process is capable of destroying a wide variety of organics. Many such organics, contaminated with radioactive and RCRA-listed materials, are currently being stored at sites throughout the DOE Complex awaiting suitable means of disposal. MEO can be used as an alternative to incineration for treating these mixed wastes by destroying the organic components of the wastes. MEO mitigates some of the perceived problems with incineration, since it operates at low temperatures and pressures, and since the radioactive materials are contained in an aqueous media.
The primary electrochemical process requires a number of secondary processes to recycle the reagents. The secondary processes include removal of water and inorganics (including radionuclides) from the electrolyte, regeneration of the acid and mediator recovery. A conceptual design for the integrated process has been completed, and work has been initiated on the preliminary design to be installed in the Mixed Waste Management Facility (MWMF) at LLNL. MWMF is expected to demonstrate the treatment of low-level mixed wastes using the MEO process starting in 1998.
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CEMENTATION AND SOLIDIFICATION OF MISCELLANEOUS MIXED WASTES AT THE ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE

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ABSTRACT

The Rocky Flats Environmental Technology Site produces a variety of wastes which are amenable to micro-encapsulation in cement. Portland cement is an inexpensive and readily available material for this application. The Waste Projects (WP) group at Rocky Flats evaluated cementation to determine its effectiveness in encapsulating several wastes. These included waste analytical laboratory solutions, incinerator ash, hydroxide precipitation sludge, and an acidic solution from the Delphi process (a chemical oxidation technology being evaluated as an alternative to incineration). WP prepared surrogate wastes and conducted designed experiments to optimize the cement formulation for the waste streams. These experiments used a Taguchi or factorial experimental design; interactions between the variables were also considered in the testing. Surrogate waste samples were spiked with various levels of each of six Resource Conservation and Recovery Act (RCRA) listed metals (Cd, Cr, Ba, Pb, Ni, and Ag), cemented using the optimized formulation, and analyzed for leach resistance using the Toxicity Characteristic Leaching Procedure (TCLP). The metal spike levels chosen were based on characterization data, and also based on an estimate of the highest levels of contaminants suspected in the waste.
This paper includes laboratory test results for each waste studied. These include qualitative observations as well as quantitative data from TCLP analyses and environmental cycling studies. The results from these experiments show that cement stabilization of the different wastes can produce final waste forms which meet the current RCRA Land Disposal Restriction (LDR) requirements. Formulations that resulted in LDR compliant waste forms are provided. The volume increases associated with cementation are also lower than anticipated. Future work will include verification studies with actual mixed radioactive waste as well as additional formulation development studies on other waste streams.

INTRODUCTION

The Rocky Flats Environmental Technology Site (Rocky Flats) currently generates and/or stores mixed wastes which are subject to Resource Conservation and Recovery Act (RCRA) Land Disposal Restriction (LDR) treatment requirements. Cementation is a technically viable, economic solidification treatment process for some of these wastes and is a widely accepted waste management method (1). It is one of several solidification technologies proposed to bring these wastes into compliance with

RCRA-LDR requirements. Cement is inexpensive, readily available, solidifies under ambient conditions, and is capable of immobilizing RCRA hazardous metals in the wastes. Cements have been used for approximately 2000 years and show good durability to weathering (2-4) making this an attractive technology for waste encapsulation. Cementation of low-level radioactive and mixed waste forms (wastes containing both chemically hazardous and radioactive constituents) offers long term stability at low cost. This technology has been proven for several waste streams at the Rocky Flats Site including a spray dried salt/brine waste and multiple analytical laboratory waste solutions. These low level mixed wastes (< 100 pCi/g) are currently being cemented at a production scale to produce certifiable waste forms. This report summarizes cementation studies performed on five waste streams at the Rocky Flats Environmental Technology Site in Fiscal Year 1994.

DISCUSSION

Cementation development activities focused on five waste streams: analytical laboratory solutions, silver nitrate solutions, fluidized bed incinerator (FBI) ash, hydroxide precipitation sludge, and an acidic solution from the Delphi process. Activities on each of these wastes are discussed separately below. Please note that the term "percent" or "%" refers to weight percent unless noted otherwise.

Analytical Laboratory Solutions

The analytical laboratory waste solutions at Rocky Flats are generally acidic and consist of an aqueous solution of salts with a low level of plutonium (< 100 pCi/g). They also contain various RCRA listed toxic metals including silver (Ag), nickel (Ni), cadmium (Cd), chromium (Cr), barium (Ba), and lead (Pb). The waste must be neutralized prior to cementation; the neutralization process causes the dissolved metals to precipitate as oxides or oxyhydroxides (5) which reduces the solubility and mobility of the metals in the alkaline cement matrix (6). The treated waste must be RCRA-LDR compliant and pass the Toxicity Characteristic Leaching Procedure (TCLP) test. The cement,s alkalinity further neutralizes acidic solutions which could degrade the cement structure and increase the leachability of these RCRA constituents. After the waste is neutralized, the solution is poured into a 55-gallon drum two-thirds full of type I/II cement and Ramcote 1200 and is manually stirred in a production-scale operation. Each drum is capable of treating approximately 80 to 90 liters of waste solution.

Waste Projects conducted two sets of bench-scale experiments to evaluate the effectiveness of the process. The first experiment was a TCLP evaluation of the cemented waste form prepared using a spiked surrogate waste solution. The surrogate waste was a mixture of water, sodium chloride, potassium nitrate, heavy metal compounds (Ag = 5 mg/L; Ni = 10 mg/L; Cd = 50 mg/L; Cr = 300 mg/L; Ba = 300 mg/L; and Pb = 400 mg/L) , and sodium hydroxide. The surrogate composition was based on characterization data from the actual waste. The surrogate solution was prepared by dissolving appropriate quantities of the heavy metal compounds in 500.0 grams of distilled water. Sodium chloride (90.0 grams) and potassium nitrate (80.0 grams) were then added, followed by sodium hydroxide (2.0 grams). The temperature and pH were measured at each stage. The solution was then divided into five lots and added to the cement/Ramcote mixture.

These samples were prepared using surrogate waste spiked with "worst case" levels of the metals based on characterization data from actual waste laboratory solutions. This testing was used to confirm that the production-scale process produced a waste form with adequate leach resistance (7). A second series of experiments investigated the capability of the production-scale process to treat wastes contaminated with the six RCRA listed heavy metals but at significantly higher levels. The surrogate waste solution was spiked with Ag, Ni, Cd, Cr, Ba, and Pb at levels of 1,000, 5,000, and 10,000 mg/L each, and mixed with the cement/Ramcote mixture as before.

The samples were mixed by repeatedly inserting and withdrawing a metal spatula to simulate the production process. The samples were cured for 24 hours in an oven heated to 50C, and then transferred to the Environmental Technologies (ET) group within Rocky Flats Technology Development for TCLP testing. ET prepared the samples per standard TCLP requirements with leach levels of metals measured with an Inductively Coupled Plasma (ICP) spectrometer. The numeric average (xavg) and standard deviation (s) or pooled standard deviation (sp) of the leachability for each element were calculated to determine the upper confidence limit (UCL) of the leachability at a 90% confidence level using the following equations (8,9):

xavg = xleachability/n,

(1)
where, n = number of data points,
90% UCL = avg + (1.2804 s), or (2)
90% UCL = avg + (1.2804 sp).
(3)
The 90% UCL defines the upper leachability limit assuming the leachability follows a normal distribution. The numeric average and upper limit can be directly compared to the maximum allowable limits established by the Environmental Protection Agency (EPA) (10). The Colorado Department of Public Health and the Environment (CDPHE) is responsible for enforcing RCRA in Colorado. CDPHE expects Rocky Flats to adopt the Universal Treatment Standards (UTS) as established by the EPA.
The TCLP results from the 1,000 ppm spike, 5,000 ppm spike, and 10,000 ppm spike are shown in Table I. These results show excellent leach resistance for all metals tested up to 10,000 ppm with the 90% UCL being below the treatment standard for all three experiments. Data from the initial experiment also passed the TCLP testing but have not been included.
TABLE I
Silver Nitrate Studies
At Rocky Flats, silver nitrate is a small volume waste stream which is difficult to treat. This waste consists of several bottles of radioactively contaminated solutions with silver concentrations as high as 368,000 ppm. Recovery of the silver is not feasible due to radioactive contamination. This waste is scheduled to be processed in the previously described production process for treating analytical laboratory solutions. Surrogate solutions with a similar composition as the analytical laboratory solution were prepared. These solutions were spiked solely with silver in the form of $AgNO_3$ at concentrations of 100, 1,000, 10,000, 50,000, 100,000, and 500,000 mg/L, respectively. The required quantities of silver nitrate were dissolved in 500.0 grams of distilled water after which 90.0 grams of NaCl and 80.0 grams of KNO_3 were added. Finally, 2.0 grams of NaOH were added to neutralize the solution. Each solution was added to the cement/Ramcote mixture, mixed, cured at 50C for 24 hours, and transferred to ET for TCLP testing.
The results of this TCLP testing are shown in Table II. Analysis of the data shows all samples to pass the UTS for Ag at levels up to 100,000 ppm. Samples spiked with 500,000 ppm Ag, however, failed the TCLP test. These samples had a brown interior and a metallic-like coating which was presumably silver metal from a reduction of Ag^+ ions to Ag. It is believed that the NaCl reacted with $AgNO_3$ to form AgCl which is generally less soluble (11). As the amount of $AgNO_3$ was increased to the higher levels, however, there was insufficient NaCl available in solution to form AgCl. This resulted in excess Ag being available in the 500,000 ppm spike (as a metal or soluble compound) which caused this sample to fail the TCLP test. The 90% UCL was also noted to be unusually high for samples spiked from 100 to 100,000 ppm. This was due to excessive scatter in the TCLP data which skewed the pooled standard deviation.
TABLE II
Fluidized Bed Incinerator (FBI) Ash
Rocky Flats produced a Fluidized Bed Incinerator (FBI) ash from the fluid bed incineration of compressor oils, crankcase oils, office trash, solid waste, and diesel fuel. This ash also contains a significant amount of sodium carbonate (Na_2CO_3), which was used as an acid sorbent in the bed, and spent chromia-alumina (Cr_2O_3-Al_2O_3) oxidation catalyst.
For the cementation of incinerator ash, a Taguchi experiment was designed to evaluate fly ash loading (0%, 12%, and 23%), waste loading level (20%, 25%, or 30%), water/cement ratio (0.5, 0.6, or 0.7), and cement type (type I/II or type V). These levels were based on preliminary testing with a surrogate ash which had similar physical properties to the actual ash waste. Attributes to be analyzed were mix viscosity, waste form density, waste volume increase, and resistance to degradation after freeze-thaw cycling. This experiment did not consider interactions between the variables. Nine experimental runs were required to complete this test phase.
A "best case" (low catalyst-high carbonate) and "worst case" (high catalyst-low carbonate) surrogate ash samples were prepared by the Technology Development/Waste Projects group based on process knowledge. These formulations are shown in Tables III and IV. Samples of each were prepared and spiked with heavy metal compounds of Ag, Ba, Cd, Pb, and Ni; the spike level for each was 1,000 ppm of each heavy metal.

Cr was spiked into the surrogate in the form of the chromia-alumina catalyst. These surrogate wastes were mixed into the cement using a 30% waste loading, cured at 50C, and submitted for TCLP analysis. The results from these tests showed all heavy metal concentrations, except for Cr, to be below the treatment standard limits (refer to Table V). Additional samples were prepared with only the catalyst and cemented using waste loadings of 20% and 30%. The results from these tests showed the 20% waste loading yielded a waste form which met the treatment standard of 5.0 ppm for Cr (refer to Table V). The 30% waste loading sample again failed the TCLP test for Cr. The cause for this is still being investigated. In future studies, ash will be mixed with cement using a 20% waste loading and a water/cement ratio of approximately 0.6. This work is scheduled to be conducted in mid-1995.

TABLE III

TABLE IV

TABLE V

Hydroxide Precipitation Sludge

The hydroxide precipitation sludge was created from a process in Building 374 which consisted of adding magnesium sulfate, calcium chloride, iron(III) sulfate, and diatomaceous earth to an aqueous waste stream entering the building. The waste stream originated from several diverse locations such as storm water runoff and the analytical laboratories, process aqueous waste. Potassium hydroxide was added to the waste until the pH was greater than 10.5 which created a precipitation of magnesium, calcium, and iron ions along with most other metallic cations in the waste stream. The sludge was produced at a rate of approximately 18.5 m3 per year when the process was operational (pre-1990). Because the sludge is a mixed waste, it must be treated to meet RCRA-LDR standards. Process knowledge indicated that the waste stream had no appreciable concentrations of ions except for the various nitrate salts. Therefore, a surrogate waste was developed which was based almost entirely on the chemical additions to the waste stream and represents the composition of the waste prior to any pre-treatment. The composition of the surrogate waste is shown in Table VI.

TABLE VI

For cementing the sludge, a Taguchi experiment was designed to evaluate the effect of the following independent variables: class F fly ash loading (0% or 23%), waste loading (20% or 30%), water/cement ratio (0.5 or 0.6), and cement type (type I/II or type V). Interactions between specific variables (fly ash loading-waste loading, fly ash loading-water/cement ratio, and waste loading-water/cement ratio) were also considered. Waste loading considered the waste as a dry powder. The attributes tested were mix viscosity, waste form density, waste volume increase, and sample weight change after 12 freeze-thaw cycles. Determining the effect of changing the independent variables on the attributes required calculating the degrees of freedom (DF), response total (RT), correction factor (CF), total sum of squares (SSTotal), individual factor sum of squares (SSFactor), and mean square (MS). These are defined below:

DF = n - 1 (DF= 2 for each factor and n = number of trials), (4)

RT = Summation () of the individual responses,
(5)

CF = (RT)2/n,
(6)

SSTotal = (Response2) - CF,
(7)

SSFactor = (Factor at level 1 - Factor at level 2)2/
Number of responses,
(8)

MSFactor = SSFactor/DF
(9)

The pooled error is used to calculate the F-Ratio per the following equation:

F-Ratio = MSFactor/DFPooled error
(10)

F-Table values were chosen at a pre-determined 95% confidence level.(8)

Analysis of variance (ANOVA) tables were prepared to analyze these attributes to determine the significance of the independent variables and the interactions. For viscosity, the viscometer reading at 300 revolutions per minute (rpm) was selected as the point of reference as the cement tends to gel at lower viscometer settings

thus increasing the measurement error. Freeze-thaw cycling consisted of freezing the samples at -30C for 24 hours followed by thawing the samples at 50C for 24 hours to complete one freeze-thaw cycle. This process was completed for 12 cycles for the samples. The sample weight percent change was chosen as the parameter to be evaluated during freeze-thaw cycling because the samples were kept in the plastic vials during the tests. Any dimension changes in the samples would be masked by the presence of the plastic shell. ANOVA tables for cement density, waste percent volume increase, and final percent weight change (12 completed freeze-thaw cycles) are shown in Tables VII-IX.

TABLE VII

TABLE VIII

TABLE IX

Statistical analysis showed that none of the independent variables had a significant effect on the cement mix viscosity. The variation in the viscosity was insufficient to produce a significant variable. Observations indicated that batches with higher waste loadings and lower water/cement ratios yielded a more fluid mix when fly ash was incorporated. This is due to the lubricating properties of the fly ash particles. As expected, additional water in the mix made it less viscous, while higher waste loadings caused the mixture to thicken. The only batch which did not yield a workable mixture was that with a low water/cement ratio and a high waste loading. This produced a tacky viscous mixture which had poor flow characteristics. Some variables, while not significant with respect to mix viscosity, are known to improve the mixing consistency. The water/cement ratio will be important during scaled mixing using conventional mixing equipment. The Hobart mixer used in laboratory studies is capable of adequately mixing all batches with little or no problem since it is a robust mixer which can tolerate wide ranges of viscosities. Scaling the process to a larger mixer may require compositional adjustments as the larger mixers are not usually as rugged as the Hobart. Poor mixing would result in a non-homogeneous waste form; therefore, a minimum water/cement ratio of 0.6 is recommended for future testing.

The statistical analysis of the waste form density showed three variables to be significant (refer to Table VII). These variables were fly ash addition (A), waste loading (B), and water/cement ratio (C). Cement type (D) was not significant. These results allow one to estimate and predict the waste form density based on these results. This is accomplished by averaging the density obtained for a given level 1 value and also at the level 2 value. For example, the level 1 average for the 0% fly ash was 1.91 gm/cc whereas the average value for the level 2 experiments (23% class F fly ash) was 1.83 gm/cc. These results are shown graphically in Fig. 1. One drawback to this modeling approach is that there are only two data points; therefore only linear relationships are considered. A confirmation test showed moderate agreement with the experimental results (refer to Fig. 1).

Fig. 1.

For waste loading, the average density obtained for a 20% waste loading was 1.84 gm/cc, while the average density at a 30% waste loading was 1.90 gm/cc. These results are shown graphically in Fig. 2. Confirmation tests show a linear relationship and excellent agreement between the two sets of data. Increasing the waste loading increases the waste form density, the packing efficiency, and the final weight of the waste container. The differences observed would result in a minimal increase in weight for a 55-gallon waste drum. Finally, the additional water causes the cement mix density to decrease. The average density obtained at a water/cement ratio of 0.5 was 1.91 gm/cc while the average density at a water/cement ratio of 0.6 was 1.83 gm/cc.

Fig. 2.

For waste volume, the statistical analysis of the data showed only one variable, waste loading (B), to be a significant (refer to Table VIII). Volumes were extrapolated from known quantities or calculated using density and mass measurements. These calculations were made using the following equations:

Waste Form Volume = Total Mass/Average Waste Form Density, and (11)

% Volume Change = (Final Volume - Initial Volume)/Initial Volume x 100. (12)

Increasing the waste loading from 20% to 30% substantially decreased the volume of the cemented waste form. The average volume increase at a 20% waste loading was 133% when considering the volume of the waste as a slurry (waste plus the addition of 40% water) while a 30% dry waste loading resulted in a volume increase of only 50%

(refer to Fig. 3). Data obtained during confirmation tests showed excellent confirmation of these results. These predictions can be used to minimize the volume increase of the final waste form to maximize storage space without sacrificing the durability of the waste form. The volume increases observed in this testing are sufficiently low to make cementation a viable waste stabilization technology for the sludge waste.

Fig. 3.

The statistical analysis of the waste form weight change after 12 freeze-thaw cycles showed two variables to be significant at a 95% confidence level (refer to Table IX). These variables were waste loading (B) and water/cement ratio (C). The results predict that increasing the dry powder waste loading from 20% to 30% decreases the weight loss. Data obtained during this experiment showed the weight loss to decrease from 2.76% to 1.45%. This is shown graphically in Fig. 4. The reduced weight loss is due to the reduced moisture content that results from increasing the waste loading; the weight loss was attributed primarily to evaporation from the sample and not sample degradation. Likewise, increasing the water/cement ratio caused the weight loss to increase which again was due to evaporation from the material even though efforts were made to keep the vials sealed during testing. The data obtained during this experiment showed the weight loss to increase from 1.83% to 2.53% by increasing the water/cement ratio from 0.5 to 0.6.

Fig. 4.

Visual examination of the freeze-thaw samples showed no noticeable physical degradation. These freeze-thaw samples were sealed in plastic vials during cycling to minimize evaporation. Visual examination of samples exposed to the atmosphere showed minor surface cracking on samples containing class F fly ash and a 20% dry powder waste loading. It was possible to fracture these samples by hand with some effort. This was not possible on samples prepared using cement only. This is probably due to the high fly ash loading level which lowered the cement hydration and decreased the strength of the samples. The exposure cracking should not present a problem based on current LDR requirements with the waste form sealed in a drum or crate (12). There was also significant salt migration to the top and side surfaces of samples not containing fly ash. The presence of the fly ash in the mix tends to reduce the pore pressure within the samples and thereby decrease the migration of salt and moisture to the outer surface.

As part of the evaluation of the sludge, specimens were prepared and submitted for TCLP analysis. Heavy metals (Cd, Cr, Ba, Pb, Ni, and Ag) were spiked at a level of 1,000 ppm based on an aqueous sludge waste (sludge powder with 40% added water). This represents an extreme value as no appreciable heavy metal ions are in the waste based on process knowledge. An initial set of five samples was prepared and mixed by hand using a 30% dry powder waste loading and a water/cement ratio of 0.55. The higher water/cement ratio was used to yield a more fluid mix. The results (refer to Table X) show good leach resistance for all metals but lead. Five of the six TCLP data points for lead were well below the established treatment standard. It was believed that the one non-compliant point was an anomaly, so the test was repeated and all data points were well below the treatment standard value. This test was repeated using a 20% waste loading and a 30% waste loading with fly ash partially substituted for cement (refer to Table X). When the standard deviation was pooled and the isolated data point for lead was excluded, the leachability for lead was below the allowable limit of 0.37 ppm.

TABLE X

Samples of the cemented surrogate bypass sludge waste were analyzed by x-ray diffraction (XRD) and scanning electron microscopy (SEM) to detect the possible presence of expansive crystalline phases in the cemented waste. These analyses were conducted approximately 6 months after the experiment had been completed since these phases typically require significant time to form. XRD detected the presence of a small amount of ettringite (6CaOAl2O33SO432H2O) which could be problematic if present in sufficient quantities. The use of a type V cement could minimize the expansion problems associated with the presence of this crystal phase. The quantity of the crystalline phase in the waste was determined to be insignificant at this time (12). Samples will be routinely examined (approximately every two months) so that corrective measures can be taken if the amount of ettringite increases with time.

DETOX Solution from the DELPHI Process

The final waste stream studied is a strong acidic solution used in a chemical oxidation process being developed by Delphi, Inc., of Albuquerque, NM. This technology is being tested as an alternative to incineration for the destruction of organics in various wastes. The solution, known as the DETOX solution, is prepared by adding 60% anhydrous ferric chloride (FeCl3) to a 1.5 M hydrochloric acid solution. This strong acid, in the presence of a platinum-palladium-ruthenium catalyst, oxidizes combustible waste and reduces the waste volume. This acid must be neutralized and treated as a secondary waste prior to disposal. Since it is an aqueous waste, cementation is an appropriate technology to evaluate for stabilizing and immobilizing the RCRA listed species which may be present in the waste solution. The phase I cementation evaluation was conducted using a factorial experimental design evaluating fly ash loading (0%, 14.4%, and 28.8%) and waste loading (35% and 50%). The attributes tested were mix viscosity, waste form density, waste volume increase, and freeze-thaw cycling. The DETOX solution was mixed according to instructions provided by Delphi, Inc. This yielded a highly acidic solution (pH < 0) which was neutralized with a 1.5 N NaOH solution to a pH of 7-10. The neutralized solution was estimated to have a specific gravity slightly greater than water. During neutralization, the solution generated considerable heat; in fact, the solution boiled when attempting to neutralize the solution with 5 N and 10 N NaOH. The gas coming off the solution was a mixture of HCl and water which will require a scrubber in the production process. The use of the mildly basic solution resulted in a large volume increase as approximately 3-liters of NaOH solution were required to neutralize 250 ml of DETOX (volume increase of 1200%). Cement was added to the solution and mixed using a Hobart mixer and the samples were cured in an oven at 50C for 24 hours. The results from this experiment show the waste loading to be significant for each attribute evaluated. This is not surprising as the neutralized waste was mostly water with NaOH and solids precipitated from the DETOX solution. The phase II cement evaluation was conducted to test the leach resistance of a cemented spiked solution. A sample of DETOX solution was spiked with Cd (50 ppm), Cr (150 ppm), Ba (250 ppm), Pb (2,500 ppm), Ni (150 ppm), and Hg (50 ppm). These levels were based on the anticipated waste reduction factor of 300:1 provided by Delphi. Samples were mixed with cement and cement/fly ash using a waste loading of 50%. These results have been completed on all elements but mercury but have not yet been received. All elements tested were below the EPA treatment standard limits (13).

Experimental Conclusions/Recommended Formulations

1. Based on the tests conducted with the surrogate waste, the analytical laboratory solution waste can be cemented into a monolith using the following: 42.3% type I/II cement, 22.8% Ramcote, and 35% waste solution. The waste form produced has excellent leach resistance to Ni, Ba, Cd, Cr, and Pb at concentrations up to 10,000 ppm.
2. Based on the tests conducted with the silver nitrate waste, these solutions can be cemented into a monolith using the following: 42.3% type I/II cement, 22.8% Ramcote, and 35% waste solution. This waste form has excellent leach resistance to Ag spike levels up to 100,000 ppm, but failed TCLP testing at 500,000 ppm.
3. Cementation of the incinerator ash produces an LDR compliant waste using a formulation of 20% waste, 53.3% cement, and 26.7% water based on the tests conducted with the surrogate waste and the older treatment standard which was applicable at the time of testing. Additional tests are required to further optimize the cement formulation for this waste stream.
4. Based on the tests conducted with the hydroxide precipitation sludge surrogate waste, the sludge should be cemented into a mixture consisting of the following: 30% dried sludge (50% in slurry form), 21.8% type I/II cement, 21.8% class F fly ash, and 26.25% water (water/cement ratio of 0.6). This can be mixed using standard mixing equipment and has adequate resistance to freeze-thaw cycling. This waste product has leach resistance to Cd, Cr, Ba, Pb, Ag, and Ni at concentrations up to 1000 ppm.
5. A waste loading of 30% of the sludge in cement produces a waste form which meets pertinent waste acceptance criteria. The volume increase of the waste form is only 50%, making cementation a practical waste stabilization technology for this waste stream. Increasing the waste loading to 33% further reduces the volume increase to approximately 35%.
6. Class F fly ash, partially substituted for cement in the sludge waste formulation, improves the fluidity of the cement-waste mixture and can be used to reduce the water loading requirements. The substitution of fly ash for 50% of the

cement (total loading of 23% class F fly ash) showed no noticeable degradation on the sample after 12 freeze-thaw cycling tests. The presence of fly ash in the mix also reduced salt migration to the exterior surface of the samples due to a reduced internal pore pressure.
7. In tests conducted with the surrogate sludge waste, the presence of ettringite may require the use of type V cement (versus type I/II) to improve the resistance of the waste form to sulfate degradation.
8. Based on the initial tests conducted on the surrogate DETOX waste solution, cementation appears to be an effective stabilization technology for this waste. Initial tests with the neutralized acid have demonstrated waste loadings of 50% (from a neutralized acidic waste) can be cemented to produce an acceptable waste form. Samples have been submitted for TCLP analysis to confirm this. Additional refinements in the processing (i.e., neutralizing and drying the waste) will result in a reduced overall volume increase and produce a waste similar to the hydroxide precipitation sludge.

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42-1

RADIOACTIVE WASTE AND DECOMMISSIONING - PUTTING THE RECORD STRAIGHT

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ABSTRACT

Radioactive waste and decommissioning are often perceived as major impediments to public and political acceptance of the case for nuclear power. This is true, not only in the United Kingdom, but elsewhere. It is often argued that the issues of radioactive waste and decommissioning pose major problems for the nuclear industry which threaten public safety and the environment and which give unacceptable

financial burdens to present and future society. It is further argued that the industry does not know what to do, when to do it or how much it will cost.
This paper sets out to put the record straight on these issues by reviewing briefly the status of waste and decommissioning developments in the UK and overseas. This review covers aspects including the environmental impact, the technical and development status, economics and financing and overall strategies for the management of waste and decommissioning.
This paper shows that environmental impacts are very small and that technical solutions and strategies exist and are in place or are planned in many countries for all aspects of safe waste management and decommissioning. It goes on to show that waste and decommissioning costs are only a small fraction of nuclear electricity generating costs and that they are being provided for out of current revenues. In other words, we do know what to do, how to do it safely, when to do it, how much it costs and we are paying for it now.
It is concluded that the nuclear industry's greatest challenge in waste management and decommissioning is to bridge the gap between public perception and reality on these issues.

INTRODUCTION

Radioactive waste and decommissioning are often perceived as major impediments to public and political acceptance of the case for nuclear power. This is true, not only in the United Kingdom, but elsewhere. It is often argued that the issues of radioactive waste and decommissioning pose major problems for the nuclear industry which threaten public safety and the environment and which give unacceptable financial burdens to present and future society. It is further argued that the industry does not know what to do, when to do it or how much it will cost.
This paper sets out to put the record straight on these issues by reviewing briefly the status of waste and decommissioning developments in the UK and overseas. This review covers aspects including the environmental impact, the technical and development status, economics and financing and overall strategies for the management of waste and decommissioning.

ENVIRONMENTAL IMPACT

Gaseous and liquid radioactive wastes have been discharged to the environment from nuclear sites for many decades. Strict regulation is applied to limit the discharges and their impact, and routine environmental monitoring confirms that the risk to members of the public and to the environment is extremely small. In the case of UK power stations the annual radiation dose to the most exposed groups of the public from waste discharges is only a few percent of the natural background dose and to the general public it is negligible. From the nuclear industry as a whole, the dose to the UK population from waste discharges is less than 0.1% of natural background. A similar situation exists in other countries.
For solid radioactive wastes, storage, transport and disposal pose even smaller risks to the general public and to those living close to storage or disposal sites. The criteria set for such disposal sites in various countries, including the UK, are such that the risk to the public and to the environment will be virtually zero in practical terms.
The decommissioning of nuclear power stations and other nuclear plant is subject to the same stringent safety criteria as are in force during operation. Because the first steps in decommissioning include removal of fuel, radioactive wastes and other potentially dispersable radioactivity for safe storage or disposal, the potential risks remaining are drastically reduced. In the case of our gas-cooled power stations, for example, the removal of fuel reduces the radioactivity by more than a factor of a thousand. The remaining radioactivity is almost all in solid indispersable form as active steel, graphite and concrete contained within substantial structures. This radioactivity reduces naturally with time by many orders of magnitude. During the whole period of decommissioning including final dismantling and site clearance the radiological impact on the environment is so small that it could not be detected.

TECHNICAL AND DEVELOPMENT STATUS

All countries with nuclear industries have radioactive wastes to manage as do many countries without nuclear industries since radioactivity is used extensively in medicine and industry. Extensive international cooperation has existed for many years on the technical and engineering aspects of safe radioactive waste management. National organizations have been established in most countries to manage, store and

dispose of wastes and there is wide international agreement on the best techniques for the safe management of the many forms and types of wastes that exist.
An important factor in the ability of the industry to deal safely and economically with its radioactive wastes is that the volumes are, in industrial terms, relatively small. This can be illustrated by the fact that the total radioactive waste volume requiring storage and deep underground disposal in the UK up until the middle of the next century, and including all past accumulations from the civil and defence industries, amounts to only about 400,000 m3. This can be compared to the annual volume of toxic industrial waste arising in the UK of 4 million m3.
It is agreed internationally by various Agencies and governments that the end point of safe radioactive waste management is disposal in engineered repositories at or near the surface or deep underground depending on the type of waste and, in particular, on its longevity. Wastes containing radioactivity which reduces rapidly with time can be safely disposed of near the surface whereas those which persist for thousands of years must be isolated by geological barriers as well as engineered barriers and hence must be placed deep underground in suitable geology.
In the UK, a surface engineered repository is in operation at Drigg in Cumbria and is capable of disposing all the country's low level waste until at least the middle of the next century. Similar repositories are in operation in other countries including France, Spain, Japan and the USA. Underground repositories are operational in Sweden and Finland for similar wastes.
UK Nirex Ltd is currently investigating the suitability of a site near Sellafield for the UK's intermediate level wastes. The repository would be at a depth of about 650 meters and could be operational by about 2010. Only one deep underground repository is currently in operation, at Morsleben in Germany, but many countries plan to construct deep repositories for operation early next century. These include repositories for the disposal of high level (or heat generating) wastes and spent reactor fuel in Sweden, Germany, France, Belgium, USA and Canada.
As with radioactive waste management, decommissioning has been the subject of extensive international collaboration over many years mainly under the auspices of the IAEA, OECD and the CEC. The collaboration includes research and development, demonstration projects and studies of the costs of decommissioning. Although nuclear plants in different countries vary in design and national regulations also vary, there is agreement on the techniques and plans for dismantling and dealing with the wastes. The technology involved has been developed and demonstrated and no major research or further development is required other than for optimization of the decommissioning process.
There are some 80 power reactors around the world in various states of decommissioning. Some have been totally dismantled (e.g. Shippingport in the USA and JPDR in Japan) and others are in the process of being dismantled (e.g. WAGR in the UK, KKN and KRB in Germany and Fort St. Vrain in the USA). These 'demonstration projects' coupled with experience of major component replacement in many plants give confidence that reactor decommissioning can be carried out using today's technology. A similar situation exists with regard to other nuclear plant used in the nuclear fuel cycle such as reprocessing and enrichment plants and plutonium facilities. BNFL in the UK have significant experience of decommissioning such plant as have others in Germany, France and the USA.
This worldwide experience gives confidence in the feasibility of decommissioning. It also demonstrates the safety of decommissioning and enables reliable cost estimates of future decommissioning tasks to be made. Far from being an unsolved problem, decommissioning can be done and the techniques are available to do it safely. The challenge that remains for the industry is the optimization of the strategy for decommissioning each individual plant taking account of local factors, safety and costs.
ECONOMICS AND FINANCING
As mentioned above, international experience and exchange of information on waste management and decommissioning is extensive. In particular, information on the costs of all aspects is widely available so that nuclear plant owners are able to make estimates of their own costs and future liabilities. These estimates are made using conventional engineering project costing techniques including contingencies and risk margins as appropriate to the particular situation. Taking account of local differences in cost elements there is good international consensus on the overall costs of waste management and decommissioning. Although these costs are often high

in real terms, when levelized and discounted over the appropriate period spanning the incidence of expenditure, they represent only a small percentage of the cost of nuclear electricity.

By way of illustration, the levelized and discounted waste management and decommissioning costs for Nuclear Electric's power stations are shown in Table I. There are two main reasons why these levelized costs are small when facilities such as repositories may cost many hundreds of millions of pounds and each power station decommissioning costs a similar amount. Firstly, much of the expenditure occurs much later than collection of the revenue from electricity and future expenditure is discounted, albeit at the modest real rate of 2% per annum. Secondly, the total electricity production associated with the facilities needed and the power stations which must be decommissioned is very high. For example, the present value of the total nuclear electricity from the UK nuclear power stations' program (2.5 million million kWh) is about 150 billion retail or about 60 billion at the current Regulator-capped pool price.

Since much of the expenditure on waste management and decommissioning occurs in the future and nuclear operators accept that the 'polluter pays' principle must be applied, arrangements are made to ensure that funds are available to meet the costs when needed. In most countries, including the UK, this is done by establishing provisions to charge the costs against current revenue. The provisions are handled by slightly different methods in different countries but nearly all nuclear utilities invest the provisions in their businesses or set aside segregated funds so that all future liabilities can be met. At the present time, Nuclear Electric's provisions are partly invested in capital assets, partly in indexed linked gilts, partly as cash deposits with the Government and the remainder will be covered by the nuclear levy.

STRATEGY

UK radioactive waste management strategy conforms to the overall Government policy which sets the framework within which to operate. Simplified, Government policy is for low and intermediate level wastes to be disposed of as soon as possible and for high level waste to be vitrified and stored for at least 50 years before disposal. The industry's strategy is thus to make full use of the existing near-surface repository for low level waste at Drigg and, through UK Nirex Ltd, to develop a deep underground repository for intermediate level waste as soon as possible. Pending the availability of this repository all intermediate level waste is being accumulated safely in surface storage facilities generally on the sites of origin of the wastes. High level wastes, which arise principally at Sellafield, are currently being vitrified and stored in stainless steel containers in the air cooled store at Sellafield. The industry plans to dispose of these in a purpose built deep underground repository in the second half of the next century.

For decommissioning, each nuclear plant operator has a strategy setting out what is to be done and when for each particular installation. For example, for UK gas cooled reactor power stations, a strategy referred to as Safestore is proposed. This involves early removal of the fuel from the reactors and early dismantling of the conventional plant and buildings. Dismantling of the reactors themselves will take place around 130 years or so later in order to take advantage of the much reduced radioactive inventory at that time. The defuelling stage is already completed at Berkeley ahead of program and at a cost well below the provision set aside. At Hunterston A and Trawsfynydd the defuelling stages are progressing well.

PUBLIC PERCEPTION

Public perception of radioactive waste and the decommissioning of nuclear plants is a long way removed from the situation described in this paper. This is not just a UK phenomenon, the situation being very similar in other countries. It is widely believed that radioactive waste is a large unmanageable problem posing threats to present and future generations and to the environment. Furthermore it is perceived to be a problem which will lead to unacceptable financial burdens on future generations. Decommissioning is seen in a similar light since it is regarded as just another radioactive waste problem.

A major challenge to the nuclear industry is thus the bridging of the gap between public perception of radioactive waste and decommissioning and the true situation. To date, the efforts of the industry to inform the public of what has been achieved in waste management and of the experience of decommissioning have failed.

The industry must therefore make greater efforts to inform the public about waste

management and decommissioning. We must publicize the achievements (e.g. operating waste repositories and the demonstration decommissioning projects) by encouraging media coverage and visits. Billions of pounds are committed to the technical activities associated with waste and decommissioning -perhaps we should consider whether we are doing enough on familiarizing the public with them.

CONCLUSIONS

The risks to the public and to the environment from radioactive waste and from decommissioning are extremely small. Technical solutions for the safe management of all radioactive wastes exist and are in place in various countries. Decommissioning nuclear plant has been demonstrated to be feasible and safe. The costs are known and represent only a small fraction of the cost of electricity and money is being set aside to pay for future liabilities.

Contrary to popular belief on waste and decommissioning, the industry does know what to do, how to do it safely, when to do it, how much it costs and is paying for it now. The industry's biggest challenge on these issues in the UK and in other countries is to bridge the gap between public perception and reality.

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THE DECONTAMINATION AND DECOMMISSIONING DEBATE

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ABSTRACT

For those concerned with radioactive waste, few issues loom larger than those relating to the decontamination and decommissioning of radioactively contaminated sites. Emerging developments in this area involving the Nuclear Regulatory Commission (NRC), the Environmental Protection Agency (EPA) and other governmental entities will soon result in generic regulatory programs determining who must decontaminate, to what levels, how much radioactive waste will be generated by these activities and what will be the conditions for disposal of such waste. Currently, NRC and EPA are in the process of developing separate regulatory regimes governing decontamination and decommissioning (D&D). If these efforts are to be successful, the two agencies must avoid overlapping and inconsistent requirements, and each set of regulations must be based on sound scientific principles.

INTRODUCTION

As we approach the start of the twenty-first century, one issue that promises to assume increasing importance among those concerned with radioactive waste is the decontamination and decommissioning of radioactively contaminated sites. The sharpening focus on this issue can be linked to a number of factors. The maturing of the nuclear power industry means that a growing number of older facilities, including facilities associated with the nuclear fuel cycle, will be coming off-line or ceasing operation, which raises a number of critical questions vis-a-vis the decontamination of these facilities, the disposal of wastes generated as a result decontamination, and the ultimate disposition of these sites. Similarly, ongoing clean-up efforts at Department of Energy (DOE) and Department of Defense (DoD) facilities has sparked intense public interest and increased regulatory concern over decontamination and decommissioning (D&D) activities at federal facilities, generally. In addition, there is a heightened awareness, among both regulators and the public, of the pervasive nature of naturally occurring radioactive material (NORM). Widespread NORM contamination may profoundly effect efforts to clean up radioactively contaminated sites, and will have a substantial impact on the disposal of radioactive wastes generated through site decontamination efforts.*

The scope of the decontamination and decommissioning issue is potentially enormous. Nuclear Regulatory Commission (NRC) regulations currently define the term "decommissioning" to mean "to remove nuclear facilities safely from service and to reduce residual radioactivity to a level that permits unrestricted use and termination of the license. Decommissioning activities are initiated when a licensee decides to terminate licensed activities" (1). Decontamination is one element of the decommissioning process. It is designed to reduce or eliminate radioactive contamination of equipment, facilities and the surrounding environment, so that the facilities, equipment and sites can either be released for future unrestricted use, disposed of, or placed under controls intended to minimize risks to public health and safety.

Currently, there are approximately 22,000 licensees under the Atomic Energy Act (the AEA, 42 U.S.C. 2014 et seq.), licensed either by the NRC or by Agreement States pursuant to AEA section 274. NRC estimates that more than 2,000 of these sites will require "substantial" decontamination when their licenses are terminated (2). Overall, the Environmental Protection Agency (EPA) estimates that there are at least 5,000 radioactively contaminated sites in the United States that will potentially require D&D attention. This figure does not include the approximately 1.5 million sites that are potentially contaminated with NORM (3). According to EPA, the oil and gas and geothermal energy sectors by themselves produce more than 50 separate waste streams amounting to several billion tons of NORM waste generated each year (4). Clearly, D&D issues will become increasingly prominent as more and more radioactively contaminated sites reach the end of their productive lives and as clean-up efforts at these sites begin in earnest. Not surprisingly, given the importance of these issues, a number of state and federal agencies have moved to develop regulations governing various aspects of decommissioning and decontamination. Chief among these efforts are two regulatory initiatives currently being undertaken by NRC and EPA.

CURRENT REGULATORY INITIATIVES

As discussed below, NRC and EPA are separately developing regulations to govern D&D activities. NRC is proceeding under the authority of the AEA to develop regulations that will apply to all licensed facilities under the AEA, as well as to other facilities under NRC's jurisdiction (other than high-level and low-level waste disposal facilities and uranium mill tailings piles). EPA is also acting pursuant to AEA authority, which was transferred to it by Reorganization Plan No. 3 of 1970 (5). EPA's D&D regulations would apply to all federally owned or operated sites and sites licensed by NRC (except for high level waste disposal sites, mill tailings piles and Superfund sites with signed RODs).

Although they are proceeding separately, NRC and EPA have agreed to cooperate with one another in developing consistent regulatory regimes. Specifically, on March 16, 1992 the two agencies entered into a Memorandum of Understanding (MOU) which is intended to establish a basic "framework for the agencies to resolve issues of mutual concern" and which sets forth principles and procedures designed to avoid unnecessary duplication of regulatory requirements (6). As both agencies have acknowledged, the MOU applies to the proposed EPA and NRC decommissioning and decontamination standards. The two agencies anticipate that EPA will review the D&D standards promulgated by NRC, and if they are found sufficiently protective, EPA will not attempt to impose its own D&D requirements on NRC licensees. According to EPA, if the Agency "determines that NRC's [D&D] regulatory program achieves a sufficient level of protection . . . EPA will propose that NRC regulated sites will be exempted from EPA radiation site cleanup regulations" (7).

Nevertheless, although NRC and EPA have committed to work together to develop consistent, albeit separate, regulatory regimes governing D&D activities, it is not at all clear that the two agencies will ultimately be successful in this effort. Indeed, as discussed below, the preliminary rulemaking efforts of each agency display a disturbing degree of overlap and inconsistency. Perhaps even more troubling, however, is that, in their rush to develop regulations in this area, NRC and EPA appear to be regulating on the basis of policy judgments that have been made without solid scientific foundation.

NRC'S PROPOSED D&D RULE

In 1989, the General Accounting Office (GAO) issued a report on NRC's regulation of D&D activities. The report, titled NRC Decommissioning and Decontamination Procedures Need to be Strengthened, RCED-98-119 (May 26, 1989), was critical of NRC's failure to establish specific, enforceable D&D standards and procedures. Partly in response to that criticism, NRC initiated an "enhanced participatory rulemaking" process aimed at developing more effective D&D regulations. This enhanced participatory process, which involved numerous public workshops and "scoping" meetings, reached partial fruition in August of 1994, with NRC's publication of a proposed rule to establish radiological criteria for the decommissioning of sites licensed under the AEA (8). Briefly, NRC's proposed rule would establish numerical criteria for determining whether a site has been sufficiently decontaminated to allow termination of the license and release of the site for other uses. The Commission is proposing a risk-based limit for residual radioactivity, keyed to the total effective dose equivalent (TEDE) that an average

member of the "critical group" would receive following decommissioning. (The "critical group" is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances.) This risk/dose based limit would be augmented by a requirement in the regulations that the licensee reduce residual radioactivity on the site to levels that are as low as reasonably achievable (ALARA).

Under the regulations proposed by NRC, a site would be eligible for release for unrestricted use if the TEDE for the average member of the critical group is 15 mrem/yr or less from residual radioactivity distinguishable from background. For purposes of the proposed rule, the TEDE for members of the critical group would be calculated over 1,000 years. In addition, as indicated, the proposed regulations would also require the licensee to demonstrate that residual radioactivity at the site had been reduced to ALARA levels before the site would be released. In an earlier draft of the rule, NRC staff had proposed a "goal" of 3 mrem/yr to demonstrate compliance with ALARA. In other words, under the NRC staff draft, the ALARA requirement would have been deemed satisfied in instances where the TEDE from residual radioactivity distinguishable from background had been reduced to 3 mrem/yr. This numerical "goal" of 3 mrem/yr appears to have been dropped from the proposed rule (although there is some indication in the preamble to the proposed rule that NRC still views 3 mrem/yr as the de facto "goal" for ALARA); the qualitative ALARA requirement remains in place.

Another important feature of the proposed rule is a provision which would permit a decommissioned site to be released for restricted use in instances where it would be unreasonable to remediate the site to achieve the 15 mrem/yr limit for unrestricted use. As NRC explains in the preamble to the proposed rule, "in those few cases where reducing the residual radioactivity to the levels required to comply with the 15 mrem/y TEDE limit for unrestricted use are either not technically achievable, would be prohibitively expensive, or would result in net public or environmental harm, . . . the proposed rule provides the licensee with the option of requesting release of the site with restrictions placed on its use" (9).

Under the terms of the proposed rule, a site may be released for restricted use if provisions are made for "institutional controls" on the site which provide reasonable assurance that the TEDE from residual radioactivity distinguishable from background will not exceed 15 mrem/yr for the average member of the critical group. In addition, residual radioactivity at the site would have to be remediated such that, in the event that institutional controls on the site were no longer effective, there would be reasonable assurance that the TEDE from residual radioactivity would be as low as reasonably achievable, and in any event would not exceed 100 mrem/yr. Finally, NRC is proposing that decommissioned sites may be "reopened" and that further remediation may be required if, based upon "new" information developed after a site has been decommissioned, NRC concludes that the level of residual radioactivity at the site poses a significant risk to the public (10). According to NRC, "new" information which could lead to reopening of a site under this provision includes information pertaining to additional contamination as well as information that has a bearing on the risk and/or health assumptions underlying the criteria established in the proposed rule (11).

There are a number of important weaknesses in the approach adopted by NRC. Although it is not possible, in this paper, to provide an exhaustive analysis of the proposed regulations, some of the more troubling aspects of the rule are discussed below.

Appropriateness Of NRC's Proposed Dose Limit

The central feature of NRC's D&D proposal is the dose limit that would be established for unrestricted release of a decommissioned site. Most of the significant provisions of the rule are keyed to this limit in one way or another. Despite the importance of this parameter, however, NRC has completely failed to justify, in a rigorous and scientifically valid manner, its choice of 15 mrem/yr as the proposed dose-limit.

In the first place, NRC has not demonstrated that the 15 mrem/yr limit is needed to address any real world risks to public health or the environment. The Commission offers no evidence, either factual or theoretical, to support a conclusion that 15 mrem/yr, or even higher doses, presents a significant potential health risk. Indeed, current scientific knowledge supports the contrary conclusion -- that doses at the 15 mrem/yr level do not represent a significant threat to public health or the environment. As NRC itself acknowledges, in its support documents for the D&D rule,

15 mrem/yr is at the low end of the range of doses that are received from natural background radiation, and it is an order of magnitude lower than the average background dose received in the United States. According to NRC, "[b]ackground is comprised of various sources of ionizing radiation which collectively produce an average total effective dose equivalent of about 300 mrem/yr to a U.S. resident. Radiological doses from background typically range between 100 mrem/yr and 1,000 mrem/yr in the United States. Although greater radiological doses are possible for people living in houses with very high radon concentrations, 1,000 mrem could be taken as a practical maximum . . . For comparison, the estimate of the average U.S. radiological dose from background is similar to the world average estimate of 240 mrem/yr" (12). Thus, the 15 mrem/yr limit proposed by NRC represents roughly one seventh of the dose received from low-end natural background and one twentieth of the average dose received from background radiation in the United States.

It is widely accepted that natural background levels do not pose a significant threat to human health or the environment. For example, the National Research Council's committee on the Biological Effects of Ionizing Radiation (BEIR) has concluded that "studies of populations chronically exposed to low-level radiation, such as those residing in regions of elevated natural background radiation, have not shown consistent or conclusive evidence of an associated increase in the risk of cancer" (13). Similarly, the National Council on Radiation Protection and Measurements (NCRP) has noted that because the human race has developed acceptably in an environment with ubiquitous natural background radiation, the risks (such as they are) associated with natural background can be considered a "normal factor of life" (14).

Since the available evidence indicates that exposure to natural background levels of radiation does not present a significant risk to public health, it follows that doses that are at the low end of natural background (and that are substantially lower than the average dose received from natural background) would also not pose a significant risk to public health. Indeed, according to the International Atomic Energy Agency (IAEA), " [a] level of dose which is small in comparison to natural background can be regarded as trivial" (15). In other words, NRC's proposed limit of 15 mrem/yr represents a "trivial" level of exposure; there is no indication that such a dose presents a significant risk to human health or the environment.

This raises the obvious question of whether it is appropriate for NRC to attempt, through regulation, to mitigate "risks" that are at most trivial. The answer that the law seems to provide to this question is "no;" agencies should not exercise their regulatory powers to address trivial risks. One of the seminal legal decisions in this area comes from a Supreme Court case known as the Benzene case. Industrial Union Dept. AFL-CIO v. American Petroleum Institute, 448 U.S. 607 (1980). At issue there was whether certain regulations which had been promulgated by the Occupational Safety and Health Administration (OSHA) to control workplace exposure to particular carcinogens were enforceable as a matter of law. The regulations at issue had been promulgated pursuant to OSHA's "carcinogen policy," which was predicated on the assumption that carcinogens have linear, non- threshold dose response curves (or put another way, there is no "safe" level of exposure to a carcinogen) and that, therefore, regulatory controls on carcinogens should be made as stringent as possible. In effect, OSHA took the position that because there is no "safe" level of exposure to a carcinogen, such exposure should be limited by regulation to the extent feasible, regardless of whether the levels being regulated pose a significant risk to worker health and safety. (As we discuss below, this is essentially the same position that NRC has taken in the D&D proposal with respect to exposure to residual radioactivity --namely, that it should be reduced to the maximum extent possible, regardless of the level of risk being mitigated.) The Court in the Benzene case disagreed with this approach, noting that "the government's theory would give OSHA power to impose enormous costs that might produce little, if any, discernible benefit" (16). In effect, the Court concluded that government agencies should not attempt to regulate trivial risks; instead, they should only regulate risks that are "significant." As already discussed, NRC's proposed dose-limit of 15 mrem/yr represents, at most, a trivial risk to health or the environment. Under the principles announced in the Benzene case, exposure to radioactivity at that low level should not be the subject of regulation.

NRC's approach to developing the 15 mrem/yr standard is flawed not just from a legal perspective, but also from the standpoint of sound science. In the preamble to the

proposed rule, NRC notes that the proposed standards are predicated on a linear, non-threshold dose effect model for radiation. It is widely accepted that when attempting to regulate a linear non-threshold risk, optimized decision-making requires an ALARA analysis that seeks to maximize the net benefit resulting from regulation (17). Without this type of analysis, it is impossible to determine whether the risks that are sought to be mitigated are amenable to control, and whether the controls that are imposed by regulation involve the most efficient application of resources in relation to the risk reduction benefits to be achieved. Instead of conducting a net-risk/net-benefit analysis, however, NRC proceeds from the linear non-threshold assumption directly to the conclusion that "the overall objective for decommissioning should be the return of the facility to levels approximating background" (18). There is no analysis of whether this return-to-background goal is consistent with the objective of obtaining the maximum net benefit from the controls being imposed.* Indeed, a net-risk/net-benefit analysis of the 15 mrem/yr standard would almost certainly reveal either a net benefit of zero or even a net increase in risk, since such an analysis would have to balance the trivial incremental risk reduction achieved by the 15 mrem/yr dose limit against the increased risks (both radiological and non-radiological) associated with a dose limit that approaches natural background levels -- which includes risks associated with removal, transportation, treatment and disposal of additional wastes generated in attempting to comply with the dose-limit.

The dubious validity of NRC's 15 mrem/yr dose-limit is also reflected in the fact that the standard selected by NRC is inconsistent with limits that have been developed by independent scientific organizations. For example, the NCRP and the International Commission on Radiological Protection (ICRP) have both recommended an annual effective dose limit of 100 mrem from manmade sources, excluding medical exposures. NCRP states that "[f]or continuous (or frequent) exposure, it is recommended that the annual effective dose not exceed 1 mSv [100 mrem] . . . [and] a maximum annual effective dose limit of 5 mSv [500 mrem] is recommended to provide for infrequent annual exposures" (19). Background sources of radiation are excluded from these recommended limits. Id. Similarly, the ICRP recommends that "the limit for public exposure should be expressed as an effective dose of 1 mSv in a year" (20). Neither of these organizations has suggested that general public dose limits should be set below 100 mrem/yr; however both groups take the position that the full 100 mrem dose should not be allocated to a single source. On these basic points NRC generally appears to agree; the Commission has adopted 100 mrem/yr as a general exposure limit and it has taken the position that the full 100 mrem/yr should not be allocated to a single source.

Based on the foregoing principles, it is probably appropriate, in setting a dose-limit for a single source, to establish a limit at some fraction of the full 100 mrem/yr -- which is essentially what NRC has done in the proposed D&D rule. What NRC has failed to do is provide some reasoned basis for choosing 15 mrem/yr (i.e., 15% of the annual dose) as the appropriate limit or to provide any evidence that there is even a remote possibility that the 100 mrem/yr dose limit would potentially be exceeded by a single decommissioned source. By way of comparison, NCRP suggests the following approach in attempting to address the single source issue: where the potential exists for an individual to receive greater than 25% of the annual effective dose-limit (i.e., 25 mrem) from a single site or source, the site operator should ensure that the annual exposure (excluding medical exposure) to the maximally exposed individual does not exceed 100 mrem/yr on a continuing basis (21).* This approach is almost identical to one developed by the Health Physics Society (HPS), which recommends a 100 mrem/yr dose limit with a 25 mrem/yr "screening" level. Under HPS's approach, "[i]f the mean annual TEDE to the critical group is likely to exceed 25 mrem, an evaluation should be made to ensure that no individual is likely to receive an annual TEDE exceeding 100 mrem (1 mSv) from all site-specific, nonoccupational exposures, excluding indoor radon" (22).

The wide disparity that appears to exist between the dose limit proposed by NRC and similar limits developed by the scientific community provides a further indication that NRC's proposal was not developed in a manner consistent with sound scientific principles.* If this artificially low limit is in fact adopted by the Commission as a final rule, the implications will be extraordinary -- particularly with respect to the additional amounts of radioactive waste that will be generated, treated, and disposed of in order to comply with the limit, and the risks attendant with such a

clean-up effort. Moreover, the potential economic costs associated with such a massive clean-up effort could be staggering.

Harmonization With Other Pertinent Regulatory Programs

Another fundamental difficulty with NRC's proposal is that it fails to consider in any depth whether (and to what extent) the proposed D&D regulations are consistent with requirements imposed under other radiation control programs, which has the potential to result in conflicting and overlapping regulatory requirements. One example of this is the "public participation" provision of the proposed rule.

As already discussed, the proposed D&D rule provides that a site may be released for restricted use if certain criteria are satisfied. Under the proposed rule, a licensee that requests restricted release of a site would be required to convene a "Site Specific Advisory Board" (SSAB), which is supposed to "[r]eflect the full range of interests in the affected community and region" and is to be comprised of representatives from state and local government, citizens residing in the vicinity of the site, and representatives from environmental and environmental justice groups, among others (23). The purpose of the SSAB is to provide advice and recommendations to the licensee regarding planned D&D activities at the site. Specifically, the SSAB is to consider whether residual radioactivity on the site can be reduced to permit unrestricted release, whether the institutional controls proposed by the licensee will achieve the 15 mrem/yr dose limit without imposing undue burdens on the community, and whether the licensee has provided adequate financial assurance to permit a third party to carry out any required maintenance at the site (24).

Regardless of the merits of the public participation provision, it is evident that one practical effect of the SSAB requirement will be to delay substantially the implementation of decommissioning activities. The broad representation required on SSABs and the controversial nature of the issues the Boards are directed to consider make such delays almost unavoidable. At the same time, NRC's Timeliness in Decommissioning regulation (25) sets out an accelerated schedule for decommissioning, with ambitious deadlines established for both initiation and completion of D&D activities. It is apparent that in many if not most instances in which release of a site for restricted use is sought, the delays associated with the SSAB requirement will make it literally impossible to comply with the deadlines set out in NRC's Timeliness regulations. Consequently, licensees may be placed in the untenable position of being forced to violate the provisions of the Timeliness rule in order to comply with the SSAB requirement contained in the D&D rule. This is just one example of the types of conflicts that may arise between the proposed D&D rule and other pertinent radiation control requirements.

NRC's failure to consider rigorously the interplay between the D&D rule and other radiation control programs also has the effect of undermining even further the validity of the proposed 15 mrem/yr dose-limit. As discussed previously, sound science requires that, in setting a permissible dose for the release of a decommissioned site, NRC perform an ALARA or net-risk/net-benefit analysis to determine what dose-limit is appropriate. This type of analysis involves an evaluation of both radiological and non-radiological risks, and requires that NRC factor in the real-world risks associated with D&D activities (including risks associated with transportation, demolition and disposal of wastes), to ensure that the limit established in the D&D regulations minimizes the overall risk associated with D&D activities and not merely the risks posed by residual radioactivity at a site (26). NRC's failure to consider the impacts of other pertinent regulatory programs makes it impossible for the Commission to develop an accurate assessment of the net risk posed by its D&D standards. Because NRC is not in a position to evaluate the extent to which these other regulatory programs might be implicated by the D&D rule, the Commission cannot evaluate what additional increment of risk (if any) will be associated with compliance with those other regulatory requirements. For example, EPA is currently developing regulations to govern the management of radioactive wastes, including wastes generated during site decommissioning activities. NRC, however, has not attempted to assess the extent to which those waste management regulations will be implicated by NRC's D&D requirements and, more particularly, by the large volumes of waste that would be created as a consequence of the proposed 15 mrem/yr dose-limit. Without considering how these different regulatory regimes will overlap, NRC cannot begin to assess what risks might flow from compliance with any implicated EPA waste management regulations. In other

words, it is impossible for NRC to conduct an appropriate ALARA or net-risk/net-benefit analysis for developing a dose-limit unless the Commission evaluates how the proposed D&D rule will impact other regulatory programs and requirements.

Restricted Use Provisions

There are several troubling features of the proposed restricted use provisions; two are worth examining here. First, the regulations appear to underestimate the quantities of waste that will potentially be created as a result of D&D activities. In addition, the proposed restricted use provisions fail to provide for adequate controls needed to restrict access to and limit the use of restricted use sites.

In the preamble to the proposed D&D rule, NRC indicates that release of a decommissioned site for restricted use is intended to be the exception rather than the rule under the D&D regulations. According to NRC, the option of restricted use is available "[i]n those few cases" where it would be unreasonable to reduce residual radioactivity to the 15 mrem/yr limit for unrestricted use (27). For sites where large volumes of waste would be generated as a result of D&D activities it may not be feasible -- or reasonable, in terms of net risk -- to attempt to achieve the 15 mrem/yr dose-limit required for unrestricted release. For these sites, some form of on-site disposal and stabilization coupled with restrictions on site use and access may be most appropriate. Although NRC appears to recognize this possibility, the Commission also appears to underestimate the number of sites for which this alternative may be necessary.

Because NRC's proposed dose-limit is well within the range of normal background variation, it is almost inevitable that D&D activities directed at achieving this 15 mrem/yr limit will generate large volumes of radioactive waste. This is particularly true for sites at which substantial quantities of NORM are present. Yet, it is not at all clear that the Commission has developed a realistic assessment of the number of sites that may require on-site stabilization and disposal. For example, in the preamble to the proposed rule the Commission notes that "there may be several existing licensed sites (no more than a few tens) containing large quantities of materials contaminated with low level radioactivity where health and environment are best protected by onsite stabilization and disposal" (28). Although the proposed regulations would apparently permit licensees in this position to apply for an exemption from the D&D requirements, the Commission's assumption that only a few tens of sites may require on-site disposal seems ill-founded. Indeed, it is unlikely that NRC could have performed any sort of rigorous analysis of the need for on-site disposal without considering the impacts of other pertinent radiation control programs --such as EPA's pending radioactive waste management regulations.

In addition, although NRC's proposed regulations clearly contemplate the use of institutional controls (such as land use restrictions) to achieve the 15 mrem/yr dose limit for restricted use sites, it is not clear whether, and to what extent, the regulations allow the use of passive engineering controls. For example, as currently drafted, the proposed rule provides that a licensee seeking restricted use must provide for "institutional controls that provide reasonable assurance that the [15 mrem/yr limit will be achieved];" and the "institutional controls must be enforceable by a responsible government entity or in a court of law" (29). In addition, the licensee must demonstrate that if the institutional controls fail, there is a reasonable assurance that the 100 mrem/yr limit will be achieved. However, "calculations used to show compliance with this [100 mrem/yr] provision may not assume any benefits from earthen cover or other earthen barriers unless specifically authorized by the Commission" (30).

It is unrealistic for NRC to expect that most (if not all) restricted use sites will be able to achieve the 15 mrem/yr and 100 mrem/yr standards without the use of passive engineering controls, including earthen cover and earthen barriers. Indeed, because the proposed 15 mrem/yr limit would likely result in a significant increase in the amount of waste generated at sites undergoing D&D, the use of passive engineering controls may be required at a substantial number of sites to be released for restricted use. The restricted use provisions of NRC's proposed rule seem ill-designed to address this need for engineering controls.

One final concern regarding the restricted use provisions pertains to the types of institutional controls that NRC anticipates will be implemented at restricted use sites. Although the proposed rule does not purport to require or proscribe the use of any particular type of institutional control, NRC seems to take the position in

this rulemaking that the most desirable forms of institutional control (and therefore the controls most likely to be approved by the Commission) involve private ownership of decommissioned sites, with devices such as zoning laws, deed restrictions, covenants and easements used to control access to and use of the decommissioned site (31).

What is problematic about this approach is NRC's apparent reliance on private ownership of sites following decommissioning. Private ownership puts restricted use sites in the hands of entities that are motivated by economic concerns and pressures, which may at times conflict with concerns for long-term radiological safety. Moreover, in a system of private ownership, the same entities that own a decommissioned site will also be in a position to influence political processes at the state and local level to undermine the restrictions on site use imposed by institutional controls. Finally, there can be no assurance that the private owner of a restricted use site will remain in existence over the extended time frame required for control and maintenance of the site. Thus, private ownership of restricted use sites is inherently unstable, particularly when compared to the option of government ownership. Government ownership of a restricted use site effectively eliminates any concerns over the adequacy and durability of institutional controls implemented at the site. Yet, almost without explanation, NRC has rejected the option of government ownership (32). Clearly, this issue warrants additional consideration by the Commission.

EPA'S DRAFT D&D RULE

EPA has not progressed quite as far as NRC in the development of its D&D regulations. In October of 1993, the Agency issued an advance notice of proposed rulemaking (ANPRM) on the D&D issue (33). Subsequently, on May 11, 1994 EPA released a draft notice of proposed rulemaking on radiation site clean-up regulations (the EPA "Draft Rule"); however, as of this writing, EPA has not yet published the "final" version of the proposed rule in the Federal Register. Because EPA's site clean-up rule is only in draft form, many of its specific provisions have yet to be developed by the Agency and therefore cannot be evaluated in depth here. Accordingly, our examination of the EPA rule must be limited to an assessment of the central features outlined in the Draft Rule, focusing on details of the proposal where such details have been included in the draft released by the Agency.

As indicated previously, EPA's Draft Rule would apply to all federally owned or operated sites and all sites licensed by NRC, except high-level waste disposal sites, mill tailings facilities and Superfund sites with RODs. The Draft Rule is similar to NRC's proposed D&D regulations in a number of important respects -- which is not unexpected, since each agency was an active participant in the other's rulemaking proceedings. As with NRC's proposal, the Draft Rule establishes a dose limit of 15 mrem/yr from sources other than background, calculated over 1,000 years. This limit is expressed in terms of the dose received by the "reasonably maximally exposed" (RME) individual, which EPA defines to mean an individual whose exposure falls within the 95th percentile, or above, of the population near a site (34).

In addition, the Draft Rule provides that in circumstances where it would be inappropriate to release a site for unrestricted residential use, the site may be released for restricted use if the owner or operator can demonstrate that the site will achieve the 15 mrem/yr standard through some combination of remediation and active controls. (For purposes of the draft rule, "active" controls include institutional controls such as land use restrictions, as well as engineering controls that require active maintenance (35).) As with the NRC proposal, EPA's Draft Rule does not require or proscribe any particular types of active controls. However, the Draft Rule appears to address, at least partially, two of the deficiencies contained in the NRC proposal. First, EPA's proposal admits of the possibility that institutional controls might include government ownership of the decommissioned site (although many of the sites that would be subject to EPA's rule are already government owned and/or operated sites) (36). In addition, the Draft Rule acknowledges that engineering solutions can be included in the mix of controls used to bring a restricted use site into compliance with the applicable dose-limits. Under the EPA draft, a site owner/operator seeking approval for the release of a site for restricted use is not required to convene an SSAB to provide advice and/or recommendations on the D&D effort. Instead, EPA's proposal requires that the public be notified of the proposed clean-up and allowed a chance to comment. The draft does not contain sufficient details regarding this notice and comment procedure to

evaluate whether EPA's proposal is likely to result in fewer delays in implementing D&D activities than would be caused by the SSAB approach. As with the SSAB process, however, complicated notice and comment requirements may result in substantial delays in initiating D&D activities -- which is inconsistent with the goal of achieving prompt clean-up of radioactively contaminated sites and which may directly conflict with NRC's recent Timeliness in Decommissioning rule.

Although there are a number of areas of relative congruence between the EPA and NRC proposals, in many respects EPA's Draft Rule is more restrictive than the corresponding provisions of NRC's proposed rule. It is conceivable (if not likely) that in those areas where the EPA rule is more restrictive, the Agency will conclude that NRC's requirements are not sufficiently protective. As discussed previously, this would serve as the predicate for imposing the more restrictive EPA standards on sites that would otherwise be subject to NRC jurisdiction.

One example of conflicting EPA and NRC standards is the "default" dose-limit for unrestricted sites. Under EPA's Draft Rule the owner or operator of a site that is to be released for restricted use must demonstrate that the site has been sufficiently remediated such that, in the event that active controls at the site were to become ineffective, the reasonably maximally exposed individual would not receive a dose greater than 75 mrem/yr. The corresponding "default" dose-limit under the NRC proposal is 100 mrem/yr. In this instance, NRC's standard is clearly more easily achieved than EPA's dose-limit while it is probably equally protective of human health and the environment. Because of the tension between the two standards, it is not clear, at this point, which would actually be applied to NRC-licensed sites undergoing decommissioning.

EPA's proposal, as currently drafted, also contains a potentially crippling "reopener" provision, under which decommissioned sites would be reevaluated every certain number of years following decommissioning (the specific interval is to be set out in the regulations), to determine whether the applicable dose-limits are being met, whether protective measures implemented at the site (such as active engineering controls) are still effective, and whether additional remediation activities will be required (37). This provision is troubling because it robs D&D activities of finality and inserts another layer of uncertainty into the decision-making process facing site owners and operators who are contemplating decommissioning a site. In turn, this uncertainty may lead site owners and operators to delay the initiation of decommissioning activities for as long as possible -- which is contrary to the goal of achieving prompt decontamination and decommissioning of contaminated sites. This uncertainty is compounded by the fact that Draft Rule does not specify any standards for evaluating whether further remediation efforts will be required at a site.

By contrast, the NRC proposal provides that "[a]fter a site has been decommissioned and the license terminated in accordance with the criteria in this proposed rule, the Commission will require additional cleanup only if, based on new information, it determined that residual radioactivity remaining at the site could result in significant public risk" (38). Admittedly, the Commission has not defined what constitutes "significant public risk;" however this significant risk standard provides at least some objective framework against which NRC can evaluate whether further clean-up at a decommissioned site is needed. EPA's proposal as currently drafted does not offer a clue to site owners or operators regarding the circumstances under which additional clean-up could be required. Here again, conflict between the approaches advocated by NRC and EPA lead to uncertainty regarding which standard will be applied to an NRC-licensed site which is potentially subject to the both sets of rules.

As a final example, EPA's Draft Rule would establish a separate groundwater standard that would be applied in addition to the overall 15 mrem/yr dose-limit. Specifically, sites undergoing decommissioning would have to be remediated to a level that would assure that radioactivity in any groundwater that is a current or potential source of drinking water would fall within the applicable maximum contaminant levels (MCLs) established for radionuclides under the Safe Drinking Water Act. Currently this translates to a total effective dose of 4 mrem/yr from groundwater that is or may be a source of drinking water.

According to EPA, "[w]hen MCLs for radionuclides are changed or added in the future, the Agency intends for those new MCLs to be the groundwater protection requirements used for the purposes of setting remedial objectives" (39). In other words, under

the EPA proposal, as MCLs for radionuclides are either added or modified under the Safe Drinking Water Act, those new MCLs will be incorporated into the site clean-up regulations as groundwater standards. Again, this approach tends to undermine the finality of decommissioning activities, and it creates uncertainty for owners and operators of sites -- raising the specter of the agency "reopening" the decommissioning to require compliance with modified groundwater requirements. The NRC proposal would also establish a separate groundwater dose-limit as part of the overall 15 mrem/yr standard; however, the NRC rule would simply incorporate the MCLs for radionuclides as they exist on the effective date of the D&D rule. There is no indication that NRC intends to change the site clean-up standards for groundwater whenever an MCL is added or modified under the Safe Drinking Water Act, which means less uncertainty for site owners and operators and therefore less incentive to delay the initiation of site decommissioning activities (40).

As these examples demonstrate, NRC and EPA need to work together to develop consistent regulatory regimes that are legally defensible and rooted in sound science; unfortunately, the current rulemaking efforts of the two agencies appear to fall short of these goals. Inconsistent, overlapping and open-ended regulatory programs that add complexity and uncertainty to the decommissioning process will almost certainly cause site owners and operators to delay the initiation of site decommissioning activities, which is inconsistent with the objective of achieving prompt decontamination and decommissioning of inactive sites. Indeed, to the extent that the regulatory regimes developed by the two agencies are either incompatible with one another or not supported by sound science, direct challenges to those standards can be expected, resulting in further delays in the D&D process.

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U.S. DEPARTMENT OF ENERGY PROGRESS AND ACCOMPLISHMENTS IN SURPLUS FACILITY DECOMMISSIONING UNDER ENVIRONMENTAL RESTORATION

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ABSTRACT

The mission of the U.S. Department of Energy (DOE) environmental restoration program is to ensure that risks to human health and safety and to the environment, posed by contaminated inactive waste sites and surplus facilities, are either eliminated or reduced to prescribed, safe levels. The program is concerned with all aspects of assessment and remediation activities where sites and facilities are no longer associated with active nuclear-related production or research operations. Remedial action comprises assessment and remediation of inactive waste sites and correction of a release or spill problems. Decommissioning includes assessment, decontamination, reuse, or dismantlement of surplus contaminated facilities that are no longer active.

DOE is implementing a national program, directed by the Office of Environmental Restoration (ER) within the Office of Environmental Management (EM), to ensure that all decommissioning objectives are accomplished. The national program will promote the safe and efficient decommissioning of DOE facilities and strive to achieve timely and cost-effective results, based on a reasonable set of priorities, focusing on protection of the environment, improved worker and public safety, and conservation of valued resources.

This paper describes the six components that comprise the framework for the national decommissioning program under environmental restoration. The paper also describes the scope of the program and provides descriptions of decommissioning projects that are included as decommissioning program performance measures for fiscal year (FY) 94 and FY95.

NATIONAL PROGRAM FRAMEWORK

A national decommissioning program is being established within the environmental restoration (ER) program to ensure that decommissioning projects are conducted in a safe and efficient manner, while enabling Headquarters and field personnel to share new technology applications, lessons learned and the latest regulatory and policy directions of the program. The framework for the national decommissioning program comprises: Policy framework, interface, ER decommissioning inventory, program management, technology, and communications.

Policy Framework

A policy framework is required for prioritizing, funding, and conducting decommissioning activities so that they are consistent with remediation activities. A key issue is the relationship of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) to decommissioning activities. The U.S.

Department of Energy (DOE) and the U.S. Environmental Protection Agency (EPA) established a working group to analyze and resolve issues related to CERCLA and decommissioning and to develop a national policy for applying CERCLA to decommissioning. ER chairs the DOE/EPA working group. The working group expects to have a draft of the policy ready for EPA and DOE management review in February 1995. After that, DOE will develop an implementation framework for applying CERCLA to decommissioning projects. In addition, ER will issue and maintain the Decommissioning Resource Manual to assist field personnel in meeting requirements.

Interface

National decommissioning program staff interface with other organizations to ensure that surplus facilities are decommissioned safely and efficiently and that decommissioning projects are consistent with Departmental and external requirements. For example, ER is working jointly with the Office of Environmental Management (EM) Office of Facility Transition and Management in the development of policies and procedures for surplus facility transition. The ER program also interacts with external organizations, such as EPA, the Nuclear Regulatory Commission, and the Defense Nuclear Facilities Safety Board, to gain insight into activities of interest to, and to work towards a consistent national direction for, decommissioning. DOE actively participates in the exchange of decommissioning technology, project experience, and lessons learned at national and international forums.

ER Decommissioning Inventory

A means of identifying, documenting, and tracking the inventory of surplus facilities and decommissioning projects is needed to define the scope of effort, track progress, and ensure future accountability for their cleanup. ER is refining an inventory of decommissioning facilities and projects in the environmental restoration program. Although still in the final stages of data verification, the inventory currently contains approximately 760 facilities that are either undergoing, or require decommissioning. The inventory will be updated periodically to reflect newly identified surplus facilities and completed decommissioning projects at facilities on the inventory.

Program Management

To ensure that decommissioning activities occurring across the DOE complex are being planned and performed in an efficient and consistent manner, the national program needs an effective program management component. This component will ensure that policies for prioritizing, funding, and conducting decommissioning operations are implemented and that project managers have the tools they need. One priority for this year is a decommissioning cost benchmarking study so that managers can better project costs and review estimates. The benchmarking study is about to get underway.

Technology

The national decommissioning program is establishing a technology development and transfer initiative to meet the needs of current and future decommissioning projects. Mechanisms are also being developed to exchange beneficial decommissioning technologies and relevant project experience internally and with other government cleanup programs, the nuclear industry, and internationally. ER is participating in decommissioning technology focus groups, to identify and assess technologies in use in the private and government sectors and their potential application in the DOE complex, to implement the interface with the EM Office of Technology Development on the identification and development of needed decommissioning technologies, and to evaluate application of new technologies and their effectiveness in the private and government sectors. In addition, ER will maintain the Decommissioning Handbook to ensure that the most up-to-date information is available to the field.

Communications

Productive and continuous communications and information exchange mechanisms are being employed for transmitting lessons learned within the decommissioning community and informing interested parties of decommissioning activities and status. To help Headquarters and field personnel get to and stay at the cutting edge of decommissioning processes and techniques, for example, DOE is presenting a televideo course on decommissioning for training and education of personnel involved with decommissioning activities. This course will air six programs, beginning February 8, 1995, featuring these topics: Decommissioning overview, surveillance and maintenance, project planning, decommissioning methods and alternatives, decommissioning operations, and policies and issues.

DECOMMISSIONING PROGRAM SCOPE AND PROGRESS

DOE has identified hundreds of surplus contaminated facilities requiring decommissioning as part of environmental restoration. Approximately 760 facilities already are slated for eventual decommissioning and another 1,200 may be in the pipeline.

Of the 17 major environmental restoration projects, 14 currently are conducting or plan to conduct decommissioning activities. There are hundreds of facilities at 26 DOE installations in 13 states that are associated with these 14 major projects. To measure significant accomplishments, DOE has been developing performance measures for each fiscal year. Within the decommissioning program, ER targeted 5 project completions for fiscal year (FY) 94 and 12 project completions for FY95. The program actually completed 9 projects in FY94 and will strive to surpass its target set for FY95. The following section presents a brief background on the decommissioning projects that were completed in FY94 or are targeted for completion in FY95, grouped by DOE installation.

Santa Susana Field Laboratory

The Santa Susana Field Laboratory (SSFL) is located 29 miles northwest of downtown Los Angeles, California. Within SSFL is the Energy Technology Engineering Center (ETEC), which is operated by Rockwell International under contract to DOE. Specific programs conducted at ETEC for DOE and its predecessor agencies involved the engineering, development, testing and manufacturing operations of nuclear reactor systems and components. Decommissioning of facilities began in the late 1960s and continues as DOE phases out individual DOE-sponsored projects.

Decommissioning projects were completed at three ETEC facilities in FY94: Building 5, Building 23, and Building 64. Building 5 was used to fabricate uranium carbide fuel in the late 1960s. At the conclusion of the fuel fabrication project, uranium contaminated equipment and surfaces were either removed or decontaminated. Additional decontamination activities were initiated in 1978 and completed in phases, ending in 1992. Activities included cleaning and/or removal of contaminated floors, equipment, duct work, drain pipes and storage tanks.

Building 23 was used from 1976 to 1982 to test the transport of radiological contamination in sodium loops. Major contaminants included cesium-137 and cobalt-60. Most of the contamination identified in the building involved the radioactive liquid holdup tank and the associated drain lines and sink. Facility decontamination included the removal of the sodium loop, holdup tank, drain lines, sink, a fume hood and the ventilation exhaust system and remediation of an area of the floor where the sodium loop had been located.

Building 64 was used primarily for storage of packaged items of source material and special nuclear material of various forms and configurations. The material stored in this facility included natural uranium, depleted uranium, enriched uranium, uranium-233, thorium and plutonium. Packaged soil contaminated with cesium-137 was also stored in the facility. Interior surfaces were determined to be contaminated from the uranium repackaging process. As part of decommissioning activities at Building 64, equipment and fixtures in the building were decontaminated and/or dismantled and removed and the interior decontaminated to allow for future release to the owner after the completion of decommissioning of other Rockwell-owned buildings in the area.

Idaho National Engineering Laboratory

The Idaho National Engineering Laboratory (INEL) is located 42 miles northwest of Idaho Falls, Idaho. INEL is a multipurpose laboratory supporting the engineering and operations efforts of DOE and other federal agencies in areas of nuclear safety, reactor development, reactor operations and training, waste management and technology development, and energy technology/conversion programs. Three decommissioning projects at INEL are described below.

Test Area North (TAN) is located in the northern portion of INEL and was established in the 1950s to support the U.S. Air Force Aircraft Nuclear Propulsion Program. The TAN-607 Decontamination Shop was used for radiological decontamination of tools and small equipment from INEL and non-INEL facilities. The area began operations in 1957 and was shut down in 1987. The shop consisted of a main decontamination room, a change room, and a high energy particulate air (HEPA) filter room. Contamination consisted of low-level radioactive equipment and debris and asbestos. Decommissioning tasks included removal of all contaminated equipment, including tanks and piping, and decontamination of the space within the larger TAN-607 facility to allow for reuse of the area. Decommissioning of the TAN-607

Decontamination Shop was completed in FY94.
The Auxiliary Reactor Area III (ARA-III) facility at INEL was used by the U.S. Army from 1960 to 1965 to test gas-cooled reactors and from 1966 to 1987 as laboratory facilities for component and instrument testing. Contamination at ARA-III consisted of low-level radioactive soil, equipment, and debris (concrete and metals). Decommissioning tasks include excavation, removal, and sectioning of three wastewater tanks and associated piping; dismantling, sectioning, and disposition of ventilation/exhaust stacks and associated ducts; and reactor pit decontamination and pipe removal. Once these activities are complete, the facility will be released for unrestricted use within the INEL complex. All the buildings except one will be made available for reuse.
The Central Facilities Area-669 (CFA-669) Hot Laundry is a cinderblock building that was used as a laundry facility for both radioactive and non-radioactive contaminated clothing and contained low-level radioactive soil, equipment and debris. Decommissioning tasks include decontaminating the interior of the building, removing and disposing of equipment, excavation of underground piping, and complete demolition of the building. The ARA-III and CFA-669 decommissioning projects are scheduled to be completed in FY95.
Including the three projects described above, INEL has excessed and internally recycled or externally sold significant quantities of materials from the decommissioning activities. These materials include 363,000 pounds of scrap metal that were sold to a metals recycle vendor, 2,360 pounds of lead that were internally recycled, and 48,000 pounds of concrete debris that will be recycled on site with INEL's new concrete recycling machine.
Oak Ridge Y-12 Plant
The Alpha-4 Production Facility, designated Building 9201-4, is a 4.5 acre structure in the Oak Ridge Y-12 Plant, two miles from downtown Oak Ridge, Tennessee. It was constructed in the 1940s to house process equipment for the separation of uranium isotopes. During the 1950s and 1960s, the building was used for the separation of lithium isotopes through a mercury-solvent extraction process known as "colex." Two Building 9201-4 decommissioning projects completed in FY94 are the 9201-4 Mercury Draining project and the 9201-4 Electrochemical Machining (ECM) Area project. As a result of the mercury-solvent extraction process, piping, auxiliaries, and structural brick and mortar were contaminated with mercury. The Mercury Draining project involved the draining and storage of residual mercury from process equipment and piping for possible recycle or reuse. Decontamination of the ECM area involved the cleanup of all known sources of non-fixed contamination in the ECM Area. The project involved decontaminating more than 24,000 square feet of surface area. Primary contaminants included depleted uranium oxides, mercury, beryllium, and polychlorinated biphenyls (PCBs), as well as elevated radiation contamination levels. Nearly 350,000 pounds of material were decontaminated and the project was completed safely, ahead of schedule, and under budget.
Both projects were accomplished under Y-12's enhanced surveillance and maintenance (S&M) program. The Y-12 S&M program is characterized by aggressive removal and decontamination projects which ultimately reduce the scope and cost of long term S&M and eventual decommissioning. DOE is encouraging use of the Y-12 approach to S&M at other Oak Ridge decommissioning projects.
Decommissioning activities at Building 9201-4 planned for FY95 include steamline asbestos removal, decontamination of the ECM support systems, and comprehensive facility characterization to support final facility decommissioning activities. Complete decommissioning of 9201-4 will involve advanced mercury decontamination technologies, development and demonstration of a recycle process for aqueous solutions, and construction of a landfill for disposal of decommissioning wastes.
Oak Ridge K-25 Site
The K-25 Site, formerly known as the Oak Ridge Gaseous Diffusion Plant (ORGDP), is located 13 miles west of the city of Oak Ridge, Tennessee. The main mission of the ORGDP had been the enrichment of uranium by gaseous diffusion. In 1989, the entire site was placed under DOE's EM program. K-25 Site facilities are primarily contaminated with uranium oxides, heavy metals, organics, PCBs, and asbestos. Decommissioning of several process buildings and related facilities has been completed or is underway.
Activities completed in FY94 include the removal and storage of roughly 10,000 capacitors and transformers. All of these electrical components contain PCBs. The

storage areas, built in accordance with the Toxic Substances Control Act (TSCA), will capture and prevent the spread of any leaking fluids. K-25 PCB Electrical Compliance projects and other Phase I decommissioning activities will reduce the scope and cost of the K-25 S&M program and will provide for a safer working environment. Activities to be completed in FY95 include the packaging and shipment of non-radiologically contaminated PCB-containing capacitors for off-site disposal, the demolition of five K-25 Powerhouse auxiliary facilities, and the removal of critical uranium deposits from enrichment process piping.

The Powerhouse demolition project is being conducted as one of two EPIC (Environmental Project Integrated Contract) projects at the K-25 Site. This innovative contracting arrangement is based on a fixed price contract and provides incentives for the contractors to perform the work safely, on schedule, and under budget. The contractors will be required to absorb any cost overruns. DOE intends to accomplish more work under the EPIC model and other innovative contracting concepts. DOE is also considering several studies and pilot projects aimed at reducing the total cost of decommissioning the K-25 process buildings.

Battelle Columbus Laboratories

Various projects are undergoing decommissioning at the Battelle Columbus Laboratories King Avenue site, which is located in downtown Columbus, Ohio. Since 1943, the Battelle Memorial Institute has continuously performed energy research and development work at its Columbus Laboratories for DOE and DOE predecessor agencies. King Avenue Building 3 (KA-3) was decommissioned in FY94 while King Avenue Building 4 (KA-4) is scheduled for FY95 completion.

KA-3 was the largest and most contaminated of the nine buildings being decontaminated at the King Avenue site. KA-3 was used for various research activities involving uranium and beryllium and housed a powder metallurgy facility, a melt facility, a metallography facility, a ceramics research facility, and a uranium-235 processing facility. Work on the first nuclear submarine, the Nautilus, was conducted in this building. Decommissioning efforts involved removing contaminants from 191 rooms, involving 174,000 square feet of surfaces; 3,200 linear feet of drain lines; and 8,300 cubic feet of subfloor soil. The work entailed removing ducts, electrical conduit, water lines, and contaminated laboratory equipment. Chemicals, equipment, and other materials were removed, separated, and packaged for reuse, recycle, or off-site disposal.

KA-4 housed a radiochemistry laboratory, a metallography laboratory and an encapsulation facility for highly enriched uranium. Decontamination of only the radiochemistry laboratory was completed in 1980. Decommissioning activities include relocating staff from the building, removal of uncontaminated furniture and laboratory equipment, decontamination, disposal of radioactive waste, and final restoration of the building for release to Battelle.

Fernald Site

The Fernald Site, formerly known as the Feed Materials Production Center, is a 1,050-acre site located in a rural agricultural area about 17 miles northwest of downtown Cincinnati, Ohio. High purity uranium metal products were produced for DOE and its predecessor agencies from 1952 to 1989. All production activities at Fernald ceased in July 1989 to allow the site to concentrate on environmental cleanup and restoration. Various decommissioning projects are being completed at the Fernald Site, including Plant 7, Building 4C, and the Fire Training Facility.

Fernald Plant 7 production activities occurred from 1954 to 1956 and consisted of a uranium conversion process. The plant evolved into a storage facility for low-level radioactive materials and other debris. The majority of the process equipment was removed over time. Plant 7 was a 112 feet high, 7 floor building, consisting of a structural steel frame on a reinforced concrete base, with steel decking floors and transite siding.

A multiple stage decontamination was performed on Plant 7, consisting of a washdown for gross decontamination, followed by the application of an acrylic latex coating to "lockdown" any remaining loose surface contamination. This process significantly reduced the radioactive contamination of the structure. It was proposed that Plant 7 structural support columns be cut using controlled detonation. By using a specialized steel-cutting method of linear-shaped charges with sequential charge detonation, the weight and shape of the building was anticipated to fold within seven seconds. The steel would then be size-reduced, monitored for radiological contamination, and packaged once on the ground. Prior to Plant 7 demolition, the

walls, asbestos-contaminated materials, equipment, and elevator were removed and packaged. Plant 7 was reduced to a simple structural steel skeleton and floor decking.
In September 1994, the demolition contractor detonated 156 linear shaped, explosive charges placed in 50 locations intended to take the building down. The first two floors of the building collapsed as planned. However, splice plates that had been precut on the third and fifth floors did not separate as anticipated. The building dropped approximately 25 feet instead of the planned 78 feet. Following the partial takedown by controlled detonation, the area was secured. Following an extensive examination of the partially fallen Plant 7 structure, a decision was made to use explosive shaped charges to complete the takedown. The successful takedown occurred one week later and utilized 260 shaped charges placed in 120 locations. Even with the second explosive charge, the project will save nearly $5 million from the original decommissioning plan and will finish seven months ahead of schedule. The next objective of the project is to recycle a significant portion of the demolished material, including 700 tons of structural steel and 1.5 tons of lead flashing.
To provide sufficient open area for Plant 7 to fall, the maintenance building of adjacent Plant 4 (designated Building 4C) was removed at no additional cost to DOE. Demolition activities were conducted on this one story building in accordance with procedures approved in the Plant 7 CERCLA Work Plan. Building 4C, measuring 60 feet by 60 feet and 14 feet high, was constructed of a structural steel frame supported by a reinforced poured concrete base, with transite siding and roof panels. Building 4C housed the maintenance activities for the Green Salt Plant and other chemical process areas and was the base for asbestos removal operations at the Fernald site.
The Fire Training Facility (FTF) was constructed and operated as a training facility for the Fernald Site fire department and surrounding community fire departments from 1966 to 1990. The FTF was estimated to have been used an average of 60 days per year. During its use, various types of combustible substances were burned to practice fire fighting techniques; some of the waste solvents burned at the FTF may have been contaminated by radionuclides.
The FTF was composed of four structures. The main two-story block building consisted of first and second story floors constructed of reinforced concrete and a roof constructed of wooden joist covered with plywood and an asphalt roofing. The building was sealed after it was determined that is was no longer safe for use. A skid tank and an open top tank constructed of carbon steel were filled with waste solvents of fuel oil and set on fire to practice fire fighting techniques. The fourth structure was a cylindrical tank (former pressure vessel), constructed of 1.5-inch plate steel that was filled with smoke so that personnel could practice confined space ingress and egress under smoke-filled conditions.
The FTF was decommissioned as part of a CERCLA removal action. The project consisted of four phases: 1) demolition of the block building, 2) removal of water/liquid from associated ponds/pits, 3) removal and size reduction of tanks, and 4) excavation of contaminated soils. The building was demolished using a front end loader and a dump truck to haul the debris for packaging. Liquids were removed and containerized using a suction pump. The liquids will be sampled and processed with on-site facilities. Tank size reduction was performed with cutting torches and shears. Approximately 360 cubic yards of soil was excavated. The combined CERCLA Removal Action Final Report and the Resource Conservation and Recovery Act (RCRA) Closure Action Final Report are scheduled for submittal in July 1995.
Argonne National Laboratory-East
Decommissioning of the Experimental Boiling Water Reactor (EBWR) at the Argonne National Laboratory-East facility, 22 miles southwest of downtown Chicago, Illinois, is scheduled for completion in FY95. The EBWR was operated from 1952 to 1967 to demonstrate the use of a direct cycle boiling water reactor as a heat source for power plant steam production. After all reactor systems were flushed and drained and the water treated, the radioactive contaminants that remained at the EBWR facility were present in the reactor and the bio-shield. The primary radionuclide of concern was the activation product cobalt-60. Other contaminants included small amounts of iron-55, nickel-63 and trace quantities of miscellaneous radionuclides.
In the 1980s, asbestos insulation was removed throughout the facility, as well as all reactor system piping, components, and associated contaminated equipment. Throughout FY94, the reactor vessel internals, the reactor vessel itself, and the biological shield surrounding the reactor vessel were removed and size reduced using

a specially manufactured reactor vessel cutting machine, an underwater plasma arc torch, an electro-hydraulic remote controlled impact machine, and other specialized equipment. The components were then packaged for disposal.

Rocky Flats Plant

Several decommissioning pilot projects are being conducted at the Rocky Flats Plant, which is located 16 miles northwest of Denver, Colorado. The Plant is situated on 400 acres and, together with its buffer zone, encompass approximately 11 square miles. The Plant's primary mission prior to January 1992 was the production of nuclear weapons components fabricated from plutonium and other materials. The mission has now changed to environmental cleanup.

The first of several decommissioning pilot projects being conducted entails the removal of three 200-gallon polypropylene sodium hydroxide tanks and associated piping. Two of the three tanks are insulated with asbestos; therefore, the removal was conducted through the Rocky Flats asbestos abatement program. The tanks were connected to acid scrubbers located on the south side of Building 123. Building 123, located on the west side of the Plant within the main complex of buildings, was used for 30 years in the conduct of bioassays and environmental monitoring sample analysis. The sodium hydroxide used to scrub the acid that collected in the fume hoods was stored in the three tanks. The removal action consisted of removing the transfer lines from the tanks to the scrubbers, neutralizing the sodium hydroxide in the tanks, removing the tanks and packaging them in a waste crate, and transporting them to an approved storage facility while awaiting shipment to an approved disposal facility.

CONCLUSION

DOE is committed to the safe and cost-effective decommissioning of contaminated inactive facilities across the DOE complex. The examples described above illustrate the progress being made in this area. DOE's efforts will continue to become more efficient with the establishment, implementation and coordination of the components of the national decommissioning program.

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DECOMMISSIONING OF SURPLUS FACILITIES AT LOS ALAMOS NATIONAL LABORATORY

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ABSTRACT

Decommissioning Buildings 3 and 4 South at Technical Area 21, Los Alamos National Laboratory, involves the decontamination, dismantlement, and demolition of two enriched-uranium processing buildings containing process equipment and ductwork holdup. The Laboratory has adopted two successful management strategies to implement this project:

Rather than characterize an entire site, upfront, investigators use the "observational approach," in which they collect only enough data to begin decommissioning activities and then determine appropriate procedures for further characterization as the work progresses.

Project leaders augment work packages with task hazard analyses to fully define specific tasks and inform workers of hazards; all daily work activities are governed by specific work procedures and hazard analyses.

State-of-the-art technical strategies are also contributing to the success of the project:

Nondestructive assay methods, including the use of a long-range alpha detector, support characterization work.

A demonstration of an innovative ductwork lining technique will be undertaken.

Sampling and dismantlement techniques recently developed at Oak Ridge National Laboratory were used to treat perchlorate-contaminated items without delaying the schedule.

Together, these strategies have allowed the Laboratory to accelerate the work schedule by one year and reduce fixed costs by $1.4 million.

INTRODUCTION

The purpose of this paper is to discuss procedures for facility decommissioning at Los Alamos National Laboratory and identify lessons learned. Specifically, this paper examines the progress of several decommissioning projects currently being

carried out at Technical Area (TA) 21, where the Laboratory's plutonium processing operations were based until the late 1970s.
Many of the facilities at TA-21 were deactivated at that time and subsequently used for radiochemistry studies. Current decommissioning projects at the site involve 1) demolition of the enriched-uranium processing facility designated as Buildings 3 and 4 South and 2) characterization of plutonium-contaminated process exhaust filter buildings. The Buildings 3 and 4 South decommissioning project began remediation work on September 20, 1993. The project is one of the largest Department of Energy-sponsored decommissioning projects currently under way. The filter building characterization project began in December 1994 and will finish in May 1995. Decommissioning is scheduled to begin in July 1995.

SCOPE

Buildings 3 and 4 South Decommissioning

The first project to be discussed involves the decontamination, dismantlement, and demolition of two enriched-uranium processing buildings covering a total area of approximately 10,000 ft2. The buildings were constructed in 1945 as part of the original plutonium processing site at the Laboratory. The facility was converted from plutonium processing to enriched-uranium recovery in the late 1940s. It operated from that time until 1984 and specialized in the recovery of weapons-grade enriched uranium from scrap, unused fuel elements, and hard-to-recover items. The facility used chemical processes to convert the scrap to purified oxide or uranium metal which could be used in nuclear research programs, reactors, and nuclear weapons.

Building 3 served as the chemical concentration plant, where the scrap uranium was converted to high-quality uranyl nitrate. This building had been partially deactivated in 1987, at which time all solution tanks and associated piping were removed. Left behind were chemical dissolution hoods, a rotary calciner, a hydrofluoric acid cubicle, grinders and mills inside of gloveboxes, and a rag incinerator. An area on the north side of Building 3, Room 308, was not used for uranium chemistry; instead, it operated as a plutonium research laboratory until it was decommissioned in the late 1970s. Residual plutonium contamination was expected to range from 200,000 to 1 million disintegrations per minute per 100 cm2 in several areas.

Building 4 served as the final recovery and purification plant, where pure oxides and metal were produced. The building contains numerous furnaces, uranyl nitrate piping, collection and precipitation systems, and hydrofluorination systems (see Fig. 1).

The superstructure of each building is approximately 98 ft x 39 ft, with a height of 18 ft to the eaves and 26 ft to the center peak of the roof. The buildings are prefabricated, truss-and-purlin steel structures supported by concrete columns. Additional structures attached to Building 3 housed sitewide utilities and a nuclear material storage vault.

At the start of the project, the buildings contained surplus process equipment, approximately 300 linear feet of hoods and gloveboxes, and extensive utility systems. Ninety-three percent enriched uranium was distributed throughout major systems, as shown in Table 1 below.

TABLE I

Additional potential hazards included residual acids and bases, asbestos, and inaccessible contaminated areas such as perimeter utility tunnels.

Filter Buildings

The second project involves the filter buildings at TA-21 and associated process exhaust. They have been identified as near-term candidates for decommissioning on the basis of current facility conditions and potential risk. Prior to actual remediation, detailed information on facility condition, utilities, radioactive contamination levels, and potential chemical contaminants is required. Such information is necessary to comply with hazardous waste requirements, properly characterize waste for transportation and disposal, protect workers and the environment, and support the selection of decommissioning methods.

The filter buildings provided process exhaust to Buildings 2, 3, 4, 5, and 21 at TA-21. Exhaust is still active in Buildings 3 and 4 North; however, all but one hood has been disconnected from the system. The process exhaust filter system consists of three buildings: 21-329, the firescreen; 21-146, the filter building; and 21-324, the filter house. Figure 2 depicts the filter buildings and associated structures.

Fig. 2.

Ductwork exits Buildings 3 and 4 North and runs along elevated stanchions until it reaches the firescreen. The exhaust stream enters this structure, which is an elevated, sheet-metal-enclosed building containing screen filters and washdown equipment. A transparent glass line exits the sheet metal enclosure and discharges into a liquid waste transfer line, which runs to the onsite liquid waste treatment plant. The exhaust then enters Building 146, a concrete block building which houses a large, circular high-efficiency particulate air (HEPA) filter array and a glovebox assembly for changing out the filters. The HEPA filter array consists of a hexagonal filter bank containing six sets of three filters housed in a drum. The drum assembly rotates so that new filter faces can be presented to the airstream, thus reducing by a factor of six the downtime needed for changeout. The exhaust stream then enters Building 324, the filter house, which was added to the flow path in 1973. It contains twenty HEPA filters in parallel. Exhaust then releases through the stack at the north end of the building.

Decommissioning of the filter buildings will involve removal of the process exhaust ductwork from Buildings 3 and 4 North; the elevated ductwork which runs into Building 146; the HEPA filters and glovebox and drum assemblies in 146; the firescreen and all ductwork and stacks in 146; the HEPA filters in Building 324; and all ductwork and stacks in 324. Both buildings will then be demolished.

PROJECT ORGANIZATION AND WORK PLANNING

The Laboratory is responsible for overall project management, health physics, environmental compliance, criticality engineering, and waste management. Subcontractor oversight in the areas of engineering and health and safety also are performed by the Laboratory. Dismantlement is performed by Johnson Controls World Services, Inc., the onsite maintenance subcontractor. Johnson Controls also provides industrial hygiene services and develops work packages.

For Buildings 3 and 4 South, site characterization began in 1992 and was completed in September 1993. This effort consisted of environmental review and approvals required by the National Environmental Policy Act and radioactive air emissions standards; development of project plans, including those for project management, health and safety compliance, waste management, and quality assurance; and historical and radiological data collection.

Characterization of the entire facility was not conducted. Instead, the Laboratory follows the "observational approach" for decommissioning projects. Rather than extensively characterizing the entire project, enough data is collected to begin activities. Detailed procedures are developed as the work progresses, and additional information is collected as necessary. This process avoids efforts which can be rendered useless due to newly discovered problems, but it requires flexibility in scheduling and completing activities. The following section discusses the application of this approach to the filter buildings.

Because the emphasis is on minimizing characterization activities, the bulk of work planning is done during remediation. The removal of utility tunnel piping is a good example of this work planning process. Buildings 3 and 4 have interior, 4- x 4-ft utility tunnels which traverse the building perimeter and are located below the floor slab. The tunnels contain steam, condensate, industrial waste, and acid waste lines and are contaminated due to system leaks over the years. Access to the tunnels is through hatches outdoors; however, current safety requirements do not permit entry. Tunnel decommissioning followed a methodical process to initially identify radiological conditions and general piping configurations through visual inspections and coring samples. Work then proceeded to remove the top of the trenches using a large floor saw. Finally, the tunnels were thoroughly surveyed and dismantled.

FILTER BUILDING CHARACTERIZATION

The characterization activities necessary to support decommissioning of the filter buildings include historical records reviews and interviews with former users, collection and revision of applicable drawings, utility identification and review, and radiological and chemical sampling. In some cases, sampling will be postponed until actual decommissioning, for accessibility or safety reasons. The data acquired from the characterization will principally support demolition activities and waste characterization requirements.

Engineering data requirements consist of utility and structural information. Specifically, the locations of all utilities and any necessary reroutes must be identified. Structurally, the characterization effort must ascertain whether the

Building 146 drum assembly will rotate. The drum has not been turned since the 70's, and seal integrity and the opposite filter banks are items of concern. Existing drawings will be collected for reference and will be annotated to identify the as-left facility condition. Historical records will be reviewed to identify any abandoned utilities and any facility modifications which could affect decommissioning.

Knowledge regarding types and quantities of contaminants is essential for decommissioning operations and waste handling. Radioactive waste may be either low-level or transuranic, while chemical contamination may result in hazardous or mixed waste. Potential contaminants have been identified from the remedial action work plan, operating summaries, decommissioning summaries, and historical interviews. Radionuclides of concern are U-235, U-238, Pu-238, Pu-239, Tc-99, Am-241, Am-243, Np-237, Th-232, and Pa-231. Chemical concerns include asbestos (146 HEPA filters), metals (146 HEPA filters), PCBs (lighting ballasts), perchlorates (ductwork, 146 filters), and picric acid.

Because the data address waste management and safety concerns, exact readings are not as important as bounding readings. The data should identify thresholds for waste categories or personal protective equipment requirements.

A significant amount of data currently exist for this project (Marshall 1994). This information was obtained during a Laboratory-wide project to quantify special nuclear material holdup in ventilation systems. These data indicate that sizable portions of the process exhaust would be classified as transuranic waste. The project will decontaminate the ductwork during decommissioning to minimize the volume of transuranic waste. Accordingly, during decommissioning the removed ductwork and decontamination waste will be characterized for waste disposal purposes. This approach also recognizes the difficulty and expense of sampling exhaust systems prior to removal. Likewise, HEPA filter sampling is best left until actual removal, at which time the filters may be sawed into pieces and samples obtained more easily.

Additional data will be collected to measure radioactivity in systems not addressed during previous holdup measurement campaigns. Measurements will be made using nondestructive assay methods with Sodium Iodide and Germanium detectors. Items likely to be free of contamination will nevertheless be surveyed to verify that no unexpected radioactivity is present. Appropriate engineering controls will be used during decommissioning to protect uncontaminated materials.

Except for one small spot of contamination on the floor of Building 146, there have been no historical releases within either Building 146 or 324. The walls and floors should be free of contamination. The long-range alpha detector (LRAD), an experimental system developed at Los Alamos (Rawool-Sullivan et al. 1994), and conventional gas-proportional instruments will be used to systematically survey the structures to verify that the material is uncontaminated.

Facility processes did not involve hazardous wastes listed under the Resource Conservation and Recovery Act (RCRA). The RCRA facility investigation work plan does identify metals as a potential contaminant of concern, so the Building 146 filters will be sampled for metals. Sampling for metals, like the surveys for radioactive constituents mentioned above, will be performed at the time of filter removal. Building 146 has been sampled for perchlorates. This sampling will be repeated after the drum has been turned. Historical records suggest that picric acid was used for some experiments. Building 146 will be tested for picric acid before and after turning the drum. During disassembly, duct systems will routinely be tested for perchlorates.

The HEPA filters contain asbestos, and the roofs of both buildings may contain nonfriable asbestos-contaminated material. The roofing material will be tested for asbestos. Lighting systems will be inspected for PCBs during disassembly, and fluorescent bulbs will be handled as hazardous waste.

TABLE II

HEALTH AND SAFETY

The Laboratory and project adhere to rigorous health and safety practices. A system is followed in which we develop work packages consisting of technical work procedures and special work permits, i.e., radiation work permits (RWPs), augmented with task hazard analyses (THAs). The THA fully defines specific tasks, identifies protective measures, and informs workers of hazards. These work packages supplement the project plan and site-specific health and safety plan. The process also employs

a graded approach in procedure development.
For example, the work package for routine decontamination activities consists of an RWP and a THA. More difficult operations, such as hood and glovebox removal, also have a work procedure in addition to the aforementioned permit and THA. Because the exact hazards and sequence of activities can vary among hoods and gloveboxes, the THA is modified when warranted.
The last type of work package addresses unique, higher-hazard systems, for which a detailed work procedure is prepared. Again, the procedure is augmented with RWPs and a THA. Because the Laboratory reviews and approves all work procedures, but not all THA revisions, this last category invariably receives additional scrutiny from project managers. Regardless of type, all daily work activities are tied to the specific governing work procedure-THA package, and this information is recorded by the site safety officer. Thus far, the project has recorded one lost workday case due to infection of a small cut.
The Laboratory provides an extensive radiological protection program for the project. The program consists of radiological engineering, dosimetry, bioassay, and health physics coverage of operations. An ALARA (i.e., as low as reasonably achievable) Committee reviews all relevant work procedures. Total dose received by all personnel to date is approximately 40 mrem.
The TA-21 Decommissioning Health and Safety Plan also is followed for filter building characterization activities. Characterization activities requiring documentation are 1) the perchlorate and picric acid testing procedure, THA, and RWP; 2) the drum turning THA and RWP; and 3), the structure survey procedure, THA, and RWP

DECOMMISSIONING TECHNIQUES

The project follows current, accepted industry practices. Because alpha contamination is the radioactive hazard in question, remote systems are unnecessary, but health, safety and environmental practices are nonetheless rigorous. For example, at the start of the project, the exhaust system for Building 3 was cut off, the stack capped, and portable HEPA systems were brought in to control general and process ventilation. These steps were required, since the building lacked an efficient filtration system, and dismantlement activities could have resulted in increased emissions. The net result was that during decommissioning, site emissions were significantly reduced by means of a filtration system that was more efficient than that used at the facility during forty years of operations.
Also, at the start of remediation, an extensive hood and glovebox residual-uranium assay was performed. The data was necessary for reasons relating to safeguards and waste transportation and disposal. Measurements were performed in situ by the Laboratory's safeguards assay group using portable field instruments. Air sampling stations were installed around the site perimeter to collect data on fugitive dust emissions.
The general decommissioning sequence consists of system electrical disconnects, acid line and piping removal, hood and glovebox removal, exhaust system removal, asbestos removal, concrete floor sawing, utility piping removal, final system disconnects (i.e. electrical and fire protection), and building demolition.
For the demolition of Building 3, the Laboratory rented a hydraulic shear and trackhoe. Although renting equipment is common in the commercial sector, the practice had not been used before by the Laboratory due to the potential for contamination. The risk of contamination was low, and the advantages of increased productivity, improved worker safety, and effective waste segregation and minimization clearly outweighed the contamination risk. Building 3 was demolished using the shear in August 1994, and the equipment was cleaned and returned without incident. Figure 3 shows the shear in operation. Additional air monitoring stations were placed to monitor emissions.
Fig. 3.
Building 4 presents several unique challenges. The process exhaust ductwork runs the exterior length of the building on both sides and is constructed of fiberglass-coated stainless steel. Each side contains approximately 300 g of enriched uranium. Removal will require secondary containment and work at a height of 15 ft to 25 ft above the ground. One portion of the utility tunnel is reputed to have a very deep pit, currently covered with steel plating and asphalt, formed by a long-term, low-volume hydrofluoric acid leak.
Some new technology is being developed and applied to the project; however, only

proven technologies are considered. As previously noted, portable nondestructive assay methods are used for determining glovebox and ductwork holdup. Additionally, the LRAD is being used for concrete characterization and for the free release of items.

The Laboratory has issued a contract to demonstrate an innovative duct lining technique to either fix contamination inside ventilation systems or remove the contamination. A fabric liner soaked with commercially available strippable coating is everted into the duct using a pressurized canister. After curing, the liner may be removed to decontaminate the duct. Our goal is to decontaminate transuranic waste ductwork to levels low enough to allow recycling.

WASTE MANAGEMENT

Wastes generated on the project are classified as low specific activity or nonradioactive for transportation purposes and as low-level waste (LLW) for disposal purposes. The project waste volume estimate is 2000 yd3 of LLW. Additionally, radioactively contaminated asbestos has been generated (50 yd3), as have small amounts of mixed waste. Waste is disposed of at the Los Alamos LLW disposal facility. During the demolition of Building 3, most waste was placed into 20- x 8- x 8-ft standard shipping containers. Considerable compaction was achieved, and the entire container was disposed of.

Laboratory and Department of Energy policy precludes the free release of any material with detectable activity above background, even when the surface contamination is below release guidelines. While some materials have been released to a municipal landfill following demonstration of no detectable activity, waste minimization activities primarily emphasize volume reduction through on-site compaction and recycling of contaminated scrap metal. Concrete will be cleaned using a shot vacuum system, and the remaining slabs will be crushed and used as onsite fill. Through recycling, steel decontamination, and concrete crushing, low-level radioactive wastes from decommissioning Building 4 should be reduced by 60% compared to those generated at Building 3.

Soil remediation is coordinated with the Laboratory's remedial action project. Sampling and other activities also are coordinated to ensure data applicability and cost effectiveness.

COST AND SCHEDULE

The Building 3 and 4 South decommissioning was originally scheduled to last for three years, with a total project cost, including assessment, of $7.5 million. By accelerating the remediation schedule to two years, fixed costs are reduced significantly. Currently, the project is scheduled to be completed in September 1995, with a total project cost of $6.4 million. This does not include the costs for disposing of LLW, as the Laboratory waste management groups are funded directly. The filter building characterization effort will cost approximately $250,000.

LESSONS LEARNED

The main lesson learned to date involves the discovery, during system disassembly, of perchlorates in the Building 3 process exhaust system. Use of perchloric acid had not been identified during the assessment phase, when many records were reviewed and former operators interviewed. Perchlorates are shock and temperature sensitive, although they may be handled safely when wet. Experts from Oak Ridge National Laboratory were called in to assist in solving the problem. Considerable time and money was saved by using their proven techniques instead of developing solutions internally (Phillips et al. 1994). Perchlorate sampling and analysis were performed in the field using a portable ion-specific electrode system. Dismantlement required steaming two 3- x 3- x 20-ft sections of ductwork that joined to form a central upsweep that ran through the building attic and onto the roof. The system was disconnected in the attic and lowered to the floor, and the large section of ductwork was cut into smaller sections and rinsed in tanks of water.

Another important lesson learned is that the observational approach is very effective from both cost and schedule perspectives. By minimizing characterization activities, initial expenses and time to completion are reduced. Moreover, involving the people who will be doing the physical work in the work planning simplifies the techniques used and guarantees the feasibility of the chosen techniques.

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APPLYING A MATRIX EVALUATION TOOL FOR DEVELOPMENT OF DECONTAMINATION AND DECOMMISSIONING CRITERIA FOR THE WEST VALLEY DEMONSTRATION PROJECT

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ABSTRACT

The West Valley Demonstration Project (WVDP) is located at the Western New York Nuclear Service Center (WNYNSC), approximately 40 miles south of Buffalo, New York. The WNYNSC reprocessed over 600 metric tons of irradiated fuel between 1966 and 1972; this reprocessing operation also produced more than two million liters (600,000 gallons) of high level wastes stored in subsurface tanks. In 1980, Congress passed the West Valley Demonstration Project Act authorizing the Department of Energy to conduct a high level waste management project on the WNYNSC site. The two million liters of liquid wastes are to be processed and solidified by vitrification, then transported to a federal repository for disposal.

An analytical tool has been developed for use in the decontamination and decommissioning (D&D) planning process. This tool is being used specifically to develop viable facility closure scenarios that, through pathway analysis, will allow identification and selection of D&D criteria for the WVDP and the WNYNSC. The criteria development process is unique because the WNYNSC is home to the only commercially licensed irradiated fuel reprocessing facility ever to operate within the United States. This process is more complex because the Department of Energy and the State of New York both have defined roles in the site decommissioning, through the 1980 Act and the Operating License. To complete the project, the Act requires the DOE to decontaminate and decommission facilities used for the WVDP to criteria prescribed by the Nuclear Regulatory Commission. To terminate the operating license, New York State must also meet prescribed criteria set forth by the NRC.

INTRODUCTION

Background

The Western New York Nuclear Service Center (WNYNSC) near West Valley, New York was the site of the only commercial nuclear fuel reprocessing facility ever operated in the United States. The reprocessing facility operated from 1966 to 1972 and generated approximately two million liters of high level radioactive waste (HLW), which is stored in underground tanks at the site. The plant was shut down in 1972 for modifications and upgrades, but was never brought back on-line.

The processing facility was licensed by the Nuclear Regulatory Commission (NRC) to operate under Operating License CSF-1, Docket No. 50-201, with the New York State Energy Research and Development Authority (NYSERDA) the licensed owner of the site and facilities, and Nuclear Fuel Services (NFS) the licensed operator. In response to a request by the AEC (Atomic Energy Commission), the New York Atomic Research and Development Authority (predecessor to NYSERDA) amended the operating license application, indicating the recognition and acceptance by the State of New York of the perpetual responsibility for caring for the waste stored at the site.

In 1980, the United States Congress passed Public Law 96-368 called the West Valley Demonstration Project Act (the Act), authorizing the DOE "...to carry out a high-level liquid nuclear waste management demonstration project at the Western New York Nuclear Service Center in West Valley, New York." The West Valley Demonstration Project (WVDP) was established to demonstrate that HLW from the reprocessing of spent nuclear fuel can be managed safely in the United States. The Act directs the

DOE to: (i) solidify the HLW; (ii) develop containers suitable for permanent disposal of HLW; (iii) transport the solidified HLW to a federal repository for permanent storage; and (iv) dispose of the low level radioactive waste (LLRW) and transuranic (TRU) waste produced during the project. It also requires the DOE to decontaminate and decommission (D&D) the tanks and other facilities used during the solidification process, and "...any materials and hardware used in connection with the project in accordance with such requirements as the Commission may prescribe."

Purpose

The purpose of this paper is to describe a tool for use in developing D&D criteria for completion of the WVDP. The WVDP Act requires DOE to decontaminate and decommission project facilities in accordance with such criteria as the NRC may prescribe. With respect to the WVDP, it is important to assure that the standards developed are applicable to the decommissioning responsibilities of the DOE and New York State, and that both parties are subject to manageable and realistic criteria. During 1993, decommissioning planning for the WVDP was initiated by defining plausible regulatory options for the DOE to satisfy its project responsibilities, as well as identifying possible regulatory options for NYSERDA, the site owner. Although the DOE is not subject to the specific regulatory requirements that are imposed upon NYSERDA, there are agreements in place to ensure that, upon Project completion, there is an orderly transition from DOE operation to NYSERDA operation. It was decided to begin D&D planning by "scoping" the possible end results of decommissioning, thereby identifying areas of concern and uncertainty that could be factored back into the earlier planning process. A summary of last year's scoping effort is presented in the following section.

REGULATORY OPTIONS

A key aspect of project decommissioning is the prescription of D&D criteria; this task is the responsibility of the NRC. The NRC will execute this responsibility in a manner consistent with its rules, regulations, and policy. Since there is a Part 50 operating license for the facility, the NRC is most likely to identify WVDP D&D criteria consistent with the potential regulatory options for the facility. Four regulatory options have been identified as those most likely and available for implementation at the WVDP. They are:

- License Termination
- Amending the Existing License
- License Conversion
- Rulemaking

These four options represent possible long-term regulatory solutions for completion of the WVDP, and provide a general "target" for D&D planning. The long-term solution will be dependent on the Record of Decision (ROD), which will be issued after completion of the Environmental Impact Statement (EIS), scheduled for October 1996.

D&D SCENARIO AND CRITERIA DEVELOPMENT APPROACH

Results of Regulatory Option Review

The four regulatory options can be evaluated relative to possible closure scenarios for the diverse facilities at the WVDP, and the resulting combinations in terms of the regulatory options can be assessed. For example, license termination would require that waste and facilities be removed from the site to satisfy site release requirements. However, at this time there is no repository available to receive much of this waste, rendering this regulatory option somewhat impractical. To more thoroughly evaluate the interrelationships between individual facility closures and possible final regulatory solutions for the WVDP, a matrix evaluation tool has been developed to permit the systematic evaluation of facility-closure combinations, referred to as D&D scenarios.

Matrix Overview

The matrix is a tool for qualitatively evaluating the various factors influencing a process. The matrix looks at environmental, programmatic, regulatory, and administrative impacts. In addition, the matrix process allows for the site to be evaluated at the facility/equipment level, instead of the entire site as a unit. For example, in the matrix, an evaluation for shipping HLW to a federal repository for permanent disposal would be based on the feasibility of such a facility in the decommissioning phase, rather than on the desirability of that option based on environmental concerns. If the facility/closure option is not feasible under the matrix evaluation, only the facility is impacted, rather than the entire site. The matrix also reflects available information from the EIS and other pertinent

documents in the evaluation of facility - closure option combinations. Use of the matrix results in plausible D&D scenarios, which can then be evaluated further using pathway analyses. The resulting outcome of the pathway analyses is data that can be used to develop site-specific D&D criteria.

APPLICATION OF THE MATRIX TOOL

A matrix approach provides an alternate and practical solution for evaluating potential closure options for the WVDP following vitrification. The matrix approach involves selection of facilities and closure options for evaluation using predetermined screening criteria. The criteria address elements of significance to the project, e.g., safety, compliance, cost, community acceptance, etc. Each of these elements can be given a relative weighting when compared with other elements. This is followed by assigning relative ranking for a particular option. These values (weights and ranking) are applied to select facilities and options in a matrix format. Iterations and summation of the numerical scores lead to development of a short list of realistic options for individual facilities or for the project as a whole.

Although specific facilities, closure options, and screening criteria have been identified for the current application, it should be noted that each of these parameters can be modified to meet project needs. Trial runs with the matrix approach indicate that it represents an economical and viable solution for the current WVDP needs. The approach is flexible enough to cater to changes in the project environment and evolving regulatory requirements.

Following is an example to illustrate how this process can be utilized for selection of candidate D&D scenarios for the WVDP. Evaluation using this matrix approach follows the following sequence:

- Identify facilities for consideration
- Identify potential options for closure
- Decide upon the key factors that influence each option
- Assign numerical weighting factors (optional)
- Evaluate and determine the numerical value for each cell
- Tabulate results, round off, and simplify
- Compare results, select options with best scores
- Review and eliminate impractical options
- Prepare a short list of scenarios for further evaluation (pathway analysis)

Identify Candidate Facilities

Applying this approach at the WVDP, the first step in the matrix evaluation process was selection of facilities potentially requiring cleanup and/or removal. Eleven facilities were selected for evaluation in the matrix. These facilities include site buildings/areas having the greatest potential for radiological impact. These facilities were either used for vitrification of HLW or will have a significant impact on any D&D decision made for the site. Facilities included in the current analysis are:

- Process Building
- NRC-Licensed Disposal Area (NDA)
- State-Licensed Disposal Area (SDA)
- Low Level Waste Treatment Facility (LLWTF)
- High Level Waste Tanks
- Vitrification Building
- Liquid Waste Treatment System (LWTS)
- Chemical Process Cell Waste Storage Area (CPC-WSA)
- Radwaste Treatment System Drum Cell
- Lag Storage Areas
- Construction and Demolition Debris Landfill (CDDL)

Identify Candidate Closure Options

A closure option is determined by what activities are to be performed on a particular facility and what the outcome to the facility would be following completion of those activities. Ten closure options were chosen for evaluation. These options cover the range of possibilities and yet keep the number manageable:

1. Complete removal from site
2. Complete removal to new on-site storage facilities
3. Complete removal to existing on-site storage facilities
4. Partial removal to new on-site storage facilities

5. Partial removal to existing on-site storage facilities
6. Complete removal to new on-site disposal facilities
7. Complete removal to existing on-site disposal facilities
8. Partial removal to new on-site disposal facilities
9. Partial removal to existing on-site disposal facilities
10. Monitor and maintain

There are now eleven facilities to be evaluated against ten closure options, which results in numerous facility/closure option combinations, such as complete removal of one facility, partial removal of another facility, monitor and maintain a third facility, and so on. However, it is not cost-effective to develop D&D criteria for all 110 possible facility/closure option combinations. A method to reduce the number of facility/closure option combinations to those that are most plausible involves the use of screening criteria.

Screening Criteria

Screening criteria are the elements to be evaluated for a given facility and closure option. The major factors considered in the selection of screening criteria are environmental impacts, regulatory and institutional constraints, public desires, congressional support, technical feasibility, and costs. From these factors, nine screening criteria were selected for use in evaluating the facility-closure option combinations. These criteria were considered appropriate because they address issues of mutual interest to the DOE and New York State, the active role of the state and the local community in the decision-making process, and the safety and cost-related aspects that are considered in all evaluations. The nine screening criteria are:

- Safety
- Compliance
- Long-Term Effectiveness
- Short-Term Effectiveness
- Reduction
- Implementability
- Cost
- State Acceptance
- Community Acceptance

Prior to beginning the evaluation, it is critical to ensure that the screening criteria are thoroughly defined and understood by all participants. Screening criteria may be weighted in the matrix based on their relative importance to each other, or may be evaluated without weighting factors.

Ranking

Ranking determines the importance of a screening criterion in a particular facility-closure option combination. For example, of all the facilities considered for complete removal, the Process Building represents the potential for highest occupational radiation exposure; therefore, this closure option would be rated poor (low score) for safety in this particular situation.

Each facility/closure option combination was ranked for each of the nine screening criteria. The screening criteria were assigned a value between 0 and 5 (0 not being viable and 5 being good), based upon the importance and applicability to a particular situation. The ranked values for the nine criteria for each facility/closure option combination are then totaled to determine the overall score for that combination. Any criterion given a 0 for a particular facility/closure option combination (for example, implementability for "complete removal from site") results in a zero for the entire combination, thereby dropping that combination from further consideration.

Matrix Completion

Figure 1 shows a portion of a matrix completed during a trial run by the process described in this report. The first step in completing the matrix was to list facilities and closure options in a row and column format as the variables. A numerical value (ranking) was noted in the matrix cell; the upper left matrix cell represents a numerically assessed comparative value of "0" for "Complete Removal From Site" of the "Process Building." This illustrates the rejection of a facility/closure option combination that is not viable because of a zero assigned to one of the screening criteria. This also applies to complete removal of the NDA. Fig. 1.

The first matrix cell in the column for "Monitor and Maintain" the "Process Building" was assigned a numerical ranking of "18." This ranking is the sum of

assessed values for each of the nine screening criterion described previously. For example, this cell shows a numerical value of "2" for Safety, a value of "1" for Community Acceptance, and so on. The rationale for arriving at this value was provided in the previous section. The argument to assign a low value of "2" for Safety was that this option could involve greater potential for exposures, industrial accidents, resulting contaminations, etc. Similarly, a low value of "1" for Community Acceptance represents the unacceptability of this option to the local community; we have assumed the community would not be very receptive to any waste or facilities associated with waste to be maintained on-site.

Although a value assigned for each criterion listed in a cell is subjective (dependent on the individual evaluator), it tends to be normalized under the given constraints and in comparisons with the other closure options. As the evaluation progresses, the evaluator may decide to revise a previously assigned value, thus leading to further normalization. It is this iterative process, involving reasoning, that eventually leads to convergence of individual assessment results.

Scenario Development

The process starts with a tabulation of the results in each cell and a summation to get a total picture. The table leads to a short list of closure options that are viable for each facility. Those options with the highest scores are evaluated further. These highest ranking closure options are reviewed for each facility and a determination is made as to whether the option is prudent to implement. For example, it will not be prudent to demolish a facility because it is shown to be an economical choice, and then construct another to support monitoring and maintaining of another facility. Thus the evaluation involves a "reality check;" evaluations take a realistic view of the surrounding conditions and environment.

Theoretically, the "preferred closure option" for each facility should be that option having the highest value in the matrix. However, in developing scenarios that would include similarly ranked facility/closure option combinations for each facility, we elected to eliminate those facility/closure option pairings that required actions that were not in concert with the preferred closure options taken for other facilities on site. This approach enabled us to focus on those options that made sense in the overall scheme of site D&D.

In summary, the highest ranking options are reviewed for each facility and a determination is made as to which options make sense considering what is being done with the other facilities on site. The result of such an interactive approach is that the closure options with the higher rankings will not be chosen for a particular scenario if the choice does not make sense in the overall D&D process. The compilation of plausible facility/closure options results in a "D&D scenario" that represents a possible site configuration, or goal, for decommissioning. D&D scenarios are further evaluated using pathway analysis. The resulting outcome of the pathway analyses is data that can be used to develop site-specific D&D criteria. As previously discussed, a preliminary matrix run has been completed, and the resulting D&D scenarios are being used to set up and test the pathway analysis program. The current plan is to perform a matrix run using a wider range of participants from other organizations that were not involved in developing the preliminary matrix, and to use the resulting scenarios for pathway analysis.

CONCLUSION

Using a matrix approach to analyze complex issues involving regulators and federal and state agencies is neither new nor uncommon. Experiences to date in developing and piloting this matrix tool have shown a number of benefits in using this type of an approach. Some of these benefits are as follows.

Benefits

Easy to Use: A process employing a matrix approach is quite simple to understand and easy to use.

Implementable: Because of its simplicity, adaptability to varied situations, and cost-effectiveness for the evaluation, it is easier to implement than other more quantitative methods.

Ownership: Since the evaluators have direct first hand knowledge and involvement in the process, they have a sense of ownership and are more likely to support the results and implementation.

User Friendly: By developing a computer program to run the analysis, we have further simplified it and made it user friendly.

Wider Range: It addresses several areas of interest to evaluators with different

backgrounds, and it can be used by a wide range of individuals.
Involvement: It is one of the few methods where project stakeholders can directly provide their input or run the analysis for themselves. It provides a vehicle for discussion and can be used to solicit input in a timely manner. The decision makers are more likely to approve the results because of the stakeholders' early and continued involvement and understanding of the process.
Economical: The process requires neither advanced computer software nor technology to implement. The costs associated with the evaluation are minimal.
Iterative: Several iterations can be performed to suit varied needs or to adjust to changes in the regulatory or political environment more quickly and at a fraction of the cost of quantitative methods.
Acceptance: This approach has been used frequently by federal, state, and regulatory agencies, and by commercial groups; it finds wide acceptability with users and regulators.

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FLAMELESS THERMAL OXIDATION FOR LOW CONCENTRATION VOC REMEDIAL WASTESTREAMS: DESIGNS FOR PLANNED DOE DEMONSTRATIONS
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ABSTRACT
Environmental restoration efforts in the government and private sectors are currently challenged with remedial decisions for a variety of hazardous and mixed waste matrices. Soils and sludges require the application of remedial technologies to separate out hazardous organics, metals or radionuclide contaminants to meet applicable regulatory cleanup criteria. Destruction technologies which typically complete thermal, vapor extraction and stripping technologies chemically alter organics to destroy them through oxidation into carbon dioxide , water and in some cases acid gases. The most commonly used oxidation processes are incineration, catalytic oxidation, regenerative oxidation and flameless oxidation. Flameless thermal oxidation is an attractive VOC destruction alternative because it combines many of the advantages of the various flame-based methods but avoids most of the problems inherent to those methods. Its packed bed design operates by preheating an inert ceramic matrix to 1600-1800F and then passing the premixed organic vapor stream (plus additional air and fuel as needed) through the bed to effect uniform and nearly complete oxidation. Because the destruction matrix is inert and non-catalytic, halogenated and sulfonated compounds do not affect system life and since it is flameless, its efficiency of oxidation is not impacted by a flame's inherent instability The large thermal mass of the oxidizer bed accommodates significant moisture content and fluctuations in organic stream concentrations without effecting process efficiency.
This paper presents several designs for remedial systems which can be enhanced by incorporation of flameless thermal oxidation for very effective VOC abatement and heat recovery for process reuse. Specifically discussed are enhanced designs for 1) low temperature thermal desorption, e.g., for the removal of volatile (VOC) and semi-volatile (SVOC) organics from soils, sludges or other solid matrices; 2) in situ vapor extraction (SVE) e.g., for the removal of spilled organics from the vadose and saturated zones of soils; and 3)heated air/steam stripping, e.g., for removal of volatile organics from surface and ground waters. Towards the realization of such enhanced remedial systems, several planned small-scale demonstrations at DOE sites are discussed, specifically 1) a demonstration of a Thermatrix ES-300H oxidizer in an SVE chlorinated organic destruction application at the Savannah River

Site; 2) a demonstration of a Thermatrix GS-100M oxidizer as the secondary combustion chamber of the Plasma Arc Furnace at DOE's CDIF facility in Butte, Montana; and 3) a proposed technology development program through DOE's Morgantown Energy Technology Center to produce an innovative, non-incineration thermal treatment system to remediate mixed-waste contaminated solids at DOE facilities.

COMMON REMEDIAL TECHNOLOGIES AND THE VOC WASTESTREAMS THEY GENERATE FOR TREATMENT OR DISPOSAL

Environmental restoration efforts in the government and private sectors are currently challenged with remedial decisions for a variety of hazardous and mixed waste matrices. Soils and sludges require the application of remedial technologies to separate out hazardous organics, metals or radionuclide contaminants to meet applicable regulatory cleanup criteria. Surface and ground waters that have become contaminated by contact with hazardous, radioactive or mixed waste constituents must also undergo processing to achieve "clean water" status as defined by regulatory contamination limits. In nearly all cases where mixed waste is involved, either the hazardous or radioactive constituent must be removed from the waste, e.g., hazardous organics removed to leave a radioactive-only waste, to allow for compliant disposal at either a RCRA or radioactive landfill.

The present paper promotes the enhancement of several remedial technologies which have found widespread application in hazardous and mixed waste treatment at site remediations: 1) low temperature thermal desorption, e.g., for the removal of volatile (VOC) and semi-volatile (SVOC) organics from soils, sludges or other solid matrices; 2) in situ vapor extraction, e.g., for the removal of spilled organics from the vadose and saturated zones of soils; and 3)heated air/steam stripping, e.g., for removal of volatile organics from surface and ground waters. Furthermore, this paper describes the role that recuperative flameless thermal oxidation can play in not only enhancing the operation and performance of these "separation" technologies but also in the very high destruction and removal efficiencies (DRE) for volatile organics it can achieve.

VOC DESTRUCTION OPTIONS FOR REMEDIAL ORGANIC WASTESTREAMS

Destruction technologies which typically complete thermal, vapor extraction and stripping technologies chemically alter organics to destroy them through oxidation into carbon dioxide , water and in some cases acid gases. The most commonly used oxidation processes are incineration, catalytic oxidation, regenerative oxidation and flameless oxidation.

Flame-based incineration has been a widely used method of VOC destruction but has recently come under greater regulatory scrutiny and constraint. With incineration, liquid or vapor organics pass through a highly reactive flame zone and are converted to non-hazardous products or easily neutralized species. Due to the inherent instability of flames however, this process is often incomplete, with the formation of undesirable products of incomplete combustion (e.g., phosgene, dioxins).

For many applications catalytic oxidation is an improvement over flame-based incineration. Organic destruction takes place more rapidly in the presence of a catalyst than in a homogeneous gas stream. Or alternatively, the same destruction rate can be achieved at lower temperatures. Catalysts can perform well with a wide variety of hydrocarbons, although most are poisoned by sulfur and halogen containing compounds. While incineration is typically used with rich organic streams, catalytic oxidation is often preferred for the destruction of streams with moderate to lean fume content. Streams with varying richness may cause sintering or deactivation of the catalyst surface.

Regenerative oxidizers are used primarily in low-concentration organic stream applications. They typically feature a flame-based combustion chamber interconnecting two or more fixed beds. The inlet and exhaust flows are alternately diverted across the beds through complex valving. The incoming organic stream is preheated in one bed while the heat released during oxidation is captured by the other bed(s) as the exhaust gases exit the system. An oxidation chamber connects the beds and is heated by flame burners or electric elements. Regenerative oxidizers typically achieve 80 -98% heat recovery which aids in the efficient destruction of highly dilute organic streams. Due to the alternating nature of the flow distribution network and the design of the oxidation chamber, DREs are typically <99%. In order to prevent overheating, regenerative oxidizers must add large amounts of dilution air to process rich streams.

Flameless thermal oxidation (FTO) is an attractive VOC destruction alternative

because it combines many of the advantages of the various flame-based methods but avoids most of the problems inherent to those methods. Its packed bed design operates by preheating an inert ceramic matrix to 1600-1800F and then passing the premixed organic vapor stream (plus additional air and fuel as needed) through the bed to effect uniform and nearly complete oxidation (See Fig. 1.). Because the destruction matrix is inert and non-catalytic, halogenated and sulfonated compounds do not affect system life and since it is flameless, its efficiency of oxidation is not impacted by a flame's inherent instability The large thermal mass of the oxidizer bed accommodates significant moisture content and fluctuations in organic stream concentrations without effecting process efficiency. The following table presents performance data from commercial and laboratory unit testing.
Fig. 1.
TABLE I
The recuperative design takes flameless thermal oxidation one step further. Instead of introducing organics directly to the bed, the organics enter the oxidizer through a series of tubes running vertically through the bed. Upon exiting the tubes near the top of the bed, the organics enter a reaction zone and are oxidized. As the oxidized vapor is directed back down through the bed toward the exhaust outlet, its heat is imparted to the incoming organic vapor rising vertically through the tubes which increases the efficiency of the overall process. This "single pass" design overcomes two major problems associated with regenerative thermal oxidation, specifically the absence of any volume of unreacted fume to degrade DRE and the elimination of a need for complex valving or indeed any internal moving parts. Our experience with a 6500 scfm scale recuperative oxidizer in the paint applications industry confirms this performance expectation with respect to heat recovery, high DRE (>99.99%) and trace to non-detectable criteria air pollutants emissions. (<2 ppm NOx; <10ppm CO).
Recuperative flameless thermal oxidation (RFTO) offers several other advantages which significantly enhance remedial applications. RFTO is designed to accommodate low concentration, moderate to high flow vapor streams which are typical of many remedial separation technologies, e.g., SVE and air/stream stripping of groundwater. RFTO as a destruction method routinely achieves greater than 99.99% DREs for the halogenated (and non-halogenated) hydrocarbons typically found in ground-spilled hazardous organic remediations. Along with excellent DREs, RFTO produces only trace to non-detectable oxidation byproducts (e.g., carbon monoxide and nitrogen oxides), easily meeting current air emission standards. Depending upon the concentration of organic vapor entering the recuperative oxidizer, the efficiency of heat recovery would allow for excess heat utilization in such applications as heated air/steam generation for enhanced SVE or groundwater treatment or in the case of thermal desorption, preheating and inerting of the kiln interior for safe, efficient separation of organics from soils and sludges.
Before discussion of enhanced remedial system designs, it will be useful to look more closely at the recuperative FTO design and the benefits to be derived from it.
RFTO DESIGN AND ITS BENEFITS
A typical recuperative flameless thermal oxidizer consists of a refractory lined alloy steel shell, a refractory lined tubesheet, heat exchange tubes, a preheat burner, a gas introduction port and an exhaust port. The vessel, above the tubesheet, is filled with ceramic packing. The ceramic packing is located both inside and outside of the heat exchange tubes. (See Fig. 1.)
In operation the recuperative reactor is first preheated. This is accomplished by firing a burner into the top chamber of the reactor. The burner is fired only long enough to heat the top 16 inches of the packing to above 1400F. The alloy tubes within the packed bed are kept cool during the preheat process by flowing a small amount of air through them. At the end of the preheat stage a temperature gradient exists in the packed bed, from >1400F at the top of the bed, to several hundred degrees at the bottom of the bed.
After preheating, a mixture of air, fuel, and organic compounds to be destroyed is introduced into the reactor, in the area below the packed bed called the plenum. The plenum is separated from the packed bed by a refractory lined tubesheet. The plenum area is kept cool by the continuous flow of cool reactant gases into this area. From the plenum, the gases are directed into the heat exchange tubes. As the influent gases flow through the tubes, they pick up heat from the hot exhaust gases flowing counter current on the outside of the tubes.

As mentioned above, the tubes are filled with ceramic bed media. The bed media within the tubes serves two functions. Its first purpose for the packing inside the tubes is that it aids in the heat exchange process. Since the gases are optically thin, heat transfer by radiation would be very inefficient. However, heating the gases by forced convection makes the heat exchange to the gases much more efficient. The second purpose for the bed is to act as a flame arrestor. This is accomplished by increasing the gas velocity to a value higher than the flame velocity of the mixture, and also by quenching any flame that may be present.

The overall heat exchange mechanism starts with the hot exhaust gases heating the packing on the outside of the tubes. This is occurs primarily by forced convection, although radiation and conduction do help to a much lesser degree. Heat is then radiated and conducted from the packing to the outside of the tube walls. Heat is conducted from the outside tube walls to the inside tube wall, where it is radiated and conducted to the bed media within the tube. The heat is then absorbed by the cool incoming gases through forced convection.

When the gases reach the top of the tubes, they have reached combustion temperature. Combustion occurs as the gases emerge from the tubes and continues into the open chamber above the tubes. This creates a pool of chemical radicals in the top chamber, which leads to the establishment of a well stirred reactor region. Due to the presence of the high concentration of radicals in this area, uniform combustion occurs very rapidly.

The products of combustion then flow down through the packed bed on the outside of the heat exchange tubes. As the hot exhaust gases flow through the packing, they give up some of their heat to the packing through forced convection. The exhaust gases then exit the reactor at the exhaust port and are directed to the stack where they are discharged to atmosphere.

Thus inherent to the recuperative design is the heat retention, transfer and reuse potential which enhances the efficiency of processing organic vapor streams of low concentration. The 60 - 65% heat recovery in operation also allows for excess heat to be available from such low concentration streams for use in the separation phase of the remedial process. And since thermal processing is maintained at 1600 - 1800F, conversion of organics is essentially complete while formation of nitrogen oxides is minimized. It should be mentioned also that non-recuperative ("straight through") designs of flameless thermal oxidizers also offers significant heat recovery and reuse. Thus for moderate to high concentration remedial organic streams which exceed the desired conditions for processing with the recuperative system, the addition of a heat recovery unit (e.g., waste heat boiler) to a straight through thermal oxidizer can effectively process the stream while providing heat recovery and reuse potential.

ENHANCED REMEDIAL SYSTEMS UTILIZING RFTO

The above described benefits of recuperative flameless thermal oxidation can not only provide very effective VOC abatement but also increase the efficiency of the remedial processes themselves. The following presents several designs for remedial systems to demonstrate the benefits of incorporating the recuperative oxidizer design with the realization that the straight-through-with-heat-recovery design can be substituted if appropriate (e.g., higher contaminant concentrations; low system flow conditions).

Enhanced Indirect Fired Thermal Desorption

Thermal desorption is typically an ex-situ remedial process for the physical separation of organic contaminants from soil, sludge, sediments or other solid matrices. In this process thermal energy is applied either directly or indirectly to volatilize organic compounds from the contaminated solids. Operating temperatures vary (typically between 500-850F) and the thermal processing is designed for separation rather than destruction of organic contaminants. Once volatilized, the organics as a vapor stream are processed through some type of air pollution control device, e.g., captured by adsorption on carbon or condensed, or subjected to further treatment to convert them into environmentally acceptable species. Perhaps the most common thermal desorption design is the rotary kiln or dryer. This design consists of a horizontal cylinder heated either directly (e.g., by blowing hot gas over the waste medium) or indirectly (e.g., by blowing hot gas over the rotary drum walls) and operate either on an incline or by rotation. Processing capacities of 5 - 55 tons per hour are reported. The effectiveness of desorption processing is a function of the chemical composition of the waste feed, its physical characteristics, as well

as feed rate, temperature and residence time.
In the proposed design organic-contaminated solids are fed into an indirectly fired rotary kiln and heated to 500 - 850F. (See Fig. 2.) The hot desorbed organic vapor stream is processed for particulate and the captured solids transferred to the treated solids handling area. The filtered organic vapor stream then enters the recuperative oxidizer where supplemental air is added if necessary. Within the oxidizer the organics are converted to CO2 and H2O. The hot oxidizer effluent vapor stream is then split, the majority going through a gas cleaning process (e.g., scrubbing of acid gases) if necessary and then discharged to atmosphere while a split stream at 700 - 1000F is returned to the kiln where it is used as a preheating and inerting carrier gas in the ongoing desorption processing. In this way the heat value of even low concentration organic contaminants is recovered and used to increase the efficiency of the overall process.
Mobile Enhanced Soil Vapor Extraction
Soil vapor extraction (SVE) is a process for the removal of VOCs and some SVOCs from the subsurface soil zone located between the land surface and the top of the water table. When operating, SVE transfers organic contaminants from soil and interstitial water to air and entrained and condensed air streams which require further treatment.
During SVE air is drawn through the soil, passing through the soil particle pore spaces, following paths of least resistance and carrying contaminating organic vapors with it. In turn, condensed-phase organics will vaporize and replace the exiting organics. Thus this dynamic vaporization process establishes a dynamic equilibrium of condensed and vapor- phase organics until all of the condensed-phase organics have vaporized and pass through the higher permeability soil volume. This dynamic vaporization process can be enhanced through the use of heated air or steam in the injection wells.
In practice SVE operates with two sets of wells. One set, the vapor extraction wells, are operated under negative pressure and provide the motive force for vapor transport through the soil. The other set, the air vent or injection wells serve to replace the removed air and enhance the flow and control of vapor transport. The extraction wells are powered by vacuum pumps or blowers with in-line liquid collection or vapor condensation capacity. The removed, organic-laden vapor is typically processed in one of three ways: adsorbed on activated carbon; thermally destroyed by incineration or catalyzed oxidation; or condensed through refrigeration.
The design proposed here is for a mobile enhanced SVE system with several unique features. (See Fig. 3.) Organic vapor removed through extraction wells is destroyed in a recuperative style oxidizer, the heat of combustion recovered and used to generate heated air or steam in a waste heat boiler. Ongoing vapor extraction can then be enhanced by using this heated air or steam in the injection wells. In addition, if organic condensate is separable from the aqueous phase in the knock-out pot, it can be vaporized in a heat exchanger and reintroduced into the oxidizer for destruction. A significant body of past performance data supports the expectation that the air emissions from the oxidizer will be 99.99% DRE or better for contaminant organics, with NOx , CO and products of incomplete combustion at trace to non-detectable levels.
Fig. 2.
Fig. 3.
This mobile unit as designed would allow localized remediation of small organic spills or contaminated areas at site remediations. Using the heat of oxidation to generate steam or hot air for injection makes efficient use of the heating value of the organic contaminants, enhances the efficiency of volatile organic removal and promotes the extraction of more difficult organics (e.g., dense non-aqueous phase liquids).
Enhanced Groundwater Extraction and Treatment
Groundwater contaminated with volatile organics is typically remediated by first pumping water from a recovery well into the top of a packed stripper column or stacked tray assembly. In the packed tower design an air stream is directed up from the base of the column countercurrent to the water which is falling down from the top. As the air rises and comes in contact with the water, the organics transfer to the air stream and are carried out the top while the treated water drains out the bottom of the tower. The mass transfer of organics from water to air is driven by

the mechanical energy imparted by the forced air and the large surface area. The ultimate efficiency of removal is dependent upon the specific stripper design and dimensions, as well as physical and chemical characteristics of the organics involved. Increased efficiency of removal can be effected to some degree by preheating the water and thus enhancing the removal of less volatile compounds. In the case of organics with low miscibility with water, e.g., halogenateds, steam stripping is commonly used. Steam stripping is also more effective than air stripping when the water contains very high concentrations of volatiles, e.g., when DNAPLS are present. Steam stripping columns are usually run counterflow steam to water and allowance is made for immiscible phase separation and decantation both in the stripper overhead products and column bottoms. Further advantage is gained in steam stripping by the formation of minimum boiling azeotropes. Such azeotropes form at temperatures below the boiling point of the organic compounds present and are removed with the steam vapor exiting the column. For example, 1,1,2-trichloroethylene forms an azeotrope with water that boils at 86C while the boiling point of the pure compound is 113.7C.

Either style of flameless thermal oxidizer can be coupled with air or steam stripping contaminant separation methods as the VOC abatement step but the coupled system with the greatest overall advantage would be the recuperative oxidizer/steam stripping design. The recuperative oxidizer would be preheated to prepare it to receive organic vapor and to generate heat for the waste heat boiler and in turn to generate steam for the preheater and stripper. During operation, groundwater is pumped from recovery wells through the preheater and into the top of the stripper column. Steam generated in the waste heat boiler is delivered typically at 1 atm and 100C with a 10 - 30 mole percent of feed steam consumption. The overhead vapor exits the stripper, undergoes decanting and then enters the recuperative oxidizer where the organics are oxidized. The heated effluent from the oxidizer passes through a waste heat boiler prior to being quenched and scrubbed for acid gases and finally released to atmosphere. The recovered heat is used to continue generating steam for the preheater and stripper while the treated groundwater effluent from the stripper can serve as the water source for steam generation. Any free phase organic liquids pumped from recovery wells or resulting from liquid/liquid separation can be reintroduced to the stripper column and processed through the oxidizer. This coupled groundwater extraction and treatment system thus offers efficient heat recovery and reuse as well as effective VOC abatement performance to enhance groundwater remediation.

PLANNED DOE TECHNOLOGY DEMONSTRATIONS

Several small scale demonstrations of coupled remedial/FTO technology are planned in the near future at several DOE sites. The objective of these demonstrations is to gain field performance data from which the benefits of such coupled technology can be compared with other remedial alternatives.

Coupled SVE/Flameless Thermal Chlorinated VOC Abatement at Savannah River Site

A demonstration of a Thermatrix ES-300H oxidizer in an SVE chlorinated organic destruction application is planned this spring at the Savannah River Site in Aiken, South Carolina. A skid mounted, electrically heated 5 scfm oxidizer will be coupled with a rotary lobe blower at a preexisting vapor extraction well in the A/M Area of the site. After system optimization, up to 10,000 ppmv of combined perchloroethylene, trichloroethylene and trichloroethane will be processed through the oxidizer continuously over a several week period. Gas samples from the inlet and outlet of the oxidizer will be taken for analysis. Analytical data will indicate the DRE achieved for each compound, the presence of any products of incomplete combustion and the quantity of acid gas (HCl) generated.

Coupled Plasma Arc Furnace/Flameless Thermal Oxidizer at the CDIF Facility

The Retech, Inc. plasma arc centrifugal treatment system installed at DOE's Component Development and Integration Facility (CDIF) in Butte, Montana will be the site of another demonstration in late spring. The plasma arc system is designed to destroy organic material through high-temperature oxidation and to immobilize inorganic material into a low-leachable glass ceramic matrix. Currently, off-gas containing acid gases, moisture and particulate from the primary processing chamber passes through a gas-fired secondary combustion chamber, high energy scrubber, a series of filters and finally catalytic reactor to reduce NOx emissions to the atmosphere. The objective of the planned demonstration is to evaluate the

feasibility of a Thermatrix flameless thermal oxidizer replacing the secondary combustion chamber. It is anticipated that total hydrocarbons, CO and NOx emissions will be reduced below current emission levels due to the demonstrated superior performance and control of the flameless thermal oxidation process over flame processes.

Proposed Design of "Non-incineration" Thermal Treatment System to Morgantown Energy Technology Center

Thermatrix has proposed a technology development and verification program to the Morgantown Energy Technology Center, in response to the Research Opportunity Announcement entitled "Applied Research and Development of Technologies for Environmental Restoration and Waste Management." The objective of the proposed program would be to produce an innovative, non-incineration, thermal treatment system to remediate mixed-waste contaminated solids at DOE facilities. The first phase of testing would verify the capability of Thermatrix's flameless thermal oxidation technology to effectively treat low concentration hazardous organic vapor streams similar to off-gases from solid, low-level mixed waste (LLMW) organic separation treatment processes like thermal desorption and soil vapor extraction. If successful, the second phase of testing would develop a pilot scale thermal treatment system by coupling a rotary kiln thermal desorber with a recuperative style flameless thermal oxidizer and wet scrubber. After this system has been verified functionally by processing clean sand, it would be demonstrated in a simulated LLMW (non-radioactive) mode using non-rad soil or sand spiked with hazardous liquid organics and then with actual low-level mixed waste soil or solids. If successful in these demonstrations, a final effort would design full scale thermal desorber/Thermatrix FTO systems scaled up from pounds-per-hour pilot design criteria to tons-per-hour processing capacities appropriate to specific DOE waste inventory volumes.

CONCLUSIONS

Remediation of organic-contaminated soils, sludges, surface and ground waters currently utilizes separation technologies which are energy intensive and typically leave a residue requiring further handling, e.g., destruction or off-site disposal. Flameless thermal oxidation, especially with heat recovery, offers an innovative approach to enhancing the efficiency of routine remedial technologies as well as superior VOC abatement performance. Designs for enhanced thermal desorption, mobile soil vapor extraction and groundwater steam stripping extraction and treatment are presented which by incorporating a recuperative oxidizer into the operating system promises significant heat recovery, effective destruction of organic contaminants (>99.99% DRE) and only trace to non-detectable criteria air pollutant emissions. Such enhancement of routine remedial methodologies can provide the remedial manager with more cost effective, energy efficient and better performing tools with which to tackle the many hazardous and mixed waste environmental restoration challenges.

43-2

EVALUATION OF TWO NEW GROUT MATERIALS FOR IN SITU BARRIER CONSTRUCTION

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ABSTRACT

Grout injection tests employing permeation grouting methods are underway to evaluate two grout materials for in situ barrier construction. Performance goals have been established to determine whether permeation of the materials within unconsolidated soils will provide competent subsurface barriers of sufficient lateral extent and hydraulic integrity to meet the U.S. Department of Energy (DOE) environmental restoration needs. The grout materials are a montan wax-based particulate grout developed in Germany and a sodium silicate chemical grout developed in France. Field tests are underway at a site in Richland, Washington, adjacent to the DOE Hanford Reservation. The results of earlier laboratory and field tests are presented and the ongoing field study in Washington is described. The preliminary results suggest both

grouts may be suitable for addressing containment/stabilization needs within the DOE complex under certain conditions. The suitability of the grout depends on the site conditions and the performance requirements (durability, residual hydraulic conductivity, etc.). Traditional grout permeation methods are unsuitable for introducing the montan wax material under most soil conditions due to the relatively high viscosity of the material and difficulty in controlling the breaking of the emulsion. Permeation grouting with montan wax may be suitable for stabilizing buried wastes with high void ratios. Other application technologies such as jet grout and soil mixing were not evaluated in this study but are being used in Germany and are the preferred method for introducing montan wax grout. The penetration of the sodium silicate material using permeation grouting was more successful but questions still exist as to the competency of the barrier produced. The post-injection characterization studies planned for the Spring of 1995 should address the remaining uncertainties.

INTRODUCTION

Two grout materials of European origin are being studied at a site near Richland, Washington to evaluate their ability to permeate unconsolidated soils and reduce the hydraulic conductivity. The two grout materials are a montan wax emulsion developed in Germany and a glyoxal-modified sodium silicate grout developed in France.

The materials were first identified in 1992 by the International Technology Exchange Program (ITEP), an Office of Technology Development program (1). One of the objectives of ITEP was to identify innovative environmental remediation and waste management technologies in other countries for possible import to the United States to expedite the clean up and waste management needs within the DOE complex. As part of this effort, a survey of German environmental technologies was performed by Golder Associates Inc., under contract to Sandia National Laboratories and the DOE Office of Technology Development.

The lack of enforced environmental regulations in the former German Democratic Republic and the concentration of industry in the western part of the country has resulted in serious environmental problems throughout the country. In 1991, the Federal Environmental Agency estimated there were 146,400 suspected contaminated sites in Germany; 99,300 in western Germany and 47,100 in eastern Germany (2). The sites consist primarily of old waste dumps, industrial areas and military bases. The agency estimates that 10 to 20% of the sites will require remediation to reduce public health risks to acceptable levels.

Because of the numerous sites that require corrective measures, remediation will not be completed for many years. As a result, containment and other mitigation measures that could be put in place quickly are being considered to prevent problems from getting worse until a final solution is found. Based on this situation, a number of companies, institutes and joint ventures are developing technology for constructing in situ containment barriers (low permeability regions) to reduce or arrest contaminant flow and transport. Two of the concepts under development and testing are a montan wax emulsion developed by Vereinigte Mitteldeutsche Braunkohlenwerke AG (MIBRAG) and a glyoxal-modified sodium silicate grout developed by the French chemical company Socit Franaise Hoechst.

The idea to use montan wax to construct containment barriers came about because of the serious environmental problems in the Bitterfeld area. There are several industrial waste disposal pits where leachates are entering the local aquifer contaminating the drinking water supply. Site remediation may take many years to accomplish because of the volume of material involved and the large associated costs. In an attempt to address this problem, MIBRAG began testing the feasibility of creating containment barriers by injecting a montan wax emulsion in unconsolidated soil adjacent to contaminated areas using permeation grouting methods (3). The early results were promising and additional government funded demonstration projects are underway.

MIBRAG was privatized in 1994 and is now owned by a consortium of US and British companies. The wax production company, Romanta GmbH, was not part of the privatization scheme and is currently under the control of the ministry of finance. The environmental division from MIBRAG responsible for developing the subsurface barrier technology was transferred to Romanta.

Montan wax is a fossil plant wax with properties similar to natural plant waxes such as those found in carnauba palms. It is a hard, high-melting point, non-toxic material used in carbon inks, emulsions, polishes and lubricants. It is actually a

mixture of waxes, resins and asphaltene-like materials composed of C-24 to C-32 carbon chain esters of long chained acids and alcohols. Montan wax grout is a suspension-type grout consisting of a stable emulsion of montan wax, water, and an emulsifier. The emulsion is formed by injecting a stream of melted (95-98oC) wax and emulsifier into water at the same temperature and then rapidly cooling the mixture to 50oC. Both ionic and non-ionic emulsions can be formed. To break the emulsion in a field application, a small amount of bentonite clay is added (approximately 2-5% by weight) just prior to injection, which binds the emulsifier and allows the emulsion to break. The resulting wax/bentonite mixture forms a highly viscous material that can significantly reduce the permeability of the soil matrix.

The modified sodium silicate grout developed by Socit Franaise Hoechst has been extensively tested in the laboratory at the Hoechst facilities in Paris and at the Technical University of Clausthal (Germany) but has not previously been field tested as a barrier material. However, the material (sold under the trade name of Klebogel) has been successfully used (particularly in near-surface tunnel construction) in Europe and the U.S. as a soil strengthener in unconsolidated soils.

The sodium silicate grout consists of four components: an alkaline liquid consisting of silicon dioxide, sodium oxide, and water; an acidic liquid consisting of glyoxal and additives; an aqueous suspension of non-agglomerated silica particles in an alkaline medium; and water. The grout is prepared by successively the components to water. The setting time and viscosity of the grout can be controlled by altering the proportions of the four components.

The DOE Office of Technology Development began evaluating montan wax and the glyoxal-modified sodium silicate grouts in 1992. The project is funded under the In Situ Remediation Integrated Program and performed by Golder Federal Services, Inc. and Sandia National Laboratories. Initial activities involved laboratory tests to assess the effectiveness of the grouts in reducing hydraulic conductivity in unconsolidated soil and their resistance to common types of industrial and hazardous wastes. Field testing was initiated in 1993 at Sandia National Laboratories in Albuquerque, New Mexico. Single-borehole injection tests using permeation grouting techniques were performed to evaluate the penetration characteristics of the grouts over a range of soil types and conditions. Additional field tests are currently underway at a site in Richland, Washington near the DOE Hanford Reservation.

This paper describes the results of the laboratory and field tests and provides a preliminary assessment of the potential applications of the materials within the DOE complex. Emphasis is given to describing the most recent field test at the Richland, WA site.

RATIONALE

A recent estimate of the quantity of buried waste within the DOE complex is 2.1 million m3. Various types of buried waste are present at the Hanford Site, the Savannah River Site, the Idaho National Engineering Laboratory, Oak Ridge National Laboratory, and the Rocky Flats Plant. Approximately half of all DOE buried waste was disposed of before 1970. The regulations at that time allowed the commingling of different wastes (i.e., transuranic, low-level radioactive, and hazardous waste). As a result, much of the waste is believed to contain both hazardous and radioactive components. The waste is present in different forms and conditions, including trenches, pits, storage pads, and other structures.

Containment systems, or subsurface barriers, are a potentially important component of DOE's goal of bringing all DOE facilities and sites into compliance with applicable laws and regulations for protecting public health and the environment. While DOE's goal is to eventually clean up all contaminated sites, the remediation of some sites may be delayed due to technical difficulties, cost, or schedule considerations. In these situations, it may be necessary to use temporary measures to mitigate the impact of the contamination until a permanent solution can be implemented. In some cases, subsurface barriers may be applicable for containing known or potential contamination long enough to allow planning, scheduling, and budgeting for remediation or the development of new technologies. Chemical and particulate grouts are of interest since they provide a means of creating an in situ containment system that will retard further movement of contaminants until they can be safety and effectively removed.

In addition to containment, barrier technology can also be used as a backup measure for cleanup activities and as a means to enhance the performance of other remediation technologies. Subsurface barrier materials can also be used to improve

the efficiency of other remediation technologies, such as pump-and-treat, chemical fixation, and vapor extraction systems, by directing and focusing the containment plume to the treatment area and/or limiting the volume of the soil or ground water to be treated.

TEST RESULTS

The grout study is being performed in three phases of increasing scale and scope: laboratory tests; single-borehole field tests; and multiple-borehole field tests. Permeation grouting is the only delivery method used in the study, i.e., the performance of the grouts when delivered via jet grouting, soil mixing, etc., was not evaluated. Permeation grouting was the recommended method for injecting both types of grout at the initiation of the project. Recently, jet grouting has replaced permeation grouting as the recommended method in Germany.

The first two phases have been completed and the results are summarized below. The multiple-borehole tests are ongoing. Grout injection was completed in late 1994 and the post-injection characterization activities are scheduled for early 1995. The completed activities are described and preliminary observations concerning grout performance are provided.

Laboratory Tests

The two objectives of the laboratory tests were to 1) characterize the extent the grouts reduce the hydraulic conductivity of unconsolidated soils and 2) assess the compatibility of the grout materials with typical waste forms.

Soil samples from three sites within the DOE complex were used for the tests: the Buried Waste Integrated Demonstration (BWID) site at the Idaho National Engineering Laboratory; the Mixed Waste Landfill Integrated Demonstration (MWLID) site at Sandia National Laboratories, and the Underground Storage Tank Integrated Demonstration (USTID) at Hanford. The results are documented in a report entitled Laboratory Test Results for Evaluating the Performance of Montan Wax and Sodium Silicate Barrier Materials in Unconsolidated Soils (4).

Standard methods and procedures were used to determine the grain-size distribution, bulk density, permeability, and mineral composition. A preliminary estimate of the groutability of the soils was made based on general groutability criteria (Table I), grout viscosity (Table II), and the laboratory permeability measurements (Table III). The groutability of the BWID soil was expected to be poor, based on the low hydraulic conductivity (10-5 cm/sec) and high percentage of silt (>>90%). The USTID and MWLID soils, with their relatively high conductivity (10-2 cm/sec and 10-4 cm/sec, respectively) and lower percentage of fines (approximately 5% for both soils) were expected to be groutable with the lower-viscosity montan wax formulations and the sodium silicate formulation.

TABLE I

TABLE II

The results of the hydraulic conductivity tests for the ungrouted and grouted samples are shown in Table III. Grout was injected into the base of 15 cm diameter soil samples and allowed to cure for 48 hours before testing. The montan wax and sodium silicate grouts were able to permeate the USTID and MWLID samples, however, neither grout could permeate the silty clay samples from the BWID site because of the high percentage of fine material in the samples. Both grouts significantly reduced the hydraulic conductivity of the USTID and MWLID soils. The source of the variability in the post-injection conductivity has not been identified. One possible explanation is sample disturbance; the grouted samples were transferred from a stiff-walled triaxial cell to a flexible-walled cell and the handling of the sample may have resulted in creation of pathways (e.g., along the grouted-soil/sample-sleeve interface). Another reason may be incomplete permeation of the samples by the grout. This could result from soil heterogeneity and/or small differences during sample preparation. Additional laboratory tests are underway with revised procedures that minimize the potential for sample disturbance and sample variability.

TABLE III

The compatibility of the grout with the waste forms and leachates that may be encountered is an obvious criterion for a subsurface barrier. A test program to evaluate the resistance of the grouts to chemical degradation was undertaken based on earlier studies by the EPA. Spooner et al. (6) evaluated the compatibility of a number of grout materials with various classes of chemicals. The chemicals used were considered to be representative of a majority of the leachates present in hazardous

waste disposal sites. The EPA study evaluated the effect of the chemicals on the setting time and durability of the grouts.
The chemical compatibility testing in this study was limited to an evaluation of grout durability. It consisted of immersing grout/sand cylinders in chemical baths and observing any changes. The chemicals are listed in Table IV along with the functional groups they represent. The samples were constructed by mixing approximately 20% grout with 80 % sand and compacting the mixture to a final density of 2.8 g/cm3. The specimens were immersed in the chemical solutions for 28 days. Visual observations were made on a daily basis to detect physical changes (e.g., swelling, shrinkage, softening, color changes).
TABLE IV
With the exception of the specimens in the ethylene glycol, the samples containing montan wax grout were largely unaffected by the chemical groups. Most showed some deterioration or softening, primarily around the periphery limited to the first few mm of the surface. The results suggest that, with the exception of the glycols, the durability of the montan wax would should be favorable for most industrial wastes. The sodium silicate specimen also performed well in all but the sodium hydroxide solution. The latter results are consistent with the EPA study.
Single Borehole Tests
Single-borehole field tests were performed at the MWLID site at the Sandia National Laboratories. The objective of the tests was to obtain information on the injectibility of the montan wax and sodium silicate grouts over a range of soil conditions. Eight boreholes were used to perform the tests, four boreholes for each of the grout types. Grout injection was attempted at multiple-levels within each of the boreholes in order to intersect a range of soils types from low permeable silts to highly permeable coarse sands and gravel deposits.
The movement of the grout within the soil was monitored using downhole geophysical tools including resistivity and neutron moisture probes. Typical response curves for each tool are presented in Fig. 1. In this example, resistivity decreases and moisture increases are associated with the arrival of the grout front at the geophysical monitoring borehole. The penetration distance of the grouts from the injection borehole typically ranged between 1 m and 2 m for the sodium silicate and montan wax grouts.
Fig. 1.
The tests results suggest that both the sodium silicate and montan wax grouts can be injected in unconsolidated soils with conventional grout permeation methods, provided the soils have sufficiently high conductivity. Montan wax grout permeation was limited to the higher-conductivity zones. Attempts to inject the grout into lower-conductivity soils resulted in unacceptable injection pressures, or the grout traveling up the borehole annulus to the surface. The minimum hydraulic conductivity for permeation grouting appeared to be approximately 5 x 10-4 cm/sec for both the sodium silicate grout and the montan wax grout. The results of the single-borehole tests are presented in a report entitled Preliminary Results of Permeation Tests Using Montan Wax and Sodium Silicate Barrier Materials in Unconsolidated Soils at the Mixed Waste Landfill Integrated Demonstration Site (MWLID) (7).
Multiple Borehole Tests
The objective of the multiple-borehole tests was to demonstrate the feasibility of constructing subsurface barriers under field conditions that met certain performance goals. The primary goal was to produce a continuous horizontal barrier with a hydraulic conductivity of 10-6 cm/sec. The appropriateness of this level of performance for actual barrier applications will depend on the site-specific needs and the thickness of the emplaced barrier. For some applications, such as the use of grouts to redirect groundwater flow, the contrast in the hydraulic conductivities between the grouted and ungrouted soils is more important than absolute values, and hydraulic conductivities of 10-5 cm/s or higher may be adequate.
The test site is located in an area adjacent to the municipal landfill for the City of Richland, Washington. The site is located approximately 2 km northwest of Richland. Reconnaissance tests suggested the site provided a zone of sufficient permeability and lateral continuity for the implementation of the multiple-borehole injection tests. The test site was located outside and immediately north of the permitted portion of the landfill. The on-site evaluation and preliminary soil sampling was completed in April 1994. Laterally continuous, poorly-graded medium grained sands were located at a depth of 5 m below the ground surface. The medium

sands were underlain by finer sand and silt. Initial laboratory grout permeation tests were performed on soil samples from the site and hydraulic conductivities of the medium sand and fine sand were measured at 9.1 x10-2 cm/sec and 7.7 x 10-3 cm/sec, respectively.

Geologic data for the vicinity of the test site were available from the Richland Landfill Development and Closure Plan (8). Near-surface soils at the site are part of the Pleistocene Hanford formation, and are sand-dominated. The total thickness of the Hanford formation at the site is approximately 15 m. The depth of the water table is approximately 38 m below surface.

Figure 2 illustrates the general layout of the test site. The site is divided into three areas, one for preliminary injection tests to test the formulation and grout injection equipment, and two other areas for the montan wax and sodium silicate multiple-borehole tests. The location of the injection boreholes and geophysical monitoring holes are shown in Fig. 3. Drilling was initiated with four boreholes at the outer margins of the test site drilled to a maximum depth of 30 ft. Split spoon samplers were used to complete a detailed stratigraphic log of the site. The results were used to select the target depth for grout injection. The interval between 3 and 4 m depth was selected. Following drilling, the boreholes were converted to geophysical monitoring access holes. Four grout injection boreholes and sixteen geophysical monitoring holes were completed in each of the two test locations. The grout injection boreholes were completed with 3.8 cm diameter tube manchettes with grout ports at 0.3 m intervals between the 3 and 4 m target depth.

Fig. 2.

Fig. 3.

Baseline logging of the geophysical monitoring boreholes was performed prior to the grout injection tests, to provide additional site characterization data and baseline information for comparison with later geophysical responses caused by the introduction of grout. Downhole logging tools used included neutron moisture and conductivity tools.

The tube manchette method was used to inject the grout into the soil formation (Fig. 4). The grout delivery system consists of PVC casing with grout ports drilled at selected locations and covered with an expandable rubber sleeve. The casing is placed in a borehole and the annular region backfilled with a cement-bentonite slurry to provide intimate contact between the casing and the surrounding soil. A grout tube with a pair of straddle packers is used to isolate ports at the desired depth. The grout injection pressure is increased until the annular grout surrounding the injection ports is fractured providing a pathway for the grout to enter the soil. The grout injection pressure is reduced to prevent further fracturing of the soil.

Fig. 4.

In general, grout was injected in one borehole at a time to facilitate the monitoring of grout movement. Grout "take", pressures, and flow rates were continuously monitored. Following initial fracturing of the annular grout, injection pressures were limited to approximately 2.1 kPa per m of depth to avoid uplift and fracturing of the soil formation. This pressure is approximately equal to the lithostatic pressure and is commonly used by the grouting industry for permeation grouting.

Geophysical monitoring in adjacent boreholes was performed frequently using the neutron and conductivity tools during the early stages of injection. Grout was injected until: 1) geophysical data indicated that the target permeation distances (radius of 5 m) and thickness (0.75 m) had been met; 2) 5300 l of grout were injected; 3) the pressure required to inject grout exceeded the limit; or 4) grout appeared at the ground surface. There were 15,000 l of each grout type available for the tests.

Food-grade powdered dyes were added to both the montan wax and sodium silicate grouts to identify the materials during post-injection sampling and excavation. The dye colors were varied during injection. In general, two color changes were made in each borehole to later identify the movement and interaction of the grout fronts. In some cases, injection was performed in more than two phases and additional dye changes made. The initial color was different in each of the boreholes.

Permeation grouting with the montan wax grout was largely unsuccessful. Grout was observed flowing from the surface along or near the borehole annulus. The annular region around the borehole was resealed with additional cement-bentonite slurry and

allowed to cure. This resulted in the grout flowing to the surface along new pathways. Following several additional attempts to prevent grout movement to the surface, additional injection boreholes were drilled. New sleeves were installed with injection ports at 0.3 m intervals to provide additional locations to inject the annular sealing material. The new sleeve design and installations procedure resulted in increased take however the montan wax eventually flowed to the surface. The upward movement of the montan wax grout was observed in all but one of the injection holes. Hydraulic fracturing of the soil is suspected in this case.

The break time of the montan wax emulsion depends on the type and amount of bentonite clay that is added to the emulsion and the amount of mixing that occurs. The presence of natural clays in the soil can also affect its performance. Once the emulsion breaks, the grout viscosity increases dramatically and further movement through the soil is not possible, even in the coarse sands at the site. Prior to the field tests, laboratory tests were performed to select a bentonite clay that would result in a sufficient set time for the grout to move radially 2 - 3 m from the borehole. A sodium bentonite clay was eventually selected, however, the break time (for all the clays tested) was highly dependent on the amount of agitation the mixture received. Under field conditions, it was very difficult to control the mixing time and amount of agitation. As a result, the break time was likely to have varied and the lack of permeation may be the result of the grout setting soon after it was injected into the soil. Similar problems have been encountered in the German montan wax grout program where they have largely abandoned permeation grouting for other application methods, e.g., jet grouting.

In general, the sodium silicate grout was successfully injected using the tube manchette method although the direction and distribution are uncertain. There were two occurrences where grout flowed to the surface but the situation was corrected by resealing the borehole annulus. Grout was detected by the geophysical measurements at distances of over 1 m from the point of injection, however, there was no indication that the target distance of 3 m was achieved. This suggests that there was a significant vertical component to grout movement unless the geophysical tools failed to detect the presence of the grout. Visual inspection of the grouted region during excavation of the area in the Spring of 1995 is expected to resolve these issues and provide additional information on the behavior of the materials. In addition to visual inspection during site excavation, a number of hydrologic tests will be performed. They include small-scale permeability tests with Guelph permeameters and, in areas where the grouted soil area is of sufficient extent, large-scale permeability tests will be performed using a sealed double-ring infiltrometers.

CONCLUSIONS AND RECOMMENDATIONS

The field tests at the site in Richland, Washington are ongoing, however, the initial data suggest the injection of the montan wax grout using permeation grouting was largely unsuccessful for creating a laterally extensive barrier. Data from the sodium silicate grout test are more promising but the geophysical monitoring devices did not detect the presence of the grout in several locations. Post-test characterization to evaluate barrier continuity, size, and hydraulic performance, is planned for Spring 1995.

The inability to construct a subsurface barrier with montan wax grout using permeation grout does not necessarily preclude the grout from further consideration for applications within the DOE complex. As mentioned earlier, the German program has largely abandoned permeation grouting for jet grouting. Jet grouting is already being used in the US to construct groundwater barriers, for example, 20% of the jet grouting applications performed by Hayward Baker are for water control including horizontal bottom sealing of powerplants (9). Soil fracturing techniques, such as the FRACTOOL technology developed by Golder Associates, and the soil-saw developed by Haliburton are other promising application methods that should be considered and evaluated as delivery systems for montan wax grouts. Montan wax should also be considered as an additive for other types of grouts such as cement grouts. A grout consisting of cement, sand, fly ash and montan wax has successfully been used in a metro tunnel in the United Kingdom to control groundwater inflow.

The application of geophysical methods to monitor grout emplacement has been largely successful and the tools should be considered when designing similar types of studies. The methods are attractive due to their relatively low costs and the ability to provide real-time information on the location of the grout front.

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43-4

DIRECT CONVERSION OF PLUTONIUM METAL, SCRAP, RESIDUE, AND TRANSURANIC WASTE TO GLASSDIRECT CONVERSION OF PLUTONIUM METAL, SCRAP, RESIDUE, AND TRANSURANIC WASTE TO GLASS

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ABSTRACT
A method for the direct conversion of metals, ceramics, organics, and amorphous solids to borosilicate glass has been invented. The process is called the Glass Material Oxidation and Dissolution System (GMODS). Traditional glass-making processes can convert only oxide materials to glass. However, many wastes contain complex mixtures of metals, ceramics, organics, and amorphous solids. Conversion of such mixtures to oxides followed by their conversion to glass is often impractical. GMODS may create a practical method to convert such mixtures to glass.
Plutonium-containing materials (PCMs) exist in many forms, including metals, ceramics, organics, amorphous solids, and mixtures thereof. These PCMs vary from plutonium metal to filters made of metal, organic binders, and glass fibers. For storage and/or disposal of PCMs, it is desirable to convert PCMs to borosilicate glass. Borosilicate glass is the preferred repository waste form for high-level waste (HLW) because of its properties. PCMs converted to a transuranic borosilicate homogeneous glass would easily pass all waste acceptance and storage criteria. Conversion of PCMs to a glass would also simplify safeguards by conversion of heterogeneous PCMs to homogeneous glass.
Thermodynamic calculations and proof-of-principle experiments on the GMODS process with cerium (plutonium surrogate), uranium, stainless steel, aluminum, Zircaloy-2, and carbon were successfully conducted. Initial analysis has identified potential flowsheets and equipment. Major unknowns remain, but the preliminary data suggests that GMODS may be a major new treatment option for PCMs.

INTRODUCTION

The Glass Material Oxidation and Dissolution System (GMODS) is a recently invented process (1,2) for the direct, single-step conversion of plutonium-containing materials (PCMs) to glass. PCMs include excess weapons components, scrap, residue, and transuranic waste. GMODS can convert mixtures of metals, ceramics, organics, and amorphous solids directly to glass.

The end of the cold war has resulted in excess plutonium metal, scrap, residue, and the prospect of significant future (as well as significant current) quantities of transuranic wastes from operations and decommissioning of facilities. Some of the high-plutonium-assay PCMs may be stored should the plutonium be needed for national security or energy. Other PCMs will be declared waste. Consequently, multiple incentives exist to convert PCMs to glass:

Storage and disposal risks are minimized by converting PCMs to a glass that is stable, insoluble, and nonburnable.

Criticality risks are minimized by converting PCMs to a glass containing neutron poisons so nuclear criticality is avoided by chemical composition, not by geometry.

Nuclear materials accountability and safeguards are simplified and made more reliable when plutonium is stored as homogeneous glass in numbered containers. (Measurements of plutonium quantities in heterogeneous PCMs is difficult.)

STRAWMAN CRITERIA

If PCMs are to be converted to glass for storage or disposal, a set of glass performance criteria are required to define goals. For disposal, there are two sets of criteria: 1) waste management criteria and 2) safeguards criteria. Waste management criteria depend upon the disposal site (3,4). There are four sets of waste management criteria: 1) mechanical (package dimensions, handling mechanisms, etc.), 2) chemical (chemical waste form behavior, hydrogen gas generation, etc.), 3) allowable heat generation rates, and 4) nuclear criticality. The safeguards criteria are designed to prevent unauthorized diversion of plutonium. Storage criteria are similar except long term performance (>100 years) is not required.

Mechanical, thermal, and chemical waste form performance criteria have been developed for the acceptance of HLW glass (3). These criteria provide a conservative basis for defining appropriate criteria for a PCM borosilicate glass.

Nuclear criticality and safeguards criteria depend upon plutonium concentrations; thus, the requirements for the final glass are also dependent on plutonium concentrations. Plutonium concentrations at which different criteria come into effect are not currently fully defined and will depend upon both technical and institutional factors.

The nuclear criticality waste acceptance criteria (WACs) are designed to prevent nuclear criticality over geological time. If the plutonium concentration in the waste is sufficiently low, nuclear criticality is not a concern. If the plutonium concentrations are high, criticality criteria may be satisfied by the addition of depleted uranium to the final glass to ensure long-term avoidance of criticality (5,6).

In a repository, plutonium is generally immobile, but nuclear criticality is a concern over geological time frames because plutonium decays to highly enriched 235U. Selective dissolution and precipitation of uranium in groundwater over geological time creates the potential of concentrating uranium and initiating criticality events if the original uranium is highly enriched. Geological evidence shows that past naturally occurring nuclear reactors have been created by these mechanisms when the uranium enrichment levels exceeded several weight percent 235U. In contrast, no evidence exists of naturally occurring reactors at lower enrichment levels. The former scenario can strongly limit the quantities of plutonium per waste package (5) and the spacing of waste packages in a geological repository.

Plutonium in light-water reactor (LWR) spent nuclear fuel (SNF) is allowed in geological repositories, although 1 % of the heavy metal in a SNF assembly is plutonium and the total fissile content of the SNF heavy metal is typically 1.4 %. The basis for plutonium acceptance is that it is mixed with depleted 238U. This prevents nuclear criticality. When the plutonium decays to 235U, it is born into a world of 238U and is isotopically diluted by that 238U. For PCMs with higher concentrations of plutonium, the repository licensing and acceptance philosophy for SNF may be used as a basis for repository acceptance of highly loaded plutonium waste forms by incorporation of depleted uranium into the waste form. A waste with a ratio of about 1 part plutonium to 50 to 100 parts uranium will have nuclear

characteristics similar to
LWR SNF
The second plutonium concentration related set of criteria are for safeguards. Three categories of safeguards may exist (these are currently being defined) for PCMs.
1. The International Atomic Energy Agency (IAEA) has developed a set of criteria (7) defining when safeguards are not required for PCMs. If the plutonium concentration is sufficiently low, potential diversion of the waste for the recovery of plutonium is not credible.
2. If the plutonium concentration in the waste is higher, some form of standard safeguards are required.
3. If the concentration of plutonium is sufficiently high, arms control concerns may require mixing the PCMs with radioactive materials to make plutonium recovery difficult. The United States Academy of Sciences (8) has proposed that if excess plutonium is to be disposed, it should meet a spent fuel standard; i.e., excess plutonium should be no more accessible than plutonium in SNF. This set of recommendations is part of the ongoing discussions concerning international control of nuclear weapons and nuclear weapons materials. Several methods to achieve this goal have been defined. The United States Department of Energy is currently determining if or at what concentrations of plutonium in a waste should the spent fuel standard be applied (9). This implies that at some concentration of plutonium in the PCMs, some quantity of radioactivity may be required to be added to the waste to make plutonium difficult to recover.
The criticality criteria suggest that as the plutonium concentration of the wastes increases, at some point low-enriched uranium may need to be added to produce an acceptable waste form. The safeguards criteria suggest that as the plutonium concentration of the wastes increases, at some point sources of radioactivity may need to be added when producing a waste form. There are two sources of depleted uranium in the United States: depleted uranium at the uranium enrichment plants and low-enriched SNF. In this second category, there exist 2100 t of Hanford-N SNF with an initial enrichment of 0.95-1.08 % which is badly degraded and may require treatment before it can be accepted for geological disposal. If criticality criteria are the only concern, adding the clean depleted uranium to PCMs during waste processing may be appropriate. If nuclear criticality criteria and spent fuel standard safeguards criteria are concerns, adding Hanford-N SNF or an equivalent to the PCMs during processing may be necessary. The Hanford-N SNF provides both the depleted uranium and the radiation source.
GMODS PROCESS DESCRIPTION
The basic concept of GMODS is based on adding unprocessed PCMs (with plutonium in multiple chemical forms), glass frit, and a sacrificial oxide directly to a glass melter. GMODS may be operated in a batch mode (Fig. 1) with all waste-processing operations preformed in sequence in a heated pot. The starting point for conversion of PCMs to high-quality glass is the melter filled with molten lead-borate glass in a ratio of 2 mol of lead oxide or more per mole of boron oxide.
Fig. 1. GMODS batch processing of PCMs to borosilicate glass.
In the first step, PCMs are fed to a molten lead-borate glass bath. Conventional molten glasses dissolve wastes made of oxidesbut not wastes containing metals or organics. Molten lead-borate glass has unique properties. Metals in the feed chemically react with the lead oxide in molten lead-borate glass, thus forming metal oxides and lead metal. These metal oxides dissolve into the lead-borate glass. The boron oxide in the molten glass assists the rapid dissolution of metal oxides, including any protective oxide layers on the metals into the molten glass. Organics are oxidized to carbon oxides and steam with lead metal as the by-product. Example chemical reactions in the molten lead borate glass are:

$$Pu + 2PbO \rightarrow PuO_2 + 2Pb \quad (1)$$

$$Zr + 2PbO \rightarrow ZrO_2 + 2\,Pb \quad (2)$$

$$2Al + 3PbO \rightarrow Al_2O_3 + 3Pb \quad (3)$$

$$3Fe + 4PbO \rightarrow Fe_3O_4 + 4Pb \quad (4)$$

$$C + 2PbO \rightarrow CO_2 + 2Pb \quad . \quad (5)$$

The lead metal from these chemical reactions separates from the molten glass and sinks to the bottom of the melter. There it forms a separate layer of molten lead. Dense reactive metals (e.g., plutonium metal when added to the molten glass) may partly react and dissolve into the glass and may partly sink through the glass and descend into the lead. Plutonium melts at a low temperature (638C). If the molten

plutonium metal enters the molten lead, it dissolves into the lead. The dissolved plutonium is then oxidized at the lead-glass interface and extracted in oxide form back into the glass.
The lead-borate glass, with its dissolved PCMs, is not a high-quality glass. To create a high quality glass, two more steps are required. Glass additivesprimarily silicon oxide (SiO2)are fed to the melter where they mix with the molten lead-borate glass and then produce a lead borosilicate glass. This is in the same family of glasses as is crystal used in elegant, expensive glassware.
After formation of lead borosilicate glass, excess lead oxide in the glass is removed by adding carbon to the glass to create a borosilicate glass that is chemically similar to HLW borosilicate glass and which has similar desirable properties in terms of waste management. The resulting chemical reactions yield gaseous carbon oxides exiting the melter and lead metal separating from the molten glass and sinking to the bottom of the melter. Lead oxide may be removed from the dissolution glass for multiple reasons: a) better final waste glass, b) reduction of the volume of the waste glass, and c) avoidance of the costs to provide added sacrificial lead oxide. Studies (10) indicate that many glasses that contain lead are not hazardous. Some lead can remain in the glass to improve glass properties. The product borosilicate glass is poured from the melter into cylinders or converted into marbles.
After removing the product glass from the melter, boron oxide and lead oxide are added to the melter to replace materials that have gone into the final glass product. The lead at the bottom of the melter is reoxidized back to lead oxide by the addition of oxygen.
2Pb + O2 2PbO .
(6)
The resultant lead oxide mixes with the residual glass to create a new batch of molten lead-borate dissolver glass. The production cycle is then repeated as new wastes are added to the melter.
PRODUCT DESCRIPTION
The product composition will depend upon the feed material. Typical glass compositions for high plutonium/uranium glasses are shown in Table I. These numbers report experimental work at Savannah River Laboratories (11) in development of actinide glasses.
TABLE I
The glass-product form can be logs or marbles. Logs provide a well-defined product form that fills a product canister. Marbles allow inspection of the glass product and recycle of off-specification materials. Marbles can also be converted into glass logs with small, simple glass melters or by filling the containers with marbles and heating the marbles to consolidate them at temperatures below normal molten-glass conditions. This option also allows separate production of marbles from different PCMs and combining the marbles and then melting them together within the final waste package.
GMODS EQUIPMENT DESCRIPTION
Cold-wall, induction-heated glass melters appear to be the preferred choice of vitrification equipment for most GMODS applications. In a cold-wall melter, a thin layer of frozen glass separates the molten glass from the melter wall. This separation prevents glass corrosion of the furnace lining. The higher temperatures (1400C) possible with such melters allow the use of glass formulations with higher waste loadings and/or the production of more durable glasses. The high waste loading per unit volume minimizes cost.
Cold-wall, induction-heated melters are used industrially to melt titanium and super alloys. They are being modified for radioactive waste processing in France (12) and Russia (13). The French have an operating melter with a throughput of 50 kg/h and are developing a 200-kg/h melter. The French melters have also been used to melt Zircaloy hulls and hardware into ingots. The melters have the capacity for bottom gas injection and may be emptied by tilting the furnace or using bottom freeze valves. Similar melters for hazardous flyash are being developed with capacities to 800 kg/h.
THERMODYNAMICS
The thermochemical properties of the PbO-B2O3 system are the basis for GMODS. The abilities to oxidize metals and to dissolve metal oxides are directly related to the thermochemical activity of PbO and B2O3. This system was assessed by Slough and

Jones (14). These authors used the Gibbs energy values reported by Kapoor and Frohberg (15). In this work, we also have used the data of the latter.
The processing of metals in GMODS involves both oxidation and dissolution into a PbO-B2O3 glass. The overall free energy change for the process will include the free energy change for the oxidation, plus the free energy change for the dissolution. The final dissolution product will be at least a ternary system involving PbO-B2O3-metal oxide. No data are available about such systems on which to base a calculation on the distribution of species for the entire process. It may be possible to perform calculations on ternary and higher systems using thermodynamic solution models. Until that work is done, the best guide to treatability of a material by GMODS is the oxidation of the metal by PbO.
The best way to compare the relative ease of oxidizing a given metal is to use oxygen potentials. Oxygen potential is described in units of energy and is defined as RT ln PO2. Figure 2 shows plots of oxygen potentials for forming a number of metal oxides. This figure shows the important feature that any metal oxide that is higher than another can oxidize the metal in a lower position. Here, we are concerned about the ability of PbO to oxidize metals. Figure 2 shows that any metal, Sn, Fe, Zn, Cr, U, Pu, Al, etc., that lies below the PbO:aPbO = 0.1 line will be oxidized when the activity of PbO is 0.1. In this figure, only Cu2O lies above the PbO lines. However, Zhou et al. (16) have shown that copper in the +2 oxidation state, as in CuO, is stable in 2PbO.B2O3 glass. The stability of oxidized copper in this glass must be due to the combination of oxidation and dissolution and not oxidation alone. Other metals, such as Ag, Au, Pt, and Pd do not form oxides that are stable at high temperatures, and they are not included here. Noble metals dissolve into the lead. The oxygen potentials for the formation of ZrO2 are similar to those of Al2O3.
Fig. 2. Oxygen potential diagram.
Some metals, such as U and Pu, can form more than one oxide. The highest oxides are thermodynamically stable as far as oxidation is concerned. However, the overall oxidation and dissolution may result in a lower oxide in the borate glass. Tests of UO2 oxidation and dissolution resulted in a black product which is most likely to be U3O8 rather than UO3, which is red.
Carbon may be introduced into GMODS either as waste to be treated or to reduce PbO to Pb with the production of carbon dioxide (Eq. 5). The equilibrium constant for this reaction is ~3 108 at 1000C; therefore, even when the PbO activity is quite low, carbon will be oxidized.
EXPERIMENTAL RESULTS
A series of small-scale experiments have been initiated a) to demonstrate process chemistry and b) to provide process and equipment design data. The data include solubility and density, viscosity, and chemical reaction rate measurements.
The solubility of various oxides in lead-borate glasses with molar compositions from PbO:B2O3 to 4PbO:B2O3 was investigated. High solubility is required for high waste loadings in the glass and rapid dissolution of oxide layers on metals in the feed. For uranium oxide, the highest solubilities in the lead borate dissolution glass were with an initial glass molar composition of 2PbO:B2O3. This particular lead-borate glass, when heated to 1020C, dissolves in excess of 20 wt % UO2 + 20 wt % ZrO2. Other experiments showed similar solubility for Al2O3.
Preliminary chemical reaction tests demonstrated the conversion of uranium, Zircaloy-2, aluminum, cerium, and stainless steel to dissolved oxides in the dissolution glass with production of lead metal. Experiments have also demonstrated oxidation of carbon with production of metallic lead and gaseous carbon oxide release. Dissolution rates are being investigated systematically for candidate metals and materials such as carbon. Quantitative data are incomplete; however, qualitative observations for Zircaloy, aluminum, stainless steel, and carbon confirm that satisfactory dissolution rates in the 2:1 PbO:B2O3 glass can be expected. Of these, only carbon has been measured. Under static conditions, the rate of mass loss from a cylindrical carbon rod was 0.1 g/(cm2*h) at 1000C.
Measurements of the mass of the lead regulus that results from the oxidation-dissolution reaction indicate similar rates for aluminum and Zircaloy. The metal cerium, which is chemically similar to plutonium, also produced a similar amount of lead.
Glass viscosities and densities were measured over a range of temperatures and viscosities in a platinum viscometer. Low viscosities are required to ensure rapid

mixing and reaction of PCMs with the lead-borate glass. The results (Table II) indicate that the lead borate system is a low-softening-point glass with low viscosities in the range of 700 to 900C. This implies that the actual dissolution-oxidation step can be conducted below 1000C.

TABLE II

Added experiments are planned to measure glass physical properties (viscosity, density, thermal conductivity, and heat capacity) and chemical reaction rates for various materials as a function of glass compositions and temperature. Major experimental work remains to demonstrate the process.

EXAMPLE FLOWSHEET

Based on theory and experiment, a set of flowsheets was developed for the conversion of PCM to glass. One flowsheet was for the direct conversion of plutonium metal to glass using a single vessel. Table III summarizes this particular flowsheet. The final glass is a plutonium storage glass with 10% plutonium. This plutonium concentration ensures criticality control by composition rather than geometry.

TABLE III

CONCLUSIONS

GMODS is a new process for converting PCMs in metal, ceramic, amorphous solids, and organic forms directly to a borosilicate glass. Preliminary experiments have been initiated using cerium as a substitute for plutonium. Tests have also been initiated with other materials that may be in PCMs. The preliminary experimental data indicate the potential for GMODS to be a workable process for PCMs. However, significant process development work is required and major work remains to demonstrate engineering feasibility.

If GMODS can be developed into a successful process, it creates the option to convert most PCMs into a homogeneous, transuranic glass. This has multiple benefits. Safeguards would be simplified by conversion of heterogeneous materials to homogeneous glass. Nuclear criticality safety for higher concentrations of plutonium in wastes would be assured by chemical composition. Waste management would be simplified by creation of a single, high-quality waste form acceptable in any repository.

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REJUVENATION OF SPENT FUEL IN ACCELERATOR DRIVEN SYSTEMS AS A WAY OF WASTE REDUCTION

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ABSTRACT

The recent rebirth of interest in nuclear transmutations is mainly due to a significant progress in the accelerator technology now permitting us to achieve sufficient target currents and thus neutron yields. These neutrons of adequate spectrum shape (e.g. a hard one) can be used for transmutation of nuclear materials (actinides - fertile and/or fissile ones, fissile products or spent nuclear fuel). In view of supposed safety and economical advantages the idea of a symbiotic nuclear energy system, consisted of an Accelerator-driven fuel Regenerator (AR) and a number of LWRs serviced by it, is the subject of this study. In the AR the regeneration of spent fuel with simultaneous fissioning of plutonium, both from ca. 3 LWRs can be carried out. At higher inventories, the abatement of respective minor actinides (e.g. 237Np, Am) and a partial abatement of fission products, can be performed too. In addition to the above, it should be emphasized that a reduction of environmental impact of nuclear energy with nuclear technology is of particular importance for the deployment of the former in view of its possible positive psychological effect. Summarizing, the advantage of the present concept of accelerator-driven transmutation of waste based upon then operating reactors, lies in efficient slowing-down of the rapid world wide build-up of nuclear waste at reasonable investment expenses.

INTRODUCTION

One of the causes conditioning the spread of negative social attitudes towards nuclear energy seems to lie in the unavoidable and deep discrepancy between the time scale of exhaustion of traditional energy sources and human (political, economic, social and personal) time horizons. The latter are simply inadequately short as compared with the former - distant and vague though unavoidable. Simultaneously, the apparently less important aspects of investigations, like environmental and social impacts of developing technologies only very recently were gaining their well deserved attention. A symptom of change is the search for new solutions of the very sensitive item coupled with nuclear energy - the nuclear waste. The recent rebirth of interest in nuclear transmutations (1) is mainly due to a significant progress in the accelerator technology now permitting us to achieve sufficient target currents and thus neutron yields. One proton of energy 1 GeV impinging upon a heavy (e.g. Pb) target (Fig. 1) can produce about 30 fast neutrons. These neutrons of adequate spectrum shape (e.g. a hard one) can be used for irradiation of nuclear materials (actinides - fertile and/or fissile ones, fissile products or spent nuclear fuel) (2). Thus being, the fertile nuclides can be transmuted into fissile ones, then these when being fissioned are producing energy while the radioactive fission products are transmuted into stable ones. The feasible in this way re-use of spent fuel reduces the environmental impact of the nuclear fuel cycle.

Fig. 1.

Therefore, the present work is an attempt to demonstrate the concept of

Accelerator-driven Regenerator (AR) as an environmentally benign solution, which should help in the deployment of nuclear energy.

GENERAL CONSIDERATIONS

The same premises that are substantiating so deeply future-oriented research, as that one in the field of spallation neutrons applications (the seek of new energy sources to meet future energy needs and abatement of nuclear waste) oblige scientists to more and more integral approach in these analyses. First, one must recognize the need of efficiency of undertakings and at the same time be conscious that just the economical incentives are motivating most human actions at social level. Thus in order to cope with the tasks that have been facing us, the studies should first of all, though - by necessity - roughly, deal with economical aspects of emerging nuclear energy systems.

Economical Questions

Unfortunately, in fact till now, there are no unquestionable, truly reliable and complete economical evaluations of existing energy sources, not to mention the emerging ones. That pessimistic opinion results from the observation that till now the energy prices do not include the whole of externalities which in reality are (or will be) paid by the society (in the future even by the whole mankind at global scale). The proposal of carbon tax is an example of first still not successful attempt to internalize such costs. Simultaneously, the respective dispute and very distant perspectives of its implementing indicate the gap between the existing and strict economical calculus. Thus, the results of economic studies of accelerator driven systems are so uncertain that one can doubt whether detailed analyses are now useful. The main sources of errors lay in the uncertainties of:

capital cost of long term investments, i.e. the interest and discount rates that determine this main component of costs,

price of uranium, determined by its demand which depends on the growth rate of nuclear power industry influenced, in turn, by public attitudes, various political factors etc.,

back-end costs somehow determined by the readiness of society to spent means for reduction of its true or imaginary threats,

electric power demand depending on the global (and also local) economic development and demographic growth.

The above factors can vary in a very wide range over the decades that separate nowadays from the time of commercialization of the use of accelerators for the energy purposes. Even though, since a simplified analysis is better than nothing, the approach presented below deserves an attention.

The spent nuclear fuel is to be rejuvenated, what should bring higher gains (equal to the fuel share in the electricity costs) thanks to the two benefits (2):

1. Direct reutilization of unburned fissile nuclides contained in the fuel;
2. Avoidance of one fuel cycle.

Both features merit some emphasizing, in view of their positive environmental effect. When discussing the AR economics one should not forget that the contribution of uranium raw material to the cost of electricity from nuclear power does not exceed recently 3-5%. Therefore, alone the production of fuel in accelerator systems cannot bring an income, which might compensate immense investment costs of spallators, well exceeding the ones of LWRs. Thus one can conclude, that accelerator systems must be confined neither to sole fuel nor to mere energy production, but transmute the nuclear waste too (3).

The essence of AR economics (4) is illustrated in the self-explanatory Fig.2.

Fig. 2.

Thus one can see that symbiosis of a certain minimum number S (support ratio) of LWRs with the AR is essential for the latter to be economic. In other words - the AR must resolve the problems of fuel cycle of several (say, 3) LWRs. The dependence of S on some physical parameters of the AR is discussed below in the item Reactor Physics Issues.

Environmental Aspects

In our concept of the Accelerator-driven Regenerator being used for rejuvenating of the burned-up fuel it is performing a triple function:

first - the "augean" one of a symbiont co-operating with S fission reactors for transmuting (fissioning) their actinides and (partly) fission products i.e. de-poisoning and deactivating the nuclear reactor waste;

second - of a device producing net energy from fissions;

third - of a breeder of fissile material in the fuel supplied for regeneration.
In the present discussion the most important postulate is the consideration of environmental impact of the whole energy system the AR is part of. This signifies that confining oneself to the analyzing of impacts of the sole device without taking into account those stemming from other inseparable elements of the system is a serious error that can falsify the assessment of the whole concept.
Therefore, in the case of accelerator-driven fuel regeneration one should compare the change in the impacts when to the system of S (symbiotic) fission reactors is added alternatively: -one AR or - one LWR reactor.
Then let us accept the following assumptions:
the LWR's environmental, health etc. impact is taken as unit;
the impact of the LWR fuel cycle alone amounts to 90% of the total impact, whereas the impact of an AR is equal of that of LWR;
the fuel regeneration reduces the impact of respective fuel cycle by 1/3.
The above assumptions are quite conservative. In spite of doubling the energy production per fuel cycle (i.e. per amount of the mined uranium, fabricated and reprocessed fuel), decay and transmutations of radioactive nuclides in the irradiated fuel, a reduction of the impact not to 1/2 but merely to 2/3 has been assumed. Also in spite of lack of danger of the uncontrolled supercriticality the impact of the AR has been assumed as equal to that of LWR's. In such circumstances the change in the overall impact of the whole energy system which resulted from the above discussed alternative solutions is presented for two representative values of support ratio S in the TABLE I
As it can be seen from the Table I the implementation of an Accelerator-driven Regenerator as a fuel rejuvenator can be environmentally benign.
Reactor Physics Issues
The connection between the design premises and physical properties of the transmutation system should be discussed first. One of the most important parameters of the system is the conversion ratio Cr, in general defined as a number of bred fissile nuclei per one net destroyed fissile nucleus in the system. It is to be calculated from the formula:
Eq. (1)
where:
symbolizes integration over all neutron energies and volume of the system,
F neutron flux,
cfe sum macroscopic capture cross section of all fertile nuclides in the system,
afi sum macroscopic absorption cross section of all fissile nuclides,
As it is shown below, this is just the factor directly determining the number of LWRs served by a given transmuting system. As far as assemblies of high fission contribution are concerned the connection between S and parameters of such AR system and symbiotic LWRs of equal power, is (4):
Eq. (2)
where CrR regards LWR, and usually varies within 0.6-0.67.
The formula (2) is illustrated in Fig. 3.
Fig. 3.
One can see that in order to serve, say, at least 3 LWRs by the AR, Cr should amount to 2-3. In the efforts to have an optimum AR, distinct for instance, by efficient transmutations of Transuranics (TU), one must never forget that since only fissioning is recognized as a definitive way of getting rid of TUs, the rate of transmutation per power unit is absolutely limited.
The process, thus inseparable from energy release, has a well fixed intensity (per energy unit) and thus its whole rate is directly determined by the size i.e. the power of the device.
In this view, the optimization efforts should rather consider other factors than the rate of transmutation i.e. first of all those facilitating the fulfillment of safety requirements and the optimum use of the accelerator. Among the former the most important postulate seems keeping the neutron multiplication factor keff below a certain value recognized as a safe one. In turn, the constancy of over the time of transmutation is desirable in order to have all the time the optimum load of accelerator. This state keff const can be approximately assured if the amount of fissile inventory in the system is kept unchanged, i.e. provided Cr 1.
But, as one can see it in Fig. 3, an AR of Cr = 1 can support no symbiotic LWRs at all.

Therefore, some other solution must be found.
Namely, the circumstances when Cr of the system as a whole remains equal to 1, but at the same time the value of S will be well above 1. This apparently contradictory requirement can be achieved by a diversification of the media filling the assembly. Generally, in hard spectra, a high fissile breeding rate can be expected in the spent fuel because of a relatively low mean enrichment in there. Instead, if into the system a certain additional fissile material is added, its increased burnup rate will compensate the breeding in the regenerated fuel, thus making the Cr of the system drop to 1. This might be done with Pu recovered from the burned-up fuel of symbiotic LWRs (after the second cycle, when no more regeneration is possible, because of the radiation damage in the fuel). The higher concentration of fissile material in the outer zones can simultaneously assure the desirable flattening of the power density distribution in the assembly.
Obviously a more diligent calculations might provide us with results of still higher degree of accuracy and reliability, yet these obtained at this initial phase of working out the present concept are presented below. An exact evaluation of Pu + minor actinides (MA) balance in the whole symbiotic system is foreseen in the nearest future.

CALCULATIONS AND RESULTS

Neutronics

The complexity of neutron transport phenomena makes impossible an exact description of physical reality in the respective calculations. Thus quite significant simplifications are inevitable. The problem lies in the right selection of them or in the avoidance of those which threaten with inadmissible errors in given circumstances.
The phase space of parameters of Accelerator-Driven Regenerator has been investigated in search for areas of Cr 1 at simultaneous safe k values, arbitrarily assumed not to exceed 0.95. The neutron transport calculations were performed on the model of cylindrical target/blanket system (Fig. 4) driven by a proton accelerator.
Fig. 4.
The neutron yield due to spallation has been assumed to be 30 neutrons (of fission spectrum) per 1GeV proton. In order to have flatter power distribution the concentration of fissile material has been increasing in the zone of fuel rejuvenation. The space devoted to the spent fuel + additional Pu and MA, has varied from 4% vol. in inner layers up to 28% in the outer ones. The remaining volume up to 90% has been filled with liquid lead as a cooling medium while construction materials occupied the 10% everywhere.
The amount of additional Pu and MA corresponded to its tenfold content in the fuel to be rejuvenated. The external neutron reflector of lead has been assumed.
For both actinides and fission products, the transmutation calculations for U/Pu fuels were performed with modified code BISON. This program uses ANISN code for neutron transport calculations and Bateman's method for time-dependent evolution of nuclide chains. The pertinent nuclear data 42-group set has been based on JENDL-3 library except for fission products evaluated from ENDF/B-6. In view of pretty high neutron multiplication in the system (keff ca. 0.94) the neutron transport in high energy region above 15 MeV has been neglected. The performance of the system is presented in the Table II.
TABLE II
As it has been announced above, in addition to a suitable enrichment of rejuvenated fuel, one should examine the changes in fissile products concentration, especially those radioactive ones and reactor poisons. Some radiotoxic or poisoning fission products can be transmuted into non-harmful isotopes, due to neutron capture during irradiation. Unfortunately, the lack of yields of fission products for all significant fissile nuclides (eg. 241Pu and 238U), in many cases deprive our preliminary assessments of any reliability. Thus only some less doubtful are presented in the Table III. where the initial concentrations in the spent fuel and the respective changes in result of the rejuvenation are given. They were based on the rejuvenation time assuming sufficient build up of fissile material. The 10 year cooling time of LWR's fuel before irradiation was also assumed.
TABLE III
The concentration of the most important stable poison 149Sm (and of some other ones too) was assessed to be decreased, although changes for specific nuclides differ considerably and some nuclides even are building-up. Nevertheless, it should be

reminded that FP activity, increases less than linearly with the burnup, and the rejuvenation finally results in doubled energy production from the fuel. Thus, the radioactivity per released energy is decreased, non mentioning the waste volume, just substantiating the proposed concept of reduction of the nuclear waste.

CONCLUSIONS

The proposed methodology gives rise to some remarks. In addition to the necessary completing the yields of fission products, it should be noticed in connection with the neutronic calculations, that an exact consideration of resonance self-shielding should be introduced. A correction for these effects, though a moderate one for hard spectrum media, that prevents us from an overestimation of neutron capture in 238U i.e. of fissile breeding, will result in a certain reduction in fissile materials concentration. Similarly, the component of high energy neutrons (>15MeV) should be considered, esp. in view of their contribution to the radiation damage. Also the reliability of the whole system, thus its load factor can prove lesser than the one of LWR and the price of rejuvenated fuel lower than expected. Nevertheless, since the structure and composition of the assembly are far from being optimized, a margin for further improvements in its performance still remains.

The influence of the pertinent geometry, requiring a 3-D code coupled to the burnup one and, at minor scale, also the distribution of newly bred fissile material within a rejuvenated fuel rod also demand evaluation. A desirable, more homogeneous power distribution during its future exploitation should be expected. This positive effect can be achieved due to higher rate of neutron captures in the outer layers of the rod due the captures (mostly) occurring in self-shielded resonances.

In conjunction with the economy of the AR, one should admit that it cannot be quite completely evaluated, at least until a recognized methodology of internalization of environmental costs of energy conversions is elaborated.

As for transmutations of fissile products and of actinides some deeper insight might be desirable too. In this case an objective function should be selected (e.g. the biological hazard potential) which would consider the risk associated with each particular nuclide i.e. among other its (radio) toxicity, probability of spreading out, entering the food chain etc. On such basis the final sum effect of the rejuvenation process could be reliably estimated.

All the above ways to raise the reliability of evaluations remain in the close plans of the authors.

On the basis of presented approach and results the concept of fissile fuel rejuvenation, is revitalized. Certain simplifying assumptions and a preliminary character of calculations notwithstanding, the possibility is demonstrated of a reduction of general hazard from nuclear energy system, with the help of an accelerator at reasonable investment effort. This advantage is due to a symbiosis of the AR with then just operating reactors. Therefore, the concept of Accelerator-driven fuel Regenerator, proven as environmentally benign, should help in deployment of the nuclear energy. This, in turn, should find a right reflection in the support of farther pertinent research. Besides, it should be emphasized that a reduction of environmental impact of nuclear energy with nuclear technology is of particular importance in view of its possible positive psychological effect. Summarizing, the advantage of the proposed concept lies in efficient slowing-down of the rapid world wide build-up of nuclear waste at reasonable investment.

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QUANTUM-CEP APPLICATIONS TO MIXED AND RADIOACTIVE WASTES

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ABSTRACT

Quantum-CEP (Q-CEP) is an innovative and patented technology developed by Molten Metal Technology, Inc. (MMT) that can process radioactive and mixed waste streams to decontaminate and recover resources of commercial value while achieving significant volume reduction and radionuclide stabilization. MMT is currently in the process of commercially deploying Q-CEP in the government and commercial radioactive markets. To this end, MMT has formed two relationships with market leaders: Martin Marietta Corporation and the Scientific Ecology Group (SEG), a Westinghouse subsidiary. M4 Environmental, L.P., a 50/50 partnership between MMT and Martin Marietta Corporation, was formed to deploy Q-CEP applications in the U.S. Department of Energy (DOE) and U.S. Department of Defense (DoD). A Q-CEP demonstration facility is currently under design and construction for the processing of DOE mixed waste in a privatized mode. MMT has also joined forces with SEG to deploy Q-CEP systems for processing spent ion exchange resins from nuclear power plants worldwide. A U.S. facility is currently under construction and will be operational by the end of 1995.

This paper outlines a wide range of CEP/Q-CEP's processing results including RCRA wastes, spent ion exchange resins and contaminated scrap metal. These results highlight the technology's unique features in processing radioactive and mixed waste:

1. Waste Minimization Performance: RCRA wastes are recycled into commercially valuable products: gases, ferroalloys and ceramics. Processing of mixed waste streams leads to complete destruction of hazardous contaminants and the potential formation of one or more decontaminated product streams such as fuel gas and/or metal alloys.
2. Environmental Performance: Processing of RCRA wastes leads to environmental performance surpassing regulatory standards. Destruction Removal Efficiencies (DREs) 99.9999% have been consistently demonstrated. NOX and SOX were not detected to 3 ppm (analytically limited). Dioxins/furans were not detected to the targeted regulatory limit of 0.1 ng/Nm3 toxicity equivalent (TEQ). All ceramic products passed Toxicity Characteristic Leaching Procedure (TCLP) tests.
3. Controlled Radionuclide Partitioning: Processing of contaminated scrap metal results in metal decontamination >99%, with current experimental demonstrations limited by the lower detection limit (LDL) of the analytical equipment. Commercial processing of spent ion exchange resins will result in gas decontamination factors >107.
4. Final Form Stabilization: Multiple high temperature glass compositions have been developed for Q-CEP applications. SEM/EDX results showed radionuclide incorporation into final form. Radionuclide incorporation into stable high chlorine content glass structures outperforming high-level nuclear waste reference glasses in PCT and TCLP durability testing has been proven.
5. Volume Reduction: Volume reduction from processing contaminated scrap metal is specific to individual scenarios and dependent on the initial form of the waste material. Processing of spent ion exchange resins results in volume reductions >30:1. Preliminary calculations indicate volume reductions >100:1 may be achievable while processing DOE mixed low level waste streams (MLLW), such as organic sludges and combustible debris.

MMT operates a variety of experimental and demonstration-scale CEP systems and has demonstrated long-term operability and reliability to commercial and government customers. Over 40 customer-established performance-based criteria covering environmental performance, product quality, and operability were surpassed with third-party oversight and validation by a major German chemical manufacturer. An on-stream factor >90% was achieved.

Q-CEP RESEARCH, DEMONSTRATION AND COMMERCIAL FACILITIES

Catalytic Extraction Process (CEP) is an innovative and proprietary technology that allows waste material of a wide range of composition (organic, organometallic, and

inorganic) to be recycled into products of commercial value, such as synthesis gas, hydrogen chloride, metal alloys, and specialty ceramics. At the core of CEP is a liquid metal bath which acts as a catalyst and solvent in the dissociation of the feed and the synthesis of products. Upon introduction to the bath, feeds dissociate into their constituent elements. Addition of co-reactants enables reformation and partitioning of desired products. The catalytic and solvation effects of CEP, together with the thermodynamically controlled reaction pathways, allow the technology to achieve superior environmental and waste minimization performance. Q-CEP is the application of the CEP technology to process radioactive and mixed waste streams. Targeted radionuclide partitioning leads to decontamination and recycling of a large portion of the waste components to commercial products (e.g., synthesis gas) with high resultant volume reduction through radionuclide concentration into a stable condensed phase.

MMT has engaged in technical development of Q-CEP and is currently in the process of commercially deploying the technology to the government and the commercial radioactive market. MMT has formed two significant partnerships to enable quick deployment of Q-CEP: 1) A Martin Marietta Corporation/MMT limited partnership -- M4 Environmental, L.P. --is designing and constructing a demonstration unit for application to DOE wastes; 2) MMT and the Scientific Ecology Group (SEG), a Westinghouse subsidiary, are jointly constructing a Q-CEP facility to process radioactively contaminated spent ion exchange resins.

A variety of experimental and demonstration-scale CEP and Q-CEP units have been operated for both research and customer needs. MMT's Recycling-Research & Demonstration Facility in Fall River, MA, has been the primary site for technology development and customer demonstrations. Fall River houses four bench-scale units, five pilot-scale units, seven physical models and a commercial demonstration prototype unit. The facility is fully permitted by the Commonwealth of Massachusetts for recycling demonstrations using hazardous and non-hazardous materials as CEP feeds. At Fall River, MMT has carried out a range of demonstrations on RCRA materials and has received recycling certifications from the Massachusetts Department of Environmental Protection for the processing of RCRA-listed and characteristic feeds. MMT has also carried out radioactive surrogate and common isotope testing at the Fall River facility. Testing on low-level radioactive materials, and particularly spent ion exchange resins, has taken place at MMT's demonstration facility in Oak Ridge, TN. MMT is expanding its "hot" testing capabilities by deploying with M4 demonstration-scale Q-CEP units to process low-level waste and mixed low level waste.

Technical development and commercialization of Q-CEP was partly sponsored by the DOE. From 1993 to date, MMT has been awarded a total of $19 million under a Planned Research and Development Announcement (PRDA) (1). Contract sponsored activities focused on demonstration of Q-CEP chemistry, recovery potential and final form stability, development of a conceptual engineering design, and application of Q-CEP to contaminated scrap metal and mixed low level waste. Under the agreement with SEG, MMT is constructing at Oak Ridge a Q-CEP commercial facility for processing radioactively contaminated ion exchange resins from nuclear power plants. The Q-CEP system, designed for high volume reductions (>30:1), is targeted for operation in 1995.

An extensive demonstration run was successfully completed for the recycling of biosolids waste provided to MMT by a major German chemical manufacturer. The run demonstrated the long-term operability of the CEP system with respect to on-stream factor and steady-state operation. This prototype campaign also demonstrated CEP's utility for on-line recycling of a commercial-grade synthesis gas. A number of operational performance criteria for on-stream factors, steady-state operation, and product quality and consistency were established jointly by the customer and MMT. Based upon on-site third party customer validation, more than 40 operational criteria were met and surpassed (Table I). Steady-state operation was successfully demonstrated and product quality and consistency met the needs for on-line recycling. An on-stream factor >90% was achieved (2).

TABLE I

Q-CEP PROCESSING OF RCRA WASTE

CEP has been demonstrated on a range of RCRA listed wastes as well as characteristic and RCRA-like surrogate material (Table II) including high-molecular weight aromatics, chlorinated organics, organically-bound nitrogen species (isocyanates),

plastics, and organometallics. Table II data are from waste processing using the Fall River demonstration unit. The technology's environmental performance has been demonstrated during actual waste processing to meet and surpass current and proposed regulatory standards. Specifically, DREs >99.9999% were achieved for principle organic hazardous constituents (POHCs). NOX and SOX were not detected in the product gases to detection limits of 3 ppm. Condensed phase non-leachable products, both ceramic and metal phases, of marketable composition were generated. Dioxins were not detected to the targeted regulatory limit of 0.1 ng 2,3,7,8 TCDD TEQ/Nm3.
Currently the effectiveness of hazardous waste treatment technologies is regulated by the EPA by measuring the concentration of hazardous materials in the after-process wastewater and non-wastewater. All processes have to ensure that organic constituent concentrations in individual wastewater and non-wastewater streams are lower than the regulated limits set by EPA. CEP demonstrations surpass the current limits even when compared with the Best Demonstrated Available Technology (BDAT). CEP was recently approved by EPA (on October 24, 1994) as a non-combustion technical equivalent (BDAT) for eight RCRA-listed isocyanate waste codes (K027) for which incineration had previously been mandated as the commercially available BDAT technology.
The environmental performance of CEP has also been demonstrated on chlorinated waste streams containing some of the most difficult to destroy hazardous constituents (3). The results of demonstration-scale processing of RCRA-listed waste F024 (chlorinated aliphatics) indicated that hazardous organic constituents in the feed were not detected in ceramic, metal and scrubber water, thereby surpassing the BDAT standards for all effluent streams (Table III). Destruction Removal Efficiency (DRE) on multiple organic hazardous constituents exceeded 99.9999%, which surpassed the current regulations mandating DREs >99.99% (Table IV). Trace constituents were not detected to the targeted regulatory limit of 0.1 ng/Nm3 TEQ (Table V).
TABLE II
TABLE III
TABLE IV
TABLE V
Q-CEP APPLICATION TO SPENT ION EXCHANGE RESINS
Nuclear power plants in the United States generate over 200,000 ft3 of spent ion exchange resin. Ion exchange resins are widely used in nuclear power plants for water purification. Regeneration of the contaminated ion exchange resin is not economically feasible, due to the large volume of contaminated liquids generated and other market barriers. The resin waste stream represents the largest waste stream in both volume and activity for nuclear power plants. Disposal has been hampered by the recent closure of the Barnwell, S.C. radioactive waste disposal facility to many of the nuclear power plants. A technology providing volume reduction and stabilization offers significant economic and safety benefits to nuclear power plants.
Surrogate Testing: Prior to hot testing, MMT performed demonstration runs on non-radioactive resin at the Fall River demonstration unit to study CEP efficiency in organic resin conversion. Gas-phase detection of the resin (polystyrene-divinylbenzene polymer) is limited by sampling protocol; hence, a modified method was used to calculate DRE. Using styrene as a decomposition product, the calculated DRE was >99.9999% (below the LDL of 1.2 ppb). As a confirmation, the destruction efficiency for the resin was also calculated by comparing the number of moles of benzene rings in the solid feed with the number of moles of benzene rings in the gas phase. This analysis also resulted in a DRE >99.9999%.
Bench-Scale Testing: A range of "hot" tests using commercial spent ion exchange resin have been performed at SEG's facility in Oak Ridge using bench-scale Q-CEP systems. A representative total Curie balance is shown in Table VI for the gamma-emitting radionuclides in the resin supplied by SEG. A radionuclide closure of 105 7% was measured.
TABLE VI
The bench-scale testing has demonstrated decontamination factors leaving the reactor system 104 (limited by analytical). The commercial facility is designed for decontamination factors >107. Table VII shows representative decontamination factors achieved in the bench-scale tests exiting the reactor system but excluding the gas handling train (i.e., HEPA filter). Decontamination factor is defined as DF = Activity In / Activity Out.
TABLE VII

Commercial facility: The Q-CEP-IER commercial facility, jointly operated by SEG and MMT, will be capable of processing up to 400 High Integrity Containers (HICs) or 80,000 ft3 of IER per year. The facility will operate two Q-CEP systems in a batch mode, which will prevent commingling of customer waste. A customer batch of one or more HICs will be processed and a stable final form contained in a steel container will be returned to the customer. Volume reduction is determined by operating parameters such as customer batch size, resin composition, and curie content. The design specifies that volume reductions will be at least 30:1 while on-going optimization of operating parameters is expected to lead to higher volume reduction ratios.

Q-CEP APPLICATION TO CONTAMINATED SCRAP METAL

Previously published studies have demonstrated the ability to partition radioactive components, such as uranium and plutonium, from the metal phase into a vitreous phase via melt refining (4,5,6,7). Residual concentrations ranging from 0.05 ppm to 2 ppm were achieved using diffusion of oxidizing, vitreous-forming agents to partition the radioactive components. Successful partitioning to less than 10 nCi/g levels (approximately <0.1 ppm) has been demonstrated for uranium- and plutonium-contaminated metals. Q-CEP offers the potential for superior performance, as the techniques involved in melt refining are completely incorporated and enhanced in Q-CEP technology. Specifically, Q-CEP incorporates active radionuclide partitioning through select co-reactant additions (e.g., oxygen) and enhanced mass transfer (e.g., convection), while melt refining is based upon "passive" diffusion-based partitioning. MMT has demonstrated the capabilities of Q-CEP to process contaminated scrap metal under the DOE sponsored development programs.

Bench-/Pilot-scale Testing on Surrogates: Q-CEP processing of radioactive wastes was first demonstrated using surrogates in the Fall River facility. Hafnium and cerium were chosen as the surrogates of radionuclides uranium and plutonium based on similarities in oxidation free energy, density, and method of incorporation into glass (8). Theoretically, these elements will be oxidized and captured in the ceramic phase. Bench-scale tests were carried out varying experimental parameters including metal charge (nickel and iron), glass composition (a matrix of calcium alumino/borosilicates), temperature, and bath carbon level. Initial contamination levels ranged from 350 ppm to 900 ppm. Table VIII shows representative decontamination factors from the processed metal samples. The results indicate successful removal of hafnium and cerium from the "contaminated" metal phase, with demonstrated decontamination factors >99% (LDL limited). Overall mass balance closure >97% has been demonstrated.

TABLE VIII

Bench-Scale Testing of Contaminated Scrap Metal: MMT continued Q-CEP experiments in the bench-scale units at Oak Ridge using depleted uranium (U) and uranium/cerium (Ce) mixtures. The experimental program included optimization of the ceramic phase capture of these species. The primary experimental design parameters evaluated were glass composition, and initial form/loading of the contaminant (Table IX). Samples were taken both from the metal surface and throughout the interior and analyzed for uranium and cerium content. The ceramic phase, reactor crucible, headspace insulation and gas handling train were also subjected to radiochemical analyses. Greater than 95% curie closure was achieved.

TABLE IX

Scanning electron microscopy (SEM) and Energy Dispersive X-ray Diffraction (EDX) were performed on selected glass samples to investigate the nature of incorporation into the ceramic phase. Results indicated that both U and Ce were partitioned to the targeted phase (glass). Figure 1 shows representative results demonstrating the partitioning and capture of uranium in the vitreous phase. These EDX results were originated from a Q-CEP experiment in which the initial iron metal charge contained 1000 ppm uranium. This EDX clearly shows that U is present in the glass phase along with the calcium, aluminum and silicon (note that peaks have been labeled). Uranium decontamination is further quantified by Neutron activation analysis (NAA) studies. In an iron-aluminosilicate-uranium system, decontamination of uranium was found to be 99.77% (LDL limited), which demonstrated consistent findings with surrogate tests. Decontamination measurement was limited by the analytical lower detection limit, which was on the order of 0.1 ppm (dependent upon the contaminant and metal). Thermodynamic calculations indicate that the residual contaminant concentration in the metal should be several orders of magnitude less than the detection limit of

NAA. To verify the stability of the glass formed, PCT and TCLP tests were performed on a series of synthesized glasses that contained various amounts of RCRA metals and uranium. All glasses were shown to be significantly more durable (i.e., less leachable) than high-level nuclear waste reference glasses.
Fig. 1
Q-CEP APPLICATION TO LOW LEVEL MIXED WASTE
Volume Reduction: Volume reduction calculations have been carried out based on Q-CEP of DOE MLLW streams. Major waste streams include soils, inorganic sludges, organic sludges, combustible debris, and inorganic debris. The ceramic phase, which captures the radionuclide contaminants, is expected to be the only solid phase which may require disposal. In the absence of a de minimis standard, decontaminated metal has restricted reuse applications within the DOE complex. Given an achievable de minimis standard, free release into the commercial sector would also be allowed. The synthesis gas product, which captures essentially all carbon and hydrogen in the feed, may be used within the processing system for energy generation. The volume reduction varies from 2:1 to greater than 100:1, depending on the initial form and composition of the feed material.
TABLE X
A recent DOE sponsored evaluation of Integrated Thermal Treatment Systems (9) for application to DOE mixed wastes showed that a Q-CEP system would have the lowest total life cycle costs (TLCC) compared to eighteen alternative systems including incineration, vitrification, and plasma treatment. Q-CEP offers a one-step process that simultaneously achieves organics conversion, metal recovery, and radionuclide immobilization. Q-CEP's broad feed acceptability and robust nature dramatically reduces the feed pretreatment and characterization requirements and the need for post-processing stabilization. This streamlined system design lowers capital and operating costs, and enhances overall system volume reductions.
CONCLUSIONS
Currently no single commercially available technology dominates the huge market of low level and mixed radioactive waste generated and stored in the United States and abroad. Q-CEP can be applied to a wide range of low level radioactive and mixed waste feeds, effectively partitioning and stabilizing radionuclides while destroying hazardous constituents. Furthermore, Q-CEP may be able to reduce radioactive levels in recovered materials to below background levels, which may facilitate establishment of de minimis standards. Q-CEP applications to DOE and commercial radioactive waste problems has been demonstrated and outlined in this paper. Current commercialization programs will deploy the technology to both the government and commercial sectors. A Q-CEP facility accepting spent ion exchange resin from nuclear power plants is targeted for operation in 1995. A mixed waste demonstration facility to process DOE wastes in a privatized mode is also under design and construction for 1995.
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A METHOD TO DEAL WITH BIOSPHERE TRANSFERS TO ACHIEVE DOSE CALCULATION IN HLW SAFETY AND PERFORMANCE ASSESSMENT

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ABSTRACT
Dose limits are usually provided by safety authorities to perform radioactive waste disposal safety assessments. It therefore involves a radioactive transfer modelling through the biosphere to provide dose calculation. This paper described the methodology used to identify and to characterize the biospheres consistent with long-lived radionuclide repository and the dose conversion factors associated. Present national data, reconstruction of past climates and present analogue biospheres were used for biosphere determination. Temperate and tundra biospheres were first studied.
Then, bibliographic study and statistical analysis allowed us to select transfer parameter values and their associated uncertainties for both biospheres. Comparison experiments and chemical analogue determination seemed to be a good opportunity to balance data lacks.
The biosphere model used to calculate the dose conversion factor in a temperate biosphere is shortly presented and its validation is in progress by international comparison. The possible ways of integrating the climate-dependent biospheres in a global safety calculation are then discussed.

INTRODUCTION

The french National Agency for Radioactive Waste Management (ANDRA) is studying several concepts of long-lived radioactive waste repositories in deep geological formation. French rules use indeed dose limits and dose is one of the criteria used to conduct performance assessment and to discriminate between different deep repository systems. The usual way to calculate dose is to determine a biosphere conversion factor which allows to calculate an effective dose (in Sv/year) received by man from an activity (in Bq/year) released in a river or an aquifer after migration through geologic barriers. The biosphere conversion factor comes from biosphere modelling in which biosphere transfers through main food-chains are determined. This calculation needs a lot of radionuclide-, scenario- or site-dependent parameters. The main work consists in selecting values for each transfer parameter and giving associated uncertainties for the biosphere. In a long-term perspective consistent with a HLW repository, various biospheres must be studied which are representative of climatic changes at least over a 160,000 year period (1). In this context, a food-chain associated to each of the six major biospheres has to be characterized and the transfer parameter values determined. Then the plausible ecosystems will be further integrated in a global model. Considering all these assumptions, we have developed a global method which is explained below: first our study consists in defining temperate and other biosphere scenarios, relevant parameters are then determined and the biosphere modelling allows to calculate a conversion factor.

SITES AND CRITICAL GROUPS

Context

We have to characterize the different transfers along several food chains and to specify the critical group associated. The Basic Safety Rule (2) (R.F.S.III.2.f.) edicted by the French Nuclear Installation Safety Directorate and the ICRP 46 (3)

recommend to take into account a critical group "representative of those individuals in the population expected to receive the highest dose equivalent, and relatively homogeneous with respect to the location, habits and metabolic characteristics that affect the dose received [...] The habits and characteristics of the group should be based upon present knowledge using cautious, but reasonable, assumptions."
With respect to these two hypothesis -present technological level and pessimistic assumptions-we chose an agricultural critical group and obviously an agricultural scenario. Main basic food chains which transfer radionuclide, to the critical group are indeed typical and traditional agricultural food chains (see Fig. 1).
Fig. 1.
Farmers are therefore supposed to receive the highest effective dose because of their high self sufficiency level with respect to the global population but also because of their numerous outdoor activities (ploughing, harvesting,...), which involve possible external exposure and inhalation of soil particles. The critical group is able to drink water directly from the release compartment. This assumption appears plausible in the case of an aquifer because, in France, some farmers use a well in their land from which they irrigate fields and water animals and from which they might draw water to drink.
We tried then to adjust site-dependent parameters to the chosen sites such as climatology (obviously just for temperate biosphere), pedology and hydrography. Relevant to the Basic Safety Rule, we had to consider hypothesis "homogeneous with respect to the location" so we had to study first local ecosystem and social behavior. We then adapted these data in order to be pessimistic to answer safety recommendations.

REAL DATA FROM A TEMPERATE BIOSPHERE

Level of human dietary intake have been determined from a national survey (4) and the mean data regarding farmer population were chosen in order to be consistent with the critical group described above. Mean animal feed consistent with a mean production were calculated from national data.
Agronomic parameters were reviewed with local conditions and habits.
Climatic parameters representing statistical means measured by the French Meteorology Institute were chosen as nearest as possible to the defined site.

Methodology Used for Other Biospheres

The main difference between temperate biosphere modelling and long-term biosphere modelling relies on the level of site-specific characterization such as edaphic and hydrologic features which are impossible to predict. In this context, we decided to use present analogues of past french climatic states to build the plausible food chains and the relevant critical group.
The methodology developed to determine and describe the plausible future biospheres likely to exist in France was the purpose of a previous paper. We first used past data to characterize the plausible vegetations over a glacial-interglacial cycle. Present analogues then help us to describe the food chain and the specific critical group. The tundra biosphere study was based, for example, on the reindeer breeders of Lapland (1960's).
It is also possible to characterize agricultural practices which are used in temperate environments and which may be adapted to colder biospheres. We can imagine, for example, cultures in greenhouses. New parameter values will be adapted to this specific situation. These other ways are now under study and will be compared with the other approach, i.e. the traditionally living critical group. This part of the work and the previous one are the subject of exchange between experts.

RADIONUCLIDE- AND SITE-DEPENDENT PARAMETERS

We had then to select the values of radionuclide-dependent parameters (distribution coefficients and concentration factors) and the associated uncertainties. We chose to use both bibliographical study and experiments.
Twenty five radioelements (H, C, Cl, Co, Ni, Se, Sr, Zr, Nb, Tc, Pd, Cd, Sn, I, Cs, Sm, Eu, Ra, Th, Pa, U, Np, Pu, Am and Cm) corresponding to about fifty five radionuclides for high-level radioactive waste disposals were chosen to be studied.

First Step

Thus, a complete study for temperate value selection began. An equilibrium between the different biosphere compartments was assumed and constant concentration factors were considered.
The method consisted in scanning international bibliography and selecting values

from articles and reviews. Experiments adapted to our temperate conditions were chosen. Then, for each radioelement and each parameter, a statistical analysis was conducted to advise the mean and variation interval corresponding to the best fit of the distribution. Values distribution were examined and plotted with a statistical code named Statgraphics Plus (5). The best distribution fit (lognormal or loguniform) was then selected taking into account the highest Kolmogorov-Smirnov test signification level. Kolmogorov-Smirnov test calculates maximum distance between the sampled distribution function and the fitted distribution function. This test was selected because it tests all the values and not only the mean and is valid whatever chosen fit model. The Chi-square goodness-of-fit test is not appropriate if the expected number of observations in the different classes of values is small (smaller than 5).

The geometric mean and the fit for the parameter values were therefore determined. A lot of data for temperate biosphere and for several radioelements (iodine, caesium, radium, selenium, uranium ...) were obtained but there were data lacks for a few radioelements like protactinium, niobium, zirconium, samarium or europium. There was a similar problem for other biospheres. A lot of data exist concerning caesium and strontium behavior in tundra biosphere, for example, but there is not many information about other radioelements. Another approach needed to be developed.

Data Lacks Balance

A collaboration with french National Institute for Agronomic Research (INRA) was intended to balance some data lacks and to understand radionuclide behavior performing experimental studies on soil-to-plant transfer. Soil is indeed an important compartment because of its connections to many pathways like water-to-soil, soil-to-plant, soil particle human inhalation and human external exposure. Another particularity of the soil is the possible accumulation of some radionuclides with time. We then considered radionuclide accumulation in soils because it appeared to be an important compartment which influences radioelement biodisposibility and so soil-to-plant concentration factors. Parameter sensitivity analysis have indeed shown that this biosphere compartment and river or aquifer flow rate participate more than other factors to the dose.

We therefore focused these experiments on soil-to-plant transfer and used three radioelements very different in their chemistry and soil behavior, i.e. technetium, nickel and niobium. The present available results concern only the technetium. In the biosphere and especially in aerated soils, this element is mostly present under TcO_4^-, a very mobile and poorly sorbed oxidized form. However, in the long-term, local reducing conditions and microbial activity promote the reduction of Tc, and the low mobile form TcO_2 which is easily complexed by organic matter tends to predominate. This work studied more particularly the uptake of Tc99 from freshly contaminated soils to rye-grass and winter wheat. Echevarria et al. (6) showed that the rate of soil contamination had a significant effect on the total amount of Tc99 recovered in rye-grass (the more soil Tc99 concentration is high, the more plants uptake this element) and the total uptake of Tc99 was always the same percentage which validates the concentration factor notion. Soil-to-plant concentration ratios ranged from 10 to 400 depending on the plant species and variety and also on the plant organ. Maximum accumulation of technetium in plants is generally found in leaves. Concerning wheat, high concentration factor values are found for leaves (600 kg in soil/kg in plant) compared to concentration factors for grains (less than 1) which allowed us to precise concentration factor for the edible part.

Plants showed a great potential for the removal of Tc from contaminated soils. Technetium contaminated plants could be hidden in soil after harvest and the bioavailability of this "organized" technetium has been shown to be similar to inorganic technetium.

Other experiments are conducted to compare soil-to-plant transfer under temperate biosphere conditions and under colder conditions with the same radioelements. To simulate colder conditions, we chose two soils developed under mountain climate with high organic matter content and plants are grown in pots which are placed in a growth chamber (16h light, 80% humidity, 15C day and 5C night).

In parallel with experiments, we also try to determine chemical analogues with a bibliographical study to balance data lacks. We pointed out, for example, the similar chemical behavior between palladium and nickel and between the two lanthanides europium and samarium and the actinide americium. We can therefore use nickel data for palladium data lacks and americium data for lanthanides.

The current experiments try to answer the question: are transfer parameters values influenced by climatic conditions (temperature, soil characteristics...)? If they are, we will try to determine the differences with temperate biosphere. If they are not, we will take the temperate values for tundra and other biospheres.
Tundra biosphere parameter values were determined and a comparison with temperate biosphere is conducted. A lot of data have been found for caesium and the same behavior has been shown: caesium migration depends on organic matter content in soils both in temperate and tundra biosphere.
More, we must point out important parameters with a sensitivity analysis and focus our attention on these parameters. The distribution coefficient Kd, for example, seems to be very important. As Kd is influenced by soil characteristics, we will try to incorporate this influence in the model.

BIOSPHERE MODELLING: CONVERSION FACTOR CALCULATION

Radionuclide pathways between each biosphere compartment can be modelled by simple equations of radioactivity transfers characterizing compartment concentrations which allow, with knowledge of level of consumptions and of dose factor values, to calculate ingestion dose. Dose factors are given by ICRP 61 (7).
We have to determine a conversion factor for each biosphere.

Temperate Biosphere

ANDRA has developed, in eighties, a one dimension code named AQUABIOS. This code integrates basic radionuclide transfer equations.
Results of AQUABIOS have been compared with other code calculations at BIOMOVS international exercises until two years. The BIOMOVS II Complementary Studies exercise consists of a set of test cases for the assessment of the radiological consequences of the release of radionuclides to a terrestrial biosphere. The release may occur in a biosphere deep soil, a river or a near surface aquifer. Many other codes involved in the exercise are compartment models employing first order linear kinetics to calculate the distribution of contaminants in the biosphere. The most obvious distinction is the number of "dynamic" compartments in each model. Aquabios code provides basic modelling with a low number of compartments compared to others. Only the deterministic phase of the exercise has been set up. Final results will be available at the end of 1995. Based on the same data and scenario describing a well-defined biosphere, calculations show that Aquabios results do not differ a lot with other codes. But it must be remembered that there is no way in which the participating models can be characterized as correct or incorrect on the basis of these results.

Other Biospheres

As we have explained earlier, the tundra biosphere was the first biosphere developed in the long-term perspective because of all the available data, owing to numerous studies conducted all over the world after the nuclear tests since 1950. The tundra modelling conducted to the determination of 21 transfer parameters associated to a traditional food chain and five major equations, mathematically formalizing the transfer pathways in water, fish, soil, plants and reindeer.
Some data are selected only for 19 radioelements among 25 and we are only able to characterize the whole transfer parameters for caesium.
The method developed for the temperate biosphere will be used to select the transfer parameter values and the associated uncertainties.
The use of chemical analogues and an elaborate knowledge of transfer processes in temperate biosphere will allow to balance data lacks for tundra biosphere.
The other ecosystems should be analyzed this year in order to determine their climatic characteristics and their critical group.

Integration of Successive Biospheres

The possible ways of integrating the different biospheres in a long-term safety calculation is the objective of the study of long-term evolution of the biosphere. In this context, for each typical biosphere, transfer processes must be determined, then relevant parameter values must be selected and finally a dose conversion factor must be calculated. Two approaches are then possible to integrate the biospheres: a statistical or a deterministic approach.
The first one uses regression equations with past and present observations to obtain the most probable extrapolation. Each biosphere is an entity with an occurrence probability at each time in the future. The second one implies a global system modelling (climate-biosphere) and uses equations believed to represent the long-term physical, chemical and biological processes governing the climate system.

CONCLUSION AND PERSPECTIVES

Our global methodology is described in Fig. 2. Future actions will consist in making a sensitivity analysis of the transfer coefficients to the conversion factor in order to determine the most important parameters. We will therefore be able to adjust the parameters to the site local conditions. We also have to analyze different probabilistic events like an accidental intrusion by man and other scenarios as well in a deep aquifer, exploratory drill...

Another important development to be performed will consist in integrating all the selected biospheres in a new model.

Fig. 2.

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44-2

A PRELIMINARY PARAMETRIC PERFORMANCE ASSESSMENT FOR THE DISPOSAL OF ALPHA-CONTAMINATED MIXED LOW-LEVEL WASTE STORED AT THE IDAHO NATIONAL ENGINEERING LABORATORY*

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ABSTRACT

A preliminary parametric performance assessment (PA) has been performed of potential waste disposal systems for alpha-contaminated mixed low-level waste (ALLW) currently stored at the Idaho National Engineering Laboratory. The radionuclide-confinement performance of treated ALLW in various final waste forms, in various disposal locations, and under various assumptions was evaluated. Compliance with performance objectives was assessed for the undisturbed waste scenario and for intrusion scenarios. Some combinations of final waste form, disposal site, and environmental transport assumptions lead to calculated doses that comply with the performance objectives, while others do not. The results will help determine the optimum degree of ALLW immobilization to satisfy the performance objectives while minimizing cost.

INTRODUCTION AND BACKGROUND

A preliminary parametric performance assessment (PA) has been performed by Smith et al. (1) of potential waste disposal systems for alpha-contaminated mixed low-level waste (ALLW) currently stored at the Transuranic Storage Area (TSA) of the Department of Energy's (DOE's) Idaho National Engineering Laboratory (INEL). The ALLW contains 10-100 nCi/g of transuranic (TRU) contamination and was generated primarily from 1970 through 1989.

The containers of ALLW are stored together with the containers of TRU waste. The volume of ALLW is estimated to be 24,413 m3, about 40% of the total waste stored at the TSA (ALLW plus TRU waste). The waste is awaiting retrieval from storage, segregation into TRU waste versus ALLW, treatment, and disposal.

This PA was conducted for the Private Sector Participation Initiative (PSPI), which supports the DOE in evaluating possible private sector treatment of the ALLW. The PSPI is in its early stages. Several feasibility studies on private sector treatment have been completed, but a final waste form (FWF) and a disposal site have not yet been selected.

The purpose of this study was to provide preliminary evaluations of the radionuclide-confinement performance of treated ALLW in various FWFs, in various disposal locations, and under various assumptions. The study examined the effects of several parameters on performance of the overall disposal system (consisting of the FWF, containers, engineered barriers, and the disposal site). The performance was evaluated in terms of projected receptor doses and the other performance objectives of DOE Order 5820.2A, Chapter III.

The principal emphasis was on the performance of potential FWFs for the ALLW. Three categories of FWF (cement, glass, and ceramic) were addressed by evaluating the performance of two limiting FWFs for each category.

Performance at three disposal sites was evaluated to illustrate the effects of site characteristics on the performance of the total disposal system. Several other parameters were investigated, both singly and in some combinations, to determine the effects on the calculated receptor dose. Not every combination of parameters was evaluated, however.

This study is a preliminary parametric PA. It is preliminary in that it is not the formal PA that will be prepared after a disposal site and the FWF(s) have been selected. It is parametric in that several parameters were varied to observe their effect, rather than examining in detail one FWF and one disposal site. At this early time in the PSPI, the evaluation of a wide range of options was needed to provide guidance on the selection of an FWF(s).

The study was performed in two phases. In the first phase, PA computer runs were performed using an initial set of assumptions and parameter values. In the second, the earlier results were evaluated, then additional runs were made using refined assumptions or expanded ranges for some parameter values. The complete set of earlier runs was not repeated using the revised assumptions or parameter values.

This study has several potential uses:

It will help the PSPI determine the optimum degree of ALLW immobilization to satisfy the performance objectives of DOE Order 5820.2A, while minimizing cost

The results indicate the margin of compliance with the performance objectives and how the margin depends on various parameters

The study identified key parameters for which better data are needed.

THE WASTE

Most of the ALLW was generated at the Rocky Flats Plant; the remainder, at several other DOE facilities, including the INEL. The waste consists of a wide variety of solid materials (e.g., sludges, paper, metals) contaminated with small amounts of TRU radionuclides and other actinides. Some of the ALLW, particularly the small fraction that is remote handled (RH), also contains fission products and activated metals.

Most of the ALLW is mixed waste; i.e., it contains chemically hazardous constituents per the Resource Conservation and Recovery Act (RCRA). Treatment will be required under RCRA's Land Disposal Restrictions.

Shipping records for the waste are incomplete, particularly concerning radionuclide information. As a result of this situation and the uncertain split of the waste between ALLW and TRU waste, there is some uncertainty about the inventory of radionuclides in the ALLW. Based on a set of assumptions, a simplified, limiting inventory was developed for use in the PA. For the second phase, a refined inventory for part of the waste was used for some computer runs.

Glass, cement, and ceramic were considered logical candidates for the FWF that will be chosen for the ALLW. To evaluate a wide range of potential variations within these three categories, two limiting FWFs were selected for evaluation from each category. Of each pair, one represents the more favorable characteristics in terms of contaminant immobilization; the other represents the less favorable characteristics. Table I lists the FWFs evaluated.

DISPOSAL SITES

The disposal location for the treated ALLW has not been designated. To illustrate the effects of disposal site characteristics on the performance of the disposal system, three arid Western sites were used. Characteristics that are representative

of the following sites were chosen: the Subsurface Disposal Area at the INEL, Site 14 in the central part of the INEL, and the Radioactive Waste Management Site of the DOE's Nevada Test Site. The sites are referred to herein as Sites A, B, and C, respectively.
Site A was studied because it is adjacent to the TSA. Site B was selected because its underlying sedimentary beds are among the thickest at the INEL. Site C was picked because it has virtually no groundwater flow.
The study of a disposal site in this PA does not imply that the site will be the actual disposal location for the ALLW.
TABLE I
SCENARIOS, TRANSPORT PATHWAYS, AND EXPOSURE ROUTES
To assess the performance of ALLW disposal systems, two scenarios were evaluated. The first is the undisturbed waste scenario. (See Fig. 1). After 100 yr of institutional control, members of the public were assumed to use a drinking water supply well located 100 m downgradient from the edge of the disposal unit. The full range of transport pathways was considered for this scenario. However, based on other studies, it was concluded that a) groundwater transport with a full range of exposure routes and b) atmospheric transport of radon with inhalation exposure would be the dominant contributors to dose. These pathways were evaluated (see Fig. 2).
Fig. 1.
Fig. 2.
The second scenario is the intrusion scenario, which is assumed to occur after the loss of institutional control. The scenario has three parts. (See Figs. 1 and 2.) In the intruder-drilling scenario, an intruder drills a well directly through the source zone (the waste disposal pit). The drilling crew suffers exposure to the radionuclides in the exhumed waste/soil mixture from external radiation and inhalation. In the intruder-excavation scenario, waste/soil mixture is excavated during construction activities. Onsite workers are exposed by external radiation and inhalation. In the intruder-agriculture scenario, some of the waste/soil mixture is exhumed by the drilling or excavation activities just described. The mixture is spread over the ground surface, resulting in exposure by direct radiation and inhalation. In addition, radionuclides are taken up by crops and are ingested with foodstuffs of plant and animal origin.
The treated waste was assumed to be buried in a large pit. The waste stack height was 22.7 ft (6.9 m), corresponding to eight 55-gal drums. The top of the uppermost drums was 5 ft or 9 ft below grade (two cases), with the void being backfilled with soil. An 11-ft-thick, multilayer cover was emplaced over the backfill. The cover was designed in accordance with Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) guidelines to help ensure its integrity up to 500 yr. After that time, it was assumed not to be present. (The belowgrade backfill was assumed to be present for all times.) Thus, the total thickness of geologic material over the waste was 16 ft or 20 ft in the two cases studied.
ANALYSIS OF PERFORMANCE
Conceptual models were developed to describe the dominant physical and chemical processes assumed to occur in the various scenarios and transport pathways.
The conceptual models were implemented by a suite of numerical models. For the intruder scenario, the Generation-II (GENII) package of computer codes was used. For the groundwater transport pathway in the undisturbed waste scenario, the Variably Saturated 2-Dimensional Flow and Transport code (VS2DT) was used to model the flow of water through the source zone. The Disposal Unit Source Term (DUST) code was used to model the release of radionuclides and transport within the source zone. Flow and transport in the unsaturated zone (below the waste disposal pit) and in the saturated zone were evaluated using analytical solutions. Finally, exposure routes were evaluated using the Dose Integrated Over Ten Thousand Years (DITTY) code of the GENII package. For the radon transport, analytical solutions were used.
The performance-related characteristics of the six FWFs, such as leach rates, were estimated using a combination of the limited existing laboratory data and theoretical calculations. Site characteristics were estimated using a combination of questionnaires completed by site representatives and data in published reports.
RESULTS
About 45 runs were made, using the numerical models, to evaluate performance in the undisturbed waste scenario and the intrusion scenarios. The matrix of runs is given in Table II.

The results are also given in Table II and are compared against the performance objectives listed at the top of each column. (The additional performance objectives that are not listed in the table were met in all cases that were evaluated.) The rationale for each run is summarized in Table III.

CONCLUSIONS

The key observations and conclusions from this preliminary parametric PA of the ALLW are presented below. The first group of observations and conclusions relate to the process followed in the study.

The second group relates to the results obtained. The number of cases examined in the study was rather largeabout 45, depending on how one countsbut was still limited. To broaden their usefulness, the following observations and conclusions were extrapolated somewhat to cover the results expected from cases that were not specifically evaluated but are believed to be predictable with a high degree of confidence based on the results of closely related cases.

Process-Related

Completing the preliminary parametric PA at this early time in the PSPI resulted in compiling information, developing simplified models, and performing scoping analyses, all of which are expected to prove useful to the PSPI.

The lack of formal selection of a FWF was compensated for by evaluating the performance of limiting examples of each of the three principal categories of FWF. This approach illustrated the likely range of potential behavior of the FWF.

The lack of formal selection of a disposal site was compensated for by evaluating the performance at three possible sites and varying several key site parameters. This approach illustrated the likely range of potential behavior of the disposal site.

As a check on the conservatism of the assumptions and parameter values used, they were compared against those in five related assessments. The comparison indicated that, on balance, the study was not on either extreme in terms of conservatism or nonconservatism, when the ranges of parameter values used here were considered.

TABLE IIa

TABLE IIb

TABLE IIc

TABLE IIIa

TABLE IIIb

Results-Related

Compliance with the performance objectives depends not only on the FWF and the disposal site, but also on the overall conservatism of the set of assumptions in the PA.

For the undisturbed waste scenario:

- The results depend particularly on the assumed values for the leach rate of the FWF, the infiltration rate, and the Kd values.
- The dominant radionuclide contributing to dose is the Am-241 decay chain.
- At Site C, any FWF would be expected to comply.
- At Site A or B, nearly all FWFs would be expected to comply under nearly all assumptions. Possible exceptions are the PCC and FUETAP FWFs or untreated waste under quite conservative assumptions (more conservative overall than those investigated herein).
- The performance of the untreated waste is nearly equal to that of PCC, suggesting that little benefit is expected in this scenario from cementing the waste. (It is noted, however, that conservative assumptions were used concerning the adsorption of radionuclides in cement.)
- Most of the cases at Site A were evaluated using a comparatively low infiltration rate and taking credit for travel time through the underlying fractured basalt. However, these nonconservatisms were slightly more than compensated for by the conservatism of not taking credit for adsorption of radionuclides in the source zone.

For the intrusion scenario:

- To the accuracy of these analyses, the intrusion scenario results are independent of the location of waste disposal.
- The results depend particularly on the depth of burial (belowgrade), cover longevity, and radionuclide concentration in the FWF.
- The dominant radionuclides contributing to dose are Cs-137 and Am-241 during the early period studied (e.g., 100 yr postclosure) and the Pu-239 and Am-241 decay

chains for the later period (500 yr postclosure).
- For the intruder-drilling scenario, all FWFs and burial depths studied satisfy the performance objective. (The one exception was the highly unlikely refined-inventory case in which the intruder drills into the RH waste, rather than into a homogeneous mixture of RH and contact-handled (CH) waste.)
- For the intruder-excavation scenario, all FWFs and burial depths studied satisfy the performance objective. (The one exception was the case with TRU waste, not ALLW).
- For the intruder-agricultural scenario, all FWFs fail to satisfy the performance objective for the 5-ft burial depth (16-ft cover). If the burial depth is increased to 9 ft (20-ft cover), all FWFs pass except IEB and IEB4. If the refined inventory is used, IEB passes and the dose from IEB4 is 110 mrem/yr, barely over the 100 mrem/yr limit of the performance objective.
- Under the assumptions used in this scenario, improved immobilization of the radionuclides into a more concentrated FWF degrades the intrusion performance by concentrating the source term. However, a more precise analysis of the inhalation exposure route would reflect the likely larger particle sizes of improved FWFs subjected to drilling or excavation, resulting in a lower calculated dose to intruders.

FOLLOW-ON STUDIES

In a third phase of the PA work, the preliminary parametric evaluations are being refined in several areas, some of which are described below.

FWF performance for hypothetical disposal of the INEL ALLW at two additional DOE sites, the Hanford Site and the Savannah River Site, is being evaluated. The effect of engineered features on disposal system performance is being explored. The waste inventory of nonactinide radionuclides, which strongly affect the dose in the intruder-agricultural scenario, is being refined. The effect of retardation of radionuclides due to chemical reactions in the cement FWFs is being evaluated. Finally, the particle size distribution of the FWF residue after disruption due to drilling and excavation is being explored.

As the project to provide treatment of the INEL ALLW moves forward, performance evaluations are expected to provide continuing support.

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44-3

EVALUATION OF POTENTIAL HAZARDS RELATED TO RADIOACTIVE WASTE DISPOSAL

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ABSTRACT

According to the German disposal concept, all radioactive waste has to be emplaced in a repository constructed and operated in deep geological formations. The required safety of a disposal mine must be demonstrated by a site-specific safety assessment that in particular includes the system geology, repository design, and waste packages.

To get a more detailed insight into the evaluation of the isolation potential of the site-specific geological barrier, it is meaningful to perform additional investigations into the radiotoxicity and chemotoxicity. Thus, an evaluation including the hazard potential of organic and inorganic substances being the major constituents of waste packages to be disposed of becomes possible.

INTRODUCTION

The need for safety assessments of waste disposal systems stems from the necessity to realize protection objectives. As waste disposal methods have become more technologically based, through the application of more highly engineered design concepts, through more rigorous and specific limitations on the types and quantities of waste to be disposed of as well as through experience and technical knowledge gained in disposal site operation, it follows that assessment scopes and procedures also must become more sophisticated.

In the past most investigations predominantly focuses on radioactive constituents of

the waste and on respective assessments of radiological consequences and impacts. However, the radionuclides being contained in the waste does only form the minor mass fraction. The major mass fraction is made up by non-radioactive organic and inorganic material including toxic and/or hazardous substances. Thus, the occurrence of chemotoxic substances has to be acknowledged and be taken into account in addition to hitherto existing safety assessments of radioactive waste disposal.

RADIOACTIVE WASTE DISPOSAL CONCEPT

From the very first beginning, the radioactive waste disposal policy in the Federal Republic of Germany has been based on the decision that all kinds of radioactive waste are to be disposed of in deep geological formations. Near-surface disposal is not practiced because of the high density of population, climatic conditions, and the existence of appropriate deep geological formations. The Bundesamt fur Strahlenschutz (BfS, Federal Office for Radiation Protection), among other things, is legally responsible for the establishment and operation of federal installations for disposal of radioactive waste.

Disposal Sites

According to the German disposal concept, all radioactive waste has to be emplaced in a repository constructed and operated in deep geological formations. As liquid and gaseous wastes are excluded from disposal in such a mine, only solid or solidified radioactive waste is accepted. Two sites are presently considered for disposal of low and intermediate level waste:

1. In the abandoned Konrad iron ore mine in Lower Saxony, it is planned to dispose of radioactive waste with negligible heat generation.
2. The emplacement of waste in the former Morsleben.salt mine in Saxony-Anhalt which was operated as a repository for short-lived low and intermediate level waste with low alpha emitter concentrations has been resumed.

The planned Konrad repository is assigned to accept radioactive waste with negligible heat generation, I. e. waste packages which do not increase the host rock temperature by more than 3 K on an average. Iron ore, I. e. coral oolite, at a depth of 800 m to 1,300 m is the host rock for this repository. Waste packages will be disposed of in drifts with an excavated volume of about 1,100,000 m3 allowing an em-placement of about 650,000 m3 waste package volume. Operation of the repository is scheduled at least 40 years. A total activity in the order of 1018 Bq and an alpha emitter activity of about 1017 Bq are anticipated in this facility.

In the former German Democratic Republic an abandoned salt mine located near the village of Morsleben was re-used for waste emplace-ment. From 1981 until 1991, radioactive waste with a total emplace-ment volume of approximately 14,500 m3 and about 6,700 spent sealed radiation sources were disposed of. Of this, the activity of alpha emitters amounts to 1.6 x 1011 Bq and that of beta/gamma emitters amounts to 4.8 x 1014 Bq. Subsequent to German unity the Morsleben facility has the status of a federal repository, the continuation of its former license being now limited by law until June 30, 2000. Until then, according to present planning, a radioactive waste volume of 40,000 m3 is envisaged to be disposed of. The estimated maximum activity of alpha emitters amounts to about 1013 Bq, that of beta/gamma emitters to about 1016 Bq. Operation has resumed smoothly on January 13, 1994.

Due to these planning data and marginal conditions the major bulk of low and intermediate level waste, i.e. radioactive waste with negligible heat generation, is intended to be disposed of in the planned Konrad repository. This is in particular relevant to alpha-bearing waste. The operation of the Morsleben repository will in the first instance contribute to the discharge of interim storage facilities.

Safety Criteria

The objective of radioactive waste disposal in repositories is to ensure that waste is handled and stored in such a way that the protection of man and environment from harm caused by the ionizing radiation is guaranteed. The basic aspects which must be taken into account to achieve the protection objective of disposal are compiled in the Sicherheitskriterien fur die Endlagerung radioaktiver Abfalle in einem Bergwerk/Safety Criteria for the Disposal of Radioactive Wastes in a Mine (1). These criteria qualitatively specify the measures to be taken in order to achieve the protection objective of disposal and define the principles by which it must be demonstrated that this objective has been reached.

The safety criteria permit a certain latitude of judgement. Such margins gradually diminish in the realization of a repository project. This process is predominantly determined by a site-specific safety assessment. Within the scope of such an

assessment the required safety of a repository in the operational and post-closure phase must quantitatively be demonstrated including the derivation of requirements on the design and construction of the facility as well as on the waste packages to be disposed of.
Nevertheless, the protection objective can only be achieved by an iterative process drawing together more and more detailed information obtained as the respective repository project progresses through its various phases of investigation, planning, detailed design and performance assessment, thus assuming more and more concrete forms.
Waste Acceptance Requirements
Persuant to the safety criteria (1), the safety of a repository in the operational and post-closure phase must be proved within the scope of a site-specific safety assessment. Such an assessment comprises the undisturbed performance of the planned facility, assumed incidents, the thermal influence upon the host rock, the nuclear criticality safety and the radiological long-term effects in the post-closure phase. The results of the respective Konrad safety assessment have been converted into both the design of the surface and underground facilities of this planned repository, and a system of waste acceptance requirements (2). They describe the general basic aspects and the general requirements to be fulfilled and then develop into more specific requirements on waste forms, packagings, radionuclide-specific activity limitations, documentation and delivery of waste packages to the repository. A survey on the structure of these requirements is given in Table I. The Konrad waste acceptance requirements can only be compiled in a final form after the license for this facility has been granted.
TABLE I
The structure of the Morsleben waste acceptance requirements (3) is similar to that of the Konrad requirements. Nevertheless, there are two decisive differences:
1. The operation of the Morsleben repository is regulated by the respective license granted on April 22, 1986, and by further documents pertinent to the license. This represents the legally binding framework which must be adhered to. The waste acceptance requirements includes both marginal conditions prescribed in the license and results of additional safety assessments which keep the above-mentioned framework. According to this, at first sight, the Morsleben requirements appear to be rather complicated.
2. The Morsleben waste acceptance requirements clearly distinguish between requirements on solid radioactive waste and on sealed radiation sources. Such a difference is not explicitly made within the Konrad requirements; they are formulated in a more general sense.
The fulfilment of the waste acceptance requirements ensures that the protection objective of radioactive waste disposal is met, i.e. radiation exposures will not exceed limiting values given in the Strahlenschutzverordnung/Radiation Protection Ordinance, and no undue radiological hazards will occur.
MIXED WASTE
Radioactive wastes including the full range from short-lived low to high level waste and spent fuel elements collectively present a very broad variety of chemical, physical and radiochemical characteristics, constituents and intrinsic hazards. Of this, mixed waste is defined to be waste that possesses both radioactive and non-radioactive constituents the latter being toxic and/or chemically hazardous. Well-known types of mixed low level waste are, e.g., liquid scintillation cocktails or fluids, miscellaneous laboratory chemicals, chlorinated organics and chlorofluorocarbon solvents. For many years, these non-radioactive constituents have been considered to be of minor concern. It has been implied that if mixed waste is disposed of in an adequate and safe manner in terms of its radioactive constituents then it will have also been disposed of safely in terms of its non-radioactive constituents. However, this is not necessarily true: radioactivity will decay, but toxic and/or chemically hazardous constituents may be persistent and their hazard potential may remain constant with time. Thus, implications arise and are implied on the disposal of mixed waste, using either near-surface or deep geological repositories. Further complications arise if different sets of regulations apply to the radioactive and non-radioactive constituents (4). In such circumstances, the ability to manage this waste has been somewhat limited.
Mixed waste particularly originates from research, development, demonstration and production activities in the defense area (4). Although this major source does not

exist in Germany, the arising of specific waste streams containing non-radioactive hazardous constituents cannot be excluded and must be taken into account. Such waste may originate from research establishments, medical/academic institutions, industrial facilities or the operation of nuclear power plants. Nevertheless, up to now, no necessity emerges for mixed waste to be regulated and handled separately from those wastes that are purely radioactive or hazardous. According to the German approach to radioactive waste disposal, mixed waste was considered within the planning work performed for the Konrad repository project and respective site-specific safety assessments were carried out. The results are reflected in both the Konrad waste acceptance requirements and an assessment of the chemotoxicity of that waste intended for disposal in the Konrad repository.

Thus, due to the Konrad waste acceptance requirements (2), the general basic requirement on waste packages

- "Prohibition of mixing non-radioactive waste which is to be disposed of in compliance with the Abfallgesetz/Waste Law with radioactive waste"

and the basic requirements on waste forms

- "No self-igniting or explosive materials are permitted"
- "Possible chemical reactions between the radioactive waste, the immobilization material and the packaging must be limited to a safety-related permissible level"

must in particular be fulfilled. These requirements are further detailed as well as appropriate checks and distinctive examinations are described within the quality assurance for waste packages (5). According to that the following quality assurance measures are applied:

The mixing of radioactive waste to be disposed of with non-radioactive waste is excluded according to distinctive regulations on the types and origins of radioactive primary waste to be treated using qualified conditioning processes, appropriate control measures by the waste conditioner, and subsequent independent inspections (performed according to BfS) of the conditioning process. Unallowed mixing would also in all probability be detected in the course of sampling inspections performed for waste packages not conditioned by qualified conditioning processes.

In order to specify further the basic requirements, characteristic values and test criteria have been deduced. The characteristic value for explosive materials is given by the content of explosives in the waste. Details, e.g. concerning permissible explosive materials, are regulated by the Sprengstoffgesetz/Explosive Substances Law. Furthermore, in accordance with this law, small masses of up to 3 g per 200 litre volume are considered to be permissible. It is to be anticipated that according to the distinctive regulations in the Sprengstoffgesetz permissible masses of explosives are kept, if contained in the waste. Nevertheless, if there should be any other hints, the primary waste will be examined and its contents of explosives be controlled.

Thus, the application of these specific tests ensures that only waste packages fulfilling the above-mentioned requirements will be accepted for disposal. In this way, it has been possible to develop a meaningful and practicable procedure for the safe handling and disposal of mixed waste.

RADIOTOXICITY AND CHEMOTOXICITY

In order to improve the predictive modeling capacity of safety assessments for the post-closure phase of a repository, safety assessment methods and procedures become more and more sophisticated. Thus, not only radiological aspects and radiotoxicities have to be evaluated but, in addition, the assessment of Chemotoxicities and the hazard potential of organic and inorganic constituents of radioactive waste to be disposed of has been performed.

A very comprehensive investigation into the chemotoxicity of radioactive waste (excluding vitrified fission product solution, cemented hulls and ends, and fines from reprocessing of spent fuel elements as well as spent fuel elements) was conducted for the Konrad repository project (6). Appropriate lists of chemotoxic substances, e.g. the International Register of Potentially Toxic Chemicals and compilations of the International Program on Chemical Safety and the International Agency for Research on Cancer as well as specific German ordinances on drinking water and sewage sludges were analyzed and respective organic and inorganic chemotoxic substances identified. Within a subsequent comprehensive inquiry into the actual and/or potential occurrence of those substances and into the elements and compounds which are the constituents of the waste packages to be disposed of, the

waste generators and conditioners submitted the data being necessary for an assessment of chemotoxicity. Thus, organic and inorganic compounds as well as radioactive and non-radioactive elements could be accumulated in a step-by-step procedure. Finally, an assessment of the chemotoxic organic and inorganic constituents was made considering the site-specific conditions in the post-closure phase of the Konrad repository.

Specific emphasis was given to the organic constituents. Table II lists chemotoxic organic compounds and typical representatives for which reaction types, reaction rates, and solubilities were needed to evaluate possible chemical and radiochemical-reactions in the post-closure phase.

TABLE II

It should be pointed out that in this table not only those compounds are compiled which are part of existing radioactive waste but, in addition, those chemotoxic organic compounds which may in future possibly be present in the waste. Thus, an assessment of a broad variety of chemotoxic organic constituents could be performed. As a result it was concluded that hydrolysis is the dominating factor concerning decomposition. According to the long-term safety assessment, it would take about 300,000 years for the water from the emplacement levels of the Konrad repository to enter the biosphere. This transfer time is long, even compared with the half-lives of the environmentally persistent chlorinated organic compounds. The chemotoxic organic compounds will therefore be safely hydrolyzed before reaching the biosphere, and there will be no need to limit the concentration of those organic compounds in the radioactive waste intended for disposal in the Konrad repository.(6)

Concerning the inorganic constituents, an assessment was based on the dilution of the saline and alkaline water from the repository on its way to the biosphere. It was assumed that the waste packages be dissolved in 106 m3 of host rock water ("repository water") in the post-closure phase. For chemotoxic inorganic elements only dilution in the "repository water" was allowed. Sorption and desorption, retardation effects as well as concentration-dependent solubility limits were not taken into account. As a result, it could be shown that the concentration of chemotoxic inorganic elements in the "diluted repository water", should it reach the biosphere, approaches the concentration limits of the German drinking water standards only in the case of a few elements, e.g. silver, cobalt, chromium, copper and nickel, being of common use in every day life.

In addition to these findings, comparisons were made as to the total of radioactive wastes:

- Chemotoxicity vs. its radiotoxicity.
- Radiotoxicity vs. 3 wt% natural uranium ore radiotoxicity.
- Chemotoxicity vs. the chemotoxicity of the mined repository volume.
- Chemotoxicity vs. the chemotoxicity of of the total repository area.

Time frames extend from 10 years up to 10 million years after the repository has been sealed (6). From that it may be concluded that the radiotoxicity of the radioactive wastes dominates the chemotoxicity for up to 100,000 years. After the first 1,000 to 10,000 years, the hazard potential of the radioactive wastes in the Konrad repository is lower than the hazard potential of the same repository. filled with natural uranium ore of 3 wt% U. The toxicity of the mined repository volume is less than the chemotoxicity of the radioactive wastes. The simple assessment with the "dilution model" demonstrates that no unacceptable hazards are associated with the chemotoxicity of the waste packages. Thus, the presence of inorganic and organic chemotoxic substances in the repository poses neither a new nor an inacceptable hazard potential to man.

For the Konrad repository the results of these assessments may be summarized as follows (6):

1. The chemotoxicity of the radioactive wastes stays distinctly below the radiotoxicity up to a time frame of about 100,000 years, when the radioactivity has decayed to such low levels that the radiotoxicity matches the already low levels of chemotoxicity,
2. The concentration of chemotoxic inorganic elements in the "diluted repository water", should it reach the biosphere, approaches the concentration limits as stated in the drinking water standards only in the case of a few elements.

In conclusion, it may be said that the chemotoxicity of the radioactive waste including mixed waste to be disposed of in the Konrad repository poses an acceptable risk to the public and that no new, hitherto unknown hazard potentials arise from

the chemotoxic organic and inorganic constituents of the waste. No additional requirements on the waste packages have been deduced from the above-mentioned assessments. However, it should be kept in mind that this advantageous result in particular stems from Konrad being a deep geological repository. As to a near-surface repository, appropriate limitations and requirements on the chemotoxic constituents of the radioactive waste are to be expected.

PROSPECTS

The approach to evaluate the potential hazards of waste by performing toxicity investigations shall not exclusively be reserved for radioactive wastes, but should exemplarily be applied as well and with even more justification to hazardous and/or purely chemotoxic wastes.

Thus, experience and knowledge gained in radioactive waste disposal planning work may advantageously be transferred and used in non-radioactive waste disposal activities, e.g. in order to reduce or avoid the necessity for later remedial actions.

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45-1

LIQUID RADWASTE PROCESSING WITH CROSSFLOW MICROFILTRATION AND SPIRAL WOUND REVERSE OSMOSIS

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ABSTRACT

The useful lifetime of thin-film composite (TFC) polyamide membranes used for the processing of variable aqueous waste at Chalk River Labs (CRL) by spiral wound reverse osmosis (SWRO) is about 3000 hours. This service lifetime is achievable through regular cleaning cycles which range between 70 to 200 m3 of waste treated. After 3000 hours of service the SWRO membranes deteriorate rapidly, and more frequent shutdowns are required for chemical cleaning cycles. The overall rejection efficiency of the SWRO membranes at an operating pH of about 6, and a volumetric recovery of 85%, decreased from about 99.5% with 3000 hours of service, to 95% after 4000 hours. Rapid increases in pressure drop due to increased deposition of foulants in deteriorated membrane areas were noted after 3000 hours of field service. Presently the crossflow microfiltration system is operated at pH 7 and removes 45% of the gross b/g contaminants and 70% of the a radioactivity. Iron concentrations are reduced to below 1 mg/L from 50 mg/L, which minimizes fouling due to ferric

hydroxide precipitates on the TFC membranes. About 60% of b/g in the permeate stream is present as 137Cs radioactivity.
The combined removal efficiencies for critical contaminants employing both microfiltration and reverse osmosis operations are as follows:

a : 99.9%;
b/g : 99.6%;
PO43-: 99.1%

INTRODUCTION

In the mid 1970's, AECL Research, at the Chalk River Laboratories (CRL) site, built a Waste Treatment Centre (WTC) for treating low-level solid and aqueous liquid wastes. The objective was to demonstrate processes for converting Canadian Deuterium Uranium (CANDU) wastes to a form suitable for disposal. The liquid waste streams are effectively volume reduced by a combination of continuous crossflow microfiltration (MF), spiral wound reverse osmosis (SWRO), and tubular reverse osmosis (TRO) membrane technologies. Backwash and chemical cleaning wastes from the membrane plant are further volume reduced by evaporation. The concentrate from the membrane plant is ultimately immobilized in bitumen using a thin-film evaporator in a shielded cell. The ability of the MF/SWRO technology to remove impurities non-selectively makes it suitable for the treatment of radioactive effluents from operating nuclear plants, with proper membrane selection, feed characterization, system configuration, and system chemistry control.
Currently, there are two streams routinely treated at CRL. One originates from the Decontamination Centre (DC waste) and the other collects waste from the Chemical Drain (CD waste) system. A total of about 2200 m3 of blended liquid (low to intermediate-level) waste are currently treated by the membrane plant annually. The current overall volumetric recovery of the two stage (concentrate-staged) reverse osmosis train employing SWRO and TRO is 96.6%. On a 35 m3 batch of waste treated there would be 0.7 m3 of backwash concentrate and 1.17 m3 of TRO concentrate sent to the evaporator for further volume reduction in a small evaporator, and subsequent immobilization with emulsified bitumen in a thin-film evaporator (see Fig. 1; (1)). The bituminized product from the facility occupies 0.175 m3. Hence, the overall volume reduction of the fresh feed through the integrated plant (employing both membranes and evaporation) is about 200.

MICROFILTRATION SYSTEM

In crossflow filtration (also called tangential flow or inertial filtration), pressure drives only part of the feed through the medium; the remaining feed flows tangentially along the surface of the medium, continuously sweeping particles from the medium's surface back into the feed. Generally, crossflow filters are operated as surface filters and have pores that are smaller than the particles to be removed. Crossflow microfiltration (MF) is the most recent advancement in membrane system technology. The removal efficiency of a MF membrane is significantly higher (10-100 times) than a conventional filtration process. MF systems operate as a barrier filtration device to remove particles, suspended solids, and microorganisms from a feed. By using a microfiltration membrane as the separation medium, particles in the 0.1 to 10 m range can be removed. The MF system employed at CRL is a model 40M1 manufactured by Memtec America Corporation and has been in operation since 1988 June. The original membranes were replaced after four years of service in 1992.
The crossflow MF system at CRL contains 40 filtration modules, approximately 6 cm in diameter by 50 cm long. Each filter module contains polypropylene hollow fibre membranes arranged in a shell-and-tube geometry. The total membrane surface area available for filtration in each module is about 1 m2. The nominal filter pore size is 0.2 mm. The system is configured as two individual systems of 20 m2 operating from a common surge tank. Each bank of 20 modules is further divided into two stages of 10 modules operating in series.
Feed flows tangentially across, over, and around the hollow fibre membranes at a sufficient crossflow velocity to keep the solids in suspension, thus minimizing deposition and fouling. A small fraction of the total feed flow (about 10%) passes through the membrane into the center of the tube or lumen, and exits as filtrate. The unfiltered portion of the feed is recycled to the system surge tank. The inlet feed pressure to the first stage is typically 300 kPa. The differential pressure across each stage is normally 40 to 50 kPa at a crossflow rate of 245 to 285 L/min. At a filtrate production rate of 25 L/min/bank, the transmembrane pressure (TMP) drop, which is the average feed pressure minus the filtrate pressure, is in the

range of 20 to 100 kPa.
To maintain the designed filtrate rate, and remove accumulated suspended solids in a concentrated form, the Memtec unit utilizes a patented gas backwash. Air at high pressure (700 kPa) is periodically introduced into the filtrate side of the system, and instantaneously expanded through the hollow fiber into the feed, thereby releasing accumulated solids from the membrane surface. Feedwater is then used to flush remaining solids from the system. The duration of the backwash sequence is approximately 90 seconds. Initiation of backwash is accomplished automatically by a panel-mounted timer, or manually by a push button. Chemical cleaning of the system is performed on a periodic basis, usually with an alkaline detergent. Chemical cleaning is required when the transmembrane pressure (TMP) exceeds 100 kPa at normal filtrate production rates of 15 to 25 L/min or if the crossflow pressure drop across the feed channel exceeds 80 kPa.

SPIRAL WOUND REVERSE OSMOSIS SYSTEM

Reverse osmosis (RO) is a technology that is well established for the production of potable water from brackish water or seawater. It has been successfully used to produce high-purity water for the electronics, pharmaceutical, and power industries. Because of its versatility to remove ionic impurities, particulates and colloids, organics, microorganisms and pyrogenic material from water, RO has attained a prominent role in water purification (2).
In reverse osmosis treatment, feedwater containing dissolved and suspended solids is pumped into the system at a desired feed pressure greater than the osmotic pressure of the solution. The feed stream is pumped into a pressure vessel containing one or more membrane elements connected in series. The feedwater then flows into the channels between the membrane sheets. These feed channels are composed of a plastic netting which breaks up the flow into small turbulent areas above the membrane surface.
Immediately above the membrane surface a concentration boundary layer forms whose thickness depends on feedwater ionic strength, particulate level, and flow in these small turbulent areas. The water and ions are transported by a solubility-diffusion process to the permeate water carrier. The purified water or permeate is recovered at atmospheric pressure. The remaining water, dissolved solids and particulates form the reject stream. The pressurized concentrate or retentate is dropped to atmospheric pressure through a back pressure regulating valve, immediately downstream of the system.
The performance of an RO membrane is usually described in terms of permeate flow, or "flux", contaminant rejection efficiency, and volumetric recovery. Permeate flux refers to the amount of flow across the membrane per unit area, at a particular operating pressure and dissolved solids concentration. The flow of water across the membrane is proportional to the effective pressure (applied pressure minus the osmotic pressure of the solution). Increasing the applied pressure will increase the permeate flow without increasing the solute flow.
Rejection is the relative change in contaminant concentration from the feed stream to the permeate stream. RO membranes are not absolute barriers, and some small percentage of the solute (typically about 0.5%) does pass through the membrane. The amount of solute transport is a function of the membrane type and is proportional to the differential concentration across the membrane.
SWRO membrane types can be broadly classified as cellulosic or noncellulosic. Cellulose acetate membranes are still widely used because of their resistance to fouling, and their low cost. They are, however, easily damaged by bacterial attack and have relatively low rejection efficiencies. Noncellulosic membranes, such as the Filmtec SW30HR membranes used at CRL, have a wider pH range and exhibit high solute rejection efficiencies. The SWRO configuration achieves a large specific surface area per unit volume, which is typically 1000 m2/m3. This can be compared to 165 m2/m3 for plate modules and 335 m2/m3 for tubular modules.
The ratio of permeate to feed in a RO system is referred to as recovery. To achieve high volumetric recoveries (up to 85% currently employed at CRL), it is necessary to stage the concentrate stream or recycle the concentrate for reprocessing. This is normally accomplished in a tapered system design. For instance, a two-stage system may have four pressure vessels in the first stage, feeding two vessels in the second stage. The tapered configuration compensates for feed flow loss by permeation, therefore maintaining optimum cross-flows in both stages.
The CRL system is a three-stage 5:3:1 tapered system with 10 cm diameter by 6 m long

pressure vessels. Each pressure vessel contains 6 Filmtec SW30HR membrane elements. The system is fed with a Goulds 3333 multi-stage centrifugal booster pump. Typical feed crossflows are maintained in the range of 35 to 50 L/min, with an inlet pressure of about 2700 kPa.

PRETREATMENT BY MF SYSTEM

There are many non-radioactive contaminants in the combined CD/DC waste stream. The most deleterious of these contaminants (from a downstream scaling perspective on the SWRO system) is iron. Although the MF system is not normally expected to provide significant rejection of chemical contaminants, iron removal to about 3 mg/L in the filtrate is required upstream of the SWRO to prevent excessive iron oxide fouling (3). Thus the MF system must be capable of removing iron if it is to be a suitable pretreatment option for the SWRO system.

The turbidity of the feed stream to the plant can be reduced from 200 NTU to less than 0.5 NTU, and the silt density index (SDI) can be reduced to less than 3 with the MF system alone (4. A turbidity removal efficiency of between 98% and 99.9% has been noted consistently for the MF system since the start of operations in 1988. It has been found, however, that filtrate from the microfiltration pretreatment process can be somewhat unstable for the types of wastes processed at CRL. Rapid increases of turbidity and silt density index (SDI) occur after the filtrate has been pH adjusted down to 6. It is thought that the increase of turbidity of the filtrate may be associated with instability of colloids at the operating pH of 6, and to the presence of polyvalent cations in solution.

EFFECT OF FEED pH ON MF REMOVAL EFFICIENCY

Pretreatment, such as pH adjustment, is used to maximize soluble contaminant precipitation. One of the objectives of this study was to determine the impact of feed pH on the removal efficiency of various contaminants by MF. The feed pH was altered by the addition of sodium hydroxide or nitric acid in the conditioning feed tank.

Fig. 1a.
Fig. 1b.
Fig. 1c.
Fig. 1d.

Figure 1a shows the effect of feed pH on the removal efficiency, e, of the gross a component of the combined CD/DC waste stream. It is evident that e increases substantially by raising the feed pH value from 6 to 8. The removal of gross a is virtually complete when the feed pH value remains above 8.

Figure 1b shows the removal efficiency of gross b/g for the same range of feed pH values. Although there is some scatter of the data, it is evident that e increases between pH 6 to 9, and remains invariant at about 90% thereafter. Normally it is undesirable to elevate the pH to 9 upstream of the MF system from a pH value of about 4, due to the prohibitive cost associated with adding large quantities of chemicals. Moreover, it would be necessary to reduce the pH back to 6 upstream of the SWRO system to prevent scaling by hardness ions such as calcium and magnesium. Hence, the MF system at CRL is usually operated at a pH of about 7, which results in a b/g removal efficiency of about 40% (Fig. 1b)[and an a removal efficiency of about 70%] (Fig. 1a).

The removal of b/g emitters elevated the contact radiation field on the MF cartridges. The contact radiation field with the MF cartridges increased from 10-4 to 10-1 Gy/h (10 to 104 mR/h) as the gross b/g concentration in the feed stream varied between 10 to 105 Bq/mL (2.7 E-4 to 2.7 E0 mCi/mL).

Of the longer-lived b/g emitters, the radionuclide that is most efficiently removed is 144Ce. Figure 1c shows the removal of 144Ce by the MF system in the evaluated pH range of 6 to 11. It is apparent that e increases with a pH ranging between 6 to 8. Thereafter, it is relatively constant, and virtually all of it is removed in the MF pretreatment step. A large component of the radiation field on the MF cartridge filters is due to the presence of 144Ce.

Iron oxide fouling was found to be present in the SWRO system at CRL. It is caused by the oxidation of ferrous to ferric ions and the subsequent precipitation of ferric hydroxides in the module (5). With the presence of sequesterants and chelating agents in the DC (EDTA, sodium hexametaphosphate), iron can exist in a complexed form, thereby causing it to stay in solution and not be removed by MF. Therefore, iron removal can be quite variable, and is dependent upon the concentrations of these compounds in the feed solution. Figure 1d shows the removal

efficiency of iron in the MF system, which increases from about 20% at a pH of 6, to 99% at a pH of 8. At the normal operating pH of 7, about 50% of the iron is removed, although iron concentrations to a level of less than 0.1 mg/L have been achieved. Generally, iron concentrations are lowered to less than 3 mg/L, which minimized ferric fouling downstream in the SWRO system.

Table I shows the removal efficiencies of both the MF and SWRO systems for various critical contaminants that are present in the CRL waste streams. The radionuclide most effectively removed by microfiltration is 144Ce with an average removal of 70.7%. Soluble contaminants such as 137Cs are not removed by the MF system. The overall removal efficiency for both the MF and SWRO systems ranges from 96.8% for 137Cs to 99.6% for gross b/g, and 99.9% for gross a. The overall removal efficiency of the most critical nonhazardous contaminant, PO43-, is 99.1%.

PERFORMANCE OF THIN FILM COMPOSITE MEMBRANES FOR LIQUID WASTE PROCESSING

The permeation flux was normalised to an applied pressure of 2.76 MPa and a feed temperature of 25oC using a method given by Bukay (6). The observed permeation flux (OBF) was normalised for temperature using a temperature correction factor (TCF) obtained from the membrane manufacturer, and an effective pressure using equation (1).

Eq. (1)

Fig. 2.

Figure 2 shows the overall normalised permeation flux for the plant-scale system. The data were obtained for the original thin film composite polyamide membranes installed in June 1991. These membranes were replaced in February 1994 after about 4000 hours of service. After the initial flux decline, which took place over the first 50 hours of operation, there was a subsequent gradual flux decline which took place over the next 3900 hours.

The flux decline with time observed in Fig. 2 is the result of concentration polarization and/or surface fouling. Surface fouling occurs when there is deposition of submicron particles on the surface, as well as crystallization and precipitation of smaller solutes. It is manifested when rejected solids are not transported from the surface of the membrane back to the bulk stream. In general, there are five types of fouling: namely membrane scaling, fouling by metal oxides, device plugging, colloidal fouling, and biological fouling (5). The different types of fouling frequently occur at the same time and can influence each other. Unfortunately, the interactions between types of fouling are poorly understood.

The chemical scale on the fouled CRL reverse osmosis membranes is comprised primarily of aluminum, silica, calcium, phosphorous, and to a lesser extent iron and sulphur. Calcium hydroxylapatite and octacalcium phosphate scale have been identified as a major fouling species on the SWRO membranes. Clay and aluminum silicate-based scale is thought to account for the initial large flux decline associated with concentration polarization, and represents the scale which is the most difficult to remove by standard chemical cleaning procedures. Precipitates approaching 10 m in diameter have been observed on the surface of the membrane, and these precipitates are hard to dissolve in even the most concentrated acids. This is further supported by the observation that alkaline cleaning chemicals at pH 12 are the most effective for permeate flux restoration where silica solubilizes to silicic acid (7). The precipitates in the 54 membrane elements of all three SWRO stages are similar in chemical composition.

While operating, TFC membranes take on an anion charge on the surface. This causes cationic foulants, such as aluminum and ferric hydroxides, along with cation coagulant polymers, to be attracted to it. Further, because of the high flux rates on composite TFC membranes, fouling occurs faster and is more noticeable than with other membranes (8).

The scatter of the data shown in Fig. 2 is not due to errors in measurement; rather, it is the result of permeation flux declines during a given run, and the subsequent recovery after a chemical cleaning with the appropriate solvent. A least squares curve fit was plotted through the data points in Fig. 2. It shows that the combined total flux (for all 54 elements) decreased from about 55 L/min at start up, to 25 L/min after about 4000 hours of service. The effectiveness of the cleaning solutions decreased noticeably after about 3800 hours of operation. The frequency of cleaning increased from 300 m3 to every 80 m3 of liquid waste treated as the membranes aged.

The most effective cleaning solution for the CRL scales was Memclean, an alkaline-based detergent containing EDTA. The most effective antiscalant chemical in the SWRO

system was Pretreat Plus which is manufactured by King Lee Technologies (San Diego, CA). The average throughput between chemical cleanings was maximized with the use of this antiscalant. Secondary wastes from cleaning accounted for about 5% of the annual waste feed volume to the plant. Acid-based cleaning solutions were generally not effective for removal of the scale-forming compounds. Silica scale was not effectively removed from the membranes with any of the cleaning chemicals that were investigated here, and it seems that the silica scale formed a tightly adherent layer immediately on top of the membrane surface.

During the final 200 hours of operation the performance deteriorated very rapidly. The permeation flux increased only marginally even after an aggressive chemical cleaning, and it became apparent that a change of membranes was required to keep up with the production of the waste. The system was shut down and new membranes were installed in all stages.

Chemical scaling on reverse osmosis membranes can be controlled and minimized by selecting optimal crossflow velocities through the SWRO system and by the use of a 5 mm cartridge filter upstream of the high pressure feed pumps. Pilot-scale studies at CRL have shown that a 50% increase in crossflow velocity can reduce the decline of permeation flux by a factor of 2 for the combined waste streams currently being treated. A 5 mm cartridge filter has been effective for the removal of scales such as silica and phosphorous even before volume reduction with reverse osmosis. This is due to the fact that unstable precipitates are formed after pH adjustment of the MF filtrate with acids. Antiscalant additives and pH adjustment appear to be secondary parameters in comparison to the dramatic impact of crossflow velocity.

Fig. 3.

Figure 3 shows the pressure drop across the three stages of the SWRO system over the lifetime of the original membranes, and the first 100 hours on the new membranes. During the initial 3000 hours of operation the average pressure drop across the system increased from about 500 kPa to 1250 kPa. This increase was attributed to clay and silt deposits in the brine channels of the system. In support of this hypothesis, Kronmiller (7) notes that the presence of silt and particulate (known to be present in CRL waste) are trapped in the membrane and not easily removed. He observed that as these foulants build they erode the membrane surface and break the spiral wound package.

During the final 1000 hours of operation there were large fluctuations in the average pressure drop, the amplitude of which sometimes exceeded 2200 kPa. However, there was still sufficient applied pressure to overcome the osmotic pressure of the waste water at 85% recovery. Aggressive chemical cleanings were performed to restore the flux (Fig. 2), and increase the net effective driving pressure to the system by lowering the system pressure drop (Fig. 3).

The significant increase in system pressure drop between 3000 hours and 3200 hours again is attributable to colloidal silt and scale buildup on the membranes and in the brine channels. During this period of time the concentration of silica-based colloids in the Decontamination Centre waste stream was significantly higher than usual. Permeate rates were acceptable during this period so the pressure drop was tolerated. Aggressive alkaline cleans using Memclean at 3120 hours and again at 3170 hours brought the pressure drop back to more normal values of about 1200 kPa.

Another performance indicator that can be used to assess the SWRO, is the overall removal efficiency of conductive ions. Figure 4 shows the conductivity removal in the plant-scale SWRO system over its service life. During this time the feed conductivity ranged from 100 to about 1000 mS/m, while the permeate conductivity was between 0.6 and 20 mS/m. In general, the permeate conductivity was below 10 mS/m except when there were upsets due to the over-addition of nitric acid.

Fig. 4.

Figure 4 shows that during the initial 3000 hours of operation the conductivity removal was steady at about 99.5%. In those cases where the removal efficiency was lower than 99%, the presence of excessive nitrate and sodium ions in the permeate following a chemical cleaning was the cause. These monovalent ions have low rejection efficiencies and the plant would have been temporarily overloaded with these monovalent ions immediately after a cleaning cycle. This would result in a reduced overall rejection efficiency. After these cleaning chemicals had been flushed from the system (at 1200 hours for example), the rejection efficiency would be restored to about 99.5% overall.

Conductivity removal decreased sharply after about 3000 hours of operation. During

the last 1000 hours the average conductivity removal decreased from 99.5% to about 95% (Fig. 4). A rejection efficiency of 95% is very poor for TFC polyamide membranes.

The many aggressive chemical cleanings that were required to restore the permeation flux (Fig. 2) may have had a detrimental effect on the membrane integrity, resulting in the considerable loss of membrane rejection performance. Physical abrasion of the polyamide rejecting layer over time may have also been a contributing factor. After 4000 hours of service the membranes were replaced because of the frequency of cleaning and the low overall removal efficiency being achieved.

CONCLUSIONS

Feed pretreatment upstream of reverse osmosis by microfiltration is critical for the removal of iron and suspended solids foulants. By conditioning the feed to a pH in the alkaline regime the precipitation of metals is optimized. About 70% of the alpha radioactivity is removed and about 50% of the gross beta/gamma radioactivity is rejected in the MF backwash when the system is operated at a pH value of 7.

The permeation flux of the SWRO system was maintained for about 3000 hours of service with regular cleaning cycles after every 100 - 200 cubic metres of liquid waste are treated. After about 3000 hours of service the membranes degraded, which resulted in a decrease of the overall rejection efficiency from 99.5% to 95%. Cleaning solutions account for about 5% of the waste feed processed annually through the plant. The chemical scale found on the fouled reverse osmosis membranes is comprised primarily of aluminum, silica, calcium, phosphorous, and to a lesser extent iron and sulphur. Calcium hydroxylapatite or octacalcium phosphate scale have been identified as the major fouling species on the membrane. Colloidal silica foluling contributes primarily to the initial large flux decline observed during the first 100 hours of operation.

The combined removal efficiencies for critical contaminants after treatment of the waste with both microfiltration and reverse osmosis are as follows:

- a: 99.9%;
- b/g: 99.6%; and
- PO_4^{3-}: 99.1%.

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ABSTRACT
Oak Ridge National Laboratory (ORNL) will be using chabazite zeolite in the near future to improve wastewater treatment processes. The zeolite, a nonregenerable ion-exchanger, will be used to remove both radioactive strontium and cesium from 70 million gallons of process wastewater generated per year at ORNL. This process will replace the ORNL Process Waste Treatment Plant and improve wastewater treatment efficiency, improve the quality of the wastewater discharged to the environment, and reduce secondary waste generation. A commercially available ion-exchange system was leased to perform pilot-scale testing and provide the data necessary for full-scale plant design. Results of the study indicated that the commercial system performed well with the zeolite and that previous smaller-scale column tests adequately predicted the zeolite performance achieved in the commercial- scale columns. Filtration systems and zeolite sluicing procedures were very effective in pilot-scale tests. The mass transfer zone length for contaminant removal, calculated using the Rosen model, was comparable to that determined from smaller-scale column tests. The full-scale zeolite system is likely to be constructed in the 1999 time frame and will be one of the largest wastewater decontamination applications utilizing zeolite in the world.

INTRODUCTION
In previous development studies, many different flowsheets were evaluated for removal of radioactive strontium and cesium from Oak Ridge National Laboratory process wastewater (1,2). The wastewater has a groundwater-like composition with components shown in Table I. The chabazite zeolite system was chosen because of the many process advantages it offers from the standpoint of waste reduction, safety, environmental protection, and operating simplicity. Use of the zeolite system will reduce solid waste generation by more than 40%, eliminate the use of concentrated acids and bases, improve radionuclide removal, reduce dissolved salt concentrations, and eliminate several unit operations.

Laboratory and small-scale column tests were previously performed to determine equilibrium and kinetic data for the zeolite (3). The pilot tests were performed to evaluate physical operating characteristics of the system and validate previous small-scale column test results. A commercial ion-exchange system was leased from a private-sector service company and used in treatment of actual wastewater. This test program involved determining optimum procedures for sluicing of spent zeolites from the operating vessel, demonstrating the effectiveness of prefiltration equipment, and determining the mass transfer zone (MTZ) length for the commercial system for comparison with the MTZ determined with small-scale columns.

One of the problems encountered in previous small-scale and near-full-scale column studies at the Process Waste Treatment Plant is the occurrence of "contaminant bleed." Contaminant bleed is characterized by high effluent concentrations of radioactive contaminants after a column has been returned to service following spent media removal and reloading with fresh zeolite. This is caused by elution of contaminants from residual spent zeolite which was not removed from the column during sluicing. This behavior can reduce the in-service duration of a column and can result in release of wastewater which exceeds contaminant discharge limits. The extent of contaminant bleed is largely determined by the degree of success in removal of spent zeolite by sluicing. Many factors can influence the efficiency of zeolite removal including column design, operating procedures, and the condition of the zeolite. If sluicing is not effective, extraordinary and expensive measures must be taken to remove spent zeolite from the column. The pilot tests were designed to evaluate the efficiency of zeolite sluicing and the subsequent extent of contaminant bleed.

The MTZ specifies the minimum depth of the zeolite needed to achieve the required decontamination factor for the wastewater being treated. The MTZ length is important for determining the minimum column bed depth for the full-scale system. Though the zeolite removes both strontium and cesium, the MTZ length to be used in system

design is based on strontium breakthrough. This is because equilibrium behavior of the zeolite results in lower loading capacity and quicker column breakthrough for strontium. The strontium MTZ was determined for small-scale columns (7.6 cm diameter, 91 cm length) in previous studies (3). Since the MTZ length changes for different column geometry and internal design, it was important to determine MTZ length for the pilot-scale column.

In small-column studies, granular anthracite prefilter columns were found to be effective for removal of suspended solids from process wastewater prior to treatment with zeolite. Pilot-scale columns were loaded with the same anthracite material to prevent fouling of the zeolite bed and to demonstrate large-scale performance.

Pilot Plant System Description

The pilot system, leased from a private service company, was modular in design and constructed for nuclear power plant wastewater treatment applications. A flow diagram for the system is shown in Fig. 1. The system consisted of three skid mounted stainless steel columns, each with a diameter of 0.91 m (36 in) and a capacity of 0.85 m3 (30 ft3). A skid mounted control system was also provided which included the circulation pump, flow controller, and valving manifold. The columns were designed for down-flow ion-exchange or granular media filter applications and were equipped for backwashing, air sparging, flow control, flow totalizing, and sluicing of media into and out of the columns. Two of the columns were used as prefilters and were loaded with granular anthracite while the third column was loaded with chabazite zeolite. The system was equipped with a process wastewater influent and an effluent header with tie-in for service water and service air. The control skid was housed in a shelter to protect the electrical and electronic components. The system, referred to as the Zeolite Demonstration System (ZDS), was assembled within the diked containment area surrounding the ORNL Bethel Valley Storage Tanks (BVSTs). The system was connected to one of the 1.32 106-L (350,000-gal) collection tanks used as an equalization tank for the PWTP. The two columns to be used as prefilters were connected in parallel such that either could be in service while the other was being backwashed. The column to be used for the zeolite was located downstream from the prefilter columns. The flow indicator/totalizer was installed downstream from the zeolite column so that its operation would not be compromised by suspended solids in the feed wastewater. Since collection of system effluent for filter backwash was not feasible, filters were backwashed using untreated wastewater. Suspended solids concentration of the untreated wastewater was typically low enough so as not to present significant problems from slight breakthrough when the filter was returned to service.

Fig. 1.

The two prefilter columns were loaded with 0.57 m3 (20 ft3) of granular anthracite. The third column was loaded with 0.43 m3 (15 ft3) of 20 50 mesh natural chabazite zeolite. All three columns were loaded dry and backwashed at a flow rate of approximately 3.2 L/s (50 gal/min) for about 40 min to remove fines. The system was operated at a flow of 1.26 L/s (20 gal/min) to coincide with the flow per unit area used in previous small-scale column tests, approximately 1.9 Ls-1m-2. A wastewater residence time of 5.6 min was provided at this flow rate.

TEST RESULTS

Sluicing and Contaminant Bleed Evaluation

Two of the three sluicing tests performed were successfully completed. The procedure provided by the service company that supplied the pilot system was used. In the first test, using a combination of air and water for the sluicing operation left the column virtually free of all but a few grains of spent zeolite. During the exhaustion cycle leading up to the second sluicing test, extremely high suspended solids concentration was encountered in the wastewater due to an abnormality in BVST system operation. The suspended solids broke through the anthracite prefilters and accumulated in the zeolite bed. This caused the zeolite to stick together and become difficult to mobilize from the column. With manual spraying of the column interior, removal of the zeolite was eventually accomplished. In a third sluicing test, the condition of the wastewater was back to normal and the sluice was very successful. In previous small-scale column studies, the lack of clearly visible amounts of spent zeolite did not mean that contaminant bleed would not be encountered. Traces of spent zeolite not visible after sluicing caused contaminant bleed which continued for several hundred bed volumes throughput. At the beginning of the second and third

exhaustion cycles in pilot tests, frequent samples of the system effluent were taken to determine the extent of contaminant bleed. In both cases, the effluent radionuclide content was below the discharge limit for 90Sr (37 Bq/L or 7.1 pg/L) at the beginning of the exhaustion cycle. Elution of the traces of spent zeolite left in the column did not significantly elevate the 90Sr concentration of the column effluent. This was true for the two successful sluicing operations and for the case where manual spraying and removal of the spent zeolite was necessary. These results indicate that contaminant bleed will not be encountered in large-scale columns if spent zeolite sluicing is performed effectively.

The conceptual design for the full-scale system had included an acid cleaning system for rinsing of the zeolite column internal surfaces to elute the contaminants from spent zeolite left in the vessel. Though alleviating the contaminant bleed problem, this system added several hundred thousand dollars to the project cost and would have generated a secondary waste, the spent acid solution. The findings of the demonstration test allowed elimination of the acid cleaning system from the final system design.

Prefilter Effectiveness

As shown in the demonstration, one of the keys to effective removal of spent zeolite is the efficient performance of clarification and filtration equipment in removal of suspended solids from the wastewater. If removal of suspended solids is not complete, the solids accumulate in the zeolite, causing agglomeration and adhesion of zeolite to vessel walls. For the pilot study, granular media filters using anthracite were effective in two of the three tests. In the first exhaustion cycle, the jet mixers installed in the 1.32 X 106-L equalization tank were deenergized to allow the tank to act as a crude clarifier for removal of suspended solids. This simulated the expected operation of the full-scale zeolite system, which will include clarification as well as filtration. The prefilters performed well during this exhaustion cycle, and the spent zeolite sluiced easily from the column. During the second exhaustion cycle the jet mixers were energized, which simulated poor clarifier performance for the test. Several days were allowed for displacement of the accumulated solids in the equalization tank before restarting the zeolite system. However, suspended solids content remained high and caused breakthrough of the filter columns and accumulation of sludges in the zeolite column. Accumulation of these solids causes the zeolite to agglomerate and stick to the walls of the vessel. After the normal sluicing procedure, visual inspection of the column revealed a large quantity of spent zeolite caking the interior of the vessel. The vessel interior was washed down by a mild jet spray of water while pumping the sluice line with a double diaphragm pump. This procedure successfully removed the zeolite. This test indicated that breakthrough of prefilter columns cannot be tolerated for the full-scale system. Manual spraying the interior of a full-scale column would be a much more difficult, time-consuming, and costly process. This can be avoided in the full-scale system by avoiding accumulation of sludges in equalization tanks and by providing a well-designed clarification system to remove the bulk of the suspended solids prior to the granular media filters and the zeolite system.

Contaminant Breakthrough Data

The influent and effluent sampling data from two exhaustion cycles were successfully used to determine the breakthrough characteristics of the zeolite. A total of 2.4 106 L (5633 bed volumes) of process waste were processed during the first exhaustion cycle. Column breakthrough data are shown in Fig 2. For the graph, each data point for the gross beta concentration of the feed wastewater was averaged with the previous three data points to dampen the fluctuation in concentration and give an improved indication of breakthrough. The gross beta concentration of the wastewater was effectively reduced to acceptable levels during the first several days of operation. On the third day, a sudden increase in gross beta concentration occurred at a throughput of 591 bed volumes. During the next 48 hours, the gross beta concentration gradually decreased to acceptable levels. At a throughput of 1580 bed volumes, the fractional breakthrough (effluent gross beta concentration/influent gross beta concentration) began to increase to a consistent level of 30 to 40%, where it remained until the throughput of the system reached 4630 bed volumes. At this point, the effluent gross beta suddenly increased to levels that far exceeded the influent concentration. Gross beta concentrations gradually fell to levels corresponding to 40 to 50% breakthrough, at which point the test was terminated at a

throughput of 5630 bed volumes.
Upon investigating the concentrations of other cations in the zeolite effluent wastewater (the analytical results for which were several weeks behind the radiochemical results), it was discovered that an increase in sodium concentration from 30 to 170 mg/L coincided with the sudden 200% fractional breakthrough. It is well known that sodium concentration strongly influences the equilibrium between zeolite and other cations in the wastewater (4). In this case, the increase in sodium concentration caused elution of contaminants from the zeolite, resulting in high effluent gross beta concentrations. High sodium concentration is caused by periodic addition of salts to the process wastewater feed. Based on this behavior, it will be important during zeolite system operation to carefully monitor and control sodium concentration of the plant feed wastewater to avoid sudden variations. Elimination of the current softening and ion-exchange processes at the PWTP will eliminate the periodic fluctuation in feed sodium concentration. Thus, control of wastewater sodium concentration is not expected to be difficult when the full-scale zeolite system is operational.
A total of about 12,000 bed volumes was processed during the second exhaustion cycle, though part of this throughput was accumulated during recirculation of the tank contents. Breakthrough data are shown in Fig. 3. A sudden drastic increase in effluent gross beta concentration caused by an increase in sodium concentration occurred at a throughput of 4200 bed volumes. The fractional breakthrough declined rapidly over the next several days to a level of less than 30%. At a throughput of 6000 bed volumes, a steady decrease in gross beta concentration for the system feed was noted. This behavior was due to the diversion of the wastewater feed to the alternate collection tank resulting in recirculation of wastewater in the pilot plant feed tank and the subsequent exponential decay curve. The process waste feed was diverted back to the ZDS feed tank at a throughput of about 11,500 bed volumes. Final breakthrough was confirmed at a throughput of 12,000 bed volumes. By deducting the approximate volume of water recirculated, the actual throughput processed was 8900 bed volumes.
MTZ Length
An important objective of the pilot testing was to determine the length of the MTZ and compare it with MTZ lengths determined from smaller-scale column tests. At any time in the loading process, a column can be divided into three zones: a saturated zone, an MTZ, and an unused zone. The solid loading in the MTZ is near saturation in the direction of the water inlet end of the column and near zero toward the outlet end. The MTZ moves down the column during loading, and breakthrough occurs when it reaches the end of the column. Ideally, the MTZ will occupy a relatively short fraction of the column length, so that nearly all of the column is saturated at the time of breakthrough, when the column must be taken off stream and changed out with fresh zeolite. Experiments using columns 7.6 cm in diameter and 91 cm in height gave strontium MTZ lengths ranging from 10 to 30 cm depending on wastewater velocity. The Rosen long-bed solution, presented below, adequately predicted these MTZ lengths in an earlier study (3).
The long-bed solution of Rosen is a simplified model which accounts for both equilibria and flow rate variations (3,5). This model was used to determine MTZ length for the ZDS column using the data collected in the second exhaustion cycle. Rosen solved the partial differential equations for unsteady-state column adsorption for the case of a linear isotherm (5). Distribution coefficients for strontium and cesium were constant over the concentration ranges encountered at ORNL, as long as the calcium and sodium concentrations were constant. Typical strontium and cesium concentrations were below 10-6 eq/L, while calcium and sodium concentrations were generally on the order of 10-3 eq/L (3).
Fig. 2.
Fig. 3.
The Rosen solution for long beds is
Eqs. (1-4)
The above equations were applied to the data collected from the second exhaustion cycle in the following steps:
1. The superficial velocity through the bed was calculated with a bed diameter of 0.91 m and flow rate of 1.26 L/s.
2. The film mass transfer coefficient was found. From a study by Robinson (4), the film mass transfer coefficient (kf) for this same system was determined to be 0.006

cm/s at a lower velocity of 0.0914 cm/s. Since kf is proportional to the cube root of the Reynold's number of the particle (Rp1/3), a ratio may be written to determine kf at the appropriate velocity.
3. A log-log plot of Kd versus total salt concentration in solution from other zeolite treatment studies gave a fairly straight-line relationship (shown in Fig. 4); thus Kd could be determined from the figure.
Fig. 4.
4. A time (or position) in the bed was assumed to calculate the length of the MTZ. Since the Rosen solution is valid only for "long beds," a position near the end of the bed was assumed. For the second exhaustion cycle, breakthrough occurred at about 3 million liters. At a flow rate of 1.26 L/s, the corresponding time was about 40,000 min, so t = 30,000 min was chosen. By working only with the ratios of Y:X and O:X given in the Rosen solution, the diffusivity, DA, could be eliminated.
5.Expressions for both ratios mentioned in step 4 were determined as a function of bed length, z. A bed length, z, was assumed and CA/CAo calculated from Eq. 1. Results from this exercise showed CA/CAo to be approximately 95% at z = 40 cm and 5% at z = 60 cm. Thus, the MTZ at time 30,000 min was about 20 cm. In earlier small column tests (6), the length of the MTZ between 70 and 5% breakthrough was 15 cm, which agrees reasonably well with the results obtained in this demonstration. However, the lengths of the MTZs should not necessarily be consistent between the small column and the demonstration test, since the feed water compositions and therefore the Kds were not the same.
For a system which displays equilibrium behavior characterized by a constant Kd, such as this one, the MTZ exhibits "square-root spreading" as it moves along the bed. For example, by comparing the length of the MTZ at the end of a 3-m-long bed versus a 1.5-m-long bed, the MTZ should be 20.5, or 1.414 times as long. This is in contrast to a system with a favorable isotherm, where the MTZ does not spread. Based on our results, the square-root spreading factor should be applied when scale-up design is done for the full-scale columns from the pilot-scale data. The bed depth of the full-scale zeolite system is 3 m compared to the bed depth of 0.67 m for the pilot-scale. The spreading factor of the MTZ due to the bed depth difference is 2.13 [(3/0.67)0.5] resulting in a predicted MTZ length of 42.6 cm (2.13 20 cm) for the full-scale system. This MTZ is only 14% of the total bed height indicating that good zeolite utilization will be obtained for the full-scale system.
CONCLUSIONS AND RECOMMENDATIONS
Pilot-scale testing of a chabazite zeolite system was performed to validate small-scale column test results, evaluate operating characteristics, and predict performance of a full-scale system. Sluicing tests performed after each exhaustion cycle indicated that spent zeolite can be sluiced easily and completely from zeolite vessels as long as adequate prefiltration of wastewater is accomplished. The granular anthracite prefilters performed adequately when wastewater was pretreated by clarification. Contaminants which elute from tiny quantities of spent zeolite left in the column (unseen during visual inspection) did not reach significant concentrations in the system effluent. The pilot testing also illustrated that the radioactive strontium concentration in the zeolite column effluent was greatly influenced by sodium concentration. Sudden increases in sodium concentration of the wastewater must be prevented; otherwise, premature breakthrough of strontium from the zeolite system could result. Breakthrough data and the Rosen model were used to determine a MTZ length of 20 cm, which was comparable to the MTZ for smaller-column tests. The pilot test results indicate that a full-scale zeolite system can be expected to perform well as long as suspended solids and sodium concentrations of the wastewater are maintained at low concentrations.
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45-3
IMMOBILIZATION OF SOLID WASTE USING CEMENT MATERIAL INCLUDING INORGANIC ADMIXTURE FOR HIGH FLUIDITY

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ABSTRACT
We have investigated the properties of cement paste made with a newly developed admixture to verify its suitability for solidifying incombustible radioactive waste. This new inorganic admixture is an alternative to the organic compounds used so far to raise the fluidity of cement paste. The paste obtained using this flew inorganic admixture has the low viscosity necessary for filling the voids between wastes. The distribution coefficients for plutonium and americium in the system of water and powdered cement paste containing the inorganic admixture are over 10,000 ml/g. We carried out various tests on the property of the solidified product, and the results indicate that good integrity is maintained for a long time.

INTRODUCTION

Immobilized radioactive waste must offer long-term integrity when disposed of in a repository. The immobilized waste contains radioactive nuclides, so excess release of these nuclides must be avoided. Many procedures for the solidification of waste have been developed, and some have been put into actual use in radioactive waste treatment facilities (1,2). Cement is an attractive material for this type of disposal, since when solidified it has great integrity and thus offers the potential for containment of radioactive nuclides for long periods. When cement is used for this purpose, however, it is necessary to include an admixture in the mix of cement, water, and aggregate so as to obtain a paste with high fluidity.

We have developed a number of cements for use in waste solidification (3,4). Generally, organic admixtures have been used to obtain the required fluidity, but it is thought that organic admixtures might decompose after long periods in a repository atmosphere. Carbon dioxide generated by the decomposition process may damage the immobilized product, such as by cracking of the cement phase. In order to avoid this type of concern, we had developed a inorganic admixture, for calcium aluminate system, which yields a high-fluidity cement paste (4). In the case of using this inorganic admixture for another cement material system, for example blast furnace slug, the cement paste was slowly stiffened, and lost the fluidity immediately. We have developed another type of new inorganic admixture for blast furnace slug system in order to solve the stiffening time and the immediate fluidity.

The requirements of a cement-based material used to immobilize waste are listed below.

The material should have good integrity under disposal conditions.

The cement paste must have high fluidity in order to fill the voids between particles of solid waste.

The product should be capable of containing radioactive nuclides for long periods.

The additive we selected contains no organic component, so the first requirement is satisfied. We carded out various tests, both to measure the property of the material selected and to confirm having the other requirements for cement paste.

MATERIALS

The cement formulation investigated in this work are listed in Table I. We used blast furnace slug to ensure mild heat generation. A three-components admixture is used to maintain paste fluidity and promote paste hardening.

PROPERTIES

The Effect of Inorganic Admixture on Dispersion of Dement Particle

Generally, cement particles condense in water because they lack an electric charge. The particles can be dispersed by the addition of sodium lignosulfonate. We found that the addition of inorganic admixtures containing sodium phosphate, instead of sodium lignosulfonate also dispersed the particles (Fig. 1). These additives generate no carbon dioxide under disposal conditions. We selected a combination of these inorganic compounds as an admixture for high-fluidity cement paste so as to ensure long-term durability.

Properties of Cement Paste

We investigated the properties of cement paste including the chosen inorganic admixture by measuring consistency and stiffening time. Two admixtures were chosen for these tests, one consisting of sodium phosphate only and the other containing sodium phosphate, an inorganic carbonate, and calcium aluminate. The sodium phosphate and inorganic carbonate enhance fluidity, while the calcium aluminate promotes hardening of the cement paste.

Consistency is expressed in terms of the time taken for all the paste to flow out of the test cone shown in Fig. 2. This is the method specified by the Japan Society of Civil Engineers. Rapid loss of fluidity causes the difficulties to operate the solidification system. Figure 3 shows the consistency of the two types of cement paste.

To use of three-components admixture gives longer-term fluidity than the sodium phosphate alone; it is possible to adjust the fluidity period from 20 seconds to three hours, which is adequate for use in the immobilization of solid waste.

The stiffening time of the cement paste was measured by implementing the proctor needle penetration test (as defined by the American Society of Testing Materials). Figure 4 shows the results of the test. Proctor needle values of pressure for initial and final stiffening time are used in the figure; initial and final stiffening times are defined as pressure values of 3.5 MPa and 28 MPa, respectively. Generally, rapid stiffening leads to the formation of voids or cracks in the cement product, and slow stiffening requires large curing area. To avoid this problem, it is necessary. to adopt a stiffening time of over 10 hours or so. The initial stiffening time when only sodium phosphate is used as the admixture is 54 hours. Consequently, we added calcium aluminate to promote paste hardening. With the three-components admixture, which includes calcium aluminate, the initial and final stiffening times are 21 hours and 24 hours. This stiffening time is satisfy to avoid large curing area in the plant. We thus decided that this three-component admixture containing sodium phosphate, an inorganic carbonate, and calcium alurninate was the best.

Product Properties

We investigated the two of the mechanical properties of the solidified product, compressive strength after 28 days curing and bleeding ratio, based on the cement paste described in Table I. When the water/cement ratio ranges from 0.38 to 0.47, the compressive strength after 28 days curing is from 31 to 41 MPa. These values indicate that no degradation of compressive strength occurs when the three-component admixture is added. The bleed on the waste was found zero. These results lead us to conclude that this water/cement ratio is suitable for the immobilization of solid waste. Using batch experiments, we measured distribution coefficients to investigate the radioactive containment of this cement material. Powdered cement grout was the initial solid phase. Plutonium, americium, and carbon were added to the solution. The distribution coefficients of plutonium and americium are found to be over 10,000 ml/g, and the value for carbon is 1,000 ml/g (Table II). These large values for plutonium and americium result from the cement products, which make the solution basic. They satisfy the requirements for containing radioactive nuclides in cement materials.

Immobilization of incombustible waste

Immobilization tests were carried out with various types of simulated waste in 200-liter drums. Figure 5 shows the immobilization test equipment. Pipe waste immobilization tests were implemented using pipes of 13 mm. in diameter and 800 mm.

in length. A total of 804 pipes were arranged lengthwise in a 200-liter drum. The drum was not vibrated during injection of the cement paste, and the rate of injection was 12 liters per minute. A filling ratio of 99.4 vol. % was obtained. If the drum is vibrated, a higher filling ratio might be obtained. These tests indicate that the cement containing the inorganic admixture to maintain fluidity is a superior of material immobilization for solid waste.

CONCLUSION

We investigated the properties of cement materials containing the new inorganic admixture that we have developed. The cement paste was found to have sufficient fluidity for immobilization of waste, and it has the potential to contain radioactive nuclides. Our results suggest that cement including sodium phosphate, inorganic carbonate, and calcium aluminate is suitable for the immobilization of solid waste.

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45-4

EVALUATION OF CEMENTITIOUS MATRICES CONTAINING SPENT ION EXCHANGE RESINS BY SEM AND X RAY DIFFRACTION TECHNIQUES

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ABSTRACT

This study is related with spent ion exchange resins beads (irradiated and non irradiated) embedded in cement matrices. Matrices containing sulphate resistant cement, blast furnace slag, zeolite and irradiated and non irradiated resin beads have been studied using SEM (Scanning Electron Microscopy) and X Ray diffraction techniques. EXRDA (Electron X Ray Diffraction Analysis) techniques were also used in this study.

Hydration has been developed at room temperature and has been interrupted with isopropyl alcohol at 3, 7, 14 and 28 days.

A durability study was also done. Eighteen months old samples were submerged in a synthetic sea water solution at 70 C during a period of four months.

From the results observed in these samples it is inferred that matrices containing blast furnace slag and zeolite present a better performance related to: removal of the calcium hydroxide from the matrix avoiding its undesired presence on the solidified product; diminution of the blast furnace slag content, which means that it is acting as a sacrifice material so protecting the zeolite; stability of the zeolite, allowing it to function as an inorganic ion exchanger particularly effective for Cs 137. The addition of Na(OH) in a concentration 1,5 Molar in the studied compositions produced no substantial improvement of the solidified product.

INTRODUCTION

Ion exchange resins are used for decontamination of different liquid streams in nuclear power plants. Management of spent ion exchange resins comprises the following actions: waste characterization, treatment, immobilization in a suitable matrix, waste form characterization, interim storage and final disposal in a repository. The purpose of these actions is to obtain a solidified and adequately contained end product and to meet the acceptance criteria related to: mechanical and

chemical stability, low leaching rates and radiation and thermal resistance.
With this objective, several formulations were designed. There was observed that an acceptable product is obtained using a matrix containing sulphate resistant cement and blast furnace slag in a 1:9 ratio. This matrix accepting 12 weight percent of dry resin. The blast furnace slag acts as a sacrifice material consuming the Ca(OH)2 produced on cement hydration.
Another interesting option is the inclusion of zeolites in the formulation. Zeolites are minerals naturally abundant in our country. They are natural pozzolanas of volcanic origin composed of hydrated aluminosilicates containing sodium, potassium and calcium. They present high thermal stability and resistance to high pH's. The inclusion of zeolites produces the following:

A better confinement of radionuclides, being particularly selective for caesium. As a consequence, lixiviation is lowered.

Utilizes the Ca(OH)2 set free by cement matrix hydration to form binding compounds.

Improves rheological properties of the paste.

The main disadvantage of zeolite is its reactivity in cement matrix. This is a pozzolanic reaction and can be limited by factors like temperature, calcium hydroxide availability and pore water.
There was observed that an acceptable product is obtained using a matrix containing sulphate resistant cement, blast furnace slag and zeolite in a 10:75:15 ratio.
Durability can be defined as the resistance of materials to physical, chemical or physico-chemical aggressions, being them of internal (intrinsic) or external (extrinsic) nature.
The aggressive environment can degrade the matrix by means of two different mechanisms. The first is dissolution (filtration, leaching, etc.) of soluble components. Dissolution starts at surfaces and propagates through the interior producing an emptying and a weakness of the structure. The second is chemical transformation. Chemical reactions may involve the formation of compounds of lower density that produce expansion stresses that may crack the matrix. If a chemical reaction occurs through solution the products can occupy the pores but if reaction occurs at a solid surface (solid-liquid reaction or topochemical reaction), expansive stress may deteriorate the structure.
Between the different types of chemical aggressions these can be mentioned: sulphate and chloride attacks, sea water and soft water attack, carbonation, alkali-carbonate and alkali-aggregate reactions.
Most common physical aggressions are freezing and thawing and drying and wetting cycles.
Porosity, pore structure and diffusivity of cement matrices exert a relevant control on the ingress of potentially deleterious substances.
In addition to avoiding the ingress of these substances, it is important to reduce the content of vulnerable phases in the cement pastes. Some vulnerable phases are: calcium hydroxide and hydrated tricalcium aluminate. Some of the compounds that produce expansive stresses are: Ettringite ($3CaO.Al_2O_3.3CaSO_4.32H_2O$ - very common), Gypsum ($CaSO_4.2H_2O$ - common) and Thaumasite ($CaSiO_3.CaSO_4.CaCO_3.15H_2O$).
EXPERIMENTAL PROCEDURE:
Samples of twelve different cementitious compositions were prepared. To simulate the waste, a mixed resin bed was prepared using LEWATIT M500 KR/OH (anionic exchanger) and LEWATIT S100 KR/H (cationic exchanger) in a 1:1 ratio. This resin bed was treated using $NaNO_3$ and pH was then adjusted using NaOH.
Part of this bed was irradiated using Co^{60} gamma radiation with a total dose of 108 rads. This is equivalent to the total irradiation dose expected until complete decay of the involved radionuclides. Samples were prepared using irradiated and non irradiated resin beds.
Hydration was developed at room temperature and was interrupted with isopropyl alcohol at the approximate ages of 3,7,14 and 28 days.
Table I shows the eight elementary compositions used. The other four correspond to matrices with irradiated resins that had the same composition of their counterparts with non irradiated resins. Samples with numbers 2,4,6 and 8 were prepared with non irradiated and irradiated resin beds. For each of these compositions the labeling has the subscript a) for non irradiated and b) for irradiated resins.
TABLE I
RESULTS

The following summarizes the results of this study:-Composition 1: Sulphate Resistant Cement (ARS) + Blast Furnace Slag (BFS): At the age of seven days the sample is compact. Hydration products cover ARS and BFS particles. We can see calcium silicate hydrate (CSH) of types I,II and III at the first stages and types II and III later. The diffraction pattern shows gehlenite (2CaO.Al2O3.SiO2.8H2O) and Ca(OH)2. The later increasing in content as hydration develops.

Composition 2a: ARS + BFS + Resin Bed (not irradiated): At the age of seven days the sample is compact. Three different zones can be distinguished: the bead, the matrix and the interface between them. Beads of approximately 500 mm appear separated from the matrix. Most of the beads show an irregular surface. Between the resin beads and the matrix there is a 30 mm gap, showing that a contraction followed the setting of the mix. Some of the beads are cracked and show solid paste inside their cracks, showing that the cracks existed prior to mixing.

Composition 2b: ARS + BFS + Resin Bed (irradiated 108 rads): At the age of seven days the sample is compact. Beads appear contracted with approximately 30 mm of gap between the matrix and them. Cracked beads have material in their crevices. Hydration products are CSH types I and II at the beginning of hydration and types II and III later.

Composition 3: ARS + BFS + Zeolite: Hydration is slower than in the first case (composition 1: ARS + BFS). At 14 days the sample is compact. Cement peaks (C3S, C2S, C4AF) decrease their heights as hydration develops. The principal hydration product is CSH type II at the first stages and type III later. There appears also CaCO3. Zeolite peaks correspond to clinoptilolite, heulandite and quartz. The height of the first two and the blast furnace slag broad peak decrease slowly.

Composition 4a: ARS + BFS + Zeolite + Resin Bed (not irradiated): The addition of zeolite improves workability. At the age of fourteen days the sample is compact. Blast furnace slag and zeolite peak heights diminish with time. There is a 30 mm gap between beads and matrix and the bead surfaces show less irregularities than in the composition 2a case.

Composition 4b: ARS + BFS + Zeolite + Resin Bed (irradiated: 108 rads): At the age of seven days the sample is compact. Beads appear separated from the matrix and their surface is less irregular than in the composition 2b case. Some beads appear cracked with solid paste inside crevices. Hydration products present are CSH types I, II and III. No Ca(OH)2 is detected. At the interface zone we observed needle like and sheet like structures (see Fig. 3).

Referring to recent experiences in Winfrith (AEA Technology), the effect of Na(OH) addition in the mix water was studied. According to this experiences, in a blast furnace slag plus ordinary Portland cement paste, the addition of Na(OH) would accelerate the hydration of Blast Furnace Slag. The effect of this is to reduce the availability of calcium ions which can exchange on the resin bead and thereby also reduces the amount of crystals formed which have the deleterious effect on the cement system. The concentration used was 1,5 molar in the mixing water.

Composition 5: ARS + BFS + Na(OH): At the age of three days the sample is compact. Addition of Na(OH) to the BFS + ARS mix accelerates hydration. Cement phases extinguish more quickly than in the composition 1 case. CaCO3 and Ca(OH)2 are detected. SEM observations show CSH of types I and II, prevailing type II. Needle like structures are profuse.

Composition 6a: ARS + BFS + Na(OH)+ Resin Bed (not irradiated): At seven days the sample is compact. The following phases appear: C3S, C2S, C4AF and CaSO4.2H2O. Also CaCO3 and Ca(OH)2. There is a 30 mm gap between the beads and the matrix. The interface matrix-bead is composed of a smooth and continuous surface.

Composition 6b: ARS + BFS + Na(OH)+ Resin Bed (irradiated 108 rads):At the age of seven days the sample is compact. Addition of Na(OH) accelerates hydration and beads appear with little irregularities. Crevices in some beads seldom contain paste inside. Observed phases are CSH types I,II and III and Ca(OH)2. The interface zone presents needle and globular like structures; also Ca(OH)2 hexagonal crystals.

Composition 7: ARS + BFS + Zeolite + Na(OH): At the age of three days the sample is compact. Addition of Na(OH) to the mix accelerates hydration. The CSH type III is predominant. The Ca(OH)2 appears in small quantities and extinguishes with time. The blast furnace slag and the zeolite react because their peaks diminish with time.

Composition 8a: ARS + BFS + Zeolite + Na(OH) + Resin Bed (not irradiated): At the age of fourteen days the sample is compact. Ca(OH)2 appears in small quantities.

Composition 8b: ARS + BFS + Zeolite + Na(OH) + Resin Bed (irradiated: 108 rads):

At the age of three days the sample is compact. Addition of Na(OH) to the mix accelerates hydration. A 30 mm gap appears again between matrix and pearls. Some pearls have crevices but without paste inside them. No Ca(OH)2 is detected.
Table II shows the X ray diffraction results of eight compositions (the eight shown in Table I but with irradiated resins). The most abundant three crystaline phases are shown for each composition.
TABLE II
Table III shows the X rays diffraction results of the eight compositions shown in Table I that were submerged in artificial sea water solution at 70C during 4 months. The second column shows bulk compositions while the first column shows the reacted surface.
TABLE III
According to semiquantitative composition profile determinations, at a depth of 3 milimeters, elementary compositions resemble bulk compositions.
Some examples of scanning electron microscopy photographs are shown in Figs. 1 to 6.

Figure 1 (Magnification 95 x) shows a fracture surface of a sample of composition NO 4 at seven days of hydration. Two beads of exhausted ion exchange resins embedded in a cementitious matrix. Pearls have approximate diameters of 500 mm and there is a gap of approximately 30 mm between the bead surface and matrix, indicating a contraction of the pearls.
Fig. 1.
Figure 2 (Magnification 800 x): typical microstructure of fracture surface of a cementitious matrix (composition NO 8). Cement and Blast Furnace Slag particles covered with hydration products.
Fig. 2.
Figure 3 (magnification 2800 x): sample of composition NO 8. Part of the matrix that had been in contact with a bead (bead crater).
Fig. 3.
Figure 4 (Magnification 2240 x) shows the surface of a sample of composition NO 4 after hot sea water immersion. Hexagonal crystals of Mg(OH)2 (Brucite) are homogeneously distributed over the surface. They have diameters of 8 mm and thiknesses of 2 mm.
Fig. 4.
Figure 5 (Magnification 2080 x): surface of sample of composition NO 6 after hot sea water immersion. Hexagonal crystals of Brucite (Mg(OH)2) at the left and trigonal crystals of Vaterite (CaCO3) at right.
Fig. 5.
Figure 6 (Magnification 2080 x): surface of sample of composition NO 3 after hot sea water immersion showing crystals of Gibsite (CaCO3).
Fig. 6.
ANALYSIS
The most abundant components of hydrated cementitious matrices are amorphous calcium silicate hydrates (CSH phase). The other components are crystalline phases detectable by X ray diffraction techniques and unreacted fractions of cement, blast furnace slag and zeolite.
Table II shows the X rays diffraction results for the different compositions examined. The three most abundant crystalline phases are indicated in order of preponderance.
An important characteristic feature of the X ray diffraction pattern is the extensive and broad amorphous peak due to the vitreous BFS. This vitreous BFS is the slag capable of react in alkaline media to form binding compounds. The height of this peak depends on the content of amorphous BFS in the matrix and it is observed that this peak diminishes its height as hydration develops.
This fact is interpreted inasmuch as the slag that has reacted doesn't contribute any more to the broad peak height. Blast Furnace Slag delays Portland Cement hydration and consumes the Ca(OH)2 forming more CSH phase. Samples containing ion exchange resins appear more corroded and expanded after hot sea water immersion than samples with the same composition but without resins. The addition of Na(OH) accelerates hydration.
In order to obtain concentration profiles, semiquantitative analysis of element concentrations were performed using EDAX techniques. Most abundant elements were calcium, silicon and aluminum. Less common elements were sodium, potasium,

manganese, iron, magnesium and chlorine.
Significant variations are observed in the first millimeter and at a deep of three millimeters the concentration is the same as the bulk concentration.
Generally, the superficial content of chlorine is higher and the alkaline (sodium and potassium) content is lower than the corresponding bulk concentration. This indicates a solubilization of the later in the immersion solution.
Another important observation from EXRDA analysis is the reduction of the calcium/silicon ratio at the surfaces. Quartz also appears in the diffractograms as a secondary crystalline phase. Diffractograms doesn't show calcium hydroxide at surfaces,pointing out its dissolution and the corresponding reduction of the calcium/silicon ratio.
The prevailing phase at surfaces is the brucite or Mg(OH)2 which appears forming a uniform cover of hexagonal crystals of uniform size, even though their medium dimension varies upon sample composition. Another important phase at surfaces is CaCO3 in two crystaline forms; calcite and vaterite. This phases seem to form a protective layer against further attack.
The bead contraction reduces the contact area with the matrix, so reducing the pathways for radionuclide difussion. This contraction indicates that there is a margin for an eventual disruptive expansion of the beads.Ion exchange resins inclusion in the tested concentrations reduces slightly their resistance to saline atack.
The addition of Na(OH) in a concentration 1,5 molar in these compositions produced no substantial improvement of the solidified product. The corrosion effect of hot sea water is more relevant in samples prepared with Na(OH) in the mix water.
Zeolite reduces the dissolution of alcaline elements since their relative concentration reduction is lower in samples were this component is present.
CONCLUSIONS
From the results observed in these samples it is inferred that matrices containing blast furnace slag and zeolite present a better performance related to: removal of the Calcium Hydroxide from the matrix avoiding its deleterious effect on the solidified product; diminution of the blast furnace slag content, which means that it is acting as a sacrifice material to protect the zeolite from a complete reaction and stability of a zeolite fraction, allowing it to function as an inorganic ion exchanger particularly effective for Caesium 137.
The addition of Na(OH) solution in the mix water reduces the resistance of these samples to hot sea water attack.
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45-5
NONDESTRUCTIVE ON-LINE VISCOSITY AND IMPEDANCE METHOD FOR STRENGTH CONTROL OF CEMENTED RADIOACTIVE WASTE
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ABSTRACT
Two methods impedance and viscosity were employed to predict the on-line compressive

strength of cemented radioactive waste. Influence of frequency change in the impedance method has also been studied. The two parameters used were water to cement ratio and the amount of simulated radioactive waste. The basic objective was to find mathematical correlation's by these two methods for the prediction of 7 and 14 days compressive strength. The reliability of these methods has also been studied. Two types of waste have been prepared. Waste 1 consisted of a constant amount of radwaste but varying water to cement ratio. Whereas in waste 2, water to cement ratio as well as the amount of radwaste was varied. The experiments were carried out using sophisticated equipment to get precise data.

INTRODUCTION

Immobilization of radioactive waste is an important environmental issue of concern for both public governmental considerations. Radioactive waste is primarily produced during the operation of nuclear reactors. It could also arise from medical, institutional and research power. Proper and safe disposal requires not only chemical, physical and radiological characterization to ensure safety, but also public acceptability and satisfaction.

Coolant water in a nuclear reactor gets contaminated by the radionuclides as it passes through the coolant channels. In order to reuse this water, different chemical separation techniques could be employed for the purification. The use of ion-exchange resins beds and filters are two common techniques to recover usable water.

Once these radionuclides are collected on the ion exchange resins, the use of cement to immobilize this waste has proved to be economical and safe. Different standards have been set up by different nuclear regulatory agencies, which have to be met in order to ensure public safety.

Process control has several advantages above final product testing. The fundamental idea is to control the process in such a way that the prescribed standards of the final product is ensured. In order to reduce the secondary waste streams, time, as well as costs, several non-destructive, on-line compressive strength prediction methods have been employed by different agencies. From the concrete building industries, a few well known method are: ultrasonic, rebound hammer, maturity and viscosity.

OBJECTIVES

This project was primarily designed to study the physical characterization of cemented intermediate and low level waste. Previous works done by Slate (1) and Lewis (2) have shown that the impedance and viscosity methods are most promising. Thus this project was specifically carried out to study these two methods. The basic objectives were to find the mathematical correlation's between strength and other parameters such as impedance, plastic viscosity, C/(W+R) ratio etc., so that strength could be predicted by simply applying these equations. In this manner, one can save time and cost. The secondary waste stream production can be reduced. Hence the quality of the final products can be controlled. Table I and table II indicates the samples compositions. The two parameters that have been employed were water-cement ratio and amount of radwaste. The reliability of the experimental apparatus was also an integral part of this report.

Table I

Table II

IMPEDANCE METHOD

It is a known fact that cement paste consists of a conductive substance such as water with various compounds. A cement paste also consist of hydrated products, for example gel water and solid products of hydration. If an electric circuit is complete, current will flow through the circuit and changes in the resistance can be measured as the process of hydration takes place. In the impedance method this idea has been utilized to determine any mathematical correlation between resistance (impedance) and strength of cemented radwaste. Impedance is the term used to describe the resistance offered by a substance to the flow of charges.

Mathematically: $Z = V/I$, Z= impedance, V= voltage, I= current.

The fundamental idea of using an electric circuit model to find relationship between strength and impedance came after the review of McCarter's report (3). In that report, McCarter has described investigations into a method for monitoring the temporal changes in gel-space ratio within cement paste employing wide-band frequency response. Upon realization of the fact that impedance of the cement paste could be used as an important parameter to predict strength, an impedance meter was

built at KEMA. The meter consists of a sinusoidal voltage source. It measure the voltage difference between the electrodes and the current which exists between the electrodes. From voltage and current the resistance is calculated and given in the input. The integrated circuits are designed in such a way that it treats imaginary part of the input as absolute. It achieves this via a root mean square (R.M.S) converter. An operator can select the band of frequency as well as resistance scale. In this experiment, via impedance meter, four different ranges of frequencies (0.1 kHz, 1 kHz, 10 kHz, 100 kHz) have been used.

VISCOSITY METHOD

Rheology is the study of the change in form and flow of matter. In general, rheology is concerned with materials whose flow properties are more complicated than those of a simple fluid like water (1). A non-Newtonian fluid is a fluid which does need a minimal shear stress to produce a shear rate. In other words the deformation starts as soon as the shear stress passes some critical value. Water is a Newtonian fluid whereas concrete is a non-Newtonian fluid. The Bingham plastic model is used to describe the behavior of a type non-Newtonian fluid. The general equation is of the form:

$T = g + hN$

where T= torque, g=yield stress, h=plastic viscosity, N=velocity.

A rheology meter has been used in this method. Rheology meter enables the users to measure viscosity and yield point of Newtonian and non-Newtonian fluids. An operator can pre-set the shear rate. Hence controlled shear rate is possible. The rheology meter consist of a 1.5 inch diameter cylinder. The height of the cylinder is 5.5 inch. The speed of the rotating bob can be controlled. The measuring bob experiences torque from the flow resistance of the fluid (i.e. viscosity of fluid). Hence shear stress offered by the fluid can be detected. The cylinder with fresh cement paste in it is attached to the rheology meter in such a way that the rotating bob sinks in it. Upon the activation of the rheolab, user can set different combination of speeds and time scale. Hence software is used to manipulate the obtained data. Upon the completion of test, the viscosity and yield stress can be obtained for the given sample.

RESULTS AND DISCUSSION

General

The experiments were done in three stages. Consequently the results are discussed in the same manner. The first stage was done to confirm McCarter's report and to see the overall behavior of the frequency on the resistance between two electrodes. In the second part of the experiment, the impedance method has been utilized for the prediction of compressive strength. And in the third and last stage viscosity method has been studied. Impedance and viscosity methods were done simultaneously. The relationship between strength and C/(W+R) has also been included to verify the current work.

Frequency Effects

In this stage the overall behavior of the frequency on the impedance has been studied. The range of frequency which has been used was 0.1 kHz, 1 kHz, 10 kHz, and 100 kHz respectively. Two electrodes made up of brass were inserted into the fresh cement paste, which was inside the mold. The electrodes were 93 mm apart. Current started flowing upon the completion of the circuit. Impedance was recorded for the first 24 hour of the curing stage. The sample composition for all the samples was as follows:

TABLE III

Certain phenomena have been observed. Figure 1 shows that impedance is not only a function of time but also frequency. The resistance decreases at least for the first 3-5 hours. Then gradually increases at approximately constant rate. As it is apparent from figure 1 that different bands of frequency do effect the impedance. Eight samples were made for this purpose. For each frequency, two samples have been utilized. Thus the curves shown in Fig. 1 for the four frequencies are the average of two respective frequencies. There was a significant difference among the impedance vs time curves for .1 kHz, 1 kHz, and 10 kHz. It has been noticed that as the magnitude of the frequency increases, the impedance decreases. This behavior can be explained by realizing the fact that higher frequency means higher number of charges (electrons) passing in a unit of time. Hence keeping all the other parameters constant, the current rate increases and consequently the impedance decreases. According to McCarter's report, the enhanced conductivity at higher

frequencies is due to surface conduction effects (due to enhanced charge mobility at higher frequencies) on the adsorbed gel water (3). When the cement paste was fresh (i.e. first 3-5 hours) there were lots of free electrons available for current. In other words the mobility of charge carriers was much higher. Hence impedance decreases for the first 3-5 hours. But as the chemical reaction takes place (i.e. process of hydration continues) and free ions form different inorganic substances via bonds, the number of ions or electrons decreases. The aqueous phase in the cement paste decreases as the gel absorbs water, which means that less number of carriers are available for current flow. Hence resistance increases.
Fig. 1.
Another behavior was noticed in Fig 1. For the frequency range of .1 kHz, the impedance reaches a stable state after 9 hours. In other words for that frequency range, more resistance cannot be noticed by the impedance meter. The frequency range has to be increased in order to make impedance change visible. For the case of the highest frequency (100 kHz in our case), this stage is reached after 25 hours. It is also apparent from Fig. 1 that there is a significant difference at the initial impedance values among the four frequency ranges. This suggest that for the correlation factor between strength and impedance, the ten minutes resistance can be easily used. It also indicates that any frequency could be used in the impedance method. This will enable the operator to predict the strength after 10 minutes. Hence one can save a considerable amount of time.
Another part of this experiment that was coupled with the above part (i.e. impedance vs time relationship for different frequencies) was to see the difference between the magnitude of the low and high frequencies. According to McCarter's report, the difference between the low and high frequency resistivity curves will thus represent the proportion of current flowing through the (adsorbed) gel water and will be a quantitative measure of the gel-space ratio within the paste (3).
Hence percentage frequency effect is a term describing the difference in the low and high frequencies. Mathematically:
((Pl - Ph)/Pl) * 100
where
Pl = low frequency
Ph = high frequency
In our case low frequency was 0.1 kHz and high frequency was 100 kHz. Figure 2 shows the PFE vs time relationship. Few things should be noticed. For the first hour the curve increases. This implies that the difference between the low and high frequency dominate the initial period completely. In other words the frequency effect is significant and increases due to the different nature of charge mobility and gel-space ratio at these two different frequencies. Gradually the PFE factor decreases indicating the strengthening of the cement and lower mobility of the charges (i.e. Ca2+, OH- and absorption of water) as the process of hydration takes place and hence the frequency effect gradually diminishes. After 25 hours PFE will be a constant since the resistance at low and high frequencies will be constant. Hence one can conclude from Fig. 2 that the first 6-8 hours are very important as far as the frequency effect is concerned.
Fig. 2
IMPEDANCE METHOD
Impedance is the quantitative measurement of the resistance offered to the flow of charges by a substance. Slate's report (1) has indicated that impedance value could be used to predict the strength of the cemented radwaste. Hence in this part of the experiment an impedance meter has been used. The sample composition used for this purpose was given in Table I and Table II. In order to calculate the accuracy in the results and to see the reliability and reproducibility of the method, three molds of each composition have been prepared. Hence in the category of waste 1, 21 samples were made (Waste 1 has a fix amount of radwaste but varies in water-cement ratio). And in each mold there were 3 prisms. Hence in each category of waste 1 there were 9 prisms. Five of them were used for 7 days strength and the remaining 4 were used for 14 days strength. After the preparation of each sample, the viscosity test has been performed. These tests were followed by impedance measurements. Ten minutes impedance values were noted. Seven and 14 days strength tests were performed. These values are the averages of the three samples of each category.
Fig. 3.
Fig. 4

The equations describing the curve fit for the 7 and 14 days strength prediction taking impedance (Z) as an independent variable are as follows:
S = 32.96 ln Z - 62.02 (7 days strength prediction) (1)
S = 27.87 ln Z - 38.54 (14 days strength prediction) (2)
The best curve fits for both equations were found to be logarithmic. In order to justify the logarithmic curve fit for the strength versus impedance data points, some important points need consideration. Three types of curve fit were taken into consideration. Those were logarithmic, power and linear. In order to see the best curve fit mathematically, the sample coefficient of determination r2 has been considered. (Values of r2 between zero and unity indicate the relative strength of the relationship between two given variables (4)). Consequently if the value of r2 is close to unity, it simply suggest that the regression equation will give quite accurate predictions. In our case, for the 7 days strength prediction, the following values were obtained:
Table IV
Hence the r2 values for the logarithmic equation is closest to one, thus justifying the best curve fit. Similar was the case for 14 days strength equations. Intuitively speaking, another phenomena explaining this logarithmic behavior between strength and impedance could be described by realizing the fact that relationship between the hydration process versus time and strength versus time are also more or less logarithmic. Hence one can conclude that since hydration is strongly coupled with impedance, the strength versus impedance curve will likely be logarithmic. Thus this combination of mathematical background with intuitive knowledge about cement paste behavior, justifies the above curves.
In the waste 2 category, the relationship between 7 and 14 days strength and impedance were found to be exponential. This difference between logarithmic curves in waste 1 and exponential curves in waste 2 could be explained by looking at the sample compositions of waste 1 and waste 2. In waste 2 the amount of radwaste is also changing. Thus this indicates that the change in the amount of radwaste affects the bond formation process during hydration. Different sets of chemical reactions take place due to the presence of different amounts of radwaste. Consequently this affects the charge mobility and hence the impedance. However the impedance method remains reliable. The sample correlation factor for 7 and 14 days were .909 and .918 respectively. The following equations, where Z = impedance, and graph shows the relationship.
S = 86.14 exp (-0.016 * Z) (7 days strength prediction) (3)
S = 99.34 exp (-0.017 * Z) (14 days strength prediction) (4)
Fig. 5
Viscosity Method
A rheology meter coupled with rheolab software has been used in this experiment. In the beginning of the experiment, several small tests were done to develop a general understanding about the viscosity method.
The viscosity measurement was divided into four intervals. These intervals were controlled by different shear rates. It has been noticed in the previous experimental work that the cement paste follows the Bingham model after going through a series of different shear rates. It has also been found experimentally that interval number 4 gives the best correlation factor. Hence as a part of test preparation all the samples were passed through these four intervals and only the measurement from the fourth interval was taken into account. The shear rate varies from 31.853 (1/sec) to 191.12 (1/sec). This variation of shear rate via rotating bob was pre-set for 30 seconds.
The objective here was to find the optimum mathematical model between 7 and 14 days strengths and viscosity. The relationships were logarithmic. The following two equations can be used to predict the 7 and 14 days compressive strength.
S = 36.18 ln m + 19.89 (7 days strength prediction) (5)
S = 41.76 ln m + 20.98 (14 days strength prediction) (6)
The following two figures illustrate the best curve fit with data points.
Fig. 6
Fig. 7
Both of these graphs behave reasonably well. Not only mathematically but also intuitively it makes sense. The sample coefficients, r2, were .993 and .994 for 7 and 14 days strength prediction equations respectively. Thus indicating an accurate estimate of the strength. Secondly, this best curve fit also indicates that if the

viscosity of the paste is less than approximately 0.2 (Pa.s), the strength will be zero, a likely behavior. If the viscosity of the fluid is very low, that indicates lots of fluidity (i.e. higher w/c ratio) and consequently the strength will be very small. Thus on both grounds, theoretical as well as practical, these graphs make sense.

In the waste 2 category, 7 and 14 days strength were found to be exponential again. The sample correlation factor were .945 and .962 for 7 and 14 days strength respectively. Hence both of the relationships, i.e. strength versus impedance and strength vs viscosity, were found out to be exponential. The following equations and figure illustrate this behavior.

$S = 15.36 \exp(0.45 * m)$ (7 days strength prediction) (7)

$S = 16.57 \exp(0.47 * m)$ (14 days strength prediction) (8)

Fig. 8.

It has also been found that the strength of the cemented radwaste decreases in a linear fashion as the amount of the radwaste increases. The following two equations shows the relationship, where R = amount of radwaste. Figure 7 shows this behavior.

$S = -2.61*R + 67.96$ (7 days strength prediction) (9)

$S = -2.91*R + 76.25$ (14 days strength prediction) (10)

Fig. 9.

Strength as a Function of C/(W+R)

Previous work at KEMA was done to find the strength prediction equation in terms of its contents, namely cement, water and amount of radwaste. A ratio had been developed for this purpose which is C/(W+R). KEMA's report (5), indicates the following equation and figure.

$S = 41.3 (C/(W+R)) - 17.3$ (28 days strength prediction) (11)

Fig. 10.

In order to check and compare the results obtained from this experiment with the previous work, similar graphs and equations have been produced. The only difference is that the following graph is for 7 and 14 days whereas the above graph is valid for 28 days. The equations and graph are as follows:

$S = 32.18 (C/(W+R)) - 7.32$ (7 days strength prediction) (12)

$S = 28.16 (C/W+R)) - 5.87$ (14 days strength prediction) (13)

Fig. 11 7 & 14 Days Strength Versus C/(W+R)

One can see that a similar relationship is obtained. Also, since the 7 and 14 days strengths are lower than the 28 days strength, the graph is less steep. As the number of days for the strength test increase, the graph becomes more steep, thus approaching like Fig. 10. Figure 11 verifies the validity of this experiment.

Also to understand the logarithmic nature of most of the results, two fundamental relationships have been tested. These were strength development and 7 and 14 days strength versus w/c ratio. The best curve fit for both of them were found to be logarithmic. The equation for strength development were as follow with the sample correlation factor of .953.

$S = 11.35 \ln(\text{days}) + 23.77$ (14)

Hence it indicates that most of the important relationships among cement parameters are related to each other logarithmically. As the process of hydration continues, it influences the strength, and resistance in a logarithmic fashion. Hence it is not surprising that most of the results were found to be logarithmic. Similarly, the equations describing the 7 and 14 days strength are as follows:

$S = -55.99 \ln(\text{days}) + 9.69$ (7 days strength prediction) (15)

$S = -49.25 \ln(\text{days}) + 20.55$ (14 days strength prediction) (16)

The following figures illustrate this behavior.

Fig. 12.

Fig. 13.

SOURCES OF ERROR

Even though every effort has been made to avoid any possible error, there were a few things which might have caused the deviation in the data:

1. It was noticed that during the impedance measurement the electrodes were sometimes not straight. Hence this would cause variation in the impedance value. A difference of 0.5 ohms was observed.
2. The accuracy of the weight machine was up to three decimal places. Sometimes four decimal places were required to make w/c ratio and other values as precise as possible. However this could not be obtained with the available balance. Hence this small difference in the samples might affect the overall measurements. However all

the samples were consistent
3. It has been anticipated that due to low resistance offered by the steel mold, the current might flow from there. Since the idea was to measure the resistance between the electrodes (in other words only through the cement), the impedance value will be the combination of steel resistance and cement resistance. A difference of 25% in the impedance value is possible. The basic idea behind this project was to see on the pilot plant basis whether impedance could be used to predict the strength or not. On commercial scale, this phenomena however will never happen in practice because of the larger distance between the electrodes.
4. The measuring cup in the viscosity method was sometimes tilted. Hence it could have shifted the cement paste on one side, which might have cause errors in the viscosity and yield stress.
5. Due to the tedious nature of sample preparation and measurement, the time interval between sample preparation and ten minutes impedance measurements could have varied five minutes. Hence it can affect the consistency of the ten minutes impedance measurements.
6. Random human errors could have occurred in recording weights, in samples preparation, in viscosity measurements, and in the impedance measurement.

CONCLUSIONS

This project was designed to study the feasibility of impedance and viscosity methods for the strength control of cemented radwaste. The fundamental objective was to obtain for optimum mathematical relationships between strength and other parameters on a laboratory scale. Both methods, impedance and viscosity, have shown reliable results.

In the impedance method the optimum relationship between strength and impedance was found to be logarithmic. The reproducibility of this method turned out to be very good. The average value of all the standard deviations in the impedance method is 3.23 ohms. This indicates that certainly impedance method could be used on an upgrade scale for compressive strength prediction. In the waste 2 category, the relationship between strength and impedance was exponential. Hence this shows that the variation in the amount of radwaste can play significant role chemically and electrically. Also it is important to realize that only one variable is required, namely impedance, to predict the strength.

In the viscosity method, the optimum relationship was found out to be logarithmic. Whereas in the waste 2, it was exponential. The reliability of this method is extremely good. The average value of its all standard deviations is .323 Pa.sec. Again only one variable is required to predict the strength. Both of these methods are not only efficient but also consume very short amounts of time to give results. Hence on the commercial scale, this factor can play a vital role.

The logarithmic and exponential nature of most of the results could be explained by realizing the fact that most of the fundamental parameters in cement paste are related to each other logarithmically. Two good examples will be strength development and strength and w/c ratio. Since these two parameters (strength development and w/c ratio) are strongly coupled with viscosity, impedance, it is likely that the other relationships will follow the same behavior. The strength versus c/(w+r) graph also follows the same behavior which has been obtained in the previous work at KEMA. Hence this also proves the validity and reproducibility of this research study.

RECOMMENDATIONS

1 The viscosity meter does not give accurate results for water to cement ratio below .40. A w/c ratio below 0.40 results in very stiff cement paste. During the measuring stage, the rotating bob creates a vacant space in which the cement paste does not return back. Hence the measurement will not be true. To avoid this one can shake the measuring cup to make sure that there is no space or some kind of instrument could be employed to remove the vacant space.
2 A remote technique for samples preparation and measurements will be preferable so that the consistency could be maintained throughout the experiment, more accurately.
3 In order to measure the resistance only between the two electrodes on pilot plant basis, some insulated materials for the mold should be used in order to avoid any kind of mold resistance effects.
4 Special care should be taken in order to make sure that the electrodes are straight into the samples. Some kind of clip could be used for this purpose.
5 A weight balance with greater precision and accuracy (i.e. 4 decimal places) would

be preferable.
6 More data points are encouraged in order to obtain smooth curves and to increase the range of applicability of the equations.
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45-6
THE RADIOACTIVE WASTE TREATMENT PLANT OF THE PUSPIPTEK SERPONG NUCLEAR RESEARCH CENTER (INDONESIA) AND ITS INCINERATION UNIT
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ABSTRACT
In the framework of its radioactive waste management activities, TECHNICATOME, a subsidiary of the French Atomic Energy Commission, developed, realized and commissioned a low level radioactive waste treatment plant for BATAN, the Indonesian Atomic Energy Authority.
Main processes implemented are: evaporation, compaction, incineration, cementation of liquid and solid waste.
The waste treatment plant and the different processes are described; particular emphasis is given to the incineration unit which was commissioned two years ago and which has been since that time fully in operation. The organization of the Project and work share between
INTRODUCTION
In order to reduce the amount of waste to be stored and to ensure storage satisfactory environmental safety conditions, selective treatments, adapted to the specific type of waste considered, are carried out in specialized installations.
For several years TECHNICATOME has been developing, building and putting into operation treatment and packaging processes adapted to different types of waste. By controlling the entire waste treatment process from production to storage, TECHNICATOME has been able to set up for BATAN (BADAN TENAGA ATOM NASIONAL, the Indonesian national atomic board), a radioactive waste treatment plant for its nuclear research center on the PUSPIPTEK site.
The installation was built in collaboration with BATAN, thus ensuring a transfer of the technologies required to enable the Indonesian staff to operate the installation.
PRESENTATION OF THE NUCLEAR RESEARCH CENTER
Within the PUSPIPTEK scientific center (research, science and technology center) BATAN (the Nuclear Energy Commission) is responsible for studying, developing and implementing nuclear energy processes. For this purpose, it is equipped with:
- a pluri-disciplinary reactor,
- a radio-isotope production laboratory,
- a reactor fuel production laboratory.

Waste produced by these different installations is treated by a waste treatment plant designed by TECHNICATOME and built in collaboration with Indonesian companies.
ORGANIZATION OF THE OVERALL PROJECT
This has been implemented as part of a joint project between TECHNICATOME and BATAN:
TECHNICATOME was responsible for overall engineering, supplying specific nuclear equipment, monitoring assemblies and tests.
BATAN was responsible for the infrastructure, erection and putting the equipment into operation.
INSTALLATION DESIGN PRINCIPLES

Waste Characteristics

Waste produced by the research installation located on the Center comprises:

liquid waste with an activity less than 5.4 108 Bq/m3 (2.10-2 Ci/m3),

semi-liquid waste (ion exchange resins: IER) with a maximum activity of 3.7 109 Bq/m3 (0.1 Ci/m3),

solid technological waste (plastic bags, gloves, cotton, paper), packaged in 100 l metal drums. This waste is separated by the producer:

- by type of waste (compactable, non compactable, alpha emitter),
- by activity, so that, the contact dose rates of the packaged drums do not exceed 0.25 m Gy/h (2.5 mRad/h),

biological waste with a maximum activity of 3.7 105 Bq/kg (10-5 Ci/kg) packaged in plastic bags,

items of clothing (clothing, masks) packaged in 5 to 10 kg plastic bags with a dose rate of under 2.5 10-2 mGy/h (2.5 mRad/h).

Installation Functions

The main purpose of the installation is to:

treat radioactive waste so as to produce waste packages consistent with existing storage standards,

carry out research and development work in the field of waste treatment.

The individual processes to which wastes are subjected depend on the type of waste involved. The table provided in appendix 2 shows the successive operations performed for different types of waste.

In order to carry out waste treatment of the following functions are available:

collect waste from different producers and transport it to the treatment station,

treat liquid waste by:

- evaporation (liquid waste) or incineration (combustible waste),
- packaging in a concrete matrix,

treat solid waste by:

- compacting (compactible waste), incineration (combustible waste) or placing in a concrete shell (other solid waste),
- packaging in a concrete matrix,

treat semi-liquid waste by:

- chemical pre-treatment,
- packaging in a concrete matrix,
- treating items of contaminated clothing in a nuclear laundry,

store the packages produced prior to being transported to the repository,

treat clothing.

Besides R & D studies are performed in the laboratories.

Basic Design Principles

The installation is comprised of three buildings:

the main building housing the processing equipment and administrative services,

a building containing the auxiliary units (various utilities, generator),

a building housing the package storage area.

The main building is comprised of two blocks:

a "cold block" containing the administration building, general technical facilities and the centralized control room,

a "hot" block sub-divided into distinct areas corresponding to differents types of process.

The entrances and exits to/from the "hot" block are provided with air locks:

"hot" and "cold" staff cloakrooms,

equipment air locks for each area.

TREATMENT OF AQUEOUS LIQUID WASTE

Purpose of Treatment

Treatment by evaporation makes it possible to obtain:

concentrate containing all the active matter which can be embedded in a concrete phase,

distillate free from active matter wich can be released into the environment.

Transport to the Plant

Liquid waste is transported from the storage tank (reactor, laboratories, workshops) to the treatment installation in an 8 m3 tanker.

Tanker design complies with the regulations governing the transport of dangerous materials in France. The vehicle is therefore authorized to use the public highway.

The tanker is provided with its own pumping system which ensures:

  the filling of the tanker from the producer's tanks,
  or draining into the treatment plant tanks.
A control console enables the operator to pilot transfer operations from the truck
The tanker is provided with all the necessary biological protection and safety equipment (overflow reservoir, gas filtration, ...) to ensure safe operation and the radiation protection of the operating personnel.
Waste Storage
Waste storage is ensured by four 50 m3 tanks each equipped with its own homogenization system either to collect the effluent or to perform mixing operations to ensure that the physicochemical characteristics of the waste comply evaporator requirements
Evaporation
The installation has a capacity of 750 l/h. The volume reduction ratio is 50. The dry condensate and distillate concentrations are 250 g/l and 2.5 10-4 g/l respectively.
The installation consists of:
  a tubular exchanger, with thermosiphon circulation, into which the waste to be evaporated is introduced. Live steam heating is used to ensure the vaporisation of liquid wastes,
  a setting tank at the exchanger outlet is used to separate the liquid and steam phases. The steam phase is fed into the separation column. The concentrate obtained is removed from the bottom of the tank,
  a separation column. The liquid carried over into the steam is separated from this steam by washing through the exchanger plates using a reverse flow of demineralized water,
  a condenser which ensures condensation of the steam phase and a cooling unit which is used to cool the distillates.
The distillates are stored in two 50 m3 tanks. These tanks are provided with sampling and chemical ratio measuring systems.
With the exception of the chemical reagent circuits, which are controlled locally, the system is controlled from the evaporator control room.
Evaporation operations are automatic. In the event of a malfunction, safety devices put the installation into a standby mode.
Waste Packaging
This unit is equipped to package the following waste in concrete shells:
  evaporation concentrates,
  ion exchange resins,
  solid non compactible waste and 200 l drums of already compacted waste.
Packaging is carried in a concrete matrix.
The unit is designed to operate 8 hours per day.
Operations are controlled from a console in the control room.
Concreting operations are automatic. In the event of a malfunction safety devices put the installation into a standby mode.
Storage
The following functions are ensured in the storage room:
  storage of concrete shells,
  storage of shells and drums of waste from other waste production centres,
  loading of shells and drums for transport to the repository.
Storage capacity is 1500 drums and 500 concrete shells. These are handled with a crane and pallet stackers.
TREATMENT OF SOLID WASTE
Purpose of Treatment
The purpose of solid waste treatment is to process:
  non combustible waste by compaction,
  combustible waste by incineration.
The wastes obtained (incineration ash, compact slabs) are then packaged into drums or concrete shells in a concrete matrix.
Transport to the Installation
The waste is transported by truck from its production site to the installation. It is packaged in 100 l metal drums. Wastes, which can be incinerated, are separated on the production site from other wastes.
Storage before Treatment
A special room is used for the storage of:

100 litre drums of non combustible waste,
empty 200 l drums into which the first drums are compacted,
100 litre drums of combustible waste.

Compaction of Drums Containing Non Combustible Waste

The 100 l drums of solid waste which cannot be incinerated are compacted into the 200 litre drums using a 600 kN hydraulic press.

This press is provided with a nuclear ventilation circuit to ensure containment during operations.

The installation is controlled and operated manually and locally.

Packaging the Compacted Units

After being filled with waste, the 200 litre drums are back-filled with cement grout and vibrated to ensure that the compacted waste is correctly packaged (see paragraphe 5.2.5.).

Incineration of Combustible Waste

The incineration unit burns:

solid combustible waste:

- cotton,
- polyethylene,
- PCV

combustible liquid waste:

- oil,
- scintillation liquids,
- tributyl phosphate (TBP) diluted in dodecane,

biological waste.

The installation has a capacity of 50 kg/h for solid waste and 20 kg/h for liquid waste.

The 100 l drums of combustible waste are transferred from the 100 and 200 l drum storage room to the incineration room (see 5.3.3.).

The contents of the drums are packaged in cardboard boxes containing 3.7 kg of waste. This is done in glove boxes.

The cardboard boxes are loaded automatically into the incinerator through a double air lock chamber.

The liquid waste to be incinerated is stored in a 1 m3 tank (waste not containing TBP) and in a 0.1 m3 tank (waste containing TBP). Wastes are then transferred into the incinerator after possible dilution, (to limit the quantity of TBP in the liquid waste to be incinerated).

Technological solid waste is packaged in plastic bags and stored in deep freezers. These bags are then packed into cardboard boxes in glove boxes and then fed into the incinerator.

The installation consists of:

a furnace with two chambers:

- a first chamber where oxygen reduced combustion takes place for solid waste (temperature between 800 and 1000C),
- a second post-combustion chamber where combustion occurs in gas produced in the 1st chamber during solid waste and liquid waste combustion operations. The temperature in this chamber is approximately 1000 to 1200C,

a dilution chamber to reduce the gas temperature to 180C by dilution in the ambient air,

a two-stage gas filtration unit:

- approximate filtration using a bag filter with automatic unclogging through a continuous flow of compressed air,
- filtration of radioactive particles using HEPA filters,

a gas washing unit to:

- cool the gas produced by filtration,
- neutralize the acid gases produced during combustion.

This unit consists of a venturi and a separation column.

an extraction assembly comprising:

- a gas reheater,
- gas extraction fan,

Gas activity is continually monitored before release into the atmosphere.

Waste incineration is performed in 6 day cycles. Each day includes a 6 hour period during which the unit incinerates waste and an 18 hour period during which the unit is maintained at a constant temperature (thermal standby).

Ash is removed from the furnace every day before incineration.
The unit is controlled from a control room equipped with a control panel.
The installation operates automatically. In the event of a malfunction, safety devices put the installation on standby.
Packaging Incineration Ash
Ash is removed from the first combustion chamber and the bag filters through an air chamber, where it is cooled before being recovered, with continuous confinement, in a metal drum which has been previously filled up with encapsulation materials.
Ash is embedded in cement using a drum-rotation mixer.
The drums are taken to a room (see paragraphs 5.2.5.1) while the concrete sets, before being taken to the storage room after the packages have been radiologically controlled.
TREATMENT OF COMBUSTIBLE LIQUID WASTE
This waste is delivered in carboys and transferred to the combustible liquid waste storage area (see paragraph 5.3.6.1.). Tanks containing liquid waste with TBP are separated from those without TBP.
The waste is incinerated as described previously.
TREATMENT OF SEMI LIQUID WASTE
Aims of Treatment
The aim of treating semi-liquid waste (ion exchange resins) is their cementation.
Transport to the Installation
The ion exchange resins are transferred from the storage tanks where they are produced to the installation by a 1.5 m3 tanker.
Storage of Ion Exchange Resins
Ion exchange resins are transferred by a hydraulic system into a 5 m3 tank where they are stored before packaging.
Packaging Ion Exchange Resins and Handling Packages
This is described in paragraphs 5.2.5. and 5.2.6.
TREATMENT OF LAUNDRY
Working clothing and gas masks used on the site installations, are cleaned in the laundry.
This treatment involves:
  sorting out contaminated laundry for treatment as solid waste (incineration/compaction),
  cleaning non contaminated laundry for re-use by the staff.
Laundry is sent to the station packaged in plastic bags in 100 l drums (5 to 10 kg load per drum).
Preliminary sorting operations involve measuring the dose rate in contact with the plastic bags. Bags with doses rate greater than a certain pre-determined value are sent to the solid waste treatment system.
The unit is composed of:
  washing machines,
  rotary driers,
  ironing presses,
  gloves boxes (used to open plastic bags).
Radiological controls are used to identify non contaminated clothing.
Decontaminated laundry is ironed using the ironing presses. A sewing machine is used to repair clothing.
The machines are controlled locally by an operator.
R & D LABORATORIES
The aim of these laboratories is:
  to improve waste treatment operations,
  to perform routine physico-chemical evaluation analysis.
They are equipped with standard physics and chemistry materials: PH-meters, spectrophotometers, conductivity meters, spectrometers, etc ...
Fig. 1. Functional Organization of the Station.

Session 46 -- Poster - Mixed Waste II
Co-chairs: Andrea M. Faucette, EG&G-RF;
Randal J. Roberts, WHC
46-1
REAL TIME NON-DESTRUCTIVE EVALUATION OF
MIXED WASTE AT HANFORD

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ABSTRACT

Two shielded mobile Real-Time Radiography (RTR) units were designed, constructed, and mobilized to rapidly perform Non-Destructive Evaluation (NDE) of more than 1,400 drums and 200 large plywood ("B-25") boxes of mixed solid waste at Hanford in early 1994. NDE was used to better define the contents of each container of potentially mixed waste without opening them or unnecessarily exposing personnel or the environment to hazardous materials.

The purpose of the project was to determine, on a package-by-package basis, which containers were likely to contain hazardous constituents and to document the actual type of hazard.

NDE was accomplished by utilizing high-energy, collimated X-ray equipment that penetrated the containers onto an intensifier to produce an image of the materials in the container. A bipolar metal-ceramic X-ray tube with variable energy output was used to allow permanent recordings of the images of material of varying densities. Images were stored on VHS video tapes.

BACKGROUND

Advanced Systems Technology, Inc. (AST) was requested to assist Westinghouse Hanford Company in the characterization of containerized mixed waste for the Hanford Tank Farm Backlog Waste Program. AST was responsible for the design, construction, mobilization, and operation of mobile NDE systems for the remote examination of more than 1,400 207- to 320-liter drums and 200 B-25 plywood containers. The design incorporated the use of two different state-of-the-art Real-Time Radiography (RTR) systems, one for standard metal drums and one for large wooden boxes.

The purpose of the project was to assist Westinghouse Hanford Company to safely determine the contents of individual waste packages as a precursor to waste segregation and subsequent storage or disposal as mixed or low-level waste, respectively.

DESIGN CRITERIA

During the early stages of design, AST established a set of customer-driven criteria:

1. Hardware must be packaged into a mobile system for ease of transfer to customer facilities. Because the equipment may be moved frequently, stout equipment and transport systems must be provided.
2. To obtain the clearest X-ray image possible, provisions must be provided to tilt, turn, and rotate individual waste packages both vertically and horizontally.
3. Provisions must be provided to examine one-hundred percent of each waste package.
4. To meet the letter and intent of As Low As Reasonably Achievable (ALARA) goals, waste packages must move quickly through the process with little or no hands-on handling (hence the need for remotely-operated conveyors, close-circuit television systems, and X-ray shielding to limit radiation exposure to less than 0.5 mRem at the outside surface of the units).
5. The operator must have a reference system in order to document the exact location of individual objects in each waste package.
6. All NDE data must be capable of being recorded both on VHS tapes or CD-ROM discs.
7. If requested by the customer, the ability to mark and scan in a host of other data, such as waste generator records, for inclusion directly onto tapes or discs must be provided.

EQUIPMENT AND OPERATIONS

NDE equipment was provided in two trailers, one "drum" unit to handle up to 320-liter drums and one larger "box" unit to handle 10-cubic meter "B-25" boxes. The conceptual layout of the NDE box and drum units is shown in Figs. 1 and 2. Each unit contained the following equipment: collimated X-ray machine, shielded x-ray chamber, control console, two high voltage transformers, oil-to-air cooler, bipolar/metal

ceramic image intensifier tube with variable energy output and video camera, video display, remote video camera with monitor, video recorder, character generator, a Digital Imaging Database System, and a conveyor system with turntable.
Fig. 1. Conceptual layout of NDE drum unit.
Fig. 2. Conceptual Layout of NDE box unit.
Mixed waste was moved through the drum unit by placing the waste package through a side access panel (Fig. 3). Waste package numbers were recorded and a location template placed on the side of the waste package to enable NDT technicians to pinpoint the location of individual items. The drum was then moved through the unit remotely through an "L" shaped conveyor system where it was monitored via close circuit TV systems from the control room.
Once the drum reached the shielded x-ray vault (Fig. 4), radiography was initiated. Hydraulic systems in the vault provided for both vertical and horizontal movement. After the operator was satisfied with the results of the NDE, the drum then existed through the back of the unit where it was removed from the conveyor system. Control room operators recorded drum numbers directly onto VHS tapes. Operators had the option to highlight drum internals via computer-generated characters and lines. Interpretation of drum contents was documented via by operators into a sophisticated data input and compression system.
The box unit operated similarly to the drum unit except for larger hardware requirements and the need for boxes to be placed onto a conveyor and turntable system that was partially erected outside the end of the unit. In addition, the box unit was fitted with two shielded retractable walls used to house the X-ray system and intensifier. Boxes were conveyed into a large, shielded vault and the doors shut before RTR was initiated. Due to the depth of each waste package (2 m), NDE was performed twice on each box by conveying the box back out the end of the unit where it was turned 180 degrees on a turntable; radiography was then performed a second time. The conveyor system was again reversed once radiography was completed to remove the box.
WASTE ANALYSIS PLAN AND QUALITY ASSURANCE
All work was performed in accordance with the Waste Analysis Plan agreed to between the signatories of the Hanford Federal Facility Compliance Agreement: EPA-Region X, Washington Department of Ecology, and Department of Energy-Richland Field Office (1). The Waste Analysis Plan was based upon criteria established by the customer (2) and AST.
A Quality Assurance Project Plan (QAPP) was written for this project and included all NDE procedures, software requirements, document control processes, change control/change request systems, non-conformance report formats, data compression criteria, internal quality control checks, preventative maintenance, document control, calibration procedures, and materials identification and control (3).
Fig. 3. Mixed transuranic waste entering mobile NDE drum unit.
Fig. 4. Waste package on turntable in shielded NDE vault.
PERMITS, TRAINING, HEALTH AND SAFETY
AST developed a Health and Safety Plan that was fully protective of operators, technicians, observers, and nearby customer personnel. Since none of the waste containers required opening to determine their contents during this phase of the backlog waste program and sufficient shielding provided for X-ray equipment, all ALARA goals were met.
Approvals were obtained from the State of Washington Department of Health for the use of mobile X-ray equipment, including an operations permit from the Office of Radiation Protection (4).
Non-Destructive Testing (NDT) personnel were certified per the requirements of ASNT SNT-TC-1A (5) and included both Level II and Level III NDE technicians. AST operators were used that had previously provided NDE services on more than 13,000 low-level, mixed, and transuranic wastes nationwide.
Other job-specific training focused upon radiation safety fundamentals, use of installed radiation survey instruments, engineered safety features of radiation-generating devices, non-radiological hazards, and operating procedures and practices. Site-specific training included radiation worker, OSHA 1910.120 hazardous material (40 hour), and Hanford orientation. Full medical exams, including vision and hearing, were also provided for all personnel.
OBSERVATIONS
A large campaign of mixed wastes stored in drums and plywood boxes was safely

completed by remote evaluation via mobile real-time radiography. Due to the design and operation of the mobile NDE units, clear images of the contents of the waste containers were safely and quickly obtained and stored on VHS tape. Based upon NDE data, more than 1,600 containers were quickly and economically sorted and decisions made as to their ultimate disposition.

Other observations noted from this waste characterization campaign were as follows:

1. Custom-fabricated NDE equipment were fabricated and mobilized 3,000 miles in a relatively short period of time to meet ambitious schedules and expectations.
2. Throughput easily reached a minimum of 25 drums or 6 boxes per shift.
3. No unusual occurrences or safety incidences were noted.
4. ALARA goals were met.
5. The design allowed required minimum waste handling. Once placed onto the conveyor system, waste packages required no additional handling.
6. The x-ray images produced were extremely clear due to the design of the unit and the ability to x-ray individual waste packages in all planes. As a result, the presence of hazardous materials or excess liquids were easily noted.
7. If coupled with a Non-Destructive Assay (NDA) component, a complete mobile NDE/NDA system could be readily adapted for use in transuranic waste certification programs, such as those required by Waste Isolation Pilot Plant.

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46-2

DEFENSIBLE ALPHA CHARACTERIZATION OF GLOVE BOXES AT D&D SITES

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ABSTRACT

Hundreds of glove boxes contaminated to levels above transuranic (TRU) waste limits will be taken out of service at various nuclear facilities over the next few years. If the contamination is categorized as TRU waste, the costs associated with packaging, transportation, size reduction, storage, and eventual disposal are estimated at about $396,000 per box. These glove boxes are typically cleaned to below the low-level waste (LLW) limit of 100 nCi/g before being taken out of service, but current measurement technologies have inadequate sensitivity or accuracy or both to provide defensible measurement of the remaining low-level contamination. Given a defensible measurement protocol, these glove boxes could be disposed of as LLW with an estimated savings of $391,000 per box. Considering all the boxes to be disposed, the savings add up to many millions of dollars. Long Range Alpha Detection technology, developed at Los Alamos National Laboratory for site characterization, shows great promise for accurately measuring the low levels of contamination remaining in glove boxes so as to defensibly characterize them as LLW. In laboratory tests using a 1m by 1m by 0.66m mock-up glove box constructed from sheet metal and safety glass, we have demonstrated the ability to measure contamination levels as low as 0.0008 nCi/g.

INTRODUCTION

Decommissioned glove boxes, which have been previously used for direct handling of nuclear materials, must have their residual contamination levels measured to satisfy nuclear safeguards and waste disposal requirements. Safeguards requirements can be met if the uranium or transuranic element content of the glove box is measured and

reported to the nearest gram. DOE Order 5633.3 specifies a measurement uncertainty "goal" of 25%. Waste disposal measurement requirements specify that contamination levels should be defined in terms of the alpha-emitting activity per unit weight of the glove box. Glove boxes with greater than 100 nCi/g of alpha activity are declared as transuranic (TRU) waste while glove boxes with contamination levels less than 100 nCi/g are declared as low-level waste (LLW). Draft Revision B of DOE Order 5820 requires that measurements of glove box contamination levels made in support of disposal as LLW must have a "95% confidence level" that the waste does not exceed the threshold level for TRU waste.

In preparation for disposal, decommissioned glove boxes are cleaned of residual alpha contamination before they are disconnected from the exhaust ventilation system. Cleaning methods typically remove most paint and deposits and are successful in reducing residual alpha contamination to around the 100 nCi/g level. If in situ measurements indicate contamination levels are still above 100 nCi/g, further cleaning can be performed. If credible measurements can be performed in situ, glove boxes can be cleaned efficiently to LLW levels. Once the glove box is removed from the ventilation system, further cleaning is not possible for safety and economic reasons. Glove boxes removed from the ventilation system and determined to be contaminated above LLW levels must be disposed of as TRU waste.

At Los Alamos National Laboratory, the current cost of size-reduced packaging, transportation, and burial of a glove box classified as LLW is about $5,000 per glove box. If the glove box is declared TRU waste, the current cost of packaging, transportation, and disposal amounts to about $396,000 per glove box. Thus, having a nondestructive, in situ method for characterization of glove boxes will considerably reduce the cost of disposal.

CURRENT TECHNOLOGIES

Measurement methods that are currently used to determine contamination levels in glove boxes are based on counting gamma or alpha emissions from 235U or TRU isotopes. For gamma detection, sodium iodide or another scintillating material is used. The gamma detection techniques suffer from one or more of the following drawbacks:

- Non-uniform distribution of the contaminants,
- Insufficient detection efficiency, or
- Background interference.

In practice, measuring glove box contamination by gamma counting is only feasible once the glove box is removed from the work area to a low-background area. If the glove box is then found not to be cleaned to LLW limits, the glove box must be disposed of as TRU waste.

Alpha counting methods using both solid-state and gas proportional detectors have been used for monitoring the inside surfaces of glove boxes. Alpha counting methods provide excellent measurement sensitivity but are subject to measurement errors in practice. Because alpha particles have a range of only a few centimeters in air, measurements must be performed on virtually every square centimeter of the interior of the glove box. The accuracy obtained using manual alpha surveying techniques depends heavily on the skill of the measurer and the care exercised in covering all internal surfaces. It is difficult to position alpha detectors in hard-to-reach areas yet these areas are likely to have the highest contamination levels and are the most difficult areas to clean.

We believe that the Long Range Alpha Detection (LRAD) technology (1) developed at Los Alamos National Laboratory will provide the measurement breakthrough needed to defensibly measure alpha contamination inside glove boxes destined for waste disposal. This paper describes exploratory tests using a mockup of a glove box and a prototype glove-box monitor.

REVIEW OF LRAD TECHNOLOGY

The range of typical alpha particles in air at STP is about 3.5 cm. Over this distance, an alpha particle loses its kinetic energy by creating a proportional number of ion pairs. These ions have a lifetime on the order of seconds, so by using either an air current or an electrostatic field one can transport these ions a fair distance. If the ions are transported to an electrode, they can produce a small current proportional to the kinetic energy deposited by the original alpha. Thus, total current is proportional to the total amount of activity present. This ion transport and collection is the fundamental idea behind LRAD technology. Detector systems based on LRAD technology (2) have been built and successfully used to

monitor soil surfaces and objects.

LABORATORY TEST SETUP

To investigate the problems involved in glove-box monitoring, we built a prototype LRAD glove-box monitor and attached it to an all-metal mockup of a glove box constructed from Unistruts and sheet metal. The length of the detector was 15 cm and the inside diameter was 8.9 cm (3.5 in.). It had a single ion collection grid which was kept at +300 V. The dimensions of the glove box were 1m high x 1m wide x 2/3 m deep. The front surface had two 8.9cm (i.d.) holes to represent glove ports. On one of these glove ports, we mounted the prototype LRAD detector and a fan with a variable power source. On the second glove port we attached an 8.9cm (i.d.) electrostatic filter to keep ions generally present in the room from entering the glove box. The fan generated air currents throughout the glove box to draw ions created inside the glove box into the detector. In a real glove-box monitoring situation, we envision that air drawn out of one glove port and through the detector will be returned to the glove box through the adjacent glove port, thus preventing contamination from leaving the glove box. An artist's conception of the LRAD glove box monitor is shown in Fig. 1. This figure does not show the ion filter used in the laboratory tests, but it depicts how the airflow will be circulated in the final implementation.

SOURCE RESPONSE OF GLOVE BOX MONITOR AS A FUNCTION OF AIR FLOW

For this test we used a 125,000 dpm (56.3 nCi) 238Pu source. The net weight of the test glove box was 69.5 kg, which meant that the contamination level was 0.0008 nCi/g. To qualify as LLW, a glove box must register alpha activity of 100 nCi/g or less. Therefore, a contamination level greater than 71,201 nCi would be required to elevate this glove-box mockup into the TRU waste category. On each of the six interior surfaces of the glove box we chose five positions to place the radioactive source, as shown in Fig. 2. The "front" surface refers to the side of the box with the glove ports, as shown in Fig. 1. The "right" and "left" sides refer to right and left from the point of view of a person standing in front of the glove box.

Fig. 1.

Fig. 2.

The fan shown in Fig. 1 was connected to a Variac. We took data at three different settings. The air flows at these settings were ~0.22 m3/min (setting A), ~0.35 m3/min (setting B) and ~0.57 m3/min (setting C) when measured in front of the detection grid.

From Fig. 3 one can see that at a contamination level of 0.0008 nCi/g we measured "detectable" current well above the background in all positions and that the response for a given position varies as a function of airspeed. At an airspeed of 0.35 m3/min, the overall detector response is more uniform than at 0.22 m3/min, i.e., the difference between the largest and smallest response measured was less for settings of 0.35 m3/min than for 0.27 m3/min. At the highest (turbulent) setting of 0.57 m3/min, the response was less than or equivalent to the response obtained at 0.35 m3/min. Additional data taken with the source placed on the other five surfaces of the box gave similar results to those shown in Fig. 3. The clear response to a point source placed at various locations within the glove box gives rise to the possibility that hot spots can be localized by covering various locations inside the glove box with an alpha attenuating material.

Fig. 3.

A second method for localizing contamination hot spots within a glove box is suggested by the variation in detector response as a function of the locations as seen in Fig. 3. This variation is caused by differences in ion transport and detection efficiency due to the airflow differences throughout the glove box. By directing the airflow at various locations throughout the glove box, we can make the detector response more uniform and determine additional information as to the approximate location of contamination. To determine the improvement in detector response as a function of source location that can be realized in this way, we constructed an air inlet nozzle capable of directing a stream of air into specific locations within the glove box. With the source located on the top surface at position #2 and with the air inlet nozzle directed toward the source, detector response improved from ~27% to ~85% of the response at position #1. We repeated similar measurements when the source was at position #4 on the right wall and found similar improvements in detector response.

Fig. 3. Response of the glove-box monitor to the source (0.0008 nCi/g) at the five locations on the left surface, as shown in Fig. 2. The Variac setting of A corresponds to an airflow of ~0.22 m3/min, Variac setting of B corresponds to an airflow of ~0.35 m3/min, and Variac setting of C corresponds to an airflow of ~0.57 m3/min.

SOURCE RESPONSE IN A GLOVE BOX WITH A GLASS WINDOW

After modifying the all-metal glove box mockup to include a safety-glass window, we conducted a series of tests to determine the effect of this nonconducting material on detector response. For these tests, we used the same source placed at the five locations on all inside surfaces of the glove box as before. No loss of signal that could be attributed to the presence of the glass window was seen. Although we did not repeat the airflow tests, we believe that detector response could be improved, as before, if the airflow is controlled.

EFFECTS OF LEXAN ON DETECTOR RESPONSE

To investigate the effects that a material capable of supporting a large static charge, such as plastic, will have on detector response, we replaced the safety-glass window with a sheet of Lexan. The same 125,000-dpm 238Pu source was placed on the bottom surface in the five locations shown in Fig. 3. The detector response observed was lower in all cases. In particular, no response was observed when the source was placed near the Lexan wall because Lexan acts as a sink for ions, which collect on the surface as a static charge, thus giving an overall lower response. No response whatsoever was measured when ions were generated near the Lexan wall. To demonstrate charge saturation, we increased the source strength to 500 Ci. In the mock glove box this activity is equivalent to 7 nCi/g of contamination. We placed this source at location #3 on the bottom surface next to the LEXAN wall, a location where we did not obtain a response from the 125,000 dpm source.

The result from this test with the 500 mCi source is shown in Fig. 4. It took about 2 hrs from the time the source was introduced into the mock glove box for the response to level off at 5000 fA. This long rise time was due to charging of the Lexan wall, which finally became saturated with ions. The detector response returned to its normal background level of 160 fA as soon as the source was removed. In conclusion, although we failed to detect contamination at the previous 0.0008 nCi/g level when one wall of the glove box was made of Lexan, we did manage to detect contamination at the level of 7 nCi/g, which is still well below the TRU threshold of 100 nCi/g.

Fig. 4.

CONCLUSIONS

In summary, all of our preliminary tests show that LRAD technology, once fully developed, will provide a defensible way of measuring contamination within glove boxes. Government guidelines require that we be able to reliably determine with reasonable accuracy that contamination levels are below 100 nCi/g to dispose of the glove boxes as LLW. Our test results show that we can measure much lower levels of contamination. Glove box monitors based on LRAD technology operate in situ and are nondestructive. They give results in real time, which will enable the user to get immediate feedback on how well the decontamination of a given box is proceeding. These monitors may also be able to isolate hot spots by taking data as various areas inside the glove box are covered.

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EFFECTS OF ATTENUATION PROPERTIES ON THE MULTI-DETECTOR OPTIMIZATION METHODOLOGY FOR RADIOACTIVE WASTE CHARACTERIZATION

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ABSTRACT
An optimization methodology has been previously developed to account for the spatial and statistical uncertainties associated with radioactivity assessment processes utilizing multi-response, non-destructive detection systems. The methodology is used to predict the optimal placement (or configuration profile) of detectors around a waste container in order to minimize spatial and statistical uncertainties. Radiation detectors located relatively close to a container of source materials are affected by an increased spatial uncertainty attributable to the unknown spatial distribution of source materials in the waste container. Detectors located relatively far from source materials are affected by greater statistical uncertainties arising from the random nature of radioactive decay and from fewer interactions within the detectors' sensitive volumes. The partial contributions of both spacial and statistical uncertainties to the assessment reliability depend on the measurement process and the assay system design. In this study, the optimization methodology is extended to account for source-and matrix-material radiation attenuation properties (which could be obtained from earlier chemical and physical waste characterization). The optimization methodology is based on minimizing a ratio--the End-Response Ratio--which is formulated to account for the combined effects of spacial and statistical uncertainties in the assessment process. The minimized End-Response Ratio corresponds to the optimal configuration (e.g., location of detectors) of the assay/detector system by utilizing local and global searching techniques. A computer code has been developed to facilitate application of the methodology. Use of the methodology leads to greater accuracy/reliability in the assessment results.

INTRODUCTION

Quantitative assessments of radioactivity have become widespread over the last fifty years and now exist in a number of fields. Such fields include but are not limited to industrial radiography, manufacturing quality control, power generation, diagnostic and therapeutic medicine, and radioactive waste management. The accuracy of a radioactivity assessment process is an important component to consider. Accuracy is especially important when results of the assessment are to be used to characterize radioactive waste materials before or after treatment/ handling, to make decisions related to health and safety, and/or to meet specific regulatory requirements. The accuracy of an assessment is affected by uncertainties inherent in the assessment process. Uncertainties in the assessment of radioactivity associated with nondestructive assay systems are attributable to instrumental and non-instrumental sources. Instrumental sources include uncertainties and errors in the assessment due to instrument limitations and configuration and to human factors involved in running equipment and interpreting results. Non-instrumental sources include uncertainties introduced by counting statistics, spatial and temporal distributions of the source activity in the container, and random phenomenon (e.g., involving radiation transport and interactions). Several optimization procedures have been employed to obtain better accuracy in assessment results. Shenhav and Ben-Haim (1) applied optimization principles to determine the minimum radioactive mass that can be detected by a nondestructive, passive assay system of randomly distributed materials. Yong et. al. (2) evaluated the uncertainty in radioactive waste inventory data by calculating the estimated range of activities and producing distributions for their predictions by numerical propagation of uncertain parameters through a model. Notea and Segal (3) applied a method based on developing a characterization function for the assay system that correlated measurable variables such as detector design parameters, detection limits, detector response and uncertainty. The method is limited to the expressible detector response and measurement uncertainty. Al-Ahmady and Huston (4,5) recently developed an optimization process for nondestructive assessment that is based on minimizing a function (the End Response Ratio, ERR) which accounts for both statistical and spatial uncertainties of the counting system. The value of the minimized function corresponds to the optimal assessment configuration.

In this paper, matrix and non-matrix radiation attenuation terms have been incorporated into the ERR optimization method to improve the design and performance of radioactivity assay systems for characterizing unknown and randomly distributed radioactive materials. The scenario investigated involves a single point source of radioactivity that can be at any random, unknown location within a cylindrical waste container. The contents of the container (source and matrix) result in attenuation

of the radiations being detected. A parametric study of the effects of different attenuations on the End Response Ratio for this scenario was performed. The results indicate that detector locations exist which allow uncertainties in the assessment process to be minimized (i.e., accuracy is improved). A computer program was developed to facilitate the investigation.

MATHEMATICAL CONSIDERATIONS

The ERR method incorporates both statistical and spatial sources of uncertainty in an unrestrictive assessment of randomly distributed radioactive materials. When a detector in the assay system is located relatively far from a radioactive source, statistical uncertainty in the detector reading increases because fewer interactions occur in the detector sensitive volume (leading to fewer counts). Radioactivity in the assessment process is represented by a randomly oriented point source that can be located anywhere within the confines of the container. If a source of unit activity is utilized for this treatment, the statistical uncertainty can be quantified by the standard deviation of the mean, S, of a set of n readings (counts) taken over equal time durations, t.

Eq. (1)

where si is the standard deviation of each element (i.e., reading) in the set and is given by Ci1/2. The term Ci represents the total counts recorded by the detector for reading i. The value of Ci can be estimated from,

Eq. (2)

where ti is the time interval of measurement i (assumed to be equal for all i here); n is the total number of elements in the set; and Fi is the Source-Detector Configuration SDC function. The function Fi represents the count rate expected to be measured for element i of the total set of readings and is determined by the geometry of the configuration utilized in the assay system. For the treatment examined here a simple SDC function F for a point kernel response is utilized. The detector count rate for uncollided particles (photons) for a unit-activity point source is,

Eq. (3)

where A is a constant which accounts for the detector's intrinsic efficiency and other conversion factors, N is the linear attenuation coefficient of the source-container matrix materials, D is the normal distance from the point source to the detector, a is the detector radius, a is the angle between the normal to the detector surface and the source direction, and d is the distance from the source point to the intersection point with the container surface of the line connecting the source point and detector.

The statistical uncertainty contributed to the assessment process is quantified by the ratio of the standard deviation (assuming Poisson statistics) in the overall detector(s) count mean to the count mean of the detector(s) for a complete set of readings. The complete set of readings is generated by spatial scanning over all possible locations of the unit activity source in the container volume and over all possible detector locations in the measurement volume. The measurement volume represents the space in which a detector(s) can be located. The source and measurement volumes are mutually exclusive since the testing is nondestructive. The statistical uncertainty contribution to the assessment process can then be expressed as,

Eq. (4)

where Ust is the statistical component multiplier of the ERR (5).

Spacial uncertainties are important when detector(s) are relatively close to the container surface and are caused by the unknown spatial distribution of the source material in the container. If source activity is being determined by the assessment, the value that is assigned to the activity will depend strongly on the location of the source in the container. Since the assessment is non-destructive, the point cannot be known and can theoretically be anywhere within the container. This unknown source location leads to a spatial uncertainty in the assessment process. The spatial component of the ERR was previously constructed from the ratio of the maximum to the minimum responses from the complete response set for the specific consideration. This component is based on the total detector counts for SDC functions of point kernel assay system configurations and can be represented by,

Eq. (5)

where Usp denotes the spatial uncertainty factor. By considering all possible source locations in the container and all possible detector locations (configurations), the

resulting set of counts per source location, per detector configuration is called the complete response set (4).
It is important to note that elements of the complete response set are assumed to be linearly independent of time, spatial distribution of unit activity sources, and the radioactive source mass. Linearity in source mass is valid when the radiation absorption characteristics of the non-source materials (including matrix materials, container material, and the medium between the detector and container) are constant or relatively constant over the assay time period. The linearity with the source spatial distribution exists if there is linear independence in the detector's responses to spatially distinct point sources of unit activity. The linearity in time is justified if the source activity remains constant over the testing period (4).
The statistical and spatial uncertainty components treated above are reasonably independent, therefore their contribution to the uncertainty of the assessment process can be evaluated from their product as,
Eq. (6)
where j is the number of detector configurations. If a constant time interval, t, is used for all the measurements then
Eq. (7)
The attenuation term in F can be studied to determine its effect on the ERR as a figure of merit to evaluate the best locations of detectors around the radioactive waste containers in the optimization process. The ERR can be perceived as a figure of merit which accounts for the overall contributions of spacial and statistical uncertainties. Larger values for the ERR indicate greater uncertainties; smaller values indicate less uncertainty. For a given source container, the detector configuration that results in the lowest relative value for the ERR will optimize the assessment since it will possess the least (relative) amount of overall uncertainty.
RESULTS AND DISCUSSIONS
A computer code has been previously developed (5) to facilitate implementation of the multi-response optimization methodology for radioactive waste management applications. Modifications have been introduced into the code in order to incorporate the effects of matrix material attenuations. The algorithmic format implemented in the code consists of objects that perform the following functions:
1. scan the radioactive waste container volume, as specified by the container geometry, to generate a sample set of the possible source spatial locations;
2. calculate the distance between every source point location and detector(s) location specified by the user; also calculate the attenuation distance between source to the container surface according to the SDC structure;
3. generate the complete response set {Ci} for all possible configurations according to sampled source points, number of detectors, and other considerations;
4. employ search techniques to determine Usp developed for each configuration;
5. compute the ERR for all configurations and determine the minimum ERR value in the set; and
6. employ a feedback routine to iterate over control variables until optimal values are obtained.
The modifications made to the existing program for this work involved addition of code to calculate the distance between the source point and the intersection point with the surface of the radioactive waste container located on the line passing through the point source and the point representing detector location. In this application, the code is used to perform a case study on a cylindrical radioactive waste container (drum); however, it can be adapted for other waste assessment optimization problems. Cartesian coordinates (x,y,z) were utilized in the scanning process; the waste container axis was taken to coincide with the z axis. The bottom center of the drum was taken to coincide with the origin.
Earlier analyses utilizing this optimization methodology (5) and container configuration showed that optimal placement of a detector system was achieved when detectors were placed at radially symmetric positions about the cylindrical container and at z equal to the midpoint of the drum length. The particular detector location depended on the number of detectors utilized in the assessment process. Step increments were applied in the three directions dx, dy and dz and were selected to be the same for all directions. As this increment becomes smaller the number of

calculations become larger and the code resolution increases. For scanning purposes, the container is assumed to be located within a closest fitting imaginary box when generating source points. The distance between each sampled source point and the drum center point, at each z increment, is compared with the waste container radius and rejected if it is larger. This technique excludes points which are not located in the drum, but which are sampled because of the scanning process. Due to results of previous runs (investigating optimum detector locations), the detector locations were restricted those lying in the plane given by z = 1/2 L, where L is the drum length. The code can be used to place detectors at any location in the measurement volume; however, run time increases.

Similar detectors are assumed in this example, and they are symmetrically located around the axis of the container at z = 1/2 L. The container used for this analysis was a 55-gallon drum of standard dimensions (~90 cm length and 60 cm diameter). The detector radius was assigned a diameter of 4.5 cm. The waste container cross section is uniform and is described by x2 + y2 = rO2, where ro is the radius of the container. A constant time interval is utilized for all measurements. A simple SDC configuration function F for a point kernel response is utilized. Distance D between detector point location (j) and source point location (i) is calculated by

Eq. (8)

where (xi,yi,zi) represents the source location and (xj,yj,zj) represents the detector location. The attenuation distance d in the SDC configuration function F is determined by substituting the parametric equations for a ray into the conical equations for a cylindrical surface and solving the resulting expression for the distance at which the ray intersects the surface. The ray was defined as originating at the source point and passing through the detector location.

The effect of different linear attenuation coefficients on the ERR was analyzed with the source configuration discussed above for a drum container. Linear attenuation coefficients investigated ranged from a theoretical value of zero (no attenuation) to one (the value approximately seen for photons in lead). Fig. 1 shows the resulting ERR plotted against the distance between the waste container surface and the detector for linear attenuation coefficients, u, ranging from 0.05 to 0.5cm-1. The minimum distance a detector can be located is zero, which represents a detector located immediately on the outer surface of the container. The y-axis of Fig. 1 gives the logarithmic (base-10) value of the ERR in order to show the wide range of values obtained. Although a minimum value of ERR vs. distance exists for each attenuation examined, the logarithmic scale does not clearly show it for all cases plotted. The introduction of attenuation into the analysis resulted in ERR values that spanned over more than 25 orders of magnitude for the attenuations plotted.

Figure 2 shows the relation between the linear attenuation coefficient and the log10ERRmin. The figure also includes plots of the components of the ERR, Usp and Ust. The figure illustrates the effects of attenuation on the statistical and spacial uncertainties and on the the combined uncertainties for the optimal configurations. As the attenuation increases, the spacial uncertainties (associated with the optimized ERR) become very large; however, the statistical uncertainties do not change dramatically. For the geometry examined, the spacial uncertainty component (associated with the minimum ERR) increased by over 40 orders of magnitude from zero to an attenuation of 1 cm-1. Since the ERR is simply the product of the spacial and statistical components, the log of the ERR is simply the sum of the log of the components. Since the optimal placement corresponds to the minimum ERR value of the set generated from different detector configurations, the introduction of attenuation into the system causes greater uncertainty in the assessment result compared no attenuation.

The effect of source- and matrix-material attenuation on optimal detector placement can be seen in Fig. 3 which illustrates how the optimal distance changes for different attenuations. The optimal distance to place the detectors so that uncertainty is minimized ranged from roughly 30-80 cm (for this case study). When matrix attenuation is negligible the optimal distance of the detectors is at the upper limit of the range cited (80 cm). As attenuation increases, the optimal distance is reduced as shown in Fig. 3. In generating data for this study, detectors were placed at 10 cm intervals. This protocol explains the lack of smoothness in the curve. Use of smaller intervals would have produced smoother results. The primary point to this figure is that attenuation affects the optimal placement of the

detector.
Fig. 1.
Fig. 2.
Fig. 3.
CONCLUSIONS
The effects radiation attenuation properties of matrix-materials on the deterministic ERR method of minimizing uncertainties in the assessment of unknown randomly distributed radioactive materials has been evaluated. Modification to the SDC function used in the optimization methodology was made by including an exponential attenuation term. Minimization of the End Response Ratio was performed to give optimal detector locations in the assay system. A computer program was developed to facilitate implementation of attenuation properties in the ERR method. The program consists of basic algorithms which: scan the radioactive container volume, calculate distances between the source and detector, calculate attenuation distances in the matrix materials, generate the complete response set according to the SDC configuration function, employ search routines for the minimum and maximum values of the response functions and the calculated ERR set, and finally relay feedback to determine the best assessment configuration. Matrix-material attenuations contribute to increase in the assay system uncertainty. Shorter distances between the detector and the waste container are thus required to provide lower uncertainties and consequently better assessment resolution for higher compared to lower radiation attenuations in the waste drum.
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ASSAY OF POTENTIALLY CONTAMINATED PROPELLANT
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ABSTRACT
One of the decontamination and decommissioning projects within the Department of Defense is the demilitarization of an aging stockpile of munitions. A large portion of the stockpile contains depleted uranium (DU) as an armor piercing core and so these munitions must be assayed for the presence of uranium in other components. The assay method must be fast and preferably easy to implement. The presence of DU is indicated by its alpha decay. The alpha particles in turn produce ions in the ambient air. If a significant fraction of these ions can escape the quantity of propellant, the ions can be detected instead of the alpha particles.
As a test of the feasibility of detecting alpha emissions from DU somewhere within a cartridge of propellant, we measured the transmission of ions through layers of real propellant. The propellant is in the form of graphite-coated cylindrical pellets. A 105-mm cartridge was modified for use as a pellet chamber. A check source served as an ion source. The ion detector consisted of a grid held at 300 V coupled to an ammeter. Results confirm that this is a promising technique for testing the propellant for the presence of DU quickly yet with sensitivity.
INTRODUCTION
The Department of Defense maintains a large inventory of munitions which contain

depleted uranium (DU) components. The DU component normally forms the core of the projectile assembly of the munitions, and is used because of its superior ability to penetrate enemy armor. As the stockpile of DU munitions ages, it will become necessary to safely disassemble the munitions, removing all explosive/combustible components and recovering other components for recycling. This process is known as demilitarization (demil). Of particular interest is the demil of 105-mm tank ammunition because this cartridge design has a section of the DU penetrator rod in direct contact with the propellant charge. The propellant consists of solid cylindrical pellets, each 1.5 cm long and 7.6 mm in diameter. Because the level of potential DU migration onto the propellant is expected to be low, a sensitive method of assay is required to segregate contaminated propellant. Given the large volume of propellant to be monitored, the assay method must also be expeditious-preferably in situ on the demil line-because a lengthy assay time could significantly add to the total cost of the demil project.

One indicator of the presence of DU is the presence of alpha particles emitted by the natural decay of uranium. Alpha particle activity can be detected via the ions that are created in air by the ionizing alpha radiation. The ions can survive much longer-and thus travel much farther from the source uranium-before neutralizing compared to the range of the alpha particles. For assay of a cartridge full of propellant, air is drawn through the pellets, transporting these ions to an ion detector.

An alternative assay would empty the pellets from each cartridge, spread the pellets on a conveyor, and scan the pellets for gamma-ray activity. This process is slower and might spread contamination.

ALPHA DETECTION

The ion detector consists of a metal grid across an air pipe. Ions caught on the grid as air passes through create a very small (femtoamps) current to ground. The response of the detector is on the order of seconds, the time necessary to transport the ions from the alpha particle to the detector. Detector requirements are simply a small voltage supply (300-V battery) for biasing the collection grid and a sensitive ammeter or electrometer. Similar detectors based on long-range alpha detection (LRAD) have been designed (1) and fielded by a team at Los Alamos National Laboratory (2).

The number of ions reaching the detector is proportional to the amount of DU present on the surface of the pellets. The ions can be absorbed or at least neutralized in collisions with the pellets before reaching the detector. This potential loss of efficiency at ion collection and hence of sensitivity to DU can affect the attractiveness of this nondestructive, in situ assay.

ION TRANSMISSION TEST

The transport of the ions through the propellant was tested at Savanna Army Depot Activity. Air containing fixed amounts of ionization is drawn through a steel 105-mm cartridge case. The loss of ions to attachment to propellant grains is measured as a function of depth of propellant in the case.

Test Apparatus

The test bench is illustrated in Fig. 1. The lower chamber held a 230Th check source which served as a source of ions. The front door on the chamber had a large cut-out, covered by a generic furnace air filter. The source disk was held in the center of the chamber by a stand. The TMA/Eberline source had an activity level of 6220 disintegrations per minute (dpm) in Sept. 1991 (essentially the same at present) over an active area of 4.4-cm diameter. To provide different activity levels, we made several masks. These masks consisted of sheet metal with circular punch-outs of graduated diameter. Diameters were chosen to provide variations in activity level in 660 dpm steps: assuming the activity was uniform across the original source. The data will call this assumption into question.

The pellet chamber consisted of an empty 105-mm cartridge, roughly 60 cm long. The bottom of the cartridge had a large hole, covered by a perforated metal sheet. This mechanical support for the pellets may have acted as an electrostatic filter for the ions passing through.

The pellets are graphite-coated, and therefore normally have a conductive surface. The packing fraction of the 7.6-mm diam., 1.5-cm cylinders was determined to be 60% in random pouring of the pellets. The layer of pellets would act as a grounded, conducting sponge that could filter ions. Qualified personnel transferred the propellant between storage and the cartridge during the operation, and data was

acquired in a separate bay behind 30-cm-thick concrete walls.
Fig. 1.
The 10-cm-diameter pipe monitor was bolted onto the cartridge. The signal grid was nominally 91 cm from the source disk. The sensitivity of the pipe monitor was determined in the lab to be 10 dpm per femtoampere (fA), similar to other LRAD pipe monitors (2). A 300-V battery was used to bias the grid. The battery was kept remote in the data acquisition bay in accordance with safety procedures. An Arnone electrometer (3) was secured to the monitor body and wrapped with black electrical tape. A BNC cable (and a grounding strap) was run out of the bay to a Keithley electrometer operating as a voltmeter. A PowerBook 180c ran the acquisition software.
A vacuum hose was connected with flanges directly to the top of the monitor. Approximately 1.8 m of this hose ran to a shop vac. A 10-cm-diameter chamber was inserted midway along the hose and contained a probe for measuring airspeed. There is some question as to the effect of turbulence within this chamber on the measured airspeed, as the chamber was not very long. A sliding opening assembly permitted control of the amount of vacuum and therefore the airspeed. Airflow was kept between 30 and 102 cm/s in a 10-cm diameter so that the response was relatively insensitive to airspeed (2).
TEST RESULTS
Figure 2 shows the response of the monitor with no propellant in the chamber. The data set taken at 45 cm/s airflow consists of two sets, separated by 90 minutes of delay. The lower value of the two points for 6220 dpm goes with the earlier set. Three observations are relevant.
1. The sensitivity (slope) changed during the delay, when the monitor and electrometer were moved and the electrometer was not physically secured. Ignoring the 6220 data (see #2), the slope below 3500 dpm is 66 dpm/fA, while above 3500 dpm the slope is 43 dpm/fA.
2. The source disk appears to have non-uniform deposition, with higher activity per unit area towards the periphery of the disk. Within each of the data sets, the 6220 data are consistently above what would be expected from extrapolating the lower activity data. This is also true to a lesser extent for the 4905-dpm activity, and the 4247-dpm area. Therefore, source strength must be considered nominal. We suspect that only the total activity of the 230Th source would be guaranteed to some tolerance by TMA/Eberline.
3. Better sensitivity was obtained with increased airspeed. The slope at 74 cm/s is 35 dpm/fA. Moreover, the measured airspeed was less stable at the lower setting. This may be a function of turbulence in the measurement chamber at that airspeed, or the load on the shop vac when the air vents just upstream of the vac were wide open. The half-minute to half-minute fluctuations in the data appeared to be proportional to the fluctuations in measured airspeed.
Figure 3 shows the response of the monitor when 5 cm of propellant covers the bottom of the cartridge. Data were taken for 145 fpm. The ratio of response for 5 cm vs no propellant is graphed in Fig. 4 for those activity levels common to the two sets of data. This ratio represents the average transmission of ions through 5 cm of propellant. The average is 25% with a (one-sigma) range of 21-29%. Thus one could anticipate that transmission of ions through 10 cm of propellant-an additional 5 cm-would be 25% of 25%, or 1/16. This would be close to the background of roughly 20 fA. Indeed, with 10 cm of propellant, a signal was obtained at background level even with an unmasked 6220 dpm source.
Fig. 2.
Fig. 3.
Fig. 4.
For a 5-cm layer, the airspeed was increased to 122 cm/s in an attempt to optimize the monitor sensitivity. The results are plotted in Fig. 5. Not much signal was recovered, and in fact the measurement with a 10-cm layer was checked at airspeeds approaching 152 cm/s without observation of any signal above background.
Fig. 5.
IMPROVEMENTS IN TECHNIQUE
We should be able to improve the technique enough to obtain a response of 10 dpm/fA (without propellant). The ion source/alpha source was separated by the perforated

metal sheet, which surely acted as an imperfect electrostatic filter. In real monitoring (and in the next phase of testing), alpha sources will be buried within or directly underneath any propellant.
Moreover, the use of triaxial cable would enable the electrometer to be separated from the detector head. Care should be taken in stabilizing air flow, at least for interpreting test results if not in actual disassembly usage. Lastly, test sources should be better defined as to intermediate activity levels, for example, through the use of a source set.
The next phase of testing will involve mixing a known amount of DU powder into some M30 propellant. We will learn the effect of reduced ion production by a single alpha particle due to pellets in the path of the alpha. This next phase will be carried out at Picatinny Arsenal in New Jersey.

CONCLUSIONS

A sensitive but fast method is needed to assay propellant from the demil of munitions containing DU. Alpha decay of the DU and subsequent ion production in air (the LRAD technique) provides a possible means of assay. In studying this technique, we have measured the transmission of ions through overlaying propellant.
A significant fraction of ions (25%) penetrates 5 cm of propellant. Therefore, assaying the DU within a batch of propellant is feasible through detection of the ions produced by the alpha decay of uranium. However, because no ions would penetrate a full 105-mm cartridge, the assay process must involve spreading the propellant within some container or conveyor belt. This still offers an improvement in the amount of time required to assay a quantity of propellant. Traditional alpha particle detectors detect the alpha particles directly and so are sensitive to DU contamination only on the top surface.
The next phase will use contaminated propellant to test the true efficiency of the detection method. If the efficiency is large enough, then only engineering problems relating to implementation on the demil line need be addressed.

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EVALUATION OF THE MADAM WASTE MEASUREMENT SYSTEM

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ABSTRACT

The Multiple Assay Dual Analysis Measurement (MADAM) system is a combined low-level and transuranic waste assay system. The system integrates commercially available Segmented Gamma Scanner (SGS) capability with a multienergy x-ray and gamma-ray analysis to measure these two waste forms. In addition, the system incorporates a small neutron slab detector to satisfy safeguards concerns and the capability for automated high-resolution gamma-ray analysis for isotope identification. Since delivery of the system to this facility, an evaluation of the waste measurement characteristics of the system has been conducted. A set of specially constructed NIST-traceable standards was fabricated for calibration and evaluation of the low-level waste (LLW) measurement system. The measurement characteristics of the LLW assay system were determined during the evaluation, including detection limits for all isotopes of interest, matrix attenuation effects, and detector response as a function of source position. Based on these studies, several modifications to the

existing analysis algorithms have been performed, new correction factors for matrix attenuation have been devised, and measurement error estimates have been calculated and incorporated into the software.

INTRODUCTION

The Plutonium Facility at Los Alamos National Laboratory (LANL) is a U.S. national defense facility involved in the recovery and processing of plutonium and other transuranic elements. Wastes are routinely generated from many stages of pyrochemical and aqueous processing of plutonium and uranium, and from plutonium fuels fabrication. The processing steps generate a wide variety of leaner scrap and waste forms, such as plutonium oxide from burned residues, Pu-bearing salts from production/reduction and metal purification processes, impure plutonium metal, metal reduction slags, ash, ceramics, cleaning rags, plastics, HEPA filters, and other remnants and apparatus generated from processing and cleanup tasks. Solid wastes created from this processing are divided into two broad categories: transuranic (TRU) waste containing greater than 100 nCi/g and low-level waste (LLW) containing less than this amount. TRU waste is obtained from various processing and cleanup steps within the processing gloveboxes and may be packaged in containers ranging from small cans up to 55-gal. drums. LLW is derived from locations outside the gloveboxes where contamination is expected to be light. Both solid waste categories receive nondestructive analysis (NDA) to determine radioactive content. For low bulk density materials (< 0.25 g/cc) such as paper, plastics, ash, powders, etc, the analyses are performed using x-ray or gamma-ray analytical methods. Denser materials, such as metals, leaded gloves, and some plutonium salts, are measured with passive or active neutron counter analysis.

In general, measurements of low-density LLW utilize a combined L x-ray and gamma-ray analysis in which the contributions from 235U, 238Pu, 239Pu, 241Am, and fission products are collected to determine the total activity in the waste.(1-4) Because this technique does not correct for matrix attenuation or self-absorption effects, it is suitable only for low bulk density materials with diffuse contaminations of plutonium or uranium. The sample is rotated and vertically scanned before a single NaI detector to smooth the spatial response of the signal. Correlation factors for non gamma-ray emitting species may be incorporated into the software to account for radioactive species not directly detected by the instrument (1). This analytical method provides high sensitivity and reasonable accuracy for low-density materials. However, the method has several limitations. First, it is generally insensitive to unknown radioactive contaminants. That is, contaminants such as 237Np, 243Am, or 244Cm, which are often associated with plutonium processing, may not be revealed with this method. Failure to detect their presence can lead to significant underestimation of the total activity in the waste. A second limitation in the method concerns its susceptibility to density variations in the waste. In the past, matrix attenuation of the gamma-ray and x-ray signals in low-density LLW has been discounted as a relatively minor constituent in the overall measurement uncertainty. Typically, a standard with the same density as the average LLW density has been used to calibrate the instrument. However, as the variability in waste matrix types has grown and come under increasingly stringent regulatory requirements for more accurate measurements of LLW, corrections for matrix attenuation effects have become necessary. A third limitation on existing x-ray and gamma-ray analysis of LLW relates to the inability of these instruments to detect plutonium concealed within photon-shielding materials. The concern here is not that the radioactive content in the waste will be underestimated as discussed above, but rather that comparatively large quantities of plutonium (10s of grams) may be concealed within gamma-ray shielding materials such that they are invisible to the detector. For example, significant amounts of plutonium could be contained within relatively-thin lead shielding, thereby presenting a diversion scenario of significant proportions to safeguards analysts. At present, this vulnerability must be addressed by additional safeguards measurements before the waste items leave the facility.

To surmount these limitations on x-ray and gamma-ray measurements of waste, a new instrument has been developed jointly by personnel at the LANL Plutonium Facility and Canberra Industries, Inc. From the same measurement platform, the instrument can operate in either an SGS mode to analyze TRU waste and leaner residues or can be used in a multienergy analysis mode to measure LLW. In the multienergy analysis mode, the software inspects the 235U, 238Pu, 239Pu, 241Am, and fission product gamma-ray and x-ray spectral regions with the NaI detector as indicated above, and

additionally performs isotopic gamma-ray analysis using a high-resolution detector. In addition, a neutron counter measurement simultaneously examines the waste for the presence of radioactive materials that may be shielded from the gamma-ray analysis. A series of test measurements has been performed to determine instrument sensitivity, matrix and geometry effects, stability, and a more realistic assessment of the measurement uncertainty.

This paper will first describe the MADAM system in more detail. The important characteristics of the TRU and LLW measurement platforms will be discussed. In particular, the capabilities of the MADAM LLW measurement system will be reviewed in some detail. The SGS is typical of commercially available SGS assay systems so only a short discussion of this component will be given. The results of the evaluation of the neutron safeguards detector will be presented. Finally, a discussion of the advantages and disadvantages of the LLW measurement system will be presented.

SYSTEM DESCRIPTION

The frame for the new instrument was taken from an old SGS unit that was decommissioned and retrofitted to accommodate both low-level and transuranic waste containers. Its footprint is approximately 5 ft by 5 ft, although the dewar for the HPGe detector protrudes an additional 18 in. on one side. All power supplies, amplifiers, counter/timers, motion controllers, and other electronic modules are housed in one standard 180 cm high by 56 cm wide by 76 cm deep NIM cabinet. The computer, printer, and keyboard are mounted separately on an adjacent 24 in. by 24 in. desk.

The SGS measurement platform includes a collimated coaxial HPGe detector and a multi-position transmission source holder. The system is designed to accommodate container sizes ranging from small cans up to 55-gal. drums. The detector collimator is designed to provide a segmented analysis with 2-in. segments over the height of a 55-gal. drum. However, for assays of smaller containers, the collimator can be configured for 0.5-in. segments with the use of a specially designed collimator insert. The software automatically detects the presence of the collimator insert and alerts the operator if the proper parameter file is not selected. The transmission source holder contains a multi-position rotator which holds up to six different transmission sources, allowing the system to be easily configured for assays of many different radioactive isotopes. Selection of the proper transmission source for an assay of a particular isotope is made automatically based on the information entered in the isotope parameter file. The SGS analysis utilizes the standard two-pass assay technique with transmission and live-time corrections that has been in use for many years (5). The instrument will perform either a three-peak assay in which there is an assay peak, a transmission peak, and a rate-loss correction peak, or a four-peak assay in which there are two transmission peaks in addition to the rate-loss and assay peaks. Each peak is analyzed using three fixed regions of interest, one for the peak and a background region on each side of the peak. The SGS software contains many diagnostics checks to validate the operation of the instrument, including detector full-width half-maximum, amplifier gain, sample density, and rate limits on the transmission and live-time sources.

At this facility, low-level waste is normally packaged in 1 ft by 1 ft by 2 ft cardboard boxes. The new instrument is capable of measuring waste packaged in these containers and, in addition, can accommodate LLW contained in 30-gal. and 55-gal. drums. The measurement time for the cardboard boxes is 2 min. The instrument is programmed to directly measure 235U, 238Pu, 239Pu, 241Am, and fission products using a single 5 in.-diam by 2 in.-long NaI detector optically coupled to a photomultiplier tube. Shielding around the detector consists of a specially fitted 1.75 in.-thick lead cover surrounding the sides, head, and back of the NaI crystal, except where the 2.0 in.-wide vertical collimator allows x-ray and gamma-ray access from the sample. The collimator permits the entire length of the LLW container to be viewed throughout the measurement. This collimator arrangement, plus rotational and vertical scanning, permits smoothing of the detector's spatial response to the waste package. The LLW measurement electronics, consisting of a preamplifier, amplifier, high-voltage power supply, scaler, and Canberra System 100 MCA board, form an independent, stand-alone unit that controls only the LLW measurements. Periodic measurements of a shuttered 133Ba source mounted to the platform provide gain and intensity control for the detector. Table I contains a summary of the major components in the LLW measurement system.

The system is also capable of performing gamma-ray isotope identification in LLW

containers using the SGS coaxial HPGe detector. For LLW measurements, the 2-in. collimator is rotated to a vertical orientation, which allows the entire length and width of the LLW container to be viewed by the detector. The purpose of the high-resolution measurement is to detect unexpected contaminants in the waste. An analytical balance, calibrated from 0 to 30 kg, is also connected to the system computer in order to calculate the activity per unit gram of waste.
Finally, a small 1 ft by 1 ft by 4 in.-wide neutron slab counter, containing four 8 in.-long 3He detectors surrounded by polyethylene, operates during waste measurements. The internally-mounted amplifier converts the output of the 3He tubes to a TTL pulse train directed to the counter/timer which, in turn, is interfaced to the computer. The neutron counter is attached to the side of the measurement platform; its purpose is to detect the presence of plutonium that may be transparent to the NaI detector. Because this detector measures only total neutron counts, not coincidences, it is not intended to accurately assay the plutonium content in the waste. It is meant only to detect the presence of relatively large quantities of shielded material. The neutron measurement is performed simultaneously with both the SGS and the LLW assays. The user is alerted if the neutron count rate is above a predetermined limit entered in the configuration parameter files. Motor controllers for rotational and vertical positioning of the sample and shutter regulation for the gain source support the detection apparatus.
TABLE I
EXPERIMENTAL METHODS AND RESULTS
SGS Measurements
The SGS system is a direct adaptation of a commercially available SGS to the MADAM platform. Modifications to the commercial SGS software for the MADAM platform include the addition of the safeguards neutron detector and an automatic peak identification analysis using Canberra's Spectran-AT at the completion of the SGS analysis. The system is currently configured to perform assays for 239Pu, 238Pu, 235U and 237Np. The live-time correction is made using the 88 keV gamma from a 109Cd source mounted on the barrel of the coaxial detector. Two transmission sources, 75Se and 137Cs, are used to perform the transmission correction for routine assays. The assay and transmission peaks used in the analysis are shown in Table II.
TABLE II
Since the SGS is a standard NDA technique, the measurement characteristics of the system will be only briefly discussed. The performance of the MADAM SGS system was tested using a set of 239Pu SGS standards. The standards are constructed of plutonium oxide blended into a low-density diatomaceous earth matrix. The standard cans are 11 in.-high with a diameter of 4 in. Tests were performed with the 2-in. collimator and with the 0.5-in. collimator insert. The measurement precisions obtained with 10 and 100 g plutonium standards are shown in Table III for two measurement times. The precision is defined as one standard deviation in the measurement result. In addition, the detection limits for the isotopes the system is configured to assay were determined and are shown in Table IV. Detection limit is defined here as the detector response that is equal to three times the background signal under the assay peak. It is assumed that the error contribution from the transmission peak and the live-time peak will be small compared to the error in the assay peak at the detection limits. It should be noted that these detection limits are theoretical limits based on the detector response from ideal assay samples. In real assay situations, instrument sensitivities are limited by practical considerations such as matrix composition, sample heterogeneity, and acceptable error. Typical detection limits for 239Pu and 235U assays at this facility are in the range of 0.5 to 1.0 g.
TABLE III
TABLE IV
LLW Measurements
Until recently, the measurement methods used to assay LLW leaving the LANL Plutonium Facility were identical to the methods originally developed by Umbarger and Cowder (1,3) in the original MEGAS box counter. In the latest revision of the MEGAS counter, the MADAM system, a number of changes have been made to the measurement algorithms in order to better satisfy changing regulatory requirements. The original MEGAS concept was intended to make a simple "go/no go" decision regarding the specific activity of a waste package. The new changes have been implemented in an effort to provide more quantitative measurements and to apply an error estimate to

the measurement result.
The changes to the measurement algorithms include: a background subtraction routine, calibration as a function of box weight (density) to correct for matrix attenuation effects, and an error estimate that is provided with the final measurement result. In addition, a spectrum integral of the areas outside of all analysis regions of interest (ROIs) has been added to detect any unexpected isotopes that may not fall into the analysis ROIs. The current MADAM system is configured to measure 238Pu, 239Pu, 241Am, 235U, and to check for the presence of fission products. The analysis isotopes and the energy of the analysis ROI for each isotope are listed in Table V. Detailed descriptions of the original MEGAS analysis method are available in the literature.(1,3)
Spectra are recorded by the NaI and HPGe detectors while the LLW box is rotated and vertically translated in front of the detectors. Typical measurement times range from 2 to 4 min. The background is removed from the NaI spectrum by performing a channel-by-channel subtraction with a smoothed background spectrum. The background spectrum is stored on disk and is updated frequently through a menu option using administrative controls. Then integrals of the analysis peaks are determined and the spectral sum for unexpected isotopes is made. The HPGe spectrum is analyzed only if the box contains greater than 100 nCi/g, unexpected isotopes, or fission products.
TABLE V
Low-level waste standards were prepared for calibration and characterization of the MADAM instrument by Robert Marshall of group CST-1 at LANL. The standards were prepared by pipetting a known amount of a standard solution (2 molar nitric) onto filter paper. The solution was allowed to air dry for 1 to 3 hours in an open-face hood. The filters were then placed between two strips of adhesive-backed polyester film (FasCal PX 1071 Bright Chrome 1 Mil Polyester, Fasson Co., Painesville, Ohio) with seven evenly-spaced filter papers for each assembly. The strips were then heat-sealed in two envelopes of 4.5 mil polyethylene. The final dimensions of each standard are approximately 4.5 in. by 23 in. The end product consists of a set of six 239Pu standards ranging from 0.26 to 22 mg total plutonium, a set of six 238Pu standards ranging from 1.4 to 80 mg total plutonium, and a set of six 241Am standards ranging from 5.7 to 434 mg Am. The plutonium used to produce the 239Pu standards had a nominal 240Pu enrichment of 6%, and the plutonium used to produce the 238Pu standards had a nominal 238Pu enrichment of 83%. Each set of standards is sufficient to span the specific activity range from 5 to 100 nCi/g in boxes ranging from 5 to 15 kg.
Standard boxes were prepared by placing five 2 in.-thick, 1 ft by 2 ft slabs of a variable-density polyurethane foam (Last-A-Foam FR-3700, General Plastics Manufacturing Co., Tacoma, Washington) into the normal waste boxes used at LANL. The variable-density foam was chosen for the box matrix rather than simulated waste, such as clean plastics and paper, since it is a very controllable and reproducible matrix. The slabs are held apart by 7/16-in. spacers to allow the standards to be slipped into the boxes in many different positions. Boxes with weights of 3, 4.5, 9, and 14 kg were constructed for the calibration measurements. The specific activities and box weights reported in this paper do not include the carton weight in accordance with DOE LLW regulations at this facility. The LLW boxes add an additional 700 g to the total weight of the package.
The 235U standards were prepared by blending 93% enriched uranium oxide into a diatomaceous earth matrix. The blended material was then sealed in 14 polyethylene laboratory bottles. The ratio of uranium to matrix material is such that in a 250 ml bottle filled to approximately 80% there are 0.26 g of oxide. The waste boxes for the uranium calibration were constructed from the same variable-density foam matrix described above. In this case, the foam matrix material was machined to allow the uranium bottles to be inserted into the matrix in multiple positions with up to all 14 bottles in a box. This results in a calibration range of 0.2 to 2.8 g 235U for each box. Waste boxes with weights of 5, 8 and 12 kg were constructed for the uranium calibration.
Each set of standards was measured in multiple positions in each box. A volume-averaged calculation was used for the instrument calibration. The volume of the waste box was divided into smaller volume elements around the positions used in the calibration measurements. The results of each measurement were then weighted with the appropriate volume element and summed to get a volume-averaged response. A calibration constant for each isotope was determined for each box weight. A

least-squares polynomial fit to the data was then used to determine the calibration constant as a function of box weight. This provides a correction for the increased attenuation of the low energy x-rays as the density of the matrix material increases. In the lightest box, 3.0 kg, the effect of the matrix attenuation over the volume of the box is small. The response from the least sensitive to the most sensitive position varies by 8.3%. On the other end of the weight range, at 14.0 kg, the effect of matrix attenuation on the 17 keV plutonium x-rays is very severe. The response varies by over 800% from the least sensitive position in the center of the box to the edge of the box where the response is highest.

The errors that contribute to a LLW measurement were broken down into five major categories: statistical, calibration, matrix variation, isotopic variation, and geometric effects. The geometric variation includes both the change in detector response as a function of source location and the effects of matrix attenuation for a single matrix. The error contributions are summarized in Table VI. The matrix and geometric errors are presented only for the 17 keV analysis of 238Pu and 239Pu because the errors for the americium and uranium measurements performed at 60 keV and 186 keV respectively, are small.

The statistical error was calculated by assuming a normal distribution and using the calibration constants in conjunction with the expected response from all of the isotopes of interest. The statistical contribution from the background subtraction was included in the calculation. Statistical error is estimated as a function of measured activity for each isotope. Bias in the calibration was measured by remeasuring the standards in each box after all the calibration functions had been entered into the system. The bias averaged over the entire calibration range is < 1% while the maximum calibration bias for any combination of source and box weight was 12%. The one sigma uncertainties in the calibration constants due to random variations in the calibration measurements range from 0.5 to 1.5%.

TABLE VI

The error due to changes in the matrix composition was measured by constructing waste boxes with simulated waste materials. Three mock boxes were constructed for this purpose. The simulated matrices used were shredded paper, small polyethylene vials, and rubber gloves. A 239Pu standard was placed in the center of each box and the box was assayed. The errors shown are the measured differences in the assay value when compared to a similar assay performed in the foam calibration matrix with an equivalent weight. The errors shown represent the largest possible error since the source was located in the center of the waste box where changes in the matrix attenuation have the greatest effect. Significantly smaller deviations would be expected with the radioactive material randomly distributed throughout the waste box.

The error contribution due to changes in the isotopic composition of 239Pu material was calculated based on the correlation of the measured L x-rays to the total activity for various material types found at the LANL Plutonium Facility. Errors in the measured total activity as large as 60% can result due to variations in the isotopic composition of the different material types. However, this source of error is not expected to provide a significant contribution to routine LLW measurements. A large majority of the materials processed at the Plutonium Facility have isotopic ratios very similar to that of the calibration material. Only rarely are 239Pu material types of significantly different isotopic composition expected to be encountered.

Finally, the geometric error was estimated by measuring a 239Pu source in various locations in each box, including folded lengthwise and placed in the top or bottom of the box. The %RSD reported in Table VI was determined by assuming the measured range in the response to be 2 standard deviations. Thus 95% of all measurements fall within the measured range. For example, in the 14 kg box, the measurements ranged from 12 nCi/g in the center of the box to 110 nCi/g at the extreme edge of the box with no matrix material between the source and the side of the box. The box had a nominal activity of 29.7 nCi/g. If the measurement at the edge of the box is not included, the next largest measurement is 30.2 nCi/g, reducing the standard deviation to 15%. Because the 2-in. foam slabs constrain the position measurements, it is not possible to get a uniform volume sampling of the box.

The sensitivity of the LLW measurement system was estimated based on the system response determined from the calibration and the geometric and background error statements. The sensitivity is reported as the detector response from a source equal

to three standard deviations above the background signal. In this report, the activity is defined as the total activity. A significant contributor to the total activity of plutonium is the beta decay of 241Pu. Unfortunately, this decay mode does not produce any radiations that can be directly measured by the MADAM system. The total activity must be correlated to the measured L x-ray activity using average isotopic compositions for material in the Plutonium Facility and specific isotope activities. The estimated sensitivities are shown in Table VII. These sensitivities are based on a 120-sec. measurement time and represent the expected sensitivity for a point source located in the center of a waste box, the position of least sensitivity. Thus, the results shown in Table VII demonstrate the worst-case detection limits for the instrument.

TABLE VII

Neutron Counter and Isotope Identification

The efficiency and geometric variation in the response of the neutron counter were measured using a 252Cf neutron source with a known activity. The measured efficiency of the neutron counter with a neutron source located on the center of the platform is 0.34%. Monte Carlo calculations using MCNP indicate the efficiency for a 240Pu neutron distribution should be approximately 5% higher. Also, calculations indicate that the slab is slightly undermoderated. The response could be increased by about 50% by adding additional polyethylene to the face of the slab. The geometric variation in the neutron detector response has a maximum deviation of -45% over the volume of a LLW box.

The response of the neutron detector to plutonium was measured using cans of plutonium oxide placed in the position with the lowest sensitivity. From this series of measurements, the minimum sensitivity of the neutron counter is estimated to be approximately 5 g of plutonium (6% 240Pu) for a 120-sec. measurement. The minimum detectable quantity will be somewhat higher for plutonium metal. This value is more than sufficient for the safeguards function that this slab counter is intended to perform. These measurements were made in the relatively high background, approximately 33 totals/sec, of the basement of the Plutonium Facility. In the lower neutron background of the LANL CMR building where the second MADAM unit is located, the minimum sensitivity is reduced to approximately 1 g of plutonium.

The HPGe detector will be used in combination with Canberra's Spectran-AT for isotope identification purposes. This feature will not be used routinely, but will be used for boxes that have activities above the 100 nCi/g limit and for boxes that are found to contain fission products or unexpected isotopes. When a box is flagged for one of the above reasons, a Spectran-AT compatible file is created from the HPGe spectrum taken during the LLW analysis and saved to the disk. These files can be analyzed later by the NDA lab supervisor. If the count rates are not high enough to produce a useable HPGe spectrum in the short LLW assay time, the boxes are set aside for counting at a later time. Measurements have shown that an overnight count of 10 hours is more than sufficient to measure 239Pu gamma-rays in a waste box with an activity level of 90 nCi/g to better than 5%.

In addition, the isotope identification feature has also been incorporated into the SGS analysis. The HPGe spectrum from each segment is summed to produce a spectrum accumulated over the entire container. A Spectran-AT analysis is then automatically performed on the summed spectrum after the SGS analysis has been completed. The purpose of the Spectran analysis is to identify any unexpected radioactive contaminants in the waste container.

DISCUSSION

The MADAM system has several advantages over many existing gamma-ray analysis systems. The capability to perform two different measurement techniques from the same measurement platform provides an important advantage for facilities or operations with limited floor space. The addition of the isotope identification feature using the HPGe detector and the simultaneous safeguards measurement with the neutron detector are two very useful features in facilities tasked with safeguarding nuclear materials. The most impressive feature is the capability to screen a large number of waste boxes in a short time while retaining good sensitivity. The LLW analysis system can perform an assay in two minutes with detection levels below 1 nCi/g for all isotopes of interest in the Plutonium Facility.

While the high specific activity of the plutonium L x-rays provides very good sensitivity for plutonium contamination, the low energy L x-rays interact strongly with the matrix material. As discussed previously, the analysis error increases

rapidly with box weight due to the attenuation of the L x-rays inside the waste matrix. This limits the LLW analysis to low-density matrices. Analysis of the NaI spectrum from the LLW measurement allows some determination of the type of material in the box, such as a 238Pu material type versus weapons grade plutonium or uranium. However, limitations in the NaI analysis prevent simultaneous assays of these material types. At the Plutonium Facility, these waste streams are carefully segregated so this does not pose a serious problem. A further limitation of the material type determination occurs at very low activity levels. For specific activities below approximately 10 nCi/g, differences in the low-energy signatures from 238Pu and 239Pu become very small. As the specific activity decreases below this level, the probability of misidentification becomes much greater. Additionally, with these very low specific activities, an isotopic ratio determination with the HPGe becomes impossible. Since a large fraction of the total activity in plutonium and uranium comes from beta emitters which are not directly measured, any change in the isotopic ratios from that of the calibration materials will lead to measurement errors which can be quite large in some cases. If the isotopic ratios of the material are known, correction factors can be applied to the results; however the exact isotopic ratios are not generally known for any given waste box.

CONCLUSION

A new NDA system has been developed to assay LLW and TRU waste generated from plutonium and uranium processing operations from the same measurement platform. The LLW measurements have been improved by the addition of algorithms for matrix correction, background subtraction, and error estimation. The addition of a small neutron counter to the measurement platform provides the capability for a safeguards measurement to be performed during the gamma-ray analysis to check for the presence of shielded fissile material. Isotope identification software has been added to both the LLW and TRU waste assay systems to aid in the detection of unexpected radioactive contaminants. The LLW assay system is capable of quickly screening low-density waste while retaining good sensitivity. The results of this paper show that improvements to the LLW assay system allow quantitative LLW measurements to be performed from this measurement platform with realistic error statements associated with the assay.

ACKNOWLEDGMENTS

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46-7

A NEW DEVICE FOR MEASURING THE SOLID FRACTION IN SLURRIES OR THE MOISTURE IN STABILIZED NUCLEAR WASTE

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ABSTRACT

Retrieval of nuclear waste from storage tanks typically requires the mobilization and suspension in the supernate of solids that have settled to the bottom. This can be achieved by means of a mixing pump. During this operation it is important to be able to monitor the progress of particulate suspension. A new method is reported here for determining the solid to liquid volume ratio, using a radio-frequency probe. The method is immune to the effects of pressure differentials which occur during slurry agitation and exhibits linearity over a wide range of solid fractions. The probe is compact and robust, so that it can be employed were access is limited and where there are mechanical stresses and corrosive environments. This device is also expected to be cost-effective as a reliable long-term monitor of moisture in nuclear waste.

INTRODUCTION

The retrieval of nuclear waste from storage tanks can frequently be effected by slurry pumps. In cases where solids have settled out below a liquid layer, agitation can be used to prepare slurries by means of the dynamic suspension of particulates. It is the need to monitor this process of particulate suspension that motivated the development of the device described in this paper. However, since the technique employed provides a measurement of the liquid to solid ratio of the medium, the determination of moisture is an alternative application.

Ionic conductivity provides the physical parameter for the determination of the liquid to solid ratio in a slurry. Aqueous solutions exhibit conductivity in the form of ion currents. The presence of solid particles in such a liquid reduces the cross sectional area available to ion currents. Rather than exposing conducting electrodes to probe such an electrolytic medium, an insulated radio frequency (RF) coil was used to sense electromagnetic energy loss via ionic currents. Among the advantages of using a RF probe which is hermetically sealed in a dielectric casing, are mechanical robustness and immunity to corrosion, electrolytic effects and surface contamination.

Initial tests of such an RF probe have been conducted in slurries containing weight fractions in the range 30% to 60% of iron oxide powder in saturated aqueous sodium nitrate solution. Ionic current energy losses were determine at frequencies in the range 1MHz to 20MHz using a Q meter. It was found that the best sensitivity for the slurries tested and the coil configuration used (10 turns of 3mm thick copper wire spaced over a length of 50mm on a cylinder of 50mm diameter) occurred at 9MHz. However, the frequency chosen was not a critical value and comparable measurements could have been obtained over the range of frequencies quoted above.

The volume fraction of fV and the mass fraction fM of solids in a slurry can be defined in terms of the solid volume VS and density rS, and the liquid volume VL and density rL as follows:

$$fV = VS/(VS + VL)$$

$$fM = rSVS/(rSVS + rLVL)$$

It was found that the effective Q of the coil and slurry combination decreased with decreasing solid fraction. This is what is expected as a result of the consideration above, since lower solid content in the slurry implies Q loss through increased ionic current. It was also found that within the 5% accuracy of the measurements performed, the relationship between the volume fraction fV and Q was perfectly linear over the range of slurries tested:

$$fV = m\,Q + c \text{ where } m \text{ and } c \text{ are constants.}$$

The mass fraction showed a dependence on Q as follows:

$$fM = (m\,Q + c)/([1-a]mQ + [1-a]\,c + a)$$

where a rL/rs

The relation between fM and Q is thus not linear, except in the fortuitous case when the density of the liquid and solid phases in the slurry are equal. The observed linear relationship between fV and Q appears to be compatible with the ionic current loss model.

It is expected that this simple RF probe, which can be operated directly with an off-the-shelf commercial Q-meter, will solve the need to monitor slurry formation in real time. The device is small enough to pass through restricted tank apertures and is not affected by dynamic pressure gradients such as occur in stirred slurries.

Calibration of the probe only requires a measurement to be taken in drained liquid which is free of solids. It may be convenient to install a two-coil probe in which one coil is fitted with a filter screen that admits only liquid, so that automatic calibration is provided. Such RF probes are, of course, not limited to the monitoring of stirred slurries and can also serve as long term monitors of humidity in moist sediments.

46-8
SYSTEMS APPROACH TO ASSESSING DOE FACILITIES
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ABSTRACT
Facility and system assessments are needed throughout the DOE complex, and are regularly performed for operating, shut-down, and surplus facilities to supply information for determining the best utilization of these facilities. The DOE has adopted the Capital Asset Management Process (CAMP) which includes a standardized inspection activity to assess DOE facility assets. However, to date there has been a lack of a standardized approach which could be followed in performing large-scale facility and system assessments incidental to the CAMP program. This detracts from the assessments' effectiveness, especially when an appropriate systems engineering approach is not used. Through experience attained in the preparation of large-scale engineering assessments for diverse facilities and systems, a standardized approach is presented which ensures the application of systems engineering to DOE facility assessments. This approach addresses DOE Orders and Site Contractor requirements. The recommended approach consists of a simple four step process. These four steps include 1) definition of requirements, 2) data compilation, 3) data evaluation, and 4) conclusions and recommendations. Case studies are presented which demonstrate the application of the approach to various types of assessments of nuclear facilities.
INTRODUCTION
The current need for facility assessments is apparent throughout the DOE complex. The needs prompting these assessments range from ensuring continued operations to planning for decommissioning. The DOE, and its predecessor, the AEC, constructed facilities for the production, processing, and management of various chemical and radioactive materials. These facilities were often constructed rapidly with a single mission objective, and often without adequate consideration of eventual decommissioning needs. Changes in long range plans, or unanticipated needs for certain materials, have caused many facilities to operate beyond their original design life and/or outside the originally intended operational bounds. As the DOE mission shifts from material production to environmental restoration, the use of existing facilities must be considered, notwithstanding the extreme changes in facility utilization this shift can cause. Many facilities were constructed so that consideration of optional missions is practical. However, outdated equipment and support systems may preclude their use. A well defined and executed assessment can supply information to determine, based on life cycle costs, the economic advantages of using a facility, maintaining the facility in a "cold shutdown" condition with only minimal surveillance and maintenance, or decontaminating and decommissioning the facility.
OBJECTIVE
The objective of this paper is to identify the key elements of a facility or system assessment that must be accomplished so that the assessment will be effective, and useful in developing life-cycle costs. The successful application of these elements will provide the life-cycle technical and economic results needed to make effective decisions with respect to future activities.
The approach presented was developed in the course of preparing detailed engineering assessments for a number of DOE facilities at the Hanford Site. As such, the approach is best suited to the preparation of large-scale engineering assessments. These assessments go beyond the scope of the basic asset inspections included in the Capital Asset Management Process (CAMP) that has been established by the DOE as an integral part of facilities management (1). CAMP is the ultimate vehicle by which DOE capital assets will be managed and which will support and justify budget and planning requests. Accordingly, the approach for facility and system assessments

presented here is intended to provide guidance for preparation of the complex engineering assessments that will supplement the integrated facilities management process now being established throughout the DOE complex.

APPROACH

The preparation of an effective assessment requires four elements. These four elements are as follows:

1. Definition of Requirements: This element involves defining the objective of the assessment; identifying the applicable regulations, codes, and standards; and preparing any checklists needed for the execution of the assessment. Two methods of defining the requirements are readily available at DOE facilities. The first is the use of the Standards/Requirements Identification Documents (S/RIDs) and the other is the use of a Surplus Facilities Inventory and Assessment (SFIA) checklist. To date, both of these methods have been limited to selected facilities which are identified by DOE or the Defense Nuclear Facility Safety Board (DNFSB).
2. Data Compilation: This element consists of compiling all available information pertinent to the objective of the assessment. In most cases this includes obtaining historical operating documentation and facility drawings, interviewing plant staff who were at the facilities during operations, inspecting facilities and systems visually or using non-destructive methods, and obtaining laboratory reports from any relevant sample analyses.
3. Data Evaluation: This element is performed based on the objective of the assessment. For assessments with well-defined checklists (such as those solely designed to establish compliance with regulations) the evaluation is a simple comparative analysis of the checklist against the accumulated data. In more complex assessments, this step may involve the preparation of a seismic analysis computer model, mass, energy, or chemical balance calculations, and evaluation of utility requirements, for example.
4. Conclusions and Recommendations: This element summarizes the data comparisons and evaluations performed above and provides the information needed for a life cycle analysis. The key items which must be included in the conclusions are the results of any risk or safety assessment activities, determination of compliance with environmental regulations, definition of operating and/or surveillance requirements, and any other applicable results which may affect operational or maintenance cost estimates.

This approach incorporates "systems engineering" in that the four elements have been found to specify all of the key functions and functional requirements needed to successfully complete DOE facility assessments. The relationships among the four elements and the various sub-tasks within elements are well defined in accordance with a rigorous systems approach.

Figure 1 graphically depicts the four elements discussed above and their functional breakdown.

Fig. 1.

CASE STUDIES

Three case studies are presented below which illustrate various applications of the assessment approach discussed above. Each of these assessments was performed using this approach, albeit sometimes in a iterative manner.

Case Study #1: Miscellaneous Tank Evaluation

In response to potential safety concerns, an assessment of 50 miscellaneous underground tanks at the Hanford Site was initiated. These tanks are all underground and most still contain some quantity of radioactive or chemically hazardous waste in either a sludge or liquid form. The tanks were used for a variety of purposes. Many were catch tanks used to collect spills or leaks incurred during waste transfers from facilities. Other types include vault tanks used for acidification of caustic waste prior to uranium extraction, and experimental tanks used in the support of the Hanford Site Hot Semi-works facility, and lime-filled neutralization tanks for treating acidic process condensates.

To perform an adequate assessment of equipment which could not be inspected, it was necessary to establish a methodology which would ensure that all applicable safety concerns could be addressed without introducing unnecessary conservatism into the final safety assessments. Another unique aspect of this assessment was in evaluating the appropriate ownership of these tanks with respect to both the Hanford operating contractor divisions and the applicable governing regulatory agency and statute,

i.e., CERCLA or RCRA.
Step 1: Definition of Requirements
The objective of this assessment was straight-forward: evaluate the expected inventory of the 50 miscellaneous tanks and determine the safety issues associated with this inventory. Due to the extensive work done in identifying safety issues for the large single and double shell tanks at the Hanford Site, the safety parameters were readily available.
Step 2: Data Collection
Data collection was the key step in this assessment. Many of these tanks had been interim isolated during the late 1970s and early 1980s; however, operational records prior to these periods were often incomplete or unavailable. Available historical records were collected and examined, site interviews were conducted, processes at facilities associated with these tanks (tank farms and material production and research facilities) were characterized, and process flow diagrams were prepared.
Step #3: Data Evaluation
This element involved compiling the data for each of the 50 tanks and preparing tank-specific data sheets. These data sheets included pertinent design information on the tank, such as location, size, operating life, and associated facilities; historical information on waste receipts; sample data (where available) on existing inventory; process flow diagrams to determine possible waste intrusions; and estimates of chemical and nuclear inventory. From these data sheets a safety checklist was completed. For example, tanks which received relatively large quantities of plutonium were evaluated for criticality. Tanks with a high inventory of organic materials were evaluated for toxic gas generation and flammability.
Step 4: Conclusions and Recommendations
Based on the results of the safety and environmental evaluations, a prioritization matrix was developed. This matrix gave those tanks with the possibility of having immediate risk to on-site personnel under a no-action scenario the highest priority for remedial activities. Tanks which posed environmental risk through the potential intrusion of regulated waste into soils or groundwater were given the next highest priority. Tanks which posed a risk to workers upon entry (such as opening risers) were identified for future planning. This prioritized list provided recommendations for developing a follow-on program plan for the surveillance and restoration of these tanks.
Case Study #2: T Plant Viability Assessment
The T Plant facility complex is located in the 200 West Area of the Hanford Site and has been proposed to function as the long-term decontamination support facilities for the site. T Plant is a canyon facility and was constructed in 1944 for the extraction of weapons grade plutonium. By 1956 the facility had ceased its original plutonium production mission, and had begun operation as a maintenance and decontamination support facility. The thick concrete walls and open deck configuration of this facility allowed versatility in the receipt, decontamination, and repair of large equipment items, such as pumps and other processing equipment. Mission changes at the Hanford Site from defense-related nuclear production to waste cleanup and environmental restoration drives the need for a variety of solid waste processing capabilities. As noted above, T Plant is proposed as the decontamination facility within the scheme of solid waste processing activities. To accomplish this mission, the 50 year old facility must be shown to be structurally sound and must be outfitted with adequate support systems (safety, material handling, environmental protection, and utilities). To provide this assurance, a viability assessment was performed.
As with the miscellaneous tank project, the T Plant assessment followed a four step process. However, this process used much different data collection and evaluation methods and resulted in a more detailed and comprehensive set of recommendations.
Step 1: Definition of Requirements
The objective of this assessment was to determine the viability of T Plant to perform as a decontamination facility in support of the solid waste disposal division mission. The requirements were developed through two methods. The first was the preparation of S/RIDs. The S/RIDs for T Plant consisted of an 18 volume set of requirements for all systems and safety and environmental administrative requirements. This set of requirements provided the applicable regulations, codes, and standards needed to assess the facility. The second method for developing requirements included the definition of a facility mission. Although the ideal

mission was to provide full service decontamination capabilities for the Hanford Site, it was recognized that more cost effective scenarios exist in which T Plant would operate in a limited role, and other planned facilities, such as the Waste Receiving and Processing (WRAP) modules, could be used for the remainder of the projected waste stream.
Step 2: Data Collection
Data collection was a multifaceted activity. One key element of the data collection was gathering information to estimate the projected solid waste stream requiring decontamination/treatment by T Plant over the 30 year environmental restoration phase of the Hanford Site. This was performed through an intensive effort of document and equipment drawing research, site interviews, and equipment inventory database consultation. The second element of the data collection was obtaining copies of all pertinent plant documents (drawings, operating records, seismic analyses, vendor literature, procedures, permit applications, and previous assessments). The final element of the data collection effort consisted of walk-throughs and detailed inspections of the T Plant buildings and systems. Material handling systems, building structure and seismic response, safety systems, mechanical and electrical utilities, secondary waste management systems and facility environmental compliance were inspected by experts in these various functional areas.
Step 3: Data Evaluation
This element was accomplished by making individual evaluations of each major plant system. The experts assigned to the various functional areas each prepared a system description, referencing essential drawings, operating history, procedures, etc. Next, the systems were evaluated for compliance with DOE orders, applicable codes and standards, engineering practices, and for expected viability over the 30 year working life of the facility in terms of operational, maintenance, safety, and environmental factors. A detailed time and motion analysis for receipt and shipping of solid waste, and a seismic analysis were two of the many rigorous evaluations accomplished in this element.
Step 4: Conclusions and Recommendations
Based on the system descriptions and evaluations, detailed recommendations were made for necessary system upgrades, equipment replacement, and facility renovations required to bring the systems and buildings into compliance and make them viable for extended future operations. Ongoing and previously planned upgrades were kept in retrospect. For example, substantial secondary containment and leak detection upgrades for the liquid waste handling system, canyon crane refurbishment, restoration of the modified canyon roof to its original configuration, various HVAC system upgrades, installation of fire suppression and early warning fire detection systems, etc., were recommended.
Case Study #3: 340 Facility Compliance Assessment
The 340 Facility is located in the 300 Area of the Hanford Site, where fabrication of uranium fuel elements was accomplished historically. Currently, the 300 Area is used for various research and development programs for environmental remediation. The 340 Facility manages radioactive and non-radioactive liquid wastes generated throughout the 300 Area. Wastes are transferred to the 340 Facility chiefly through two process sewer systems. Radioactive wastes are collected in agitated, underground vault tanks and periodically sampled for analysis. Within 90 days, the collected radioactive wastes are transferred to railroad tank cars and shipped to 200 Area tank farms for long-term storage. Non-radioactive wastes are collected at the 340 Facility in open retention basins, the effluent from which was formerly discharged to a local process trench, and currently discharged to the Treated Effluent Disposal Facility at the Hanford Site.
Step 1: Definition of Requirements
The objective of this assessment was to provide an environmental compliance evaluation of the 340 Facility as regards requirements for the receiving, storing, handling and transferring of radioactive and non-radioactive liquid wastes. The compliance assessment was based on the applicable requirements of the Washington Administrative Code (WAC) for hazardous and mixed waste facilities, federal regulations for airborne emissions and aboveground and underground storage tanks, and site contractor (Westinghouse Hanford Company) guidelines.
An interesting aspect of this study was the additional identification of requirements the 340 Facility would need to comply with to become a RCRA Part B

permitted facility.
Step 2: Data Collection
Data collection for this assessment centered on the identification of environmental requirements based on the site contractor's Environmental Review Compliance Assessment Inspection Checklist and guidelines given in the site contractor's Environmental Compliance Manual. Review of the checklist and manual indicated that their organization and content would be adequate to assure a rigorous assessment. In addition to the key data collection task of identifying federal, state, and site contractor requirements, relevant documents and drawings of the 340 Facility were reviewed, plant staff were interviewed, and site visits and inspections of facility systems were conducted as needed.
Step 3: Data Evaluation
Data evaluation began by filling out the checklist provided by the site contractor for facility environmental compliance. Due to the complexities of multiple federal, state, and site contractor requirements identified in the course of filling out the checklist, a matrix cross referencing the requirements was developed. This matrix listed references of applicable federal and state regulations, the references of corresponding site contractor requirements, description of the individual requirements, and then an evaluation of compliance or non-compliance with each requirement, along with accompanying comments.
Step 4: Conclusions and Recommendations
Based on the requirements compliance matrix, a list of compliance deficiencies was developed. This list was broken down by facility systems, including sampling and monitoring systems, storage tanks, storage areas, etc. Some notable deficiencies that were identified concerned continuous monitoring and sampling systems for airborne radioactive effluents, and lack of integrity assessments for underground storage tanks. Specific recommendations were made for correction of each deficiency, with rough estimates of costs for the corrective measures.
Also, conclusions were drawn as to what actions would be required to permit the 340 Facility as a RCRA facility. Major storage tank replacements with double shell tanks, sampling and monitoring system upgrades, secondary containment upgrades, etc., were shown to be required, along with preparation of additional documents for permitting purposes. Recommendations were made that these upgrades would probably not be cost effective, and that alternatives to the long-term use of the 340 Facility be considered, along with request of a waiver to operate the facility as is in the interim.
CONCLUSIONS
Preparation of effective facility and systems assessments requires a consistent approach that will ensure that adequate definition of requirements, data compilation and evaluation, and drawing of conclusions and recommendations are accomplished. A systems engineering approach as outlined has been shown to adequately meets these needs, regardless of the kind of assessment to be performed. Properly prepared, assessment results can be effectively used to develop life-cycle costs, and can provide valuable input to the DOE complex-wide Capital Asset Management Process currently being established.
REFERENCES
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46-10
SYSTEMS APPROACH, INTEGRATION, AND PROJECT MANAGEMENTFOR SUCCESS ON ENVIRONMENTAL RESTORATION PROJECTS
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ABSTRACT
Mixed waste environmental restoration projects are presenting new requirements in technical management to companies and government agencies. These requirements reflect new technical, schedule, and cost challenges which must be met in an environment far different than that previously encountered by civil or "works"

projects. Existing skills and techniques for conventional engineering design may be insufficient. This paper presents a techniques for project management for environmental restoration projects involving multidisciplinary technologies, "first time" design, and innovative equipment operation.

TIMES CHANGE - TODAY'S PROJECT FRAMEWORK

Things are not what they used to be. Environmental Restoration (ER) projects are different. They are different for the government, different for the those designing solutions, and different for those implementing solutions. The framework for these projects, as well as the life cycle of concern, has changed, and the new framework needs to be clarified. Those who implement these projects, as well as those who must procure these services must know what to expect.

Because some of the resources to accomplish ER are similar to those used in works/civil projects, (for instance land filling) specification of the approach/effort for ER is quite similar. However, because many of the solutions required for the remediation of mixed waste are leading edge technologies (for instance vitrification), it may be more appropriate to look to the system engineering model for guidance. Examining the elements of each shows why.

"Works" and Environmental Restoration Projects

Works/civil projects:

Deal with the known - roads, strength of materials, flood plains, soil structure, landfills.

Proven solutions - no development for new technologies required.

Basically have one decision maker - a city water commission, an office of the Corps of Engineers, a developer, and industrial plant manager.

Have a clear definition of job - build a bridge from A to B, dredge the channel, build a landfill for a city of 30,000.

The scope is finite - design and build (one contractor), design or build (two contractors).

Life cycle defined - structure to last 30 years, landfill to operate for 15 years.

Liability circumscribed - covered by insurance, governmental body assumption of liability.

ER projects on the other hand:

Deal with the unknown; for example at WMS '95 topics will address

- "Rebar Corrosion Due to Carbonation in Structural Reinforced Concretes for Near-Surface LLW Repositories", J. Torok
- "Alternate Conceptual Models in the Saturated Zone at Yucca Mountain", L. Lehman, T. Brown
- "Credible Future Climate for the Nevada Test Site for the Next 10,000 Years", J. Cochran, J. Emery

Require leading edge solutions - Hanford Tank Wastes.

Decision making Is fragmented - Stakeholders, government, private business, citizens

Magnitude of the problem may not be bounded - quantity of waste, toxicity time period, number of waste streams.

Life cycle takes on new meaning - half life greater than 1000 years, administrative controls capability 500 years hence.

Specter of "LIABILITY"

- Lawyers will not be stopped by contracts regarding liability,
- Our lawyers are smarter than your lawyers, and the
- Deep pockets concept.

Example - Vortec Vitrification Demonstration Plant Project

This Technology Demonstration Contract is for design and construction of a high throughput (25 tons/day) combustion melting system to be used in a 30 day demonstration of the vitrification of a low level mixed waste. The decision makers were DOE and DOE support contractors.

The final waste stream characterization of the several sources considered resulted from a system engineering definition accomplished through an iterative analysis. The design soil waste stream was the worst case which could be encountered tempered by a "cost and reasonableness" criteria relative to radionuclide levels.

Phase 1 involved developing a system concept through trade studies, surrogate soil testing, and interaction with a number of potential host sites. Phase 2 involved the detail design for a particular host site (Hanford), but late in Phase 2 Hanford

withdrew support and a new site was required. Phase 2A will involve selecting another site while Phase 3 will implement the construction and 30 day demonstration of the solution. In all phases, analyses, tradeoffs, and iteration of answers have been required. In no case have the requirements lent themselves to being resolved by a "plug in and grind out" approach.

Example - Hanford Low Level Tank Waste

In this case, the magnitude of the problem may never be bounded because there are too many unknowns, not only in terms of the waste but also for the appropriate technology to use in remediation, the business/contractual manner in which the remediation will be carried out, the life cycle for assurance of environmental protection, and that old specter-- "LIABILITY" (whose?, when?, why?). The government is in the early phases of developing parallel approaches of: 1) selecting an A&E contractor for implementing what may be a conventional design/build civil project with operation to be defined at a later date; 2) while in parallel selecting a company that will sell a "complete remediation service" covering technology development, design, build, and operation on a fixed price basis.

Approaches Have Changed

In the old way of handling civil projects, there was "design then build". When dealing with the government, the rules are quite precise. In general, the effort was accomplished on "cost plus" contract. Any modifications were fairly straightforward, but in any event usually never required innovative development.

Most of today's senior management with the engineering firms (or in government offices) involved in ER gained their experience with "works" and civil projects. They still try to define a solution with a one pass design, "Our General Spec.", and supporting standard specifications and drawings in the atmosphere of the "cost-plus" contract inefficiencies and perceived large government budgets.

Contract cost goes up because ER projects are not standard, mixed waste stream definitions are imprecise, and pressure to improve public health and safety quickly encourages a "do something" mindset.

The "old way" to civil projects may have a small role in ER but only where cost effective, proven design definitions exist.

A better way, the systems engineering approach, provides a high confidence, reduced risk method because it provides a disciplined methodology controlled by decision points based on established success criteria. It can, exercised properly, result in a lower life cycle cost for obtaining results while satisfying the real needs of the stakeholders. The major benefit is that it will produce results which solve the ER problem.

Not yet totally accepted by the A&E sector, the system approach has long been used in the aerospace community and is gradually being implemented in the industrial community. It is being explored in the ER community as the DOE Tank Waste Remediation System Program opportunity (Commerce Business Daily, 8/18/94) acknowledges.

UNDERSTANDING SYSTEMS

The Characteristics of the System

A process is not a system. Adding a computer to equipment does not define a system. Systems are more than just processes or computers. Systems consist of any or all of the following; process, methods, equipment, and software. The identifying feature of a system is that it is defined by the interfaces. This emphasis on interfaces distinguishes a process from a system. Interfaces may be physical (mechanical, electrical), informational, human interface, environmental, and timing. The importance is that success at the interfaces brings together equipment, software, and individuals into a combination that works.

System engineering provides a problem solving approach where perceived complexity can be simplified, options can be analyzed and implemented to close on solutions, and the methodology will be iterated until solutions meet requirements based on success criteria. Concepts transition into designs, designs into equipment, and costs flow from budgetary estimates into firm costs for implementation.

As illustrated in Fig. 1, integrated system design demands: establishing requirements; solving problems functionally; iterating to close on a solution; and then translating results to designs as specified by drawings, specifications, and other documentation. This approach provides higher confidence for achieving objectives because the discipline dictates that all variables, not just those that are part of the design, be addressed.

Fig. 1.
System engineering requires not only understanding the methodology, but also requires creativity, an open mind, and fresh approaches.
The Role of Creativity
Creativity is essential to modern Systems Engineering Management because the technology is changing quickly and many approaches, which previously were not feasible, can now be implemented. A positive approach to analyzing these new situations is required and particularly in ER projects because to date, mixed waste remediation approaches have not resulted in easy answers with existing technology. Roadblocks in thinking will assure that the inefficiencies of the old ways will prevail, and project success will be questionable.
Creativity, like systems, is an iterative approach. Creativity implies an absence of preconceived ideas (open mind) and positive approach (can do attitude).
An open mind will accept that there are many solutions to a problem which will meet the functional requirements. A positive approach will provide the motivation to follow through to success.
Communications, System Projects, and the Microcomputer
Excellent and speedy communications are necessary in the system engineering approach so that needs and data can be delineated in timely fashion to those who must take action.
The one practical tool that can be most important to the Project Manager and the Team, so that data and information is readily available and can be transferred quickly, is to have microcomputers for all team members. It is not necessary that the computers be networked although that will obviously improve communications; if everyone uses the system. The major benefits of the microcomputer will be that once the information is in, it will be available for use as often as required, and it can improve productivity Thus, a table with data developed for characterizing the soil (compounds, values) on the Vortec Vitrification Demonstration Plant Project is not only communicated to the design engineers, but it is also easily accessed and inserted into the monthly and final reports,, and it can be quickly available to respond to either the customer or a refractory supplier.
THE ER PROJECT- ACHIEVING SUCCESS
Two critical phases in ER projects require the most attention to achieve success. They are:
  Phase 1 - System Concept
  Phase 2 - Detailed Design
Phase 3 - Implementation, when the design is installed, operated, and the environmental remediation accomplished, while important, is not considered critical (until things fail to go as planned) because it is totally dependent on the first two phases. In this phase, correction of problems will be have the highest cost impact and can contribute heavily to schedule delays. It is therefore extremely cost effective to devote the most attention in Phases 1 and 2 to items which historically have contributed the most to problems encountered in Phase 3. In general, these have been in the area of requirements, interfaces, and organization.
To provide the greatest help to the executives and Project Managers involved with ER projects, this paper will focus on critical issues in the first two critical phases of the project.
Phase 1, System Concept Definition, is the investigation phase where the remediation problem is analyzed and approaches are synthesized to select a concept which will meet the needs of the customer. The system requirements and the major subsystem requirements are defined. Data is collected, and all functions are analyzed. Systems analysis to quantify waste characterization, quantities, and throughputs are accomplished. Interfaces and critical technical issues are identified, and a baseline system concept for achieving the remediation is developed. The baseline concept scopes the cost of remediation.
Phase 2, Detailed Design, is the pre-implementation phase that results in the drawings and specifications that define the design of the system and equipment for remediation. Where there are existing proven methods to accomplish the remediation, the exact method in which the remediation will be carried out is specified in detail during this phase.
In Phase 3, the design is implemented either by fabricating and installing the designed system or by procuring the proven method necessary to achieve the desired results.

Techniques for use in Phases 1 and 2 which will enhance the probability for success in Phase 3 will be described in the following sections.

Phase 1 - System Concept Definition

This phase is possibly the most critical part of the project because the basic requirements are established. The objective of the effort and the criteria for completion are defined. Reviews of successful projects show that either, or both, of these items are never established, or they are unclear resulting in contractual debates. Requirements may change later, but the basic needs initially established identify the analyses to be conducted and the data to be collected.

There are three major tasks which can be achieved in this phase to enhance success. They are establishing Project Manager and the Team, developing the system requirements, and bounding the technical parameters.

The Project Manager and The System Engineer

Some organizations use Project Management as an executive development path for managers, and as a result they may have non-technical individuals manage ER projects. If the executive management understands the risks, and the associated costs, then this is an excellent technique for training future executives.

A higher confidence, reduced risk approach is to have a technically strong Project Manager with the necessary people skills. Because successful Project Managers need to provide guidance in technical approaches and problem resolution required on ER projects, purely administrative experience may not provide the "horsepower required". It is not always possible to obtain both qualities in one individual, and therefore it will be necessary to choose management over technical skills and support the Project Manager with a technically strong System Engineer.

The System Engineer must understand the total requirements before addressing the hardware, software, and interfaces of the project but is guided by these items while developing the requirements. The System Engineer, while concerned with all technical aspects of the project, usually has different skills than the classic Project Engineer. The Project Engineer historically has been concerned with the equipment aspects of a project. The System Engineer must be constantly aware of all the interfaces affecting the solution and not just the design engineering factors. Having engineered a "works" project may not meet the technical skills criteria in a leading edge project where the technical requirements may demand an understanding of air pollution control, nuclear parameters, physical characteristics of barriers, chemical interactions, mass and thermal balances, as well as complete engineering documentation.

There are two intangible characteristics the Project Manager and the System Engineer must possess. Both must be goal driven and output oriented. In a project where timely communications are imperative, it does no good to provide information to those who need it only when it is "complete". System engineering is an iterative and dynamic approach. One cannot wait for "completeness."

Many times the best available information and necessary assumptions must be used pending "complete" information.

On the Vortec Vitrification Demonstration Plant Project, it was not possible during Phase 1 to define the offgas from the melter so that the air pollution control system (APCS) could be developed because a site and waste stream had not been selected. Thus, a worst case soil was defined from the available characteristics for candidate sites. The "available" characteristics were used because the waste streams at candidate sites were themselves undefined. As a result, the APCS included acid gas cleanup. This equipment was removed when the selected host site did not have chlorinated hydrocarbons in the waste.

Finally, recognize that the most qualified Project Manager for one phase of a project may be totally inadequate for another phase. For this reason, it is imperative that senior management not take the selection process lightly.

Selecting the Team

Naming the project team is also an important decision. Many emphasize the need for management commitment and support as the Number 1 reason for project success. While important, many success stories continue to illustrate the ability to achieve the project objectives on time and within budget in spite of lack of management commitment. The specific reasons are debatable. However, a strong team and Project Manager that are goal directed and output oriented seems to be one reason.

Team capability includes a liberal amount of creativity. The role of creativity was stressed earlier. If it is lacking, roadblocks in thinking will assure that the

inefficiencies of the old ways will prevail, and project success will be questionable.

For all but the smallest project, the core team (those directing the overall efforts which are continuous throughout the project), must be dedicated to the project substantially full time. In Phase 1, these people, in addition to the Project Manager and System/Engineer, are key system analysts (heat balances, pollution control, waste chemistry, or other). An important, but little noticed key contributor, is a lead designer that can rapidly synthesize approaches and translate concepts into drawings.

The System Requirements Document

The System Requirements Document (SRD) defines functional requirements, design constraints, interfaces, and the acceptance criteria. It is the guiding document for all technical needs on the project and the key element in preventing technical surprises. It contains much more information than "Our General Spec." for civil projects because of the attention to function and interfaces. Finally, it indicates when the job is finished with the acceptance criteria. Because of its importance, it must be reviewed and approved by the customer to insure that the project is headed in the proper direction. To contribute to an on-schedule project, periodic review and feedback by the customer will allow the SRD to achieve "mid-course corrections".

Formats for the SRD vary. If one has not been specified by the customer or by the contractor's internal policies, an acceptable format is one which defines "requirements from top to bottom and from left to right". That is, the document will define requirements at the system level and for each major subsystem (top to bottom). It will also contain a sequence of events (timeline based description of operation) which establishes requirements beginning with the system initial conditions and ending with the completion of operations (left to right).

Bounding the Technical Parameters - Identifying the Drivers

The process and equipment are sized based on the Mass and Energy Balance of the system. When radionuclides are contained in the waste stream, the nuclear energy effects must be included. Developing a mass and energy balance is part of basic engineering, but it takes on added meaning in ER projects because of the need to account for hazardous materials before, during, and after the process. Therefore, the Mass and Energy Balance not only is necessary for sizing equipment, it is also an auditing tool and will aid in the Safety Analysis of the system.

The other types of analyses vary according to the technical content of the project. While initial analyses can be defined at project start, many will be defined as a result of analyzing the unknowns. Enough analytical effort must be achieved to designate the technical drivers. These may not necessarily be the ones identified at the start of the project. Because the drivers are primarily associated with cost and schedule, it is important to begin obtaining budgetary subsystem and system costs to develop implementation costs.

Phase 2 - Detailed Design

The most important items which will affect Phase 3 success (other than the design) are:

- The implementation cost
- The definition of subsystem/equipment interfaces
- Design oversights

The effects of these problems can be alleviated by:

- Technical Parameter and Equipment Cost Tracking
- Actively cooperating with suppliers to obtain more cost effective solutions and shorter schedules
- Interface Control Documentation
- "Red Team Reviews" before the preliminary and final design reviews

Proven techniques, requiring no special Management Information Systems, are described for accomplishing these tasks, and they are easily implemented.

Technical Parameter and Cost Tracking

The microcomputer is the tool for simplifying the effort. It will serve two purposes (among many) in technical parameter and cost tracking which are: 1) to store the Equipment List with the necessary data in spreadsheet form; and 2) to store other data and information about the system, subsystems, specifications, and drawings. This provides suppliers with the latest design information.

The Project Manager can assess the design in near real time by tracking technical

parameters and equipment costs rather than actual documentation. It is easier to determine the level of design definition by knowing the parameters and values that have been defined than by trying to determine a level of design where the documentation is said to be 50% complete.

The tracking is accomplished with a simple spreadsheet program on the microcomputer. For those with the proper skills, a database manager can be used instead; however, consider the time required to prepare the database program if using this option. A spreadsheet can be set up easily and quickly without programming skills. The Equipment List is established in a hierarchical manner beginning at the system level. The major subsystems and their equipment are defined along with the parameters associated with the equipment. In addition to the technical parameters, the other important items are cross referencing (drawing and specification numbers), lead times, and costs.

Costs should include equipment, installation, testing, engineering, and any special costs contributing to total system/project cost. The costs can be those items identified on the Government Standard Form 1411 although this depends upon the accounting system/cost information required by either the contractor or customer. With the spreadsheet, total costs can be rolled up in columns to generate assembly, subsystem, or system costs and also summed by row to provide the total cost of the piece of equipment. This is important during cost reduction exercises when equipment additions or deletions are made. The other benefit is that it aids the team to focus on all costs other than purchase cost. For instance, procuring a motor/generator at a lower price may not reduce project cost if additional engineering is required for repackaging or changing an interface because of a new configuration. Representative parameters tracked during Phase 2 of the Vortec Vitrification Demonstration Plant Project are listed in Fig. 2.

Fig. 2.

Supplier Development

Working with the vendors actually begins during Phase 1. Because requirements are broadly defined, the activity is usually at a low level. It will benefit the project to become active in interactions with all potential suppliers. Some companies wait until all requirements and the design is neatly established before requesting quotes from suppliers. By actively interacting with suppliers, the design will proceed faster and can contribute to an increase in the excellence, and therefore quality, of the design.

Providing requirements and approaches to potential suppliers at an early date, allows them to become more familiar with the design issues and effective concerning their products. By participating in the design and costing activities with budgetary estimates during the early stages and refining these estimates as the design proceeds (See Fig. 1), an early warning can be obtained on potential cost problems. The usefulness of the microcomputer can be established during this process by having the latest documentation always available. Drawing changes or sketches are provided more quickly. Suppliers can be given key information at the end of a meeting when the computer to take notes during meetings.

Active cooperation with suppliers is a concept upon which all may not agree. There are many reasons companies state for keeping suppliers in the dark until design completion. They range from "The suppliers do not want to be involved until the design is completed" to "Our company policy prohibits it". Successful Project Managers continue to overcome these roadblocks and demonstrate that designs can be continuously improved and the costs can be reduced. This happens because suppliers want to provide a service to customers in order to obtain business and a better informed supplier can do this.

Interface Control Documentation

Definition and control of the interfaces during Phase 2 can improve Phase 3 success. In general, items of equipment from one supplier will usually operate successfully in a standalone mode. It is when equipment from different suppliers is brought together to operate as a system that the problems occur. Controlling interfaces and how they affect performance on both sides of the interface lies at the heart of system integration in ER projects.

The Interface Control Documentation (ICD) will be used to control these interfaces. While its use is primarily between major items of equipment from different suppliers (for instance a packaged Air Pollution Control System from Vendor A and the Process Monitoring and Control System supplied by Vendor B), the ICD can also be used for

other critical interfacing equipment under the design control of one vendor. The ICD results from the design drawings and specifications, but it focuses on the interfaces between pieces of equipment within the system or equipment outside the system that interfaces with the system.

Interfaces are defined, as appropriate, in the ICD prepared by the system integrator. This documentation is then reviewed and approved by all parties to the interface. It then becomes the binding document for the interface. Any changes to the interface, must result in a review and approval to changes to the ICD by all parties again. This is particularly important where control signals and timing are critical, but is also necessary where minor changes in physical interfaces can have a large impact on system performance (for instance a new seal at a pipe flange that results in only a material change but which could ultimately fail due to temperature extremes).

Red Team Review

The Red Team (independent) Review can result in raising emotions on a project. The benefits of avoiding adverse cost, schedule, and performance impacts will overshadow the emotions. Some Project Managers may initiate the independent review at critical events in the life of the project (at final design review for instance), but most result from an executive's initiative. The Red Team Review can be the Project Manager's best tool in the pre-implementation phase for assessing the design.

In carrying out the review, the reviewers are provided with the System Requirements Document, the major design drawings, a description of the system operation or flow, and any other major relevant information about the project, requirements, and design. This information should be provided at least two weeks before the review and earlier if possible. At the presentation to the reviewers, changes occurring since the documentation was provided are highlighted. All major project participants should be in attendance to hear the questions/comments of the reviewers. Subsequent to the review, the Red Team will provide written comments.

It would be nice to have a critique at each major design review, but this may not be cost effective. As a minimum though, a review should be held prior to the final design review with the customer.

SUMMARY

A new environment exists for managing Environmental Restoration projects. For projects requiring multidisciplinary technologies, "first time" design, and innovative equipment operation, the systems engineering approach provides a high confidence, reduced risk method for managing the project from the development of the concept through the implementation. Using the systems approach and the techniques and methods presented will help the Project Manager:

- Reduce total life cycle cost
- Integrate the project, and
- Anticipate and solve problems

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ENVIRONMENTAL COMPLIANCE MANAGEMENT SYSTEM DEVELOPED BY THE DOW CHEMICAL COMPANY AT TEXAS OPERATIONS

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ABSTRACT

A cross-functional team of environmental regulatory experts, plant managers, and plant engineers have been working since 1991 on the development, implementation and maintenance of the Environmental Compliance Management System. The Environmental Compliance Management System is a practical and accurate method of determining the applicability of the state and federal environmental regulations and of establishing standard and straightforward procedures to meet these requirements. The Environmental Compliance Management System allows individual manufacturing facilities to avoid the additional manpower that would be required to read, digest and decide on the applicability and plan of action to meet the requirements of all the environmental regulations.

INTRODUCTION

In 1994, a record number of enforcement actions were brought and a record number of civil and criminal penalties were collected by the EPA.*

TABLE I

Voluntary compliance through self-auditing and pollution prevention programs offer an excellent alternative to achieving environmental compliance. However, the issue of understanding what a facility must do to be in 100% compliance with the applicable regulations in an area of ever changing regulations is complex given the fact that the federal environmental regulations have approximately 2,000 changes monthly.

A complete Environmental Compliance Management System is the best insurance to fight against fines, penalties, and business failure due to environmental non-compliances. The Dow Environmental Compliance Management System is consistent with the U. S. Sentencing Commission Environmental Guidelines and with ISO 14000.

Federal Operating Permit applications will require a comprehensive regulatory applicability assessment and a determination of the compliance requirements, including plans and schedules. Deadlines in a compliance schedule must be met because the schedule is the same as having an administrative consent order and the applicants can be fined or brought to court if the dates are not met. The Operating Permit program now is a program in which facilities must prove compliance--not one in which regulators are charged with finding noncompliances.

THE DOW SOLUTION

An environmental compliance system was developed by The Dow Chemical Company as a comprehensive management approach for environmental issues in chemical manufacturing facilities. The main causes of compliance deficiencies are inadequate operating discipline, inconsistent regulatory interpretations, and limited knowledge of the requirements

Dow has invested a considerable amount of time and resources in this project to document that its operating units are fully in compliance with existing laws and regulations pertaining to the environment. This system included the knowledge of its seasoned regulatory experts with specific know-how related to the interpretation of and the impact of the law. The Environmental Compliance Management System gives managers the tools to implement a comprehensive, multi-media, uniform, cost effective, plant level compliance program. Comprehensive, multi-media (air, water and solid waste) compliance requirements specific to an operation are customized via the Environmental Compliance Charts and documented via standard computer software (MS Excel spreadsheet and MS Word wordprocessor) for maintaining and reporting compliance data. This system defines the measures of environmental performance against objectives to allow the line managers to assume responsibility for compliance, in addition to the environmental staff. Typically, monthly Environmental Progress Reports are submitted to key stakeholders. In addition, it provides the tools necessary for line managers to be fully knowledgeable about their environmental obligations and to manage their responsibility, authority, and accountability for the environmental performance of their operations. The day-to-day compliance requirements are incorporated in the routine operating discipline making manpower available to focus their efforts on more cost effective concerns.

To ensure compliance, Dow developed compliance modules for each state and federal regulation. Some 60 modules currently cover these regulations. A few of these modules are still in the development stage. They provide a vehicle to communicate between environmental and production personnel by giving a clear and consistent understanding of the implications and requirements of the regulatory laws. The Environmental Compliance Management System allows individual manufacturing facilities to avoid the additional manpower that would be required to read, digest, and decide on the applicability and plan of action to meet the requirements of each law.

A module condenses the consensus of the experts and gives feed back on resolutions of compliance concerns. They provide consistent interpretation of what is needed to document compliance and by their nature become a useful roadmap for environmental self-audits. The Environmental Compliance Management System is a practical and accurate method for determining the applicability of the law and of establishing standard procedures to meet these requirements. Standard compliance checklists help new personnel gain an understanding of the requirements with minimum training. With this system in place, process changes can be made without allowing the unit to be out of compliance.

MODULE CREATION OVERVIEW

The Environmental Compliance System was developed by The Dow Chemical Company, Texas Operations to achieve compliance with federal and state laws and regulations. The

key component of the Environmental Compliance Management System is the Environment Compliance Module. The purpose of these modules is to provide plant personnel with a practical tool to ensure alignment and consistency between their function and the overall process for maintaining compliance.
The utilization of an environmental compliance module can be divided into five facets, starting with the issue of a new environmental law or regulation up to the full documented compliance with the new requirements.
Regulation Issue Management
The Regulation Issue Management Team studies each new or proposed regulation or interpretation and applies Dow resources to develop a compliance plan that minimizes the cost impact. The approach involves representatives from the potentially regulated plants and environmental regulations experts within Dow.
Module Development
The Regulation Issue Management Team submits to the environmental compliance team all new environmental laws and regulations to be converted into workable modules. The module development team analyzes final regulations and associated documentation for the specific compliance requirements. These compliance requirements are documented into a modular format.
Fig. 1.
Field Application & Training
The field application and training facet involves customizing applicable modules and compliance requirements for the unit involved, listing those requirements on an Environmental Compliance Chart, and training employees to implement the compliance requirements.
Self Assessment
The self-assessment facet is a "spot-check" performed by plant management and plant environmental personnel to verify compliance. Self-assessment provides the assurance that the assigned personnel perform the required compliance tasks in a timely manner.
Compliance
Compliance is achieved when all the technical, monitoring, and reporting requirement are met and a working plan is in place committed to the continuance of the program.
DEVELOPING AND MAINTAINING AN ENVIRONMENTAL COMPLIANCE CHART
The Environmental Compliance Chart (ECC) is the core of this system and it summarizes the applicable compliance requirements at the plant or department level.
All required documentation (recordkeeping and reporting) applicable to the facility appears as individual tasks.
Specific form number and/or name are shown.
The regulation, permit, law, or policy requiring the task is listed.
All the paperwork supporting the completion of the task is filed and the file number is listed.
The name of the personnel responsible for the completion of the task is listed.
A column is dedicated to note the verification of records when continuous monitoring is required.
The ECC provides to management at a glance the key information to verify the facility environmental compliance status. All the compliance requirements are listed in one place with the names of personnel assigned to perform the compliance tasks. In addition, the frequency of the tasks is shown with indication of when it was last performed.
The ECC includes a frequency section (e.g. daily, weekly, monthly, etc.) showing the total number of tasks completed and not completed for each respective period. For example, one of the monthly tasks is an inspection by the department head of the status of the items due that month.
Environmental personnel use the ECC information as a basis to certify that a plant is in compliance. All the information is referenced with checklists, special forms, and documentation files.
DEFINING APPLICABILITY
The initial task of the Module Project Leader is to establish the applicability of the environmental regulation. Applicability is a process to determine the scope of the regulation and to identify what elements of the production facility are affected.
The applicability process is done via the interaction of production representatives and regulatory experts. During this process the scope of the modules is defined,

tailoring each of them to the needs of a production facility. All the requirements of the environmental regulations are listed and plant personnel identify which ones affect their site.

FIELD IMPLEMENTATION AND TRAINING

The implementation of the Environmental Compliance Management System starts with the identification of the applicable modules and compliance requirements for a given plant. The requirements or tasks are listed on the Environmental Compliance Chart and plant personnel are assigned to those tasks after a training session.

The steps of the implementation process are:

- Identify modules pertaining to the manufacturing plant
- Determine and document modules for applicability
- Assign personnel for the implementation of tasks
- Train personnel in the implementation of tasks
- Change module applicability if necessary
- Identify units, areas, equipment, subject to regulation
- Identify the compliance requirements
- Complete reporting requirements and note completion on the ECC
- Document required process changes
- Develop specific operating procedures for each required task
- Develop, conduct, and documents training sessions

SITE INSTALLATION

Applicability

Analyze the applicability of the environmental regulatory compliance requirements at the plant level.

Conversion

Customize the Environmental Compliance Modules consisting of an environmental encyclopedia with detailed operating discipline. With this system in place, line managers can look to much leaner corporate and business units with environmental staffs only for specialized expertise and service that cannot be provided cost effectively on a decentralized basis.

Training

Train plant personnel to incorporate the Environmental Compliance Management System into the company's operating discipline. The company operating culture is incorporated during the module conversion process and delivered to the company representatives charged with implementation.

Maintenance

Institute a system maintenance process capable of incorporating new and amended regulations as they are promulgated and adjusts to reviews made of capital projects regarding applicable regulations and interpretation.

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CYANIDE MIXED WITH RESIDUAL RADIOACTIVE MATERIALA UNIQUE REGULATORY COMPLIANCE PROBLEM*

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ABSTRACT

In 1981, the U.S. Department of Energy (DOE) Grand Junction Projects Office (GJPO) was assigned responsibility for the Uranium Mill Tailings Remedial Action Grand Junction Vicinity Properties Project. This project required planning, characterization, design, and remediation of more than 4,000 properties in and around Grand Junction, Colorado. These properties were contaminated with radioactive uranium mill tailings (residual radioactive material) as a result of milling operations conducted for the U.S. Government during the 1950s and 1960s.

During the remedial action activities conducted by DOE-GJPO and its contractor, Rust Geotech, a number of Resource Conservation and Recovery Act issues were identified at some of the vicinity properties. One of these vicinity properties was a machine shop whose principal operations consisted of manufacturing, repairing, and distributing drilling and other types of heavy equipment. As part of the operations,

a sodium cyanide bath was used in a metal heat-treating and case-hardening process. This process produced cyanide contamination that was commingled (mixed) with the soil and residual radioactive material surrounding the heat-treating area. This contamination posed a unique mixed-waste problem that required concurrence from the Colorado Department of Health and Environment (CDPHE) for the hazardous waste determination decisions before site remediation could be initiated.

OVERVIEW

The U.S. Department of Energy (DOE) Grand Junction Projects Office (GJPO) administers the Uranium Mill Tailings Remedial Action (UMTRA) Grand Junction Vicinity Properties Project in and around Grand Junction, Colorado. These properties were contaminated with radioactive uranium mill tailings (residual radioactive material [RRM]) as a result of milling operations conducted for the U.S. Government during the 1950s and 1960s.

Several Resource Conservation and Recovery Act (RCRA) issues have been identified during remedial action of these vicinity properties. One of these vicinity properties is a machine shop operation, which has been in operation for approximately 40 years. An area outside the machine shop was used for heat-treating and case-hardening metal parts. Commingled waste was identified in the heat-treating operation area (see Fig. 1).

Fig. 1.

The commingled waste presented an unique mixed-waste problem that required Colorado Department of Public Health and Environment (CDPHE) concurrence for the hazardous-waste determination decisions before site remediation could be initiated.

DEFINITIONS

Commingled Waste: Commingled waste is defined as material composed of RRM, or RRM and a solid waste, as defined in Title 40 Code of Federal Regulations (CFR) Part 261.2 (1), commingled with a RCRA hazardous waste or a Toxic Substances Control Act (TSCA) regulated substance (specifically polychlorinated byphenyls [PCBs]). RRM is excluded from the RCRA definition of a solid waste in 40 CFR 261.4, "Exclusions," (2).

RRM: The Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA), as amended, 42 United States Code, Section 7901 to 7925 (3), defines RRM in Title 1, Section 101, Definitions(7)(A), as "Waste . . . in the form of tailings resulting from the processing of ores for the extraction of uranium and other valuable constituents of the ores . . . ."

METAL HEAT-TREATING PROCESS

The following process describes the metal heat-treating operation conducted at this site:

Granular sodium cyanide is placed in a heat-treating vat, where it is heated to approximately 927C. Sodium cyanide is the only ingredient in the heat-treating vat.

Metal parts connected to a pull tool (or some other lifting mechanism) are lowered into the molten solution in the heat-treating vat.

The hot metal parts (after several hours) are extracted from the solution and transferred to an adjacent container of water for quenching. Sometimes this quenching bath is located approximately 3 to 5 meters away from the heat-treating vat.

Sodium cyanide residue (in the heat-treating vat) is treated on site by mixing it with chlorine bleach. The resultant substance is then sent off site for disposal or recycling.

The quenching bath is cleaned periodically. The wastewater sludge is sent to an off-site facility for recycling.

The following field observations were noted as a part of the hazardous-waste investigation:

Molten sodium cyanide was observed dripping onto the concrete slab as the metal parts were transferred from the heat-treating vat to the quenching bath.

The quenching bath container does not have a mechanism for preventing liquid drift. Drift occurs when the hot metal is placed into the quenching bath.

POSSIBLE HAZARDOUS WASTE SOURCES

Potential hazardous-waste sources that may have contaminated the soil and RRM on this vicinity property are

Spills or discards of pure, unused sodium cyanide.

Spills or drips of molten sodium cyanide solution and drift from the quenching bath.

Spills or disposal of sodium cyanide residue waste generated during cleaning of the heat-treating vat.

Wastewater or sludge from the quenching bath discarded on site.

ASSESSMENT OF HAZARDOUS WASTE SOURCE

Following is an evaluation of the applicability of the RCRA regulations to the potential hazardous waste sources identified:

Pure, unused sodium cyanide is a listed RCRA hazardous waste if spilled or discarded. Under these circumstances, its hazardous waste number is P106. In the event that pure, unused sodium cyanide is spilled or discarded in a manner that contaminates the soil, any concentration of the sodium cyanide contained in the soil will be a P106 listed hazardous waste.

The molten sodium cyanide solution used during the metal heat-treating process is not regulated as a listed waste; however, it qualifies as a characteristic hazardous waste if it is spent and it exhibits the characteristic of reactivity. Therefore, any spills from this source will not be considered to be a listed waste; however, any material contaminated by the spill will be regulated as a characteristic reactive waste (hazardous waste number D003) if it meets the regulatory definition. Specifically, reactivity is established if the waste reacts violently with water or generates toxic gases, vapors, or fumes in a quantity sufficient to present a danger to human health or the environment when exposed to pH conditions between 2 and 12.5. Reactivity can also be assessed by determining the concentration of total releasable cyanide in a waste sample. The regulatory limit for this determination is 250 milligrams of hydrogen cyanide per kilogram of waste in soil.

NOTE: Once used, sodium cyanide will not qualify as P106 (a pure chemical product).

The quenching bath water will only be regulated as a hazardous waste if it is spent or discarded and meets the definition of a reactive waste. Therefore, drift from this source will not be considered to be a listed waste. However, any material contaminated from this source will be regulated as a reactive waste (D003) if it meets the regulatory definition.

The sodium cyanide residue generated from vat cleaning could be identified as a listed waste with a waste number of F011. The residue contains spent cyanide solutions left in the vat after the heat-treating process and material generated during the actual cleaning of the vat. The hazardous constituent for which F011 is listed is cyanide (salts).

At this vicinity property, the sodium cyanide residue is treated on site and subsequently managed off site. Any spills resulting from the vat emptying or cleaning process could potentially result in soil containing a listed waste (F011) as determined by any detected concentration of cyanide (salts). If the spent sodium cyanide solution or residue is considered not listed, this same solution or residue and any material discarded or contaminated from a spill could be considered a reactive waste (D003) if it meets the regulatory definition.

Although it does not meet the definition of a listed waste, wastewater from the quenching bath could be a reactive waste (D003). Soils contaminated by spilled or discarded quenching bath wastewater will be regulated as a reactive waste (D003) if they meet the regulatory definition.

Wastewater sludge generated in the quenching bath is a RCRA-listed hazardous waste (hazardous waste number F012). If the sludge is, or has been, inadvertently spilled or discarded on site, soil that contains any concentration of the listed waste will be considered a listed waste. The hazardous constituent, for which F012 is listed, is cyanide (complexed).

ASSESSMENT OF SOIL CONTAMINATION SOURCE

Listed-Waste Sources

Possible listed-waste sources for the cyanide identified in the soil at this site include P106 from spilled or discarded pure product, F012 from spilled or discarded quenching-bath wastewater sludge, and F011 from spilled or discarded sodium cyanide residue generated during emptying or cleaning of the heat-treating vat. Process knowledge for the metal heat-treating operation does not demonstrate mismanagement of any of these listed waste streams. The owner or operator relies upon off-site recycling or disposal of the quenching-bath wastewater sludge and the discarded sodium cyanide residue generated during emptying or cleaning of the heat-treating vat.

Characteristic Waste Sources

Process knowledge for the metal heat-treating operation does not demonstrate

mismanagement of the sodium cyanide residue from the heat-treating vat. However, the molten sodium cyanide was observed dripping onto the concrete slab as the metal parts were transferred to the quenching bath. Drift during the quenching process was also observed. These sources, along with the possible discarding of the wastewater, are the probable sources of the total cyanide detected in the soil samples collected in the vicinity of the concrete slab. Therefore, the soil will qualify for management as a characteristic hazardous waste if it meets the definition of reactivity.

SITE CHARACTERIZATION ACTIVITIES

Sampling was performed to evaluate the potential of hazardous waste commingled with RRM. This sampling was restricted to locations of assessed radiological contamination determined most likely to contain hazardous waste. Factors considered in this determination included drainage areas for waste materials generated inside the machine shop (the concrete sumps); process knowledge of the activities and materials used in the heat-treating area; visibly stained or discolored soil in the heat-treating area; and anecdotal evidence about the existence of potential waste in areas of assessed radiological contamination. The metal heat-treating area was targeted for investigation on the basis of process knowledge from observation of on-site operations.

A series of site-specific sampling and analysis plans was developed and implemented with a combination of authoritative and systematic sampling techniques. Analyses performed during these investigations included Target Compound List (TCL) volatiles, TCL semivolatiles, Toxicity Characteristic Leaching Procedure (TCLP) metals, flammability, and the characteristic of reactivity with regard to cyanide content.

Phase 1

The first phase of the site characterization effort consisted of investigating the sumps inside the machine shop, the water meter pit, and the soils adjacent to the heat-treating area. Because no residues were discovered in the machine shop sumps, only liquid samples could be collected. These samples were analyzed for TCL volatiles, TCL semivolatiles, TCLP metals, and cyanide. The sump samples are designated as Locations 1 and 2 on Fig. 1.

No liquid was present in the water meter pit; therefore, a grab sample of soil was collected from the bottom of the pit and analyzed for flammability. This sample location is designated as Location 3 on Fig. 1.

Soils adjacent to the concrete pad in the heat-treating area were characterized by collecting a composite sample consisting of four aliquots. The positions of these aliquots are denoted on Fig. 1 as Locations 4 through 7. These soil samples were analyzed for TCLP metals and cyanide. A cyanide concentration of 30.6 parts per million (ppm) was detected in the composite sample collected from Locations 4 through 7. This detection prompted further characterization of soils in this area to define the areal extent of elevated cyanide concentrations. To complete this characterization, discrete samples were collected from Locations 8 through 13. The applicable reactivity test for soils collected from this site should evaluate the total releasable cyanide concentration (e.g., in an extract). The regulatory limit defines reactivity as 250 milligrams of hydrogen cyanide per kilogram of soil (ppm). This regulatory limit is specified in EPA's Test Methods for Evaluating Solid Waste, Chapter 7, Volume IC (4).

Phase 2

Because the analytical results from Locations 8 through 13 and those from the composite sample collected at Locations 4 through 7 indicated that elevated cyanide concentrations were mixed with radiologically contaminated soils, a second sampling activity took place. Locations 4 through 7 were resampled individually to quantify the cyanide concentrations at each discrete location. In addition, samples were collected from the concrete cuttings from the concrete pad in the heat-treating area and in the soils immediately beneath the pad. These samples were analyzed for cyanide content.

Results of analytes, equal to or exceeding typical contract required detection limits, are presented in Table I (Phase 1) and Table II (Phase 2). These analytical data were reviewed for compliance with laboratory quality control and data acceptance procedures. Those departures, from standard laboratory quality control measures, that did occur did not impede the effective evaluation of the analytical data for the presence of hazardous waste on this property.

TABLE I

TABLE II
Depth is reported in inches to reflect actual measurements taken at this UMTRA vicinity property.
SUMMARY OF SITE CHARACTERIZATION RESULTS
Evaluation of Characteristic Hazardous Waste: Sampled materials that met the regulatory definition of a waste and contained elevated cyanide concentrations potentially qualified as being hazardous in relation to the characteristic of reactivity. The sources of the elevated cyanide concentrations in the sampled material were identified in a previous section.
Elevated cyanide concentrations were noted in the soil samples collected from Locations 4 through 14 and in the concrete cuttings sampled at Location 14. However, total releasable cyanide concentrations in all of these samples were less than the regulatory limit of 250 ppm; therefore, these materials do not exhibit the characteristic of reactivity.
The toxicity characteristic is measured by analyses using the TCLP. Samples from Locations 1, 2, and 4 through 7 (composite) were analyzed for TCLP metals. None of the metals on the TCLP-metals list were detected in these samples as leachable concentrations exceeding contract-required detection limits.
The characteristic of ignitability was not suspected on basis of the absence of potentially ignitable-free liquids in any of the sampled areas. Although not a characteristic waste concern, a soil sample from the water meter pit (Location 3) was investigated for flammability to determine if the material had been affected by petroleum products and if it would present a hazard during any future transportation off the property. Results indicated the material tested does not present a flammability hazard.
There was no reason to suspect that radiologically contaminated materials on this site were excessively corrosive; therefore the characteristic of corrosivity was not investigated.
Evaluation of Listed Hazardous Waste: Several cyanide-based listed wastes that were potential sources of the elevated cyanide concentrations in the soils and concrete in the heat-treating area were identified. These possible sources could have included pure sodium cyanide that was spilled or discarded; spilled or discarded sodium cyanide residue generated during emptying or cleaning of the heat-treating vat; and spilled or discarded wastewater sludges generated in the quenching bath. Observations and historical research of site operations and the heat-treating process demonstrated that these waste streams are handled through off-site recycling or disposal; there was no evidence of spillage or mismanagement. Therefore, it was concluded that listed hazardous wastes were not the source of the cyanide contamination in the soils and concrete in the heat-treating area.
The contents of the two sumps inside the machine shop area (Locations 1 and 2) were analyzed to determine the chemical composition of these waste materials. If these materials were concluded to be hazardous, their leakage from the sumps into surrounding radiologically contaminated soils would have created the potential for commingled waste. The analytical results indicated the presence of detectable concentrations of the following TCL volatiles: acetone, chloroform, xylene (m, p), and bromodichloromethane. In addition, a number of tentatively identified compounds were detected during the TCL-volatile and TCL-semivolatile analyses. The minuscule concentrations of the positively identified contaminants (with the acetone from the sump [Location 1] being the largest concentration at 0.094 milligrams per liter [mg/L]) suggests that these constituents were present in the samples as a result of laboratory contamination rather than from on-site machine shop operations. The radiologically contaminated soils in the area of the sumps did not contain listed hazardous waste.
Evaluation of Toxic Substances (PCBs): No samples were analyzed for PCBs because there was no process-related information that would indicate spills or dumping of materials containing PCBs. There was also no reason to suspect the presence of any other toxic substance regulated by the EPA under TSCA.
CONCURRENCE OF STATE REGULATORY AGENCY
To enable DOE-GJPO to remediate the RRM on this property, concurrence was required from the CDPHE Hazardous Materials and Waste Management Division. A technical discussion document, prepared and presented to CDPHE, detailed the initial evaluation of this property. This evaluation included site background and process information, assessment of possible waste sources, assessment of hazardous waste

source, and the results of the Phase 1 site characterization activity. The technical discussion document scrutinized the materials and activities associated with the heat-treating process; assessed potential cyanide-based waste streams; addressed regulatory issues associated with the potential cyanide waste streams by examining the relevant classification of these wastes with regard to RCRA regulations; reported the results of the Phase 1 investigation; and recommended a course of action for a further investigation (Phase 2) on the basis of the regulatory conclusions.
This cooperative effort resulted in DOE receiving written concurrence from the section chief of the CDPHE Hazardous Materials and Waste Management Division accepting the regulatory determinations presented in the technical discussion document and approving the recommendation for conducting a Phase 2 site characterization. The results of the Phase 2 site characterization did not identify either characteristic or listed hazardous wastes. The results of the Phase 2 site investigation were submitted to CDPHE along with a recommendation to remediate the RRM on this property.
This vicinity property has subsequently been remediated of RRM. The radiologically contaminated materials were excavated and removed from the property. The disposal site for the RRM was the UMTRA Cheney Disposal site southeast of Grand Junction, Colorado.
REFERENCES
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4. U.S. Environmental Protection Agency, Test Methods for Evaluating Solid Waste, Chapter 7, Volume IC, Third Edition (November 1986).
5. 40 CFR 268, U.S. Environmental Protection Agency, "Land Disposal Restrictions," U.S. Code of Federal Regulations.

46-15
RECONFIGURATION OF THE DOE MOUND AND PINELLAS PLANTS FROM OPERATIONAL FACILITIES TO REMEDIATION AND PRIVATELY OWNED INDUSTRIAL FACILITIES
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ABSTRACT
The Department of Energy (DOE) Mound and Pinellas Plants are part of the DOE nuclear weapons complex current reconfiguration effort and have been designated for transition from operational facilities to Environmental Restoration Programs and private industry ownership. A critical element during the initial stages of transitioning is proper identification and classification of materials, equipment, and wastes. A logic process was developed and incorporated as a tool to provide direct guidance to the site's transition and environmental management programs. The intent of the logic process is to help identify and establish the correct sequence for decontamination and decommissioning (D&D) of buildings, routing of materials and equipment, and disposition of any waste generated. Each plant used the logic process to develop site-specific transition plans.
Mound and Pinellas Plants have realized benefits from establishing a formal logic process that includes only contaminated materials are processed for decontamination, better management of hazardous materials, reduced cost, improved regulatory compliance, and effective use of resources and time.
INTRODUCTION
The transition effort at the Mound and Pinellas Plants is described below to demonstrate the magnitude of the remediation, material, and waste management problems associated with transition. The logic process reduces the potential for

radioactive and hazardous cross contamination of buildings, equipment, and materials. When cross contamination does occur it can result in generation of a radioactive RCRA contaminated mixed waste. This increases the decontamination and disposal costs.

SITE DESCRIPTION

The Mound Plant identified 72 buildings for transition. The buildings are in various stages of remediation and range in size from 25,700 to 183,000 square feet. Some of the buildings are being demolished, a few are in the process of decontamination, and other buildings have already been signed over to the Miamisburg community for industrial development. These buildings contain production areas, storage areas, office areas, various chemicals, laboratories, materials, process equipment, personal property, and building systems, all requiring transition.

The Pinellas Plant used a slightly different approach and identified 110 areas to be characterized and cleaned. These areas range in size from 100 to over 21,000 square feet. Most of the areas are in a single building and are at various stages of remediation; some are completed and are currently being leased by private companies for industrial development. Included in the Pinellas areas for transition are production areas, storage areas, office areas, various chemicals, materials, laboratories, process equipment, personal property, and building systems.

PLAN PROCESS DESCRIPTION

Development of the logic process began in May 1994 in Miamisburg, Ohio. The original concept focused on the development of waste minimization (WMin) procedures to help reduce or eliminate the volume of waste generated during the transition process. The resulting process, illustrated in Fig. 1, began with selection of a representative building, a potential waste material such as a chemical or process equipment, and the building systems (air handlers, mechanical pumps, and air duct systems).

The chemicals were determined to be either non-radioactive (clean) or radioactive (dirty). The chemicals were then determined to be reusable or recyclable. The question to reuse or recycle was always asked regardless of whether the chemicals were clean or dirty. If the chemical could not be reused or recycled, the final decision was disposal.

Using the plan logic process, equipment was evaluated in much the same way as chemicals, but the greater cost of equipment encourages the user/owner to try to decontaminate, reuse, or recycle equipment before declaring it waste.

In considering a building's systems, the building manager should anticipate the building's planned future use and what processes it will house. The future use of a facility may be identical to past operations and require minimal decontamination, or the opposite may be true. Decisions concerning the building's systems are typically the same as for the equipment: planners must identify which systems are contaminated and require decontamination, and which systems can be left in place or processed for disposal.

Fig. 1.

CHEMICAL INVENTORY EVALUATION FOR DISPOSITION

Every effort should be made to reuse or recycle a material and not dispose of it as waste. A number of reuse opportunities are identified in Fig. 2. For example, if a chemical has been used in a Radioactive Material Management Area (RMMA), the user must determine if the chemical is radioactive. Sampling and analysis may be required to make this determination. If the chemical is clean, a determination can be made concerning its suitability for reuse. If the clean chemical can be reused, it may be transferred to a new user. If it cannot be reused, the next decision is to determine whether the chemical is RCRA regulated and required to meet RCRA disposal requirements.

At the RMMA decision block, if the chemical is determined to be radioactive (dirty), a determination is made whether the chemical is also RCRA regulated. If it is not RCRA regulated, it is low-level radioactive and not mixed. The next decision is to determine whether the dirty chemical can or should be decontaminated. If it cannot be decontaminated, the decision is made whether the chemical can be reused. If the dirty chemical can be reused, transfer to the new user is required. If it cannot be reused, a declaration is made that the dirty chemical is now a low-level waste (LLW) and can be processed for shipment to a DOE-approved disposal site. If the dirty chemical is decontaminated, a decision is made whether the clean chemical can be reused. If so, it may be transferred to a new user. If not, the chemical can be prepared for proper treatment and disposal.

If the dirty chemical is determined to be RCRA regulated and radioactive, it is mixed, but not necessarily a waste. A decision is now required to determine whether the mixed chemical should or can be decontaminated. If it cannot be decontaminated, a decision is made that the chemical is a mixed waste (MW). The MW is included in the site's plan for mixed waste treatment and disposal.
If the mixed chemical can be decontaminated, the next decision is whether to reuse. If the chemical cannot be reused, it is prepared for shipment for treatment and disposal at an approved treatment, storage and disposal (TSD) facility. If the chemical can be reused, it may be transferred to the new user.
Fig. 2.
PROCESS EQUIPMENT EVALUATION FOR DISPOSITION
If the process equipment is clean of radioactivity, a decision is made regarding RCRA contamination, as illustrated in Fig. 3. If there is no RCRA contamination, a decision is made whether the equipment can be reused locally. If the clean equipment cannot be reused locally, it is declared Excess Property and formal procedures implemented to offer it for reuse off site. If the clean equipment can be reused, it may be transferred to the new user. If the clean equipment is RCRA contaminated, a decision is made whether the equipment can or should be decontaminated. If the equipment can be decontaminated, it is considered for reuse or declared as Excess Property. If the RCRA contaminated equipment cannot be decontaminated, a decision is made whether it can still be reused. If it can be reused, it may be transferred to the new user. If it cannot be reused, it is prepared for RCRA treatment and disposal.
If the equipment is radioactive the next decision is to determine whether the equipment is also RCRA contaminated. If the radioactive equipment is not RCRA contaminated, it is radioactive only and the next decision is whether it can be decontaminated. If it cannot be decontaminated, the next question is whether there is a user for the radioactive contaminated equipment. If there is, the equipment is transferred to the new user. If the equipment cannot be reused, it is declared low-level waste and requires disposal at an approved DOE facility. If the dirty equipment can be decontaminated, a decision must be made whether the equipment can be reused. If it cannot be reused after decontamination, then the equipment is Excess Property. If the equipment can be reused, it can be transferred to the new user.
If the radioactive equipment is also RCRA contaminated, it is mixed but is not necessarily declared a waste. The decision is made whether it can be reused as-is. If not, then a decision is whether the equipment requires decontamination. If the equipment cannot be decontaminated, a decision is made that the equipment is a mixed waste (MW). The MW is included in the site's plan for mixed waste treatment and disposal. If it can be decontaminated, an effort is made to reuse or recycle the equipment. If the decontaminated equipment can be reused, then it can transferred to the new owner. If the decontaminated equipment cannot be reused, it is prepared for RCRA treatment and disposal.
Fig. 3.
BUILDING SYSTEMS EVALUATION FOR DISPOSITION
The same basic logic process is used with a building's systems, as illustrated in Fig. 4. If elements of a building's system (air handlers, mechanical pumps, or air duct system) are clean, the decision is made whether the system is RCRA contaminated. If the system is not RCRA contaminated, it can possibly stay in place. If the clean system is RCRA contaminated, a decision is made whether to decontaminate. If the clean system can be decontaminated, the question is whether the system can stay in place. If the clean system cannot be decontaminated, a decision is to be made whether the system is to be reused or dismantled for proper RCRA treatment and disposal. If the clean system is not to remain in place, dismantling and packaging for RCRA treatment and disposal are required.
If the building system is determined to be radioactive a decision is made whether the system is also RCRA contaminated. If it is not RCRA contaminated, a decision is made whether the system can be reused as is or whether it requires decontamination. If it can be reused as is then the system can stay in place. If the radioactive system can only stay in place if decontaminated then an effort is made to decontaminate. If the radioactive system cannot be decontaminated then the system is declared LLW and packaged for proper disposal at a DOE-approved facility.
If the radioactive system is determined to be RCRA contaminated, it is declared

mixed, but not necessarily declared a waste. A decision is made whether the system can stay in place as is or if it requires decontamination. If it can stay in place as is no further action is required. If the mixed system requires decontamination but cannot be decontaminated, a declaration is made that the system is now a mixed waste (MW). The MW is included in the site's plan for mixed waste treatment and disposal. If the mixed contaminated system can be decontaminated, a decision is then made whether the system can stay in place.
Fig. 4.

CONCLUSION

The ultimate goal of facility transition includes the proper disposition of materials, equipment and building systems. In this capacity, the logic process ensures reduction of waste, maximum reuse or recycle, and D&D only of building systems and materials that require it. The benefits of this process include improved Health and Safety, reduced cost, timeliness, efficient use of resources, and recovery of materials and equipment for reuse or recycle.

REFERENCES

1. ENVIRONMENTAL SCIENCES DIVISION, "Resource Conservation and Recovery Act," Oak Ridge National Laboratory, Oak Ridge Tennessee (Mar. 1992)
2. U.S. DEPARTMENT OF ENERGY, "5820.2A Radioactive Waste Management, Chapter V, Decommissioning of Radioactively Contaminated Facilities," (Sept. 1988)

46-16

EVALUATIONS OF NATURAL GAS HAZARDS AT THE FORT ST. VRAIN ISFSI

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ABSTRACT

Public Service Company of Colorado (PSC) is considering repowering the Fort St. Vrain generating station with natural gas-fired combustion turbines and heat recovery steam generators. The Nuclear Regulatory Commission had concerns over potential natural gas hazards associated with repowering. PSC contracted Westinghouse to make an assessment of these hazards. Natural gas hazards evaluated by Westinghouse were potential explosions of unconfined vapor clouds from postulated gas line ruptures and a detonation within the Turbine Building and their impact on the Independent Spent Fuel Storage Installation Modular Vault Dry Storage (ISFSI). A series of analyses were performed using the state-of-the-art Vapor Cloud Explosion Damage Assessment Model (VEXDAM) (1) developed by Engineering Analysis, Inc. Our results showed that both the ISFSI and repowered Fort St. Vrain station could co-exist without a nuclear safety threat.

BACKGROUND

PSC is in the process of decommissioning the Fort St. Vrain High Temperature Gas-Cooled Reactor. Upon completion in 1996, there will be only one on-site facility containing radioactive material. This is the Independent Spent Fuel Storage Installation Modular Vault Dry Storage Facility (ISFSI). This facility currently stores the spent fuel from the Fort St. Vrain Station.

The ISFSI is a massive, reinforced concrete structure that is designed to withstand a 360 mph design-basis tornado.

Natural gas hazards in the vicinity of Fort St. Vrain have been previously analyzed. These hazards were associated with nearby natural gas wells and associated collection pipelines. The results of the analyses showed that there was no compromise to the ISFSI's nuclear safety function.

PSC has obtained Colorado Public Utilities Commission approval for repowering the site with natural gas-fired combustion turbines and heat recovery steam generators. The natural gas hazards considered included explosions of unconfined vapor clouds resulting from postulated gas line ruptures and a detonation within the Turbine Building and their impact on the ISFSI. The effects of the postulated natural gas explosions on the Reactor Building were not evaluated because significant quantities

of radioactive materials will no longer be available for release from the reactor building when natural gas is introduced.
Figure 1 shows proposed routing of gas lines at the Fort St. Vrain site. The following locations are relevant to the evaluations:
Fig. 1.
Point T is the tie-in location of the 12 inch service line feeding the combustion turbines to the 26 inch main supply header. The nearest approach is conservatively estimated to be 1,400 feet south of the ISFSI.
Point A is a routing option for the 26 inch Main Supply line, estimated to be located 5,280 feet southwest of the ISFSI.
Point C is located 4,300 feet to the west of the ISFSI and represents the closest approach of the 26 inch main supply header to the ISFSI.
Point Z is the location of the Fort St. Vrain Turbine Building where a confined natural gas explosion is postulated. The detonation was modeled at 1,737 feet from the ISFSI.

METHODOLOGY

The unconfined vapor cloud explosion analyses were performed in three steps. First, the amount of gas released from the pipeline ruptures was determined. Next, the downwind transport of the natural gas was evaluated with the DEGADIS (2) dispersion modeling code for several different wind speeds (2.0, 2.2, 4.5 and 6.7 mph) to determine the worst case scenario. This yielded the shape and extent of the unconfined flammable vapor cloud for the various wind speeds selected. The mass of natural gas available for an explosion was then determined for the plume considered to pose the most severe threat. This mass was adjusted upward for the elevated vapor cloud scenarios to account for the reflected shock wave (Mach region effect). Finally, the VEXDAM code was used to model the vapor cloud explosion and its impact on the ISFSI from both overpressure and impulse.
The confined vapor explosion in the Turbine Building was performed by determining the maximum amount of natural gas that could be present in the Turbine Building in a stoichiometric mixture with air. The VEXDAM code was then used to determine the overpressure and impulse at the ISFSI, assuming ignition of the mixture and confined detonation at ground level.

WORST CASE SCENARIO

To illustrate the methodology, a detailed description of the worst case, the 12-inch service line release will be presented. The results of all the other scenarios will then be summarized.

Natural Gas Release

The worst case release analyzed is the potential rupture of a 12 inch service line feeding the combustion turbines and a postulated unconfined vapor cloud explosion. The main supply line pressure is 875 psig and the service line was also evaluated at this pressure. A 20 foot diameter hole was assumed to be created as the gas jets from the rupture through the soil to the atmosphere. After time exceeds six seconds, the service line will depressurize and a steady state flow of 10,500 ft.3/sec. will result. A rupture with a release duration of three minutes was modeled to allow for full development of the flammable cloud. During this time interval, a total of 89,920 lbm of "Northern Colorado Standard" gas is released to the atmosphere.

Dispersion Model

The jet plume module of the DEGADIS dispersion modeling code was used to determine the trajectory of the flammable cloud. The flammable cloud is the volume of gas whose gas concentration is between the upper flammability limit (UFL) (15% vol) and the lower flammability concentration limit (LFL) (5% vol). The initial task was to evaluate the meteorological conditions, i.e., stability class and wind speed, that would yield the most conservative results, (i.e., the most severe damage to the ISFSI). Dispersion analyses were performed to estimate the flammable cloud volume for the following meteorological conditions:

- 2 mph, F stability
- 2.2 mph, F stability
- 4.5 mph, F stability
- 6.7 mph, F stability

Based on those analyses, the 6.7 mph and F stability resulted in the formation of the largest volume cloud, 3.3 million cubic feet, and this cloud was located closest to the ISFSI. In all cases, the wind was assumed to be blowing directly from the rupture to the IFSI.

The natural gas exits the ruptured pipe initially at sonic velocity. The upward momentum of the gas jet is included in the dispersion model. In addition, natural gas is buoyant (calculated molecular weight of 17.7 versus 28.8 for air) and this was also modeled. Near the rupture, the gas is at a concentration much higher than the UFL of natural gas and will not explode. As the vapor cloud is transported upward and downwind toward the ISFSI, the volume of the natural gas cloud in the flammable region (between 5% to 15%) increases due to atmospheric mixing. As the gas continues to be dispersed and diffused, the flammable volume decreases, finally reaching a point downwind where the volume drops below the LFL concentration and the gas cloud will not explode.

Figure 2 depicts the trajectory of the LFL and UFL clouds. The DEGADIS code does not provide instantaneous concentration profiles of the cloud. Rather it records the maximum concentration of the LFL cloud over time resulting in a tracing of the plume path. The DEGADIS code output can be used to determine the cross-sectional concentration profile at any height. Performing successive profiles (slices) permit development of the cloud (plume) profile. Additionally, the area of each slice can be determined and the total cloud volume estimated by multiplying the areas by the differential height. The net flammable volume was estimated by subtracting the volume above the UFL from the LFL volume. A review of these estimates indicated that the F stability and 6.7 mph conditions resulted in both the largest flammable cloud with the closest approach to the ISFSI. This volume was used for the detonation-damage estimation. The flammable cloud volume was determined to be 3.54E6 ft3. Assuming the flammable mixture to have an average concentration of 10% natural gas in air, this corresponds to 12,448 lbm of natural gas in a flammable concentration. Based on the dispersion analyses, the maximum downwind distance from the release point to a flammable concentration is 965 feet at an elevation of 358 feet. The center of the LFL cloud was estimated by determining the elevation in the cloud where one half of the cloud volume was above it, and one half was below. The area of the plume slice at this elevation was then estimated to determine the downwind distance where half the area was in front and half was behind the point. Based on this analysis, the center of the flammable cloud would be at 264 feet elevation and 361 feet downwind. At this point, the cloud center is 1,039 feet from the southside of the ISFSI. Under no conditions could a flammable gas concentration enter the ISFSI's natural cooling inlets from the 12 inch service line rupture (this is also the case for the other scenarios postulated in this study).

Fig. 2.

The VEXDAM Multi-Energy Model

VEXDAM has been designed to allow the rapid evaluation of damage experienced by each structure within a facility as a result of a primary vapor cloud explosion located on the surface, and any accompanying secondary vapor cloud explosions. Its primary application has been siting analysis of petrochemical storage and manufacturing facilities. The code can also be used to evaluate terrorism and sabotage threats to a facility.

The program has the capability to model an unlimited number of structures, and each with different dimensions and structural properties, including elevated structures by means of pressure-impulse diagrams. VEXDAM utilizes recently developed dimensionless curves of overpressure and pulse duration versus range, based on the Vandenberg multi-energy method (3), to predict overpressure and impulse at each structure location.

In VEXDAM, structure shielding, based on the three-doublet advanced shielding algorithm, is calculated, as well as damage levels. VEXDAM produces output in the form of damage tables, before-damage and after-damage displays, pressure, cumulative impulse and damage contour plots, and damage-versus-distance graphs.

Damage levels from vapor cloud explosions have traditionally been expressed in qualitative terms, with the adjective "severe", "moderate", and "light" most commonly used (4).

Severe damage corresponds to "degree of damage that precludes further use of the structure or object for its intended purpose without essentially complete reconstruction. For a structure or building, collapse is generally implied".

Moderate damage represents "degree of damage to principal members that precludes effective use of the structure or object for its intended purpose unless major repairs are made".

Light damage corresponds to "degree of damage to buildings resulting in broken

windows, slight damage to roofing and siding, blowing down of light interior partitions, and slight cracking of curtain walls in buildings. Minor repairs are sufficient to permit use of the structure for its intended purpose".
In order to produce a more quantitative description of damage, a numerical value has assigned to each such level. Based on past studies [5, 6] the following values have been used for VEXDAM:
TABLE
VEXDAM was developed for vapor cloud explosions located on the surface. The elevated vapor cloud explosion under consideration occurs at an altitude which falls in the Mach reflection region, for the horizontal ranges of interest. In such a region the incident peak overpressures can be approximately twice the values produced by a surface explosion. To take this Mach reflection region effect into account, the actual mass of the vapor cloud formed from the 12 inch service line rupture was multiplied by a factor of 3.9 to double the overpressure at the ISFSI for this scenario. This produced an adjusted mass of 48,485 lbm. The mass multiplier is unique for each of the elevated vapor cloud explosion scenarios modeled.
For purposes of the explosion analysis, the vapor cloud was modeled as a single sphere with its center coinciding with the center of the LFL region described earlier. The ambient temperature was assumed to be 70F and the ambient pressure 12.0 psi corresponding to an elevation of 5,500 ft above mean sea level. An explosive strength of 10, corresponding to a fully confined detonation, was assumed. The likelihood of such a detonation for an unconfined vapor cloud is extremely remote. By comparison, the most probable magnitude of the explosive strength of an unconfined vapor cloud would be two.
Based on available structural descriptions, the ISFSI facility was modeled as 22 individual components. This model was used for all cases in this study. Each component was described by its dimensions, location, orientation and structural resistance to blast.
VEXDAM Results for 12 inch Service Line
The output of the VEXDAM software consisted of a tabulation of damage to each structure of the ISFSI, as well as graphical displays and contour plots depicting the distribution of overpressure, impulse, and damage levels. Figure 3 shows the three dimensional distribution of peak overpressure from the 12-inch service line scenario. Figure 4 presents the three dimensional distribution of the 3.0 psi overpressure for this same scenario.
Fig. 3
Fig. 4.
The total composite damage level to the ISFSI facility is given as 13.8%, corresponding to light to moderate damage. The damage levels sustained by the reinforced concrete components were minimal, ranging from 0% to 0.9%. These percentages correspond to very light damage levels, which would not prevent such components from performing their structural, cooling and shielding functions. The corrugated metal components sustained higher levels of damage, ranging from 2.4% to 94%. Such percentages constitute light to severe damage levels. Because these metal components do not perform a safety function, such damage levels would not interfere with the performance of the ISFSI.
VEXDAM results indicate that the ISFSI would be exposed to a maximum incident overpressure of slightly greater than 3.3 psi for the 12-inch service line scenario. Also the ISFSI would be exposed to a cumulative impulse of 252 psi ms.
Based on the results produced by VEXDAM, damage levels to the reinforced concrete components of the ISFSI facility should be minimal. The structural, subcriticality control, cooling and shielding functions of such components would be unaffected. Essentially, no repairs would be necessary for the reinforced concrete.
The damage levels to the corrugated metal components would be significantly greater. Portions of these components would be destroyed with much of the remainder bent or buckled. The corrugated metal components do not perform any safety function, and thus, such damage would not in any way interfere with the mission of the ISFSI. The ISFSI could continue to function, although maintenance and security personnel might experience some inconvenience or discomfort due to exposure to external weather conditions, resulting from the absence of the corrugated metal. Considerable structural repairs to the corrugated metal would be necessary to repair or replace the damaged surfaces.
SUMMARY OF ALL SCENARIOS

The calculated overpressures and impulses are as follow:
TABLE
The results of these analyses show that a potential rupture of the 12 inch service line at Point T could create the highest overpressure at the ISFSI. The maximum overpressure was calculated to be 3.3 psi. A potential rupture of the 26 inch main supply line at Point C could create the highest impulse at the ISFSI. The maximum impulse was calculated to be 267 psi ms based on the pressure-impulse diagram analysis. The ISFSI can withstand these conditions.
The results of the VEXDAM damage assessment are as follow:
TABLE
The metal cladding of the ISFSI does not perform a nuclear safety function, but rather serves as a weather enclosure for workers' comfort. Damage to concrete components in all cases was light.
CONCLUSION
Based on the above values, there is no compromise in the mission of the ISFSI to provide containment boundaries for the irradiated fuel and its fission products, maintain the fuel in a subcritical array, provide for natural circulation cooling of the fuel storage containers and shield the public and ISFSI workers from the irradiated fuel. We conclude that even with the extremely conservative assumptions made, the ISFSI can safely withstand the impact of all natural gas scenarios postulated for the repowering of the Fort St. Vrain station.
REFERENCE
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DEWCHAP - SO YOU WANT TO KNOW HOW MUCH RADIOACTIVE WASTE YOU'LL HAVE AT THE END OF THE DAY!
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ABSTRACT
The DEcommissioning Waste CHAracterization Program (DEWCHAP) has been developed to aid nuclear plant owners in projecting the amount and classification of the radioactive waste that will be generated during the decommissioning of a facility. This innovative computer program is built around statistical simulation techniques used for critical areas of data. This data includes radioisotope concentrations, quantities, volumes, and development of a level of accuracy reflecting the level of detailed information existing at the time of compilation.
The program is designed to use Monte Carlo techniques and various selectable

distribution functions to calculate waste characteristics. Default contamination levels and concentrations of radioisotopes have been built-into the program and can be used when site specific data is not available
The program allows the user to performance of "what-if" waste characterization calculations based on system, type of component or location in the plant.
Program results can be presented in a variety of formats: i.e. spreadsheets, defined reports listing radioactive waste volumes, weights, and class, or the add-on ability to design your own report.
The Decommissioning Waste Characterization Program (DEWCHAP) has been developed to aid nuclear plant owners in projecting the amount and classification of the radioactive waste that will be generated during the decommissioning of a facility. This computer program is designed to incorporate statistical simulation techniques for critical areas of data. This data includes radioisotope concentrations, quantities and volumes, to be used to develop a level of accuracy that reflects the level of detailed information existing at the time of compilation.
The program is designed to use Monte Carlo techniques and various distribution functions to calculate waste characteristics. Default contamination levels and concentrations of radioisotopes have been built-into the program and can be used when site specific data is not available.
The predictive windows-based computer program was developed using the simplified development model shown in Fig. 1. It was recognized that the information necessary to determine the waste characteristics from the decommissioning operations is heavily dependent upon plant specific data. Therefore, the program must be capable of handling a large number of variables from the plant, and requires them to be able to be changed or modified. In addition, to cover the situations where a minimum amount of information is available, a default set of data was prepared for each of the databases to allow the user the opportunity to manipulate data from a zero ground state.
TABLE I
A default set of information has been included in the following main databases:
Radiological,
Decontamination,
Component, and
Systems.
RADIOLOGICAL DATABASE
The radiological database is a table of the isotopes. It contains the isotope name, atomic number, atomic weight, and half life. An example of the database information is shown on Table I.
This information is important for several reasons. First, the half-life is essential in determining the specific activity of an isotope at the time of decommissioning. Second, it is important when determining the waste classification for the component since the isotope is used per the instructions in 10 CFR 61.55 Table I and Table II.
DECONTAMINATION DATABASE
The decontamination of a system, component or area in the plant will provide for the reduction of the radio nuclide concentration on the affected surface. The reduction of the concentration varies depending on many different parameters. A table was prepared for each of the different types of decontamination methods that exist in the industry today. Some examples of these decontamination methods are presented on Table II.
TABLE II
Each method has been given a range of decontamination effectiveness or a decontamination factor (DF). The DF range is based on published literature covering the subject and experiential data. In addition, it should be noted that the decontamination of a component or system will not reduce the total radio nuclide concentration in the plant. It will only move the contamination from one point to another. Therefore this program has been developed to keep track of the system wide decontamination method employed and to transfer the radio nuclide concentration to a designated waste volume for the particular method. An excerpt from this database is presented in Table III.
TABLE III
COMPONENT DATABASE
The components to be evaluated are imported into the program from various databases. The structure of the database is developed to optimize the information that exists

in a plant's master equipment list. An example of the structure of one of these databases is shown as Table IV.
For purposes of waste characterization, the important parts of master equipment list are the data that are used to determine weight, volume, system and location. The other information will be used in future programming to support the development of the decommissioning plan and component removal work packages.
TABLE IV
The waste characterization is developed at a component level. The process theory is based on several simplifying assumptions. First, the only radio nuclides that exist are those that are present in the reactor coolant system. This is valid for most non-transuranic elements that would be found in the plant. In addition, it assumes that any fuel failure has released a minimum amount of fuel into the system.
Second, the concentrations or specific activity of radio nuclides will vary depending on the interface between the system being evaluated and the reactor coolant system (RCS). The transfer of radio nuclides between systems can be measured in the actual plant situation. However, this information is transient and not normally available. Therefore, a System Activity Transfer Factor (SATF) has been developed for each system. This factor is used to develop the radio nuclide concentration in the component as a function the system's relationship to the RCS. The program allows for this factor to be developed using simulation techniques, thus providing upper and lower levels of confidence about the factor.
Third, radio nuclides can concentrate differently in different components. This effect is handled through the Component Activity Transfer Factor (CATF). Like the SATF, this factor can be determined through simulation techniques to improve the confidence level in the value and the calculation results.
Default values for these factors have been placed in the component and system databases.
The program proceeds to take the component, apply the SATF and CATF to determine the radio nuclide activity at the time of removal. In addition, if some method of decontamination is employed, the data is modified in accordance with the high and low decontamination factors to determine the final activity level of the component. The activity along with the component weight and displaced volume is provided in the output listing.
SYSTEM DATABASE
The system database contains the information about the radio nuclides and their concentrations as they exist in each system. The concentration can be developed by using simulation techniques, like those found in @RISK. This information is most important in developing the waste characterization and has been provided with the maximum ability for modification and change as the data associated with the end of plant life becomes available.
WASTE CLASSIFICATION
The information developed in the calculations is used to compare the activity levels and the radio nuclides present in the waste with the 10 CFR 61 requirements. The two tables used by the CFR are shown in Table V and Table VI.
The results of the evaluation are place in the output in a format determined by the user.
TABLE V
The process used in the determination is presented in Fig. 2.
TABLE VI
CONCLUSION
With the continuing uncertainties associated with the ability to dispose of radioactive waste and with the continued rise in the cost for disposal when available, the ability to accurately predict the amount of waste arising from decommissioning becomes increasingly important. The use of a program like DEWCHAP, that uses probabilistic techniques to predicting the amount of radioactive waste will aid the owner in determining the cost for its disposal. The program was designed to provide the information necessary to support the decommissioning planning of a nuclear plant. The preliminary results of the analyses has indicated that the accuracy of the information is greatly enhanced when a complete and detailed master equipment list exists. Even with this detailed list the probabilistic techniques are important in increasing the level of confidence in the accuracy of the results.
Fig. 2.

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CADMIUM CONTROL/SAFETY ROD DISPOSAL AT THE SAVANNAH RIVER SITE

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ABSTRACT
A total of 862 activated cadmium control/safety rods will be removed from four heavy water moderated reactors at the Savannah River Site. The cadmium rods are mixed waste and must be disposed of in accordance with the Resource Conservation and Recovery Act (RCRA). Determining a viable design plan for a 40-year-old facility while meeting today's standards, at a moderate cost, proved to be quite a challenge. First, a method to remove, handle and load these long, small-diameter cadmium rods from the reactor tank had to be determined. Second, a shielded cask which could fit in the small transfer pools and which could hold the control rods, had to be located or designed. Third, the waste acceptance requirements, the RCRA requirements and the logistical difficulties in meeting these requirements had to be addressed. These difficulties were resolved through close interpretation of the RCRA laws, involved management, strong communication, and detailed engineering.

INTRODUCTION

Four heavy water reactors will undergo activated cadmium control/safety rod removal. Du Pont designed and built these reactors in the early to mid 1950s. Although these Savannah River Site (SRS) reactors are forty years old, they offer several basic advantages for decommissioning. First, the equipment is still in some sort of operable state. Second, the reactor is below the floor in a large process room, which will allow access. Third, control/safety rods can be handled by existing components (fuel) handling equipment, which is a remote control robotic system. Finally, a radiologically shielded removal path exists for these cadmium control/safety rods. With these positive aspects, preliminary decommissioning steps (transitional steps) should be much easier.

The Savannah River reactors still had numerous constraints. First, the reactors are forty years old. Equipment operability and spare parts are a potential problem. Second, improvements in technology caused incompatibility problems. With improvements to technology came more stringent modern standards. Applying these modern standards to this 40-year-old facility took some detailed engineering and environmental guidance. To compound the plant's problems, the project had to comply with environmental regulations, such as RCRA.

PROCESS SELECTION

The progress of the project involved radical changes during the first couple of months. The process, the cask, and the RCRA interpretations needed to be clarified. The solution to any one item would impact the others. After initial investigation of the total project, it was decided that the rod removal process was the first problem to resolve.

The first approach was to bring a cask into the process room (or the reactor room) for loading. Problems quickly developed. Initial forecast of the cadmium rod activity indicated that a 35 ton (32,000 kg) cask would be the minimum weight for a shielded container for one reactor's compliment of rods. (Fabrication and manufacture investigations showed smaller casks that hold half or a quarter of the reactor's complement of rods was not cost effective.) Per discussions with structural engineers, the floor in the process room could not support this type of concentrated loading. In addition, the rods are lifted vertically by the components handling equipment. However, the ceiling was not high enough to vertically load a

vertical oriented cask. The cadmium rods were long and thin (approximately the diameter of a quarter) (2). The cadmium rods did not have enough structural rigidity for a single point of contact to rotate the rods from the vertical position to the horizontal orientation. Another group onsite had two very capable titanium robotic arms that could be used. However, investigations showed the arms could not be used in unison to rotate the cadmium rods from vertical to horizontal. As the more practical answers faltered, the proposed approaches became more radical. One solution suggested removing one leg of the reactor coolant loop, boring out its hole in the concrete floor, inserting a specially designed cask in the hole and loading the cask with the titanium arm attached to the components handling equipment.
After investigation and research, it was determined that the reactors used a different type of control/safety rods at one time (1). These rods were removed by the components handling equipment, were run through a canal to the disassembly basin, and then were processed through the disassembly basin, broken up, and removed from the basin via small casks. This was a possible process solution. However, cask selection became critical to the success of this proposed approach.

CASK SELECTION

The next critical link in the project chain was the cask. The cask had to be submersed in the disassembly basin pool to allow shielded loading. The cadmium rods were activated and had estimated fields up to 25 rad/rod (7). As the rods accumulated in the cask, the radiation field would become tremendous. Ideas were proposed such as cutting a hole in the disassembly basin roof and dropping a cask in the pool with a crane. Investigation of the area showed that the best approach was to use existing equipment and the existing transfer bay to remove the cask from the pool.

However, the transfer bay had several problems. The canal from the disassembly basin to the transfer bay was narrow, 1'-2' (30 cm-60 cm) wide. Three of the four reactors had a curved canal, which presented an obstacle with DOT-approved casks. (Most DOT-approved casks are loaded from the end, like a gun barrel. However, as discussed later, most SRS casks are loaded from the top, like a shoe box.) Second, the pool in the transfer bay was small. In fact, the largest pool dimension was one foot (30 cm) shorter than the length of the cadmium rods. The overhead crane was a dual point lift and was limited to 85 tons (77,300 kg). An approach for tilting the cask in the transfer bay pool was discussed. A tilted cask approach was unacceptable for two reasons. First, there were structural concerns about the floor and wall loading. Second, the cask would need to be handled in two orientations: horizontal (on the transportation vehicle) and tilted (in the transfer bay pool). Handling in two orientations presented some engineering problems. The transfer bay building was too small to allow use of a second crane of sufficient capacity. Loading the cask above water meant designing all new rod handling and cask loading equipment. This would be expensive and would impact the schedule. Plus, if the equipment broke with a rod unshielded, a repair person would receive significant dose. The best solution was to use the existing equipment in an as-normal-as-possible manner.

In parallel with the cask handling investigation and with the process selection investigation, a usable cask search was initiated. The project manager recommended a search of all the available casks in the US. A site group kept a database of the DOT-approved casks for this country. The initial search was to find a cask long enough to hold a full length rod. None was found. An alternate investigation of the characteristics of the cadmium rods (inventory count, length, diameter, make-up, etc.) had revealed the cadmium portion made up only the lower two thirds of the length of the cadmium rods (2). One offsite, DOT-approved cask was found to be long enough to contain the cadmium portion (the bottom 2/3s) of the rods. However, several obstacles still remained with the DOT-approved cask. Although this cask was large enough to contain 862 rods, it was very heavy and exceeded the transfer bay crane's capacity. This would require unshielded loading. Second, use of this cask would require cutting the cadmium rods, which violated the RCRA requirement of "no treatment". (RCRA defines treatment as cutting, bending, etc.)

A decision was made to look more closely at the Savannah River Site's (SRS) casks. The first contacts were the site transportation committee and the cask custodial group, and they assisted in locating two casks that could be used. The first cask was long enough to receive the full-length cadmium rod, so it required tilting in the transfer bay pool. Also, it was not DOT-approved and it required modification for dewatering, and it could hold only half of the total rods. Since detailed

drawings of this cask were not in the site's document control system, and the cask fabricator had not maintained drawings files over the last forty years, this cask would still require a detailed design. This meant an expensive and time consuming procurement cycle (as long as three years). This was not acceptable because the components handling equipment would further degrade and the technician staffing could not be held for a three year wait.

A search of old operating documents was initiated, and a second cask, a spent fuel cask, was located. It was compatible with the existing cask handling equipment. It could easily fit in the transfer bay pool, and it was large enough to contain all 862 rods. Also, it could be modified to meet the project needs. An inspection with the construction technicians of the cask exterior confirmed the ability to modify the cask. However, a detailed search of the document control archives failed to find any purchase order or purchase specification, so it was uncertain what the cask was capable of withstanding (factors of safety, heat transfer ability, crash strength, etc.).

The design requirements were important since the Department of Energy required a DOT-approved cask (or the equivalent). Old documents were searched again, one of which was a 1959 engineering description of the spent fuel cask (6). In this document, the two Du Pont engineers detailed their heat transfer calculations and performance testing. They discussed fabrication difficulties and structural capacity and had designed the cask to withstand a 4 g stop/crash. Also, in the event of a rolling accident, the cask was designed to withstand a complete roll of the cask attached to its railroad flat car. This was the engineering data needed to justify the safe use of this cask without having to pay for expensive engineering evaluations. In preliminary discussions, the site transportation committee felt that this cask, with this documentation, could be accepted as DOT equivalent for onsite use under controlled conditions. The spent fuel cask was a viable approach except that use of this cask would require the cadmium rods to be cut. This would again be "treatment" by the RCRA guidelines.

RCRA INTERPRETATIONS

If the SRS spent fuel cask was to be used, it must comply with environmental regulations. Since the Savannah River Site previously removed, cut, and transported its control rods (before the use of cadmium control rods), and since the site still performs similar actions with other materials, the state authorities accepted cutting the rods as normal operation. Since the site performs similar actions with existing functional equipment, no additional training was required, and no new procedures had to be writtenthe task required only minor changes to existing procedures. The rods could be cut above the cadmium portion (the bottom two-thirds) without disturbing the cadmium, and then they would be short enough to be placed in the spent fuel cask.

Now the project plan could be determined. It was presented to the site authorities and tentative approval was received. Then the plan to cut the rods above the cadmium portion was approved by the state, although with some constraints. The 90-day RCRA clock was implemented in a more conservative manner than expected. (RCRA allows the waste owner 90 days from the declaration the material is waste, until the waste is properly encapsulated and properly disposed of.)

The cadmium rods had a stainless steel outer covering swaged over the an aluminum core cladded with cadmium (2). After the top non-cadmium section of the rods was trimmed off, one could possibly argue that capillary action would draw the water between the swaged metals, ultimately reaching the cadmium. Now, the proposed spent fuel cask was something of a fortress. It had an outer wall of 0.5" (1.27 cm) thick stainless steel, 8.5" (21.5 cm) of lead for shielding, and a 0.25" (0.63 cm) stainless steel inner wall (2). Possible leak paths for this cask were the lid, four penetrations and five thermowells (2). (Since the thermowells penetrated the cask interior, and since the cadmium rods may shift during transport, welding the thermowells would provide a secondary barrier if a thermowell was damaged or broken.) If all penetrations and the lid were welded closed at the completion of the project, this would serve as "macro-encapsulation". The project team voiced concern about the difference between what is sealed, and what is contained. It was acknowledged that everything leaks. Leakage is a function of pressure and size of the molecule. Code of Federal Regulations 10 CFR 71.71 (DOT Transportation Requirements) (3) state that the cask must not leak during two inches of rain in one hour's time. ANSI 14.5 (Radioactive Leak Rates for Transportation) (4) stated

acceptable leak rates for radioactivity. The project team felt that "no leaks" would be the only acceptable leak rate.
The characteristics and details of the waste were looked at thoroughly. The radioactivity in the cadmium rods was activated cobalt. Cobalt was a trace element in the stainless steel cladding. Therefore, the radioactive cobalt would not "leak" out of the container, and ANSI 14.5 was not applicable (4). Since radioactive leakage was not a concern, and since RCRA does not provide clear guidance, cadmium (a toxic element) was studied. Cadmium leaches into water, which makes it similar to lead. If water can be kept from entering and leaving the cask, the cadmium rods will be "macro-encapsulated". To determine the pressure to seal against, calculations were performed. These calculations included the effects of ideal gas expansion of air (seasonal temperature changes), hydrolysis of water (gamma radiation causes a hydrogen atom to be driven off from the water molecule) and pressurization effects from evaporation of residual water (partial pressure changes). After discussions with the site environmental and non-destructive exam (NDE) groups, it was decided to use a conventional piping hydro test at 150% of the maximum expected pressure. Visual observation of the exterior, with no leaks, was the acceptance criteria. The penetrations used to perform the leak test will be welded and dye-penetration tested. The proposed test criteria was written up with extensive justification and was signed by engineering, environmental, and NDE authorities.
The final criteria to satisfy was "no free liquids". The site equipment engineering group was given the task of injecting 100 pounds (45 kg) of Sorbond UG (a clay based desiccant) to absorb any free liquids. The clay based absorbent was not as aggressive as petroleum based products; however, it did not degrade under heavy gamma radiation from the cobalt 60. The petroleum based products would break down and emit hydrogen gas, which would pressurize the cask. First, an absorbent was inserted in the low point drains to remove any liquid in the drain line. Second, using compressed air with an eductor and a low pressure relief standpipe, the desiccant was injected through the low point drain. Two complications encountered with the low point drain were that two long radius elbows eliminated the use of any screw type delivery, and the desiccant pile had built up on top of the low point drain entry point. To verify that these complications were overcome, the equipment engineering group made a full scale mockup to qualify its performance.
CURRENT PROJECT STATUS
The modifications to the spent fuel cask have been made. However, the welding of the lid and the final penetrations will not be complete until the cadmium rods are loaded in the cask. Before the cadmium rods were removed from the four reactors, the project team obtained verbal acceptance of the project plan from the waste acceptance custodians (8). Then, the cadmium rods were removed from the reactor tanks at all four reactors, and they are now in the disassembly basins awaiting cutting and loading into the cask. The project has received written approval from the waste acceptance custodians for the spent fuel casks as waste containers. Very soon, one spent fuel cask will proceed through all four reactors' transfer basins and collect all 862 cadmium rods. The projected excess cross-sectional area in the spent fuel cask is 27% (5). However, it is anticipated that the majority of the excess cross-sectional area will be taken by bent cadmium rods. Once the cask is loaded, welded, leak tested and desiccant injected, it will be transported to Waste Management via a railroad car. The total trip will be 15 miles (24 km) in length. The cask will be lifted off of the railroad car and placed on a transuranic waste pad (5). The cask will be disposed of when the mixed waste vaults are built.
LESSONS LEARNED
Research into previous documentation and human experience proved to be valuable in solving several key problems with the project. Also, good communication was essential. Many groups and individuals were involved with this project, and asking the right questions, and sometimes asking detailed questions, prevented derailment of the project. For future projects, verify information by checking drawings, by asking others, or by sight inspection. Many reviews should be expected because of the project's significance, because of the vast amount of unknowns, and because of the 90-day RCRA regulatory clock. Project management should work to minimize risk, cost, time, and therefore embarrassment.
Fig. 1.
Fig. 2.
Fig. 3.

Fig. 4.

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PHASE II MICROWAVE CONCRETE DECONTAMINATION RESULTS*

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ABSTRACT

We report on the results of the second phase of a four-phase program at Oak Ridge National Laboratory (ORNL) to develop a system to decontaminate concrete using microwave energy. The microwave energy is directed at the concrete surface through the use of an optimized wave guide antenna, or applicator, and this energy rapidly heats the free water present in the interstitial spaces of the concrete matrix. The resulting steam pressure causes the surface to burst in much the same way popcorn pops in a home microwave oven. Each steam explosion removes several square centimeters of concrete surface that are collected by a highly integrated wave guide and vacuum system. We call this process the microwave concrete decontamination, or MCD, process. The MCD process is fast, generates little dust, and avoids mechanical impacts. The concrete particles from this steam explosion are small enough to be removed by a vacuum system, yet less than 1% of the debris is small enough to pose an airborne contamination hazard. In the first phase of the program the principle of microwaves concrete removal concrete surfaces was demonstrated (1). In these experiments, concrete slabs were placed on a translator and moved beneath a stationary microwave system. The second phase demonstrated the ability to mobilize the technology to remove the surfaces from concrete floors. Area and volume concrete removal rates of 10.4 cm2/s and 4.9 cm3/s, respectively, at 18 GHz were demonstrated. These rates are more than double those obtained in Phase I of the program. Deeper contamination can be removed by using a longer residence time under the applicator to create multiple explosions in the same area or by taking multiple passes over previously removed areas. Both techniques have been successfully demonstrated. Small test sections of painted and oil-soaked concrete have also been removed in a single pass. Concrete with embedded metal anchors on the surface has also been removed, although with some increased variability of removal depth. Microwave leakage should not pose any operational hazard to personnel, since the observed leakage was much less than the regulatory standard.

SCOPE AND PURPOSE

The purpose of Phase II of the microwave decontamination system development was to demonstrate the ability to mobilize the equipment necessary to remove concrete surfaces from floors. This required the integration of all necessary subsystems on a mobile cart. The cart was designed to move in a straight line with limited manual steering capability. Tests were carried out on noncontaminated surfaces, simulating various conditions expected to be encountered in actual decontamination scenarios. These included painted concrete surfaces, oil soaked concrete surfaces, and concrete

surfaces with imbedded metal anchors. Different operating parameters were studied to provide a basis for the design of a Phase III prototype. The Phase III prototype will be fully robotic for deployment in a real-world decontamination demonstration. Estimated capital and operating costs for a final machine were generated from the fabrication of the prototype system and the concrete removal rate obtained with the unit.

SYSTEM DESCRIPTION

Figure 1 shows the overall MCD system used in Phase II testing. This system consists of a microwave high-voltage power supply, instrumentation, and controls.

Fig. 1.

Electrical power (480 V 3-phase, 120 A) and plant cooling water (15 gal/min) are supplied to these systems. The smaller, fully mobile cart contains the microwave tube and wave guide system along with a vacuum system and a 55-gal drum for collecting the concrete debris. The mobile cart is tethered to the main power supply that supplies the electrical power and cooling to the cart. A PC-based data acquisition system and rack-mounted controls for the cart speed and microwave power are located near the power supply.

Microwave Frequency

The frequency of the microwave generator chosen for this series of tests was 18 GHz. This is a much higher frequency than the 2.45 and .896 GHz used in previously published work (2,3) and higher than the 2.45 and 10.6 GHz frequencies used in the Phase I experiments. The higher 18-GHz frequency preferentially heats near the concrete surface compared with the wave amplitudes of all lower frequencies as shown in Fig. 2.

Fig. 2.

This causes the uppermost 5 mm of concrete, where most of the surface contamination resides, to be removed very efficiently. The 18 GHz generator was available for loan from the U. S. Department Of Energy (DOE) Fusion Program at no cost to the DOE Waste Research and Development Program. In addition, the 18-GHz, 15-kW generator was considerably more powerful than the 10.6-GHz, 10-kW generator or the 2.45-GHz, 6-kW generator used in Phase I. Therefore, higher removal rates and speeds were predicted. The rectangular wave guide size for transmitting microwave energy from the 18-GHz generator (1.78 by 1 cm) is considerably smaller than the wave guide used for 2.45-GHz generator (7.62 by 3.81 cm) and is therefore more compact and lightweight for integrating into a robot arm for positioning the applicator on floors and walls.

Microwave Applicator

The applicator used in Phase II represents a significant improvement over Phase I. Besides spreading the microwave energy over a larger area, 10 by 14 cm compared to the 3-cm diameter spot in Phase I, the applicator minimized any reflected microwave energy and eliminated the need for a bulky, expensive microwave circulator (protection device). Scattered microwave energy off the concrete floor that could constitute a safety hazard was also minimized due to the efficient design of the applicator. The microwave applicator was highly integrated with a 350 ft3/min vacuum collection system for concrete debris. In addition, the applicator design had unrestricted real-time video monitoring of the actual removal process, which allowed for greater operator control and improved understanding of the dynamics of the removal process.

EXPERIMENTS

In the Phase II dynamic tests, a mobile cart was used to remove concrete surfaces. Initial tests were carried out in the high bay area of Building 9201-2 at the Oak Ridge Y-12 Plant. An aluminum stage 5.48 m long by 78.7 cm wide by 26.7 cm high was built to support the weight of the cart (680 kg) and to elevate the cart so that cut slabs of concrete could be placed along the stage flush with the top of the stage, as shown in Fig. 1. The slabs were supported off the floor by a system of aluminum braces that allowed the underside of the slabs to be accessed. A microwave transmission diagnostic was placed in this region to measure the amount of microwave energy that might penetrate through the slabs. The slabs were 1.2 to 3.6 m long and 35 to 61 cm wide. The top surfaces of the slabs had a typical sidewalk finish, and the sides were smooth as a result of the diamond saw cutting required for slab removal. The bottoms of the slabs were rough and irregular due to the uneven gravel beds employed during the fabrication of the slabs. Because of this, the slab thickness varied from 12 to 18 cm. The age of the concrete slabs could not be

precisely determined, but they were estimated to be approximately 20 to 30 years old. The shorter length concrete slabs came from old sidewalks at ORNL, and the longer length slabs came from concrete parking pads at the Oak Ridge Y-12 Plant. The slabs did not contain any steel reinforcing rods. However, the Phase I results showed that the presence of steel reinforcement had a minor effect on the microwave removal of concrete at 2.45 GHz. This is because the steel mesh and rods lie normally 5 cm or more beneath the surface of the concrete and the microwave fields have decayed to low levels at this depth, as shown in Fig. 2. This rationale is even more true for the 18-GHz frequency. Final field testing was conducted in Building 3034 at ORNL. The MCD system was used to remove selected concrete surfaces from the building floor. The building drawings indicate that the concrete floor was poured around 1960.

RESULTS

Forward and reflected microwave power was measured to determine the net power incident on the concrete surface. An array of three detectors located around the applicator measured the scattered power leaking from around the applicator. A microwave detector on the 55-gal vacuum collection drum measured the amount of microwave power present in the drum. Power transmitted through the slab was also measured. Cart position and speed were controlled and documented for each test. A sensor monitored the vacuum in the applicator (4~6 in. of water) during operation. Also, a thermocouple measured the applicator temperature. A color video camera and pickup microphone recorded the concrete removal process during the experimental runs. All data was recorded on a 486 PC running Labview 2.5 for Windows. Data were displayed and processed with Mathcad 4.0 for Windows. Figure 3 shows a typical steam explosion.

Fig. 3.

Concrete Removal

To estimate the volume of removed concrete debris the clean trenches in the concrete were filled with fine sea sand. The sand was leveled with the surrounding slab surface so that all cavities in the slab were filled. Excess sand was carefully scraped and brushed away from the filled cavities. The sand was vacuumed up and the weight gain of the filter bag was measured. The weight of the sand was divided by its measured density to calculate the approximate volume of removed concrete. These measurements were conducted for the first few long runs. To obtain the volume removal rate the measured volume was divided by the duration of the run. To obtain the average depth of removal the volume was divided by the average width and length of the run. To obtain the area removal rate the speed was multiplied by the average width of the trench. The new Phase II applicator produced trenches 9~10 cm wide by 4~5 mm deep at speeds as high as 1.1 cm/s. This gave an area removal rate of 10.4 cm2/s and a volume removal rate of 4.5~4.9 cm3/s. The area removal rate was about 250% higher than that obtained in Phase I, and the volume removal rate was about 210% higher than it was in Phase I. The cross-sectional shape of the trenches in Phase II is flatter than that of the trenches cut in Phase I because the heated concrete surface has more uniform illumination. Other factors affecting the removal efficiency include the translation speed and the way in which the microwave power is distributed over the area to be heated.

Microwave Leakage

The microwave leakage at 18 GHz from the applicator was well below the American National Standards Institute (ANSI) standard (4) of 5 mW/cm2. This was true even if 5-mm trenches were cut between the applicator and the surface of the slab for the microwave energy to propagate. The low leakage may be explained by the high losses in concrete at 18 GHz and the fact that almost all the energy is absorbed in the first pass due to the efficient applicator design. After several dozen runs the drum detector and two of the scattered microwave detectors were removed because they did not reveal any hazardous leakage levels and their removal simplified the cart instrumentation and data acquisition. The remaining scattered microwave detector was retained as a personnel safety interlock but was seldom tripped in any tests.

Painted Surfaces

Historically, some rad-contaminated concrete surfaces have been painted to fix the contamination present at the surface. To investigate the effect of paint on the microwave removal process, a 20-cm-long section of concrete was painted with a two-part epoxy paint sealer commonly used to fix contamination to concrete surfaces. The paint was allowed to cure in order for the paint to reach its full bond

strength. The cart was started at 0.9 cm/s on bare concrete and driven over the painted section. Upon heating, the paint could be observed to soften, and then the paint and concrete were explosively removed in much the same fashion as unpainted (plain) concrete. The speed of the applicator was lowered from 0.5 to 0.7 cm/s to achieve removal depths similar to those of plain concrete. Some microwave energy was absorbed in the paint. However, the amount of energy deposited in the paint was insufficient to cause any fire or smoke from the heated paint. The concrete debris appeared to have a larger particle size as a result of the paint's binding some smaller pieces together.

Oil-Soaked Surfaces

Concrete contaminated with Polychlorinated Biphenyl's (PCBs) can be found at many DOE sites. To investigate the effect of oil on the microwave removal process, a 40-cm-long section of concrete was soaked with standard hydrocarbon pump oil and allowed to stand for 24 h. The cart was started at 0.9 cm/s on dry concrete and driven over the oil-soaked section. Upon heating, the surface oil bubbled and then the oil and concrete were explosively removed in much the same fashion as plain concrete is removed. No significant reduction in speed was required, indicating that the oil did not directly absorb significant amounts of the microwave energy. After the trench was cut, the oil was observed to have penetrated to a depth of 2~3 mm. A single pass removal of 5 mm was observed. Again, no smoke or fire was generated by microwave-heated oil.

Metal Anchors

Surface metal structures in the concrete may affect the reliability of the microwave removal process as a result of metal objects changing the applicator electric fields. To investigate this effect four 0.5-in.-diam holes were drilled into the concrete slab, and a steel bolt anchor was placed in each hole, flush with the surface, to simulate a typical floor or wall attachment to the concrete. Concrete was removed to 5 mm around two holes and partially removed around the remaining two holes. No arcing was observed with any of the holes, but water boiled from the holes during microwave heating (no water was added during the drilling). This caused a reduction in the steam generation because the hole provided an efficient pathway for the water to escape. The reduction in steam pressure reduced the reliability of the microwave removal process.

ECONOMIC ANALYSIS

The cost analyses in Tables I, II, and III give a rough estimate of the capital and operating costs associated with a potential Phase III MCD system.

TABLE I

TABLE II

TABLE III

The configurations vary with respect to the number of microwave tubes on a single mobile system and the frequency and power of each tube. These costs are relative and will be revised as the program progresses to later phases.

Capital Cost Basis

The estimated capital costs consist of two categories of costs. One category of costs includes those for subsystems obtained from various vendors off the shelf. The estimated costs for these subsystems are based on quotes from the manufacturers and include the high-voltage power supply, and high-power microwave tube(s); the vacuum system; and the robotic vehicle. These costs do not include discounts for multiple purchases. A second category of capital costs involves system integration costs. Integration systems include the wave guide transmission system and the instrumentation and control system. Costs for these subsystems are based on costs associated with the Phase II mobile system scaled as appropriate to fit the other potential configurations.

Operating Cost Basis

The estimated operating costs are based solely on operating the equipment. The costs are expressed as cost per area removed and cost per volume removed. Costs do not include maintenance costs or waste disposal costs. Phase II equipment has operated thus far without an equipment failure, so maintenance costs have not been determined. These operating costs are based on the concrete removal rates achieved by the various equipment configurations and the assumption that one person would operate the equipment at a cost of $95.73/hr. Concrete removal rates for all of the single microwave tube configurations, except the 15-kW 2.45-GHz unit, have been determined experimentally as part of this program. The removal rates of multiple

microwave MCD systems are projected rates, some of which will be verified in Phase III of the program. Volume removal rates are based on experimental results obtained by previous published work (2). This study determined that two microwave systems in parallel had a volumetric removal rate 30% greater than that removed by two single systems. The same study showed that by using three microwave systems in parallel the volumetric removal rate was 14 times the removal rate of a single system. The previous study did not show the effect of multiple microwave systems on area removal rates. Therefore, the area removal rates for multiple microwave systems are assumed to be as much as 30% greater for two in parallel versus two operating independently and as much as 60% greater for three in parallel versus three operating independently. The effect of multiple magnetrons on area removal rates will be determined in Phase III. To estimate operating costs the hourly labor rate is divided by the area removal rate. Electrical costs are negligible compared with labor costs.

Other Cost Considerations

The total cost of decontaminating concrete structures must consider other factors besides the cost of concrete removal. Currently, disposal of low-level waste at ORNL is $85/ft3. If transuranic-contaminated low-level waste disposal costs were the same, the microwave decontamination system would offer a potential savings of $43.5/ft3 in disposal cost over other technologies that produce concrete debris (powders) that do not meet the Waste Isolation Pilot Plant Waste Acceptance Criteria. Those wastes in powder form that do not meet these criteria must be immobilized (i.e., grouted), which could result in a 50% volume increase.

COST RESULTS

Based on the work completed to date, this preliminary cost analysis shows that a MCD system can easily be developed to remove 3/16 in. of concrete at a cost of $2/ft2, when company personnel are used. Vendor quotations of $2.50/ft2 for 1/16 in. concrete floor removal using conventional scabbling are typical. Phase II resulted in a greater than 200% increase in concrete removal rates over Phase I results. As the final configuration and operating parameters of the Phase III MCD system are defined, further increases in concrete removal rates are expected, and reduced operating costs are anticipated. The MCD operating cost is also well below vendor quotations of $50~$70/ft2 for removal of 1/4 in. from concrete hot cell walls.

CONCLUSIONS

ORNL has completed Phase II testing of a mobile MCD system to remove concrete surfaces. Continuous-area and volume concrete removal rates of 10.4 cm2/s and 4.9 cm3/s, respectively, at 18 GHz were demonstrated. These rates are more than double those obtained in Phase I. The mobile prototype produces high removal rates, generates little dust, avoids mechanical surface impacts, and generates no liquid waste. The concrete particles generated by the MCD process are small enough to be removed by a vacuum system, yet less than 1% of the debris is small enough to pose an airborne contamination hazard. Deeper contamination can be removed by using a longer residence time under the applicator to create multiple explosions in the same area or by taking multiple passes over previously removed areas. Both techniques have been successfully demonstrated. Small test sections of painted and oil-soaked concrete have also been removed in a single pass. Concrete with embedded metal anchors on the surface have also been removed, although with some increased variability of removal depth. Microwave leakage should not pose any operational hazard to personnel because the observed leakage was much less than the 5-mW/cm2 ANSI standard (4), and the Phase III MCD system will be operated remotely. A preliminary operating cost analysis shows that the MCD system can easily be developed to remove 3/16 in. of concrete at a cost of $2/ft2.

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46-21

CHEMICAL DECONTAMINATION: THE ROLES OF PROCESS CONTROL AND PLANNING IN WASTE MINIMIZATION

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ABSTRACT

Chemical decontamination has been used successfully by ALARON Corporation in the decontamination of metals for free-release. Initially, however, secondary waste volumes associated with these applications ranged from expected minimum volumes to significantly higher waste volumes. Operational records from the first 15 metric tons of process material revealed that waste variances resulted from only six types of errors -- some relating to specific operational difficulties while others were due to shortcomings in the initial planning stages of these projects. Errors were analyzed for root cause(s) and for effect upon both processing effectiveness and waste volumes generated. The secondary waste volumes resulting from specific errors were compared to waste volumes produced during ideal process operation, thereby quantifying the effect of each error on overall secondary waste generation. The information obtained was then used to implement effective measures for controlling and minimizing secondary waste volumes produced during chemical decontamination operations.

INTRODUCTION

Over the last several years, ALARON Corporation has used its patented fluoroboric acid (HBF4) process for the decontamination of radioactively contaminated metals to achieve free-release. The specific characteristics and effectiveness of this process have been presented previously (1,2) and, therefore, are not provided in this paper. In general, however, fluoroboric acid provides excellent corrosion rates for almost all metals and, more importantly, can be regenerated and reused, thereby minimizing secondary wastes from the chemical decontamination process. Despite these advantages, early full-scale applications of this process produced varying degrees of success. The initial problems and errors resulted in relatively wide fluctuations in secondary waste volumes produced during various chemical decontamination campaigns.

In order to understand these fluctuations and the underlying reasons for higher-than-expected waste volumes produced at that time, operational data from more than 40 of ALARON's earliest chemical decontamination campaigns, involving approximately 15 metric tons of metal, were analyzed. The metal included stainless steels, carbon steels and aluminum contaminated with mixed fission and activation products. The primary objectives of this evaluation were to:

- identify major problems and errors resulting in abnormally high secondary waste volumes
- identify the origin of each error and/or problem
- provide some quantitative measure of the relative impact of each error and/or problem on final waste volumes.

Understanding the nature and origin of these errors, as well as their respective impact on secondary waste generation, was an important first step in recommending and implementing procedural changes consistent with effective secondary waste minimization practices.

INCOMING MATERIAL

Contaminated metals are generally evaluated for chemical decontamination processing at ALARON's facility based on specific surface characteristics which reflect the overall availability of the contaminated surface area to the chemical decontamination medium and to survey instrumentation. As a general rule-of-thumb, materials exhibiting no more than 5 - 10% "permanently inaccessible" surface area are considered "good" candidates for chemical decontamination. Surfaces may be "permanently inaccessible" due to surface geometry as well as due to the presence of paints and other coatings. Although many metal surfaces are, as received, inaccessible to survey instruments, these surfaces are readily exposed or rendered "accessible" by simple segmentation techniques. Therefore, "permanently inaccessible" refers only to those surfaces that cannot be made accessible with a reasonable amount of effort and within a reasonable amount of time.

In addition to surface accessibility, contaminated metals are also screened for

relatively uncommon surface characteristics resulting, usually, from harsh environmental conditions over an extremely long period of time (e.g., 20 years). Strong oxidizing environments, thermal cycling, and other stress-inducing phenomena can result in surface and oxide chemistries which, unlike most contaminated metal surfaces, are strongly resistant to chemical decontamination. Stress cracking, chrome-rich oxide layers and other atypical conditions produce metal surfaces that are not "amenable" to acid etching -- due to some unique resistance to the acid process or, more typically, to the relatively large volumes of waste generated during the decontamination of these surfaces. Although ALARON routinely screens for these surface phenomena, historical information accompanying most metal received for decontamination is generally incomplete and, as a result, the presence of such conditions is difficult to detect prior to actual receipt of the material for processing.

For the purposes of this evaluation, operational data and records from the first 15 metric tons of metal processed through ALARON's chemical decontamination process line were compiled and reviewed. In general, this material exhibited the following characteristics:

- a range of common metal types -- from 304, 304L and 316 stainless steels to structural-grade carbon steels to various anodized and non-anodized aluminums.
- a wide variety of sizes, shapes, configurations and original functions -- including hand tools, spent fuel racks, structural I-beams, galvanized roofing material, pipes, stands, valves, angle iron, tanks and vessel components.
- a range of surface contamination levels -- from 20,000 - 1,0000,000 dpm/100 cm2 "smearable" beta-gamma contamination and from 300 - 1,000 cpm "fixed" beta-gamma contamination.
- less than 10% of surface area classified, upon receipt, as "permanently inaccessible"
- close to 100% of material classified, upon receipt, as generally "amenable" to chemical decontamination.

The material was, for the most part, processed in a series of operational "campaigns" designed to separate metal and secondary wastes by customer and metal type (i.e., carbon steel vs. stainless steel vs. aluminum). Typical chemical decontamination unit processed batch sizes were 100 - 1,000 kilograms depending upon the configuration and density of the metal processed.

PROCESS FLOW SCHEME

Formally, the term "chemical decontamination" denotes only one of several specific unit processes used collectively in the decontamination of metal to achieve free-release. The flow of incoming material (metal) and the resulting secondary waste-streams are depicted in Fig. I. The metal received for "chemical decontamination" was first sorted and segmented to remove "permanently inaccessible" surfaces. The material removed included, specifically, metal with severely damaged surfaces and/or surfaces which, due to their configurations and geometries, could not be easily modified by further segmentation to provide ready-access by the decontamination media and/or survey instruments. These metal pieces were collected, further reduced in size and, when possible, embedded within the chemical stabilization matrices produced during the solidification and stabilization of chemical decontamination process waste (described below).

Fig. 1.

Since most metals received for decontamination exhibited relatively high levels of "loose" or "smearable" contamination (generally exceeding 60,000 dpm/100 cm2), a rinse step using hot (90o C) water at 100 - 200 kg/cm2 pressure was employed to reduce "loose" contamination below 50,000 dpm/100 cm2. In some cases, the rinse-water also contained a surfactant to remove grease, oils and other residue from the metal surface. Reduction of "loose" contamination levels and removal of chemical residues enhanced the performance of the subsequent chemical decontamination process step by:

- minimizing radionuclides in the chemical bath which may, at high concentrations, "recontaminate" the metal by redepositing on exposed metal surfaces
- minimizing chemical interferences associated with greases, oils and other coatings.

Liquid waste from the rinse step was volume-reduced through evaporation and the remaining residue solidified using cement-based chemistry prior to disposal or storage. The liquid waste from the rinse step contained no appreciable levels of

chromium (from stainless steels) and, therefore, required no stabilization to render the material non-hazardous.
Following the rinse step, the pretreated metal was then subjected to the chemical decontamination unit process. Typically, the metal was immersed in 5 - 25% fluoroboric acid at 60o - 90o C and allowed to undergo an "etching" or "pickling" process for 15 - 60 minutes. The acid was then drained from the tank to allow a 5 - 10 minute rinse with fresh water. Once saturated with metal, the acid solutions was then chemically or electrochemically regenerated and reused for processing additional incoming metal. The resulting liquid waste was volume-reduced through evaporation and the remaining residue solidified using cement-based or pozzolanic chemistry prior to disposal or storage.
Metal from the chemical decontamination unit process was then surveyed for free-release. Materials that did not meet ALARON's standards for free-release were returned to the chemical decontamination or sort & segment unit processes for further processing as "rework". In general, most material exiting chemical decontamination was free-released without additional processing. Survey rejection rates during normal periods of operation averaged roughly 7 - 10% by weight.
MINIMUM SECONDARY WASTE VOLUMES (MSWV)
The data and records associated with the first 15-metric tons of metal processed through chemical decontamination at ALARON's facility were reviewed and analyzed to reveal those specific process batches and campaigns in which no errors or problems were noted and which, accordingly, produced the lower quantities of secondary wastes. These "ideal" batches accounted for roughly 35% of the total 15-metric tons considered in this evaluation.
Typical minimum secondary waste volumes (MSWV), expressed as cubic centimeters of secondary waste per kilogram metal received for processing, are summarized in Table I. MSWV ranges and median values are provided for each of the three major unit processes specified previously in Fig. I -- sort & segment, rinse and chemical decontamination. The MSWV values reflect final, solidified and stabilize waste volumes per kilogram of metal as received at the front-end of the process sequence.
The MSWV values provided in Table I reveal that, under relatively ideal operating conditions:
  total MSWV produced by the entire process sequence was roughly 47 cm3 per kilogram of incoming metal; this corresponds to approximately 1.5 ft3 secondary waste per ton (short) of metal received.
  the chemical decontamination unit process accounts for the lion's share (>70%) of total MSWV
  roughly 20% of the total MSWV resulted from removal of "permanently inaccessible" surfaces or surfaces not "amenable" to chemical decontamination during the sort & segment unit process
  less than 10% of total MSWV was attributed to the rinse unit process
TABLE I
OPERATIONAL AND PLANNING (O/P) ERRORS
Information from the remaining, non-ideal batches and campaigns was reviewed to identify and pinpoint specific operational and planning (O/P) errors or problems noted throughout the course of process operations. Of the approximately 25 batches reviewed, all were characterized by at least one of the six O/P errors presented and described in Table II.
Of the six major O/P errors identified, three were associated with the sort & segment unit process, two with the chemical decontamination unit process, and one with the rinse unit process. The six O/P error types are listed in Table II in descending order of frequency recorded -- with " volumetric contamination from torch cutting" during sort & segment representing the most commonly encountered error and "incorrect process specification" for chemical decontamination representing the least frequent error. The underlying cause(s) for each O/P error, as well as the effects on unit processes and secondary waste production, are also summarized in Table II.
TABLE II
RELATIVE IMPACT OF O/P ERRORS ON SECONDARY WASTE VOLUMES (SWV)
Quantities of secondary waste produced by the unit processes affected, respectively, by each of the six O/P errors outlined in Table II were recorded and compiled. Ranges and median values for the secondary waste volumes (SWV) produced during process batches in which each of the O/P errors occurred are summarized in Table

III. In addition, a comparison (difference) is provided for each non-ideal unit process median SWV against the corresponding median MSWV from Table I. Finally, Table III includes a comparison of each error-induced secondary waste impact as a percent of the ideal, corresponding MSWV (again from Table I):
The data provided in Table III indicate:
the more grievous errors in terms of additional secondary waste produced were due to "incorrect process specifications" for the chemical decontamination and sort & segment unit processes; both of these types of errors were due to improper or incomplete characterization of the metal surface used to specify the treatment sequence for decontamination (see Table II) -- in other words, chemical decontamination should never have been selected for these materials.
the "volumetric contamination" and "excess splatter" from torch (plasma arc) cutting during the sort & segment unit process produced significant quantities of additional secondary waste and given their relatively high frequencies of occurrence in the study, represented areas of particular concern.
"process control variance" for both the rinse and chemical decontamination unit processes had relatively little impact on additional secondary waste volumes; considering their low frequencies of occurrence, these errors were found to be of relatively minor importance when compared to errors committed in the sort & segment unit process.
TABLE III
CONCLUSIONS AND FOLLOWUP
The results of this investigation illustrate the overall importance of good planning and preparation in order to minimize secondary wastes generated by chemical decontamination operations. Prior to this evaluation, wastes generated during the rinse and chemical decontamination unit processes were thought to represent the major waste volumes associated with the decontamination of metal for free-release. Under that assumption, major emphasis in limiting waste production was placed on maintaining good process control during the operation of those two unit processes. As the data indicate, however, the more significant waste volumes were produced by errors unrelated to rinse or chemical decontamination unit process control. In fact, the more serious errors in terms of waste production were attributed to poor planning. Further, significant waste volumes were also attributed to improper metal preparation (i.e., cutting techniques) prior to the rinse and chemical decontamination steps.
Using this information, ALARON Corporation devised and implemented procedural changes which, based on more recent operational data, maintain unit process secondary waste volumes routinely consistent with the MSWV's provided in Table I. More awareness of potential problems and greater diligence in characterizing incoming materials have all but eliminated excess waste due to incorrect process specification. Technologies used in the sort & segment unit process have been modified to enhance downstream processing by chemical decontamination. Greater emphasis on non-torch cutting method (e.g., saws and "nibblers"), as well as the use of "shields" to prevent the deposition of splatter from torch cutting on metal surfaces, has contributed to greater control of excess secondary waste volumes.
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46-22
SUBCONTRACTING STRATEGY FOR THE DECONTAMINATION AND DECOMMISSIONING OF SAVANNAH RIVER SITE'S FIRST TRITIUM EXTRACTION FACILITY, 232-F
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ABSTRACT

The Savannah River Site (SRS) has been actively proceeding with the decontamination and decommissioning (D&D) of various facilities and structures which were instrumental in the success of past missions at the site. The most ambitious of these efforts involves the subcontracting of the complete D&D of the first SRS Tritium Extraction Facility, identified as building 232-F. This facility operated in the mid 1950's and discontinued operations permanently in 1958. The approach utilized for this effort attempts to invoke the novel principle of "As Commercial As Reasonably Achievable" or "ACARA". This concept of ACARA applies only the minimum essential requirements necessary to successfully perform the D&D task. Integral to this approach is the subcontractor provision for maximum flexibility in the identification of and adherence to the requirements of applicable DOE Orders, federal, state and local laws and regulations, as well as site specific procedures without violating the site contractual requirements. The technical specification prepared for this effort provides the basis for a competitively bid contract to perform the entire D&D evolution, including initial facility characterization, waste stream characterization and certification, D&D and waste disposal. Preparation and development of this specification and the subsequent Request For Proposal (RFP) was a successful team oriented endeavor. The schedule for this fast-track undertaking took three months to complete. Successful completion of this task will be the first D&D of a facility containing both radioactive and hazardous material at an operating site within the DOE Weapons Complex.
The strategy for preparing the D&D subcontract for the 232-F structure was facilitated by applying the ACARA principle. This approach resulted in the accelerated development of the specification and RFP documents, as well as minimized the complexities of proposal evaluations.♀

INTRODUCTION

The old Tritium Extraction Facility, at SRS, known as Building 232-F, was the first Process and Production Facility of its kind in the 1950's to exist in the early Atomic Energy Commissions (now Department of Energy, DOE), Weapons Complex. At a time critical to National Defense and Security, 232-F produced tritium, a vital component in the development and manufacture of thermonuclear weapons. In the 36 years since 232-F ceased production, no activity of any production significance occurred in the facility. The building has remained vacant with the primary contaminated processing system purged and flushed and, over the years, useful non-contaminated ancillary equipment scavenged for use elsewhere on the site.
In recent years, the facility has been restricted from access, except under controlled entry conditions involving respirators and/or protective clothing. This was mainly due to the radiological condition of the facility and the industrial safety concerns resulting from building deterioration. Furthermore, a report by the DOE Office of Nuclear Safety registered 232-F as a concern over "increasing spread of contamination within the facility (1)."
In February of 1994, the decision to D&D the facility by the use of an independent subcontractor was made. This was the first major undertaking of the SRS D&D Department in making progress on an ambitious, site-wide decommissioning schedule. The process by which a strategy was developed to begin the procurement activity on this subcontract, is the focus of this paper. The intent from the beginning of this activity was to maintain this effort as a commercial endeavor, invoking only the minimum essential requirements necessary to perform a successful D&D effort. An ACARA concept was conceived and pioneered during this procurement which involves allowing the subcontractor maximum flexibility in the identification of and adherence to the regulatory laws and requirements, as well as site specific procedures, applicable to the D&D effort.

FACILITY AND PROCESS HISTORY

The 232-F Facility is comprised of 1765 m2 of building area with approximately 1579 m2 being an enclosed structure. The facility consists of two butting sections, a process wing and a laboratory/administrative wing. The process wing is 40 meters by 25 meters while the laboratory/administrative wing is 27 meters by 21 meters. The remaining open building area consists of approximately 198 m2 of concrete loading dock and access bay area as well as an outside exhaust fan area with stack. The building frame is of structural steel with exterior walls of Transite. All floors are poured concrete with asbestos bearing tile in the laboratory/administrative areas. Interior walls are Transite except where radiation levels required concrete, steel and lead as shielding materials. The floors and walls in process areas are

coated with Amercoat.
The building is a one-story structure except in the center where a second story and basement were required to house sections of the process equipment. The roof is a cement slab of built up construction on top of structural bar joints. The roof surface is tar mopped on saturated tar felt and covered with gravel. Roof areas were utilized in the placement of ventilation equipment. Ductwork from the roof leads to a concrete fan platform on the southside of the main building. The fans are connected to a reinforced concrete stack 60 meters tall, 3 meters in diameter at the base and 1.5 meters in diameter at the chimney cap. (See Floor Plan, Fig. 1) The air intake ducts are galvanized sheet and the exhaust ducts are stainless steel. The heating and ventilation supply ducts are insulated with bitumen coatings.
In the 1950's, tritium processing did not require the stringent seismic codes of today's structures, therefore, structural steel, cement slabs and Transite walls were used as the common building material (2). Other building materials used during construction (later defined as hazardous) were asbestos insulation, asbestos floor tiles, lead-based paints and lead brick used for personnel radiation shielding. The process used elemental mercury in process pumps and diffusion pumps. The process pumps used were the Sprengel brand. These pumps contained some Inconel parts. Process equipment and designs were drawn from experience at the Hanford Works and tritium process flows were extrapolated from the initial tritium production conducted at the Los Alamos Scientific Laboratory.
The 232-F process equipment was enclosed in a single stainless steel cabinet with panels. As previously mentioned, the process area is a two story section with basement integrated with the single story layout. This large vertical height , over 8 meters, was required to house the thermal diffusion column, hold tanks and process pumps that were critical to the tritium production. The panels, associated with the steel cabinets, were framed and held plexiglass windows for visual inspection of equipment. The panels were mounted on tracks to permit opening for access to equipment for maintenance or replacement.
Fig. 1.
Irradiated targets, supplied from other SRS facilities, were received as canned "slugs" about 30 centimeters in length and 5 centimeters in diameter. The slugs were removed from a shipping cask by a remote manipulator. Using the manipulator, the aluminum sheeting or hull was separated from the Li-Al target slug (called decanning) through a die. The hull was crushed to conserve space and placed in another shipping cask for eventual processing and shipment. The target material was placed in a crucible and charged to one of the two extraction furnaces. The furnace was placed under a vacuum and heated in excess of the target-melting point. Tritium, Helium and small concentrations of ordinary hydrogen were extracted from the target during the diffusion heat cycle. In addition, fission products were entrained in these flows. A diffusion pump delivered the gases through a palladium metal barrier where the heavier tritium was extracted from the bottom and captured in a hold tank for analysis. The by-product line containing Helium-3, Helium-4, Nitrogen2 and trace Tritium2 was pumped to storage tanks for analysis and later pumped through a zeolite bed to a cylinder trailer for shipment.
The success of a D&D activity involving contaminated facility and process systems is dependent, by and large, on the known or documented process history records maintained during the facility's operational existence. Unfortunately, due to the short operating life of 232-F, approximately two years, and the excessive time lag between facility shutdown and this D&D effort, approximately 36 years, limited information existed to document the periodic processing of this facility during the 1950's time frame. Health Physics documentation was available from the 1950's and the proposed facility shutdown procedure was found. This documentation provided useful historical information, but was limited in assisting the development of the specification for the D&D Procurement.
To develop a working knowledge of this facility, the project utilized retired Westinghouse (DuPont) personnel that had worked in the facility to recall, document and develop a process and information history of the 232-F facility. This source of information was beneficial during preparation and execution of the contract specification.
The major radiological and hazardous chemical constituents associated with the 232-F Facility are tritium, mercury, lead and asbestos (3). Tritium is the only radioactive isotope of hydrogen. It has an atomic weight of 3 and exists by itself

as a gas or in combination with oxygen (tritium oxide) as a liquid. Tritium has a relatively short radiological half life of 12.3 years. Tritium decays to Helium-3 by the emission of a low energy beta particle. Tritium is readily dispersed in air and is a highly diffusible element that will penetrate a wide variety of substances including rubber, plastic, and metal. Tritium can be inhaled and is considered an internal hazard.
Beta-gamma radioactivity was introduced to the facility from impurities in the LiAl target slugs. Beta particles are charged electrons emitted by decay of some radioactive elements, such as tritium and strontium. Gamma radiation is a highly penetrating electromagnetic radiation of extremely short wavelength similar to x-rays. Gamma-emitting nuclides, such as those of cobalt and iridium, are a hazard both when ingested or inhaled and when external to the body. The majority of the beta-gamma impurity contaminant in 232-F are expected in the process furnaces.
Mercury is a Resource Conservation and Recovery Act (RCRA) regulated hazardous metal. Mercury was used in the process pumps and the process column to transport gases through the equipment. It is highly probable mercury contamination exists throughout the process piping and equipment. Lead is also a RCRA regulated hazardous metal. Lead bricks are used for shielding in the process room furnaces and the stack. The decanning room viewing windows consist of oil-filled, lead glass layers. Lead buttons are used on the building exterior to hold transite panels in place. Paint throughout the facility is suspected to be lead-based paint.
Asbestos is regulated under Toxic Substances Control Act (TSCA). The building exterior and interior walls are transite panels. Floor tile and mastic in the administration and process areas are suspected to contain asbestos. Wall, piping and ductwork insulation is suspected to be asbestos insulation. Electric control panels, electric boxes, breaker panels, etc. are suspected to contain asbestos components.
Polychlorinated Biphenyls (PCB's), also regulated under TSCA, may be present in lighting ballasts and transformer oils. Oil is contained between the lead glass layers of the decanning room viewing windows. Hydraulic oil may be present in pumps and other motor driven components. Freon may be present in drinking fountain cooling systems and the process cooling water system may have utilized chromated water.
During operation, the process likely generated little waste for disposal, with the exception of the radioactive spent metal (the LiAl target slug) and the day to day controlled wastes such as personnel protective clothing. Other than mercury and process cooling water, no liquids or chemicals were introduced to the process during normal operation. Mercury was used in the pumps and in the process column, and thus, there is sound basis for considering the process piping to be mercury contaminated. Tritium is practically insoluble in mercury, and therefore, any mercury remaining in the facility would be expected to have nil to trace amounts of residual tritium. Chromated water may have been utilized in the process cooling water system.
The most comprehensive radiological survey involving the entire facility was conducted in 1990. The results of the 1990 radiological survey for radioactive contamination are as follows:

Maximum transferable contamination of <500 d/m alpha and <1x105 d/m beta-gamma 0.1 m2 detected in the Decanning Room. Maximum dose rate of 5 mr/hr detected in the Decanning Room.

Swipes throughout the remainder of the building were <500 d/m alpha and 1 x 104 d/m beta-gamma 0.1 m2. Dose rate of 1 mr/hr (background) detected in the general area. Disc smears were <10 d/m alpha and <80 d/m beta-gamma except for the decanning room which was 10 d/m alpha and 1475 d/m beta-gamma 100 cm2.

The tritium survey detected minimum observable readings of 145 d/m and a maximum reading of 5.3 x 105 d/m in the Process Room.

Currently, no areas have airborne contamination of any kind. As indicated in Fig. 1, the building has some areas that do not require radiological work permits for access. No personal protective equipment is required to enter these areas.

D & D SCOPE

The Scope of this project involved the complete subcontracting services of all aspects and support functions of a D & D activity. These services include planning, preparation, equipment, engineering, material, labor, tools, and management for the D&D of the 232-F Facility. the D&D effort includes the existing building structure, exhaust stack, outside equipment, and contaminated systems within the process area. The procurement documents were to encompass all sampling and analysis, including laboratory work, waste characterization and waste certification. A formal D&D plan

was expected which would provide direction and a technical accounting of all D&D techniques to be used during the job. Various plans and reports, as well as site specific safety and health documentation, were expected from the successful subcontractor. Waste disposal of all generated waste from this activity through final release and acceptance of the remaining grounds of the facility area were also included.

PROCUREMENT PLANNING AND STRATEGY

The DOE indicated a strong desire to exercise SRS's abilities in obtaining the services of commercial D&D subcontractors to further the sites mission in environmental remediation and waste management activities. Procurement of subcontracting services of this type is also beneficial to SRS. First, these tasks permitted technically, qualified, experienced organizations to perform their services and promote technology transfer and development while at SRS. Second, these tasks allowed multiple D&D tasks to be performed at full capacity while minimizing SRS site resource involvement. Thirdly, this activity would initiate progress on a site wide D&D schedule proposed to D&D other site facilities adopting the same procurement format. The challenge presented with the 232-F procurement was time. In as much that a service procurement of this type had not been attempted previously at SRS, there where no previous examples to model. Thus, from a specification and requisition viewpoint, the development of these documents did not have the benefit of previous models or similar procurements nor was there ample time to search in great detail for industry wide sources. The time frame to develop, approve, competitively bid and eventually procure the award for this contract was four months. This ambitious schedule was proposed to support DOE-Savannah River direction for Waste Management activities.

Considering the magnitude of the task, including the cross organizational involvement required to construct the RFP, a team approach was adopted and used through the entire RFP development. One of the key successes to the effort was the team approach. A list of organizational entities integral to the SRS RFP included:

- Construction Engineering Services
- Design Engineering and Management
- D&D Project Management
- Industrial Hygiene
- Site Operations Engineering
- Safety and Health Department
- Site Facilities Management
- Environmental Protection Department
- Procurement Management
- Radiological Protection Department
- Quality Engineering
- Waste Management Engineering
- Systems Group Engineering
- Office of General Council

Frequent, scheduled meetings of all team members were held to review the initial revisions of the RFP specification and review, concur and incorporate comments into the final document. Over the course of the specifications development, nearly 1100 comments were addressed, responded to and/or incorporated into the document.

The basic requirements of the RFP involved a listing of subcontractor responsibilities and a document submitted schedule. The typical subcontractor responsibilities consisted of:

- Facility Characterization - this involves the facility Sampling and Analysis Plan (SAP); waste certification documentation; required inspections and radiological surveys to determine isotopic quantities; and surveys to quantity all hazardous, mixed, TSCA waste, sanitary, and other wastes requiring removal.
- Decontamination and Decommissioning Planning - this involves the development of the Decontamination and Decommission Plan for decommissioning the facility.
- Engineering Services
- Development of the detailed work procedures and work specifications necessary to achieve project objectives.
- Quality Assurance Program
- Waste Disposal - this involves preparing the shipping manifest, supplying all shipping containers, waste classification and certification activities, packaging, and waste preparation services required to meet disposal facility waste acceptance

criteria.

Project Performance and Controls - this required the development and utilization of a Project Management Plan to perform all direct management of subcontractor and lower tier subcontractors to achieve the project objectives.

Labor - requires the subcontractor to provide all labor to perform the D&D of the 232-F structure and contaminated systems.

Occupational Safety/Industrial Safety and Health - involves the development of a project specific Health and Safety Program (HASP) that covers the entire scope of work including potential hazards as they affect SRS employees, other subcontractors, and the public.

Environmental Protection - involves the documentation of all onsite monitoring during D&D and preparing the required waste management, sampling, analysis, and reporting in accordance with the applicable federal, state and local environmental laws and regulations.

Health Physics - requires that all health physics services including radiological surveys, personnel monitoring, radiation training, radiological records and inventories, and as low as reasonably achievable (ALARA) coordination be supplied by the subcontractor.

Contract Close-out activities consisting of a site radiological survey, final grading of any parts of the disturbed site, and a final report providing an overview of the project activities, accomplishments, final site status and lessons learned.

Document submittals for the proposal evaluation required outlines of the D&D Plan SAP, Waste Management Plan, Environmental Compliance Plan, and Safety documentation.

Minimizing excessive requirements or unnecessary site requirements and restrictions was another key factor in the success of this effort. The concept of ACARA was developed and implemented on this task. What ACARA attempts to do is apply only the minimum essential requirements necessary to safely and successfully perform a task. With respect to SRS, and this 232-F D&D, the attempt was made to treat this procurement activity as a commercial activity. This meant minimizing the influence of the many DOE Orders, existing site procedures and requirements and allowing the subcontractor maximum flexibility in the identification of and adherence to the codes, laws and regulations necessary to do the work. The experience of the subcontractor is utilized to the maximum extent. The responsibility remains with the subcontractor for identification and compliance with all governing regulations, codes and standards that are applicable to the D&D effort.

In this instance for 232-F, SRS established a base of minimum required documents that the successful subcontractor could build upon. These documents included Titles 10, 29, 40 and 49 of the Code of Federal Regulation (CFR), Chapter 61 of the South Carolina Code Laws, the DOE/EH-0256T (Radiological Control Manual), DOE Order 5400.5 (Radiation Protection of the Public and Environment) and Regulatory Guide 1.86 (Termination of Operating Licenses for Nuclear Reactors). During the writing of the RFP, and in keeping with the ACARA concept, the radiological release criteria for this D&D activity was purposely kept as an open item with the intent to have the competitive bid process develop a creative approach to this seemingly arbitrary choice. Even though a choice for release criteria was suggested in the RFP, the criteria established in the DOE RCM provided the limits best suited for the SRS Mission. This was eventually adopted as the release criteria to use for this D&D.

The overall ACARA approach provided a cost effective means to issue a complicated procurement activity in a very short period of time. These practices allowed the RFP to be prepared in less than six weeks and resulted in an approved package ready for bid solicitation in approximately twelve (12) weeks. Our 232-F specific schedules allotted perspective subcontractors twenty-two (22) days to submit their proposals. The detail of the bid requirements in the RFP (plan requirement submittals and technical requirements) provided enough information that a technical evaluation of the bids was virtually eliminated. This streamlined the bid evaluation process by providing a successful "checklist" type means of quickly and accurately evaluating proposals. This effort was solicited as a fixed price contract.

Only two (2) days were required to review the lowest competitively submitted proposal. It was determined that the proposal complied with the technical specification of the RFP and that it was a valid proposal for contract award. The entire cycle from conception to technical acceptance of a viable bid proposal was approximately 15 weeks.

CURRENT STATUS AND LESSONS LEARNED

At the present time, it has been announced publicly that the Virginia based firm, Babcock and Wilcox Nuclear Environmental Services, Inc., has been awarded this D&D contract for a fixed price of 1.6 million dollars. Activities are underway to begin mobilization of the company onto the SRS work area. Waste certification activities are the initial tasks which will take place. This will involve sampling and subsequent laboratory analysis, followed by waste characterization and certification determinations. The sampling activities are scheduled to begin in April/May 1995 with D&D activities beginning in September/October 1995.

Lessons learned from this task include the following:

The team approach to the RFP and technical specification development is critical and very positive to the success of the effort.

Education, understanding and application of the ACARA thought process is essential. Allowing a greater dependency on subcontractors to be responsible for and complying with governing standards and regulations within the work scope, in this instance, worked well and was cost effective by providing a streamlined bid evaluation process.

It must be recognized up front, that it takes time to change established practices and thought processes that the ACARA concept attempts to overcome. Valuable time can be wasted until this concept becomes a viable way of doing business.

Choose a definitive release criteria versus leaving the choice as part of the contract. The choice for this task was the limits found in the DOE RCM.

Top management must give proactive support for a task as schedule sensitive as this RFP.

Recognize that vision and flexibility are the key elements in ACARA. Ask how the task at hand would be done if there were no site regulations and requirements or what would be required for the task if the D&D facility was in an offsite "greenfield" area.

Established start-up practices, procedures philosophies be directly applied to D&D activities. Must develop this direction on an "As Required" basis.

Foster DOE counterpart support of the ACARA approach.

REFERENCES

1. Office of Nuclear Safety Field Programs Quarterly Report for Savannah River, period 10/92 through 12/92, U. S. Department of Energy dated May 25, 1993.
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46-23

DECONTAMINATION OF A MANUFACTURING FACILITY CONTAMINATED WITH THORIUM OXIDE

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ABSTRACT

Bartlett recently decontaminated a facility that had previously been a site for the fabrication of radar tubes. The facility was uniformly contaminated with loose thorium oxide contamination. In order to minimize the cost of the decontamination, minimize the volume of radioactive waste, and minimize exposure risks associated with airborne or loose thorium oxide contamination, Bartlett and the facility owner decided to apply strippable coating to all accessible surfaces and all contents of the rooms. Loose thorium oxide was reduced below detectable levels.

DECONTAMINATION PROCESS

Bartlett Nuclear Inc. (BNI) recently decontaminated a facility that had previously been a site for the fabrication of radar tubes. Several models of the radar tubes were fabricated with cathodes of thorium metal. The facility had been established during World War II and had been in use for approximately 50 years. The construction of the facility included wooden floors, some with several layers of subsequent flooring, and plasterboard walls. The owner of the facility had negotiated a contract for the sale of the property and desired to terminate the nuclear material license. The scope of the decontamination included removal of ventilation ducts and the removal or decontamination of process and sanitary drains. The facility was uniformly contaminated with loose thorium oxide contamination at levels below the action level guidelines of USNRC Regulatory Guide 1.86. Nevertheless, the owner

decided to decommission the facility by removing the loose contamination and reducing fixed contamination below the action levels as much as practicable.
In order to minimize the cost of the decontamination, minimize the volume of radioactive waste, and minimize exposure risks associated with airborne or loose thorium oxide contamination, Bartlett and the facility owner decided to apply strippable coating to all accessible surfaces, including walls, floors, and all contents of the rooms. The contents included benches, sinks, electrical apparatus, cabinets, and process equipment. The strippable coating was applied and removed in two days in a 2000 square meter facility. As a result, loose thorium oxide was reduced below detectable levels. There was no airborne radioactivity encountered, and no personnel or clothing were contaminated.
Subsequently, Bartlett performed a complete survey of accessible surfaces for elevated levels of alpha and beta/gamma contamination. We found that elevated levels of beta/gamma contamination generally indicated subsurface alpha contamination, presumably due to ingrowth of thorium progeny. Ventilation ducts, drains, electrical conduits, and process equipment, were treated on a case-by-case basis.
The total cost of the decontamination was less than $100,000.

CONCLUSION

In conclusion, the results indicated that the use of strippable coating for decommissioning of facilities contaminated with loose metal oxide contaminates could be remarkably cost effective from the perspectives of labor, cost, time, waste volume, and health protection.

46-25

DECONTAMINATION OF STAINLESS STEEL HEAT EXCHANGERS WITH OZONE-ENRICHED FOAM TO ALLOW STEEL RECYCLING

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ABSTRACT

The enhanced decontamination properties of nitric acid with Ce(iv) additive on stainless steel are well known. This paper discusses an application of the method (COMODIN process) to decontaminate five heat exchangers with recirculating foam. A sixth heat exchanger was also decontaminated by the same method, but with ozone-enriched foam. The process allows continuous regeneration of Ce(iv) without complex facilities, and its effectiveness was more than tripled. The process is a very promising development that minimizes the liquid waste volume, the dry residue requiring conditioning as an ultimate waste form, and the occupational dose rate.

INTRODUCTION

In a paper presented at Waste Management '92 in Tucson (1), one of the authors described the use of suitable foam compositions to decontaminate hollow components in the nuclear industry, notably those with complex shapes such as valves and heat exchangers.
Contamination of metal surfaces by fission products or by activation products and actinides is known to occur at two levels:

- contamination is deposited as metallic oxides are formed or develop on the surface of stainless steel and inconel alloys; the oxide compositions depend on the environmental conditions at the time of formation;
- a small fraction of the contamination penetrates into microcracks and intergranular gaps in the metal itself.

Equipment maintenance procedures call only for removal of the oxide layer and its contaminants, to avoid initiating corrosion processes in the metal. The residual dose rate continues to represent an occupational hazard. For decommissioning operations, however, the second layer of contamination must also be eliminated, implying removal of the metal surface to a depth of several micrometers. This operation not only virtually eliminates the occupational irradiation hazard, but also allows the steel to be reused for other purposes after melting in a specialized steel mill. An effective method for achieving this difficult erosion on stainless steel and inconel alloys is to employ nitric acid with Ce(iv) additive.
Even when adequate amounts of Ce(iv) are dissolved in acid mixtures, however,

in-depth decontamination of Fe, Cr, Ni steels is a very slow process that consumes large (and expensive) quantities of Ce(iv); yet acid etching of the metal substrate is indispensable to ensure a very low residual contamination level that would allow the steel to be recycled. Moreover, adding Ce(iv) progressively as it is transformed into Ce(iii) causes ions to accumulate in solution. This increases the quantity of dry residue obtained after treatment, e.g. coprecipitation of the liquid waste generated by the operation. The dry residue, which contains all the initial contamination, will be conditioned for long-term disposal, and the cost is directly proportional to the quantities produced. In addition, electrolytic regeneration of Ce(iv) from Ce(iii) in ferrous alloys is made difficult by the fact that the Fe(ii) Fe(iii) reactions predominate at the electrodes.

In this context, the use of ozone was considered to regenerate Ce(iv) from Ce(iii) directly in the decontaminating foam. The foam is an ideal active medium for ozone, which exhibits very limited solubility in water. The immensely larger liquid surface area, due to the thin films in the foam bubbles, enhances ozone dissolution and therefore its oxidizing effect.

DESCRIPTION OF CONTAMINATED HEAT EXCHANGERS

Six identical heat exchangers (Table I) were contaminated on the shell side by the pool reactor primary cooling fluid. The contamination was estimated by the Radiological Protection Department when the primary system was isolated from the reactor in December 1992: samples were taken from two of the heat exchangers and dose rates were measured around the circumference of the units. Each sample consisted of a disk with an area of 23 cm2 weighing 57 g, together with swarf from the contaminated inner face. Samples were taken from a highly radioactive portion (1.8 mGyh-1) of heat exchanger 4 and from a slightly radioactive section (0.2 mGyh-1) of heat exchanger 6. The total activity was 70.9 GBq, or an average of 11.8 mGBq per unit; the mean dose rate was 875 mGyh-1.

TABLE I

Figure 1 indicates the radioactivity spectrum for heat exchangers 4 and 6 at the time of decontamination in December 1993. The total activity and dose rate diminished significantly thereafter (200 mGyh-1) because of the short half-lives of the contaminant radionuclides.

Fig. 1.

THE COMODIN PROCESS

Foam, created in a generator where reactants are mixed with gas, is continuously circulated through the contaminated unit from the bottom up until foam flows out from the top of the unit and is recovered for recirculation. The liquid runoff from the unit and the foaming gas are recovered in the recirculation device to create additional foam. The same quantity of liquid, filtered and/or regenerated at each cycle, can thus be used indefinitely until complete decontamination is obtained. The gas contained in the foam is also partially recycled.

The COMODIN recirculation unit comprises mainly a tank and hydropneumatic pumps suitable for use with liquids and foams. The system is maintained under slight negative pressure by filtered exhaust blowers.

The contaminated heat exchangers were installed three at a time in the basement of the G2 reactor at Marcoule, which has been decommissioned to Stage 2. They were inclined on a cradle (Fig. 2) to prevent the internal heat exchanger baffles from hindering the fluid circulation. Foam was injected at the base of each unit through a safety valve, and flowed out of an orifice at the top for recovery by the COMODIN device. During each decontamination cycle, 50 ml liquid samples were taken from the foam and liquid return lines to measure their dose rates and total Fe concentrations, indicative of the extent of metal dissolution. Between each cycle, the contact dose rates were measured at three points (top, middle and bottom) of the heat exchanger shell.

Fig. 2.

Decontamination Reactants

The reactants used here were similar to those employed for the decontamination of the Rapsodie reactor primary system (2); the same AISI 304 stainless steel grade was used in both cases.

Alkaline and acid foams were used in turn, for periods of 4 hours each. The first was a 12% NaOH solution; the second was a mixture of 0.5N sulfuric acid and 2N nitric acid together with 0.04 moll-1 of Ce(SO4)2 additive. The alkaline fluid was filtered across 100 mm PALL filter cartridges, and the acid fluid was filtered to 10

mm. The filters were replaced whenever the dose rate reached 1 mGyh-1. The operation was terminated by two foam rinses. The surface-active agent (0.8 vol%) was a biodegradable mixture of betaine and glycoside covered by a CEA patent.
On completion of the operation, the filtered liquids were pumped into interim storage tanks. The alkaline and acidic liquid wastes were mixed to obtain a neutral fluid (pH 6) and sent via special flow lines to the central liquid waste treatment plant on the site. The salt resulting from this mixture constitutes an effective foam breaking agent.

Decontamination Procedure

The operation was conducted between December 1993 and February 1994, with the working temperature maintained at 20C.

Example 1. This work involved decontamination of five heat exchangers using air-generated foam. The Ce(iv) additive was regularly consumed by oxidation of the metal (about 2 g of iron were found per liter of solution) and had to be renewed continuously, for a total of 16 kg per cycle. Decontamination was satisfactory based on the drop in the dose rates (14 mGyh-1). Only a small quantity (1 m3) of liquid waste was produced for each heat exchanger.

Example 2. Heat exchanger 5 was decontaminated using ozone-enriched oxygen foam (100 g of O3 per Nm3 of O2) with 6 kg of Ce(iv). The reaction progressed some 50% faster than in the preceding test. Both acid foam steps provided better decontamination and the dissolved iron quantity was twice as high (4 gl-1). The residual heat exchanger dose rate was lower (< 8 mGyh-1), while the quantity of liquid waste was comparable to the previous test.

The ozone process uses relatively little cerium, requiring only an initial load. Any increase in the dry residue in the liquid waste is due exclusively to the larger quantity of metallic ions dissolved by the combined effects of Ce(iv) and the acid reactants.

The surface-active agents used to prepare the foam are slowly decomposed on contact with the ozone. It could be necessary to add a small quantity of surface-active agent during the decontamination operation, which may last for several hours. However, this drawback is minor in comparison with the advantage of the phenomenon: the surface-active agents are destroyed during interim storage of the liquid waste prior to transfer to the treatment station. The foaming power, which could be a nuisance at this stage, is considerably weakened.

RESULTS

Activity of the Decontamination Liquid Waste

The liquid waste generated by the decontamination operations were collected, sampled and diluted for analysis by the Marcoule Radiological Protection Department. Seven analyses were conducted as part of the waste management procedure to determine the total activity and spectrum for the following waste streams, where the alphabetic suffix A, B or R indicates the type of treatment (acid, basic or rinse) and the number of letters corresponds to the number of the treatment (AA: second acid treatment):

Analysis 1:	Heat exchanger 2:	basic treatment (2B)
	Heat exchanger 4:	basic treatment (4B)
Analysis 2:	Heat exchanger 2:	3 acid treatments (2A, 2AA, 2AAA)
	Heat exchanger 4:	2 acid treatments (4A, 4AA)
Analysis 3:	Heat exchanger 1:	2 acid treatments (1A, 1AA) & basic treatment (1B)
	Heat exchanger 2:	rinse (2R)
	Heat exchanger 4:	acid treatment (4AAA) & rinse (4R)
	Heat exchanger 6:	basic treatment (6B)
Analysis 4:	Heat exchanger 1:	2 rinses (1R, 1RR)
	Heat exchanger 3:	basic treatment (3B)
	Heat exchanger 6:	2 acid treatments (6A, 6AA) & rinse (6R)
Analysis 5:	Heat exchanger 5:	acid treatment (5A)
Analysis 6:	Heat exchanger 5:	acid treatment (5AA)

Analysis 7: Heat exchanger 5: basic treatments (5B)

The acidic liquid waste from heat exchanger 3 and the rinse from heat exchanger 5 were not analyzed. The liquid waste streams collected and analyzed therefore represent the decontamination of five of the six heat exchangers (1-6).
The analysis results are summarized in Table II. The liquid waste activity was calculated by multiplying the activity concentration (determined from the sample) by the liquid waste volume and the sample dilution factor. The total activity of the liquid waste, i.e. of the contamination removed from the heat exchangers, was thus 16.1 GBq. As the analysis results cover only 5 of the 6 heat exchangers decontaminated, the mean contamination removed from each unit may be estimated at 3.22 GBq, and the total contamination was therefore 6 3.22 = 19.32 GBq.
TABLE II
Estimated Residual Steel Activity prior to Melting
The activity remaining after decontamination was estimated from samples of the shell and internal tube bundle of heat exchangers 4, 5 and 6.
Sample Representativity. The heat exchangers were cut up into four elements (two end sections and two center portions). Samples were taken from one of the center portions of each heat exchanger. Each sample included disk obtained from the shell using a crown saw, and ten tube sections each about 10 cm long. The sampling positions were determined after radioactivity mapping around the tube bundle and on the inner portion of the shell using a Berthold 123 portable surface contamination detector to identify zones of average contamination levels, representative of the actual contamination of the given heat exchanger section.
Analysis Results. The specific activity values measured on the samples are indicated in Table III. The shell mass was 370 kg, and that of the tube bundle nearly 2330 kg; the total calculated activity in heat exchangers 4, 5 and 6 is also indicated in the table.
TABLE III
The activity in heat exchanger 5 appears to be appreciably lower than in the others. This is not unexpected, considering the dose rates observed after decontamination: the mean dose rate for heat exchanger 5 was 7.75 mGyh-1, compared with an overall mean of 14.2 mGyh-1 for all the units.
Estimated Residual Activity. The mean residual activity of heat exchangers 4, 5 and 6 was 65 MBq. The representativity of the samples means that this mean value can be applied to the other three units. The total residual activity was therefore 390 MBq, or about 0.4 GBq.
Decontamination Factor. The decontamination factor obtained by the COMODIN process was calculated from the mean activity removed from the six heat exchangers and from their residual activity:

Residual activity:		0.40 GBq
Removed activity:		19.32 GBq
Calculated initial activity:		19.72 GBq
Decontamination factor:	19.72/0.40	= 49.3

The overall decontamination factor was therefore approximately 50.
Note:The residual activity was calculated from the sample analyses performed at the end of July 1994, i.e. over four months after the decontamination liquid wastes were at the beginning of March. If allowance is made for this delay in estimating the residual activity, the actual decontamination factor drops to 41.
Heat Exchanger 5. Analyses 5, 6 and 7 are of particular interest, as they concern a single heat exchanger, No. 5. The activity removed by the decontamination process amounted to 3.49 GBq, and the residual activity was 0.022 GBq, for a decontamination factor of nearly 160, three times higher than the overall factor. Unlike the other units, ozone-enriched oxygen was used during the acid decontamination treatment for heat exchanger 5.
The radioactivity of the tube bundle was 100 times higher than the shell for heat exchanger 4, and 60 times higher for heat exchanger 6, but only 14 time higher for heat exchanger 5. As the shell activity was comparable for all the units, the residual activity of the tube bundle was therefore appreciably lower in heat exchanger 5.
Secondary Liquid Waste Spectrum
The liquid waste spectra are compared in Fig. 3. The spectra for heat exchanger 5

(analyses 5, 6 and 7) are distinguished by their very high 144Ce content, while the other spectra (for units 1, 2, 4 and 6) are more homogeneous.
Fig. 3.
Table IV indicates the concentrations of the three principal radionuclides (ruthenium, cerium and cobalt) found in the liquid waste streams. The basic treatment effluents contained significantly greater amounts of ruthenium (26 to 49%) than the acid treatment effluents (9 to 19%); basic treatments are thus apparently more effective for 106Ru decontamination. The opposite appears to be true for 60Co, which is more effectively eliminated by acid treatment. The situation is less clear for 144Ce decontamination, with comparable values for both basic and acid treatments, but with a high degree of residual cerium entrainment in the rinses.
TABLE IV
Residual Activity Spectrum
Figure 4 shows the radioactivity spectrum of the samples taken after decontamination and before melting. The mean spectrum was determined by calculating the average of the mean spectra for the shell and tube bundle, with allowance for the weight differences between them.
Fig. 4.
The most significant radionuclides were 60Co (30.4%), 106Ru/Rh (17.5%), 110mAg (32.4%) and 144Ce (8%). Note the high percentage of 110mAg, which did not predominate in the spectra recorded before decontamination.
CONCLUSION
At the request of a pool reactor operator, the CEA's nuclear facilities decommissioning unit (UDIN) submitted six stainless steel heat exchangers to a decontamination and decommissioning treatment consisting of the following operations:
  foam decontamination by the COMODIN process (with ozone-enriched foam for one of the units);
  cutting of each heat exchanger into four sections for dimensional compatibility with the specialized melting facility at Marcoule;
  melting;
  reutilization of the very low-level contaminated steel in the nuclear industry.
Radiological monitoring was conducted throughout the treatment:
  radioactivity assessment by dose rate measurements and application of transfer functions corresponding to the contamination spectrum prior to decontamination;
  quantitative laboratory analysis of the liquid waste generated by the decontamination process;
  assessment of the residual radioactivity after decontamination and before melting on samples taken from the tubes and outer shell of three heat exchangers for laboratory analysis.
The occupational dose rate was 1.2 man-mSv for the entire operation.
The COMODIN Ce(iv) foam decontamination process yielded a mean decontamination factor between 40 and 50, for an initial radioactivity level of about 20 GBq. Ozone-enriched oxygen in the process applied specifically to heat exchanger No. 5 significantly increased the decontamination factor to 160 for that heat exchanger; ozone management raised no major difficulties. Basic treatments proved more effective than acid treatments in eliminating 106Ru. The metal ingots are suitable for recycling, in view of the additional decontamination factor of 3 provided by the melting process itself.
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46-26
STUDIES OF APPLICATION OF MICROMYCETES AND CLAY COMPOSITION FOR DECONTAMINATION OF BUILDING MATERIALS
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ABSTRACT
The objective of this investigation was to examine the effect of decontamination of the building materials by simultaneous application of various species of micromycets and decontamination clay covering (DCC). Taking into account the abilities of some micromycetes to decompose "hot" particles and to adsorb radionuclides and using sorption properties of DCC the effectivity of decontamination can be greatly improved. In this study it was examined the ability of micromycetes to grow in different clay media under the condition of radioactive contamination of the tested surface of materials. More than two hundred strains of micromycetes belonging to 30 species and 10 genus were isolated, but only 28 strains of microorganisms were capable to grow on the surface of building materials, 10 demonstrated the ability to decompose "hot" particles. The best results in terms of decontamination were obtained when the species of the culture "Arthrinium mortagnei" and "Cladosporium" were used. The results of the test performed under the laboratory conditions showed that the decontamination using micromycetes decreases the contamination level from 55% to 120% compare to only DCC treatments. This corresponds to decontamination factor of 1.8 to 2.7.
INTRODUCTION
In recent years there has been observed increasing public and scientific concern with the problems of contamination and cleanup of terrestrial environments near Chernobyl Nuclear Power Plant (NPP). After the Chernobyl accident a number of the reviews and reports of the research on decontamination techniques available for cleanup of urban and supplement areas have been published (1-6).
The new composition of coatings for decontamination of radioactive deposit on buildings and structures was developed at the Ukrainian Academy of Science (7-8). Decontamination using a clay coating is suitable for large scale decontamination operations in urban environment, since it is safe, practical and cost effective. However, in many cases the residual contamination remains rather high. This is due to the contaminant level, the depth of contaminant penetration in porous substrates and the physical-chemical conditions of the surface of the building materials.
The analyses of relevant publications shows that the application of fungal organisms or their subcellular components along with associated process technology applied to manufacturing service industries and environmental management has received much attention in recent years (9,10). Several treatment approaches have been proposed for the cleanup of contaminated materials. One of the most promising is the biodecontamination using micromycetes. These fungi have been extensively studied (11-17) in the laboratory testing systems which permit observing and fixing interaction of the fungi with a radioactive source. The authors (14) have examined five species (six strains) of micromycetes isolated from radionuclide-contaminated soils and fifteen "hot" particles. These fungi are of interest for environmental cleanup research, because the combination of such properties as ability to overgrow and to destroy "hot" particles, and to adsorb radionuclides are ideally suitable for application with the decontamination clay coating (DCC). Clay minerals have been reported to influence microbial and enzymatic transformations on a variety of substances ranging from ammonium and sulfur to carbohydrates, organic phosphorus compounds. Such minerals have also been variously claimed to increase, decrease, or not affect microbial growth and activity. The literature on these subjects has been reviewed by Stotzky (18). In this study it was tested the ability of micromycetes to grow in the clay medium under different physical-chemical conditions and of radioactive contamination of the tested surfaces.
MATERIALS AND METHODS
Sampling Areas
The specificity of the Chernobyl accident and the features of release favored a widespread distribution of activity, throughout the variation of meteorological conditions and wind regimes. The levels of contamination and activity distribution by the multiple plumes on the surface varied widely. A large amount of studies of the radioactive contamination of the environment has been carried out (19-22). Objects of our monitoring include ground depositions (soil, buildings) on the exclusive zone. Data were collected from two different types of environmental

contamination regions - the city of Chernobyl and in the site areas within the range of 5-20 km from Chernobyl NPP. The measurement showed that the contamination level was formed by products discharged from the reactor, mainly in the form of "hot" particles derived from the fuel and activated matter. Radiographic, radiochemical and gamma-spectrometric analyses showed intrusion of "hot" particles having considerable specific activity 2.0-7.0103 Bq/particle and dominance of such elements as 90Sr, 134Cs, 137Cs, 144Ce and 241Am. In the other, more remote region, village of Vladimirovka of Polesky district, 65 km from the destroyed reactor the deposition of Chernobyl debris was carried in the lower parts of the atmosphere and was washed out by rain. 134Cs and 137Cs were prevalent in the radionuclide contamination.

Sampling

The most common town and farmhouses building roofing materials (clay roof tiles, roofing asbestos cement slate) were chosen as principal subjects for laboratory experiments. The orientation of the roofs was found to be important concerning the level of contamination. In the village of Vladimirovka samples of clay roof tiles were collected from the north-sides roof. Slate samples were taken from the north-west sides of the roofs of the houses in the city of Chernobyl in accordance with preliminary measurements. Complete characterization of the contamination on building materials requires a detailed sampling investigation. The degree and character of contamination were assessed by analyzing samples for b- and g-activity. Samples were analyzed for b-activity by measurements using the KRK-1 type radiometrically equipment. The activity of the samples was measured as "count per second, cps". The date of activity samples are given in Table I.

TABLE I

Gamma measurements were made using a Ge-detector spectrometry system. The gamma-spectrometric analyses of the two different samples are given in Table II.

TABLE II

Micromycetes

The micromycetes were obtained from the different sampling locations. The distribution sites are given in Table III. This species are common soil fungi of widespread distribution (23,24). More than two hundred strain of micromycetes belonging to 30 species and 10 genus were isolated. Simple techniques, such as dilution plating are commonly used for isolating micromycetes from the soil or from the surface layer of building materials. Ordinary bacteriological media has been used to prepare and cultivate the experimental suspension of the fungi.

TABLE III

Clay Minerals

The physico-chemical properties of the clay minerals including chemical composition, cation exchange capacity and swelling are shown in Table IV.

TABLE IV

EXPERIMENTAL

The objective of this investigation was to examine the effect of decontamination of the building materials by application of various species of micromycetes and DCC. Determining the most appropriate combination of the clays with electrolyte for prepared media was central problem of investigation applicable to the fungi growth. The matrix presented in Table V showed different testing combinations. All micromycetes have been studied under relatively similarly conditions.

TABLE V

An aqueous fungal solution (106 conidio/ml) was applied on the surface of each sample and covered the liquid clay medium.

Effect of Clay Medium

Experiments were made for determination of the relationships between the physicochemical characteristics of clay minerals, the ratio solid/liquid and fungi the growth activity. Initially on the surface of building samples micromycetes were added. They were covered with wet clays media. Gradually the water evaporates and that lead to a change of gel to pore texture. In this case the process of exhalation of the pore water is very slowly. Fast drying of the clay composition media is serious problem, because shrinkage of covering. For clays media the fast drying had damaged the coating surface. Swelling clay composition are dense and very hard in the dry state due to shrinkage stresses. This clay covering contain a maze of shrinkage cracks ranging from hairline to different widths and depths. Shrinkage cracks considerably influence the rate of growth of micromycetes. The shrinkage problem can be avoided when the coating is protected against water evaporation.

Factors such as the type of clay (montmorillonite or palygorskite), void ratio, method of preparation had significant effect on the water content. The sensitivity of tested micromycetes to various combinations medium show in Table VI. In this study expert values were used, because of difficulties to obtain cardinal number.

TABLE VI

As seen in Table VI all the species showed highest ability to grow in clay medium number 3-m-1. The species of Alternaria alternata, Mycelia sterilia were grown most efficiently. During their metabolism the fungus can produce acids able to degrade the contaminated surfaces of building materials. Fungal strains of Mycelia sterilia, that were producing gluconic and malic acid were found to increase more efficiently the decontamination factor.

Effect of Different Species State

The most important physical factors for micromycetes growth are: temperature, hydrogen ion concentration and various clay media characteristics: texture, structure, water and gaseous exchange. The experiments have been performed to determine the effect of different species state. The examined micromycetes were separated into two groups. The first group consisted of micromycetes which had been preliminary prepared to be in active form. The second group has been composed by dry spores. All micromycetes were examined at similar conditions using medium number 10-mp-05. The latent period also is affected by different factors, such as hydrogen ion concentration and temperature. The temperatures of 295 K and pH were common there. Clays medium were well buffered, with little variation in the pH media. For many fungi swelling of the spore accompanied the enchanted respiration. The rate of respiration is dependent on the water content. The presence of mixture montmorillonite with palygorskite in the medium allows better gas exchange for the fungi. In Table VII the effect of the growth ability micromycetes using expert values are illustrated.

TABLE VII

Cleanup

In order to determine the efficiency of the decontamination procedure the colonies micromycetes were removed using self-stripping DCC. This procedure involves covering of the contaminated surface with grown micromycetes. After drying it cracks, flakes, peels and fall off as a solid with entrained in the peelings loose surface material, removing the fungal layers. Loose contamination immobilized by clay coatings is in a form ready for collecting and disposal. Decontamination factor (DF) was defined as the original amount of radionuclide (A0) divided by the final amount (Af). The results of the test performed under the laboratory conditions showed that decontamination by using micromycetes decrease of the contamination level from 55% to 120% compare to only DCC treatments. This corresponds to a decontamination factor between of 1.8 to 2.7. The best results in terms of decontamination were obtained when the species of the culture "Arthrinium mortagnei" and "Cladosporium" were used.

CONCLUSION

It was concluded that using of micromycetes offers a potentially efficient method of removal of radionuclides from surface of building materials and is a technical feasible method. Micromycetes can function as effective immobilized and sorbent systems together with DCC. Mycelia of fungi are used as sorbent for retaining the products within their pores. The results of these studies confirm that clay minerals (montmorillonite) coating stimulated the growth fungi and activity of the fungi. Results of the studies presented here are preliminary.

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46-27

COMPARATIVE ANALYSIS OF DECONTAMINATION METHODS FOR THE REMEDIATION OF MIXED RESIDUE TANK SYSTEMS AT THE ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE

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ABSTRACT

The Rocky Flats Environmental Technology Site (RFETS) is currently in the process of closing a number of Mixed Residue Storage Tank Systems which are subject to the Colorado Hazardous Waste Act closure requirements. A study was conducted to identify and evaluate the available methods and technologies for removal of characteristic and/or listed hazardous constituents (as defined by 6 CCR 1007-3) from identified plutonium processing tanks. In addition to the listed and characteristic Resource Conservation and Recovery Act (RCRA) constituents, these process tanks are contaminated with residual radionuclides (primarily plutonium), which also were considered in the study.

The purpose of the study was to identify, and recommend, a course of action for performing the decontamination of the targeted tanks based on risks, compliance with specified requirements, and relative costs. The study involved detailed analysis of potential methods for decontamination and/or removal of the mixed waste residues. Once the methods had been identified, recommendations were to be made with respect to the most appropriate decontamination approach(es).

In an effort to provide thoroughly evaluated recommendations, a methodology was developed by which the study elements were conducted. The methodology is unique due to its multi-step, multi-faceted evaluation which continually addresses the relative risks associated with a given decontamination method as relate to; health and safety, technical feasibility, and cost and schedule.

BACKGROUND

The Rocky Flats Environmental Technology Site (RFETS; formerly the Rocky Flats Plant) is a Department of Energy (DOE) owned facility which is managed and operated by EG&G Rocky Flats, Inc. (EG&G). EG&G is preparing for systematic decontamination and decommissioning (D&D) of process tanks to remove hazardous and radioactive constituents. The Decontamination Methods Study was conducted in support of this effort in order to identify a recommended course of action for decontamination which will result in the safe and cost-effective disposal of the identified tanks in compliance with all applicable regulations.

ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE BACKGROUND

The Rocky Flats Environmental Technology Site (RFETS) is located in northern Jefferson County, approximately 16 air miles northwest of Denver, Colorado. The site area is approximately 11 square miles. The primary production mission of the RFETS, prior to shutdown, was the manufacture of nuclear weapons components fabricated from plutonium and other metals. Since January 1992, the mission of RFETS has been changed from involvement in national defense programs to environmental remediation and cleanup. The transition of buildings, systems, and components of RFETS from Defense Programs (DP) to Environmental Management (EM), which includes D&D activities, began in 1993.

As an element of the Department of Energy's Decontamination and Decommissioning objectives, the RFETS is tasked with developing D&D programs, including budgets and estimates, for the remediation of onsite facilities and areas. As the primary site support contractor to the DOE Rocky Flats Field Office (DOE-RFFO), EG&G is directly involved in the development and implementation of D&D projects for RFETS. The transition of RFETS surplus facilities is addressed in the EG&G Rocky Flats Mission Transition Program Management Plan, dated March 11, 1993.

The RFETS is currently in a cold shut-down (transition) mode and is developing D&D program processes, procedures, and protocol. No full-scale field activities for D&D programs presently are underway. The management structure to accomplish the transition and implementation of D&D programs and policies is developmental; specific programs elements are under internal review at this time. The management structure to be implemented is being developed to organize, staff, direct, and control activities necessary for managing and implementing D&D program requirements and to assist in the site-wide transition. Subsequently, the RFETS will be

transformed from its previous operational mode of nuclear weapon components production to a condition of deactivation, decontamination, and environmental restoration.

PURPOSE

The purpose of the Decontamination Methods Study was to identify and evaluate the available methods and technologies for removal of characteristic and/or listed hazardous constituents (as defined by 6 CCR 1007-3 part 261) from identified plutonium processing tanks. The specific RCRA hazardous constituents to be removed include spent solvents (e.g. carbon tetrachloride, 1,1,1-trichloroethane), cadmium, chromium, and lead. In addition to the listed and characteristic RCRA constituents, these process tanks are contaminated with residual radionuclides (primarily plutonium), which also were considered in the study.

The ultimate goal of the study was to recommend a course of action for performing the decontamination of the targeted tanks based on health, safety and environmental risks, compliance with specified requirements, and relative costs. The study involved detailed analysis of potential methods for decontamination and/or removal of the mixed waste residues. Once the methods had been identified, recommendations were to be made with respect to the most appropriate decontamination approach(es).

SCOPE

The Decontamination Methods Study addressed the decontamination and/or closure of 228 tanks located in Buildings 371, 771 and 777. This study was limited to identification of decontamination methods which may be applied to the process tanks housed in the identified buildings which have been designated as `mixed residue' storage tanks. Residue, by definition is a waste which contains plutonium in sufficient quantities to warrant treatment for material recovery. Mixed residue is defined as residue mixed with hazardous waste or hazardous waste constituents that are regulated under RCRA.

Since these mixed residue tanks contain constituents that are regulated under RCRA, the tanks are subject to the closure requirements of RCRA. Based on these requirements, it was necessary for the analyses performed in the Decontamination Methods Study to consider whether the candidate decontamination methods would satisfy the RCRA closure requirements.

BACKGROUND FOR THE STUDY

The mixed residue tanks under consideration are of three types: Raschig ring-filled tanks (or Raschig ring tanks); annular tanks; and pencil tanks. The Raschig ring-filled tanks are the largest volume tanks of the three types. These tanks provided criticality safety for processing of fissile materials by filling the tank volume space with Raschig rings, which are hollow, cylindrically-shaped borosilicate glass rings. The volume taken up by the Raschig rings within the tank space typically is approximately 30 percent of the total available tank volume. Annular tanks are tanks of double wall construction, which creates an annulus, or uniform spacing between the inner and outer tank walls. In these tanks, the process material is contained in the annulus, which provides criticality control. The annulus is created by welding the tank walls together at a typical spacing of 1.5 to 2 inches. Pencil tanks are so designated because these are long, thin tanks of limited capacity. The small diameter (typically four inches) of these tanks provides the required criticality safety.

These mixed residue storage tanks contain (or have contained) both "characteristic" and "listed" types of hazardous wastes. Characteristic hazardous waste are those wastes which exhibit the characteristic of ignitability, corrosivity, reactivity and/or toxicity. Listed wastes are so designated if the particular waste form is included in the listings in 6 CCR 1007-3 part 261.33. The only waste forms processed in the tanks addressed within this study were "characteristic" or "F"-listed hazardous wastes. Of the 228 tanks within the scope of the study, 210 tanks contained "characteristic" wastes, while 18 tanks contained "F-listed" wastes. All of the tanks which managed "characteristic" wastes are located in Buildings 371 and 771 while those that managed "F-listed" wastes are in Building 777.

The following characterizes the tanks in Buildings 371, 771, and 777 to be decontaminated.

1. There are 228 tanks to be decontaminated. Of these, there are 107 Raschig Ring tanks; 93 Pencil tanks; and 28 Annular tanks.
2. The contaminants include: plutonium, chromium, cadmium, lead, carbon tetrachloride and freon.

3. RCRA hazardous waste involve both listed and characteristic wastes. Tanks which stored listed waste are only in Building 777 and include six Raschig Ring tanks and 12 Pencil Tanks. The remaining tanks only stored characteristic wastes.
4. The contaminants are anticipated to be found as passivation films covering the wetted surfaces. Scale deposits are expected to be found at the "mean" gas-liquid interface.
5. The majority of the tanks are of stainless steel construction; some are kynar-lined. These tanks were used in highly corrosive nitric acid based systems. A limited few of the tanks are of fiberglass reinforced plastic or carbon steel construction. These tanks were used in caustic systems such as off-gas treatment systems.
6. Tank sizes vary for each type of tank. Raschig Ring tanks range in capacity the largest of which is 7,000 liters. The largest Annular tank can store approximately 525 liters. Pencil tanks are typically small volume tanks the largest having a capacity of approximately 100 liters.
7. Buildings 371 and 771 contain a sequence of typical continuous flow chemical processing systems involving plutonium recovery. Building 777 is a support facility involving Research and Development. Beyond filtering waste organics to remove particulate plutonium, no treatment was performed in Building 777. The tanks in Building 777 were used solely for waste storage. However, the chemical processes in Buildings 371 and 771 involved treatment and storage tanks which were manifolded together in complex series/parallel arrangements. These complex arrangements allowed for flexibility in operating the chemical processes.
8. All of the many interconnections are valved while only a few vacuum pumps were used to transfer solutions.
9. Non-destructive assay has shown that the plutonium hold-up in the tanks range from zero to 773 grams. However, there are only 26 tanks with plutonium hold-up in excess of 100 grams. Since the majority of the tanks have less than 10 grams of plutonium hold-up, it is postulated that the small amounts are present as wall films. It is anticipated that those tanks having hold-up greater than 10 grams contain a large amount of sludge which could not be removed by conventional tank draining methods. In order to achieve effective decontamination, these residual sludges must be removed from the tanks.

These characteristics constantly must be considered when attempting to select the more appropriate decontamination methods to be applied to these tanks.

Selecting decontamination methods involves a system approach, in that it requires considering all items necessary to perform the task rather than only the costs associated with equipment; reagents; secondary waste treatment; disposal; etc. This implies conceptualizing all of the facilities required to support a given decontamination method to include modifications to existing facilities.

STUDY WORK PROCESS

The study work process consisted of the following:

Literature Search

Perform a literature search and obtain relevant information on existing and developmental technologies potentially capable of removal of the mixed residues and contamination from the tanks;

Identify and interview key personnel at RFETS and at other selected locations within the DOE Complex to identify decontamination methods that have been utilized on similar tank systems;

Use the literature search and interviews to identify required equipment, radiological containment, and any other special requirements.

Ensure compatibility of the identified decontamination methods with DOE Orders 5480.23, 5480.24, 5480.25, and American National Standards (ANS) Section 8 (for nuclear criticality) and;

Verify the capability of the identified decontamination methods to satisfy the RCRA Closure requirements;

Evaluation of Options

Develop criteria for the evaluation of the decontamination methods, including handling techniques, radiological contamination control techniques (for both personnel and the surrounding area), throughput estimates, waste generation estimates (separate by liquids and solids, RCRA hazardous and non-hazardous), equipment required, feasibility of methods for plutonium processing, and cost estimates;

Specify evaluation criteria that includes RFETS Tank Closure Performance Standards, and the requirements for radionuclide handling and removal;

Perform separate evaluation of the identified decontamination methods for Raschig ring tanks, annular tanks, and pencil tanks;

Perform separate evaluation of the identified decontamination methods for large versus small tanks;

Include an assessment and listing of required equipment, engineering, and construction activities required to initiate the decontamination processing;

Prepare a conceptual cost estimate;

Prepare an estimate of liquid generation and waste type to be anticipated during decontamination operations; and

Identify vendors and/or consultants with appropriate expertise and experience in the evaluated technologies; including rationale for evaluation of these companies (e.g., reason for listing the company, previous work efforts, qualifications of staff, etc.).

Recommendation of Optimal Methods

Provide a formal report (with supporting documentation) on the recommended decontamination processes for the tank systems.

STUDY APPROACH AND METHODOLOGY

The basis for the study was comprised of two elements. The first involved establishing the process and basic methodology by which the study was to be conducted and the options were to be evaluated. The second element was the implementation of the process, which involved development of assumptions, selection and screening of technologies, and ranking of favorable technologies. Decisions and evaluative information were introduced throughout the process to establish, develop, and support the baseline of the study. These sequential elements of the methodology and process, referred to as the Process Ranking System, are depicted in Fig. 1.

Fig. 1.

Basic Assumptions

Once the methodology for conduct of the study was established, and prior to ranking the candidate decontamination methods, it was necessary to define the assumptions that represented the fundamental basis of the study. These assumptions were used as the general parameters for researching the literature and information sources for potential decontamination technologies. Those technologies that in no way met the general intent of these assumptions were eliminated as not applicable to this Study.

Identification of Regulatory Requirements

The next step of the ranking process involved identifying RCRA Regulatory Requirements associated with "closing" the mixed residue storage tanks. As previously discussed, the tanks under investigation were utilized for processing solutions containing plutonium and other hazardous constituents. These nonradiological hazardous constituents included both "characteristic" and "F-Listed" chemical forms. The hazardous waste constituents of concern were identified as cadmium, chromium, lead, carbon tetrachloride, freon and 1,1,1-trichloroethane. An evaluation of RCRA regulatory requirements was conducted to establish the decontamination goals with respect to these waste constituents.

Field/Facility Investigations

As a part of the preliminary definition of the existing conditions, field visits were conducted. These visits were conducted in order to obtain an accurate assessment of the actual working conditions for the project, and of the physical condition of the tanks and associated piping. Buildings which contain approximately 70% of the tanks within the scope of this study were visited and considered representative of all of the tanks that would be decontaminated.

Identification of Potential Technologies

Once the baseline criteria had been established with regard to basic assumptions and regulatory requirements, a comprehensive investigation was conducted to identify potentially applicable technologies that could be evaluated in more detail for utility in decontamination of the mixed residue tanks. The comprehensive investigation included: use of on-line computer-access data bases; library/literature searches; interviews with professional personnel knowledgeable in relevant decontamination areas; and contacts with vendor/supplier companies that manufacture, install and/or operate identified decontamination technologies.

Clarification of Assumptions

As the study progressed, interactions with site personnel and representatives from

the organization directing the study resulted in refinement and clarification of original assumptions and working parameters. These clarifying assumptions involved issues associated with the levels of contamination and space limitations as relate to skid-mounted decontamination equipment.

Applicability Check/Screen

The next analytical step in the process consisted of identifying the available technologies that may be applicable to the decontamination scenario under evaluation, and performing an applicability check. This preliminary screen was performed to eliminate technologies based upon inherent, perhaps obvious, shortcomings. These shortcomings included issues relating to complying with existing permit conditions, or the likelihood of obtaining permits, to immediately recognizable risks posed to the worker, the public or the environment.

This applicability check was a basic `yes/no' validation conducted to verify that identified technologies were worthy of further consideration, and were not so unapplicable or esoteric as to be not feasible in possible decontamination scenarios. This screening was conducted on the basis of professional judgement of a licensed professional engineer responsible for corporate decontamination and decommissioning technology analysis, development, and transfer.

The initial technology searches resulted in the identification of over 100 various technologies that had some level of potential application to decontamination of the mixed residue tanks. The applicability check/screen winnowed this listing to 59 decontamination technologies warranting further evaluation.

Technology Groupings

In order to adapt the remaining technologies into a more manageable system, groupings were selected in order to assign each technology to a category of similar technologies. Upon completion of the applicability screen, candidate technologies were grouped into categories of decontamination methods, which included physical methods, chemical methods, process enhancements, and cutting techniques.

Preliminary Screening

The preliminary screening process utilized nine prepared "go/no go" (or "yes/no") questions to eliminate technologies that had the potential to manage the waste forms in question, but that did not meet established screening criteria selection for the study. The preliminary screening process narrowed the list of candidate technologies from 59 to 47.

Numerical Ranking

Once the noncompliant methods were eliminated by the screening process, the remaining candidate decontamination methods were subjected to a semi-qualitative ranking. This process involves the second step of the Process Options Ranking System. The semi-qualitative ranking process involved evaluating each decontamination method in specific detail. Details included evaluation of the technology with respect to: the risks posed to health, safety and the environment; technical feasibility; and cost/schedule. Each of the evaluative categories were assigned numerical rankings relative to the importance of the factor to this study. The numerical ranking process involved the most rigorous evaluation of the potential decontamination methods in that numerous areas of risk associated with health and safety, technical feasibility, and cost and schedule were considered. This numerical ranking, coupled with the cluster evaluation discussed below resulted in identifying the decontamination methods presenting the least risks under consideration.

Cluster Evaluation

The cluster evaluation provided a method to select the most feasible groupings of technologies. In this technique, a `cut-off' point was assigned to the weighted, ranked technologies. This cut-off point was chosen on the basis of the statistical `breakpoints' at which clusters of technologies were situated.

Comparison Model

In order to provide a benchmark against which to compare and contrast remaining candidate technologies, a theoretical comparison scenario in the form of a `model' problem was developed. The model problem was required in order to allow expedient and orderly examination of the remaining technologies, which would have been otherwise difficult due to the fact there were three tank types and two RCRA waste types to be assessed.

The model problem allowed preferential ranking of each of the candidate decontamination methods. Without the model problem, each method would have had to be evaluated under six scenarios to accommodate each tank and waste type. This

divergent number of analyses would have been unwieldy and nearly impossible to compare on an equivalent basis

Comparative Analysis

With the technologies identified that had the highest probability of success for the study scenarios, a comparative analysis was conducted to evaluate the positive and negative features among the remaining technologies. This [somewhat qualitative] comparative analysis required a consensus of knowledgeable engineering and technological experts. The result of this evaluation was the identification of the technologies that were the most feasible for the given conditions and requirements associated with the study.

Configuration of Technology Systems

Another consideration for the study was that the combination of technologies into a treatment systems might perform superior to single technologies. For this reason, an evaluation was conducted to identify possible sequences of the preferred technologies that would result in the optimum performance criteria.

Identification of Recommended Technologies

The evaluation of individual technologies and consideration of technology systems allowed the selection of optimal treatment schemes, and the identification of the technologies that could be recommended on the basis of this study.

CONCLUSIONS

Application of the methodology presented above allows for the determination of technological approaches presenting the lowest risk approach to decontamination, or any activity, through the up-front consideration of all associated risks which may be encountered through the various approaches. This methodology is extremely versatile in that it allows the decision makers to assign the relative numerical rank weightings based upon criterion specific to their requirements. For instance, if health and safety is of utmost importance; the highest weighting can be imposed on this area with lower weight placed upon technical feasibility and/or cost and schedule. This approach is unique in that it has the ability to accommodate specific areas of relevance in the selection of preferred approaches to given problems since, rather than simply choose a pre-determined approach; the most appropriate approach can be selected upon considering all associated risks given their respective risk weightings.

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SCHEDULING AND IMPLEMENTATION OF FIELD ACTIVITIES FOR DECONTAMINATION AND DECOMMISSIONING PROJECTS AT DEPARTMENT OF ENERGY NUCLEAR FACILITIES

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ABSTRACT

The mission at the U.S. Department of Energy (DOE) nuclear weapons facilities has changed from weapons production to environmental cleanup and restoration. A number of the former production facilities/buildings are contaminated with or contain radiological and/or hazardous substances. Facilities targeted for Decontamination and Decommissioning (D&D) include uranium enrichment facilities; plutonium production facilities; metal fabrication and machining shops; laboratories; waste processing facilities; and metal recovery facilities.

In order to properly plan and implement D&D field activities at facilities within the DOE complex, it is beneficial to develop a process and precedence network/logic diagram of the generic field activities which would be performed during a facility D&D project. The sequences of and relationships between these activities are governed by criteria such as: technical feasibility; safety and health protection; minimization of contamination spread; accessibility; environmental protection and regulatory acceptability; site scheduling and operational constraints; and ultimate disposition.

Decontamination activities are to be performed on individual pieces of equipment, such as tanks, gloveboxes, entire rooms, and systems which are no longer useful (in their present state) and have been transitioned for D&D. Materials generated include asbestos; PCB's; concrete; steel; copper wire; glass; plastics; fabrics; and contaminated solutions. Some of these materials are uncontaminated and may be

selected for reuse, recycle, disposal in a sanitary landfill, or free-release/reuse. Some of the equipment and materials will be decontaminated to low-level waste or low-level mixed waste for appropriate storage or disposal. Ultimate disposition of facilities depends upon whether or not a future use has been identified for the facility or equipment following decontamination, and the success or feasibility of performing the decontamination.

Establishing a D&D program necessitates the evaluation of the integral parts encountered when conducting field activities for a typical D&D project. Past D&D projects at various DOE and U.S. Nuclear Regulatory Commission (NRC) regulated sites were evaluated to determine the critical elements of D&D field activity processes. The proposed, generic D&D network encompasses more than 100 activities, beginning with the issuance of an approved readiness review report. These activities include both decontamination and disassembly/dismantlement, and are time-phased into the major D&D elements/categories of Internal (equipment, utilities, etc.); Structural; Subgrade; and Closeout and Verification.

By utilizing facility-specific parameters such as: materials and equipment types; the type, location, and degree of contamination; and the selected ultimate disposition, facility-specific schedule networks can be prepared. By applying labor loading estimates to the activities, projected costs to perform the field activities are obtained. This process was developed as a PrimaVera schedule network of generic field activities to be performed during a facility D&D.

BACKGROUND

The U.S. Department of Energy (DOE) has focused internal efforts for the decontamination and decommissioning (D&D) of surplus facilities within the DOE office of Environmental Restoration and Waste Management (ERWM). This Office is challenged to develop a total system within the DOE complex to resolve issues associated with site and facility cleanup activities. The scope of the D&D program is to address the removal of contamination associated with buildings and equipment in order to support programs for continued use, reuse for nonradioactive missions, or release for unrestricted use. Specifically, the D&D program will consist of: 1) Recycling reusable raw materials and products; 2) Decontaminating and recycling recoverable internal building components and equipment; 3) Implementing an asbestos abatement program; 4) Managing, treating, and packing non-recoverable materials, equipment, and other building components for disposal; 5) Refurbishing buildings for future unrestricted use; 6) Dismantling surplus buildings; and 7) Properly managing the waste generated by D&D operations to limit future liabilities. Although all of these elements were considered, this poster focuses heavily on the activities required to accomplish items (2) and (6).

The basic requirements for D&D programs for radioactively contaminated facilities are contained in DOE Order 5820.2A, Chapter V; the Decommissioning Handbook (March 1994); and the Decontamination and Decommissioning Guidance Document (January 1994). These documents require that DOE manage radioactively contaminated facilities in a safe, cost-effective manner to assure that release of and exposure to radioactivity and other hazardous materials comply with Federal and State standards. In addition, facilities, equipment, and valuable materials are to be recovered and reused when practical.

Consistent with these documents, D&D efforts must:

Eliminate or reduce to acceptable levels the hazards and risks associated with facility/equipment contamination to ensure worker protection during the dismantling of surplus facilities;

Remove building components in phases to minimize the potential for spreading contamination and to ensure proper management of waste materials;

Eliminate one or more of the contaminant hazards to allow the facility, building, and/or equipment to be subject to less stringent controls and to be reused for other purposes (as much as possible)

Figure 1 presents the following major steps which are performed as part of a site D&D project:

Fig. 1.

D&D PROGRAM AND PROJECT PLANNING

Each DOE field organization must prepare and maintain a complete list of contaminated facilities (both operational and excess). In order to implement these requirements for facility inventory and to establish the D&D program goals, the responsible DOE field office must conduct programmatic actions, including:

Establishing and maintaining a master D&D schedule;
Transferring landlord responsibility from operations to D&D programs;
Developing waste management strategies;
Evaluating existing D&D technologies and developing specific applications;
Establishing free-release criteria; and
Identifying ultimate material disposition.

The surplus facility inventory will be used for establishing the phasing of overall D&D activities. The ultimate land use of the facility, utilization of buildings to support D&D operations or other restoration programs, and the need to maintain utility services for on-going plant operations need to be considered when developing the master schedule.

Following the establishment of program components, the development of plans for implementation of individual subprojects can commence. These efforts consist of plans preparation and completion of critical pre-D&D Operations activities, including the Subproject Management Plan; Facility Operation History; Baseline Characterization Plan; and Facility Characterization.

Next, D&D Engineering and Implementation Planning is performed. These efforts include the preparation of the Decommissioning Subproject Plan for the facility; Regulatory Integration; and Procedures Preparation. These activities form the basis for completion of planning activities prior to the initiation of D&D operations and field activities.

Specific procedures must be prepared that are specific to the facility and actions in question. These procedures could include:

Procedures for residual waste removal activities for wastes contained in process equipment, tanks, piping, sumps, drains, and other locations;
Procedures for implementation of decontamination techniques including contact and non-contact methods for equipment, building structures and surfaces, piping, tanks, and other ancillary equipment;
Procedures for decontamination verification sampling for equipment and building surfaces, rinsates, and decontamination fluids for both radioactive and non radioactive contaminants;
Procedures for dismantlement and size reduction procedures for building structures, vessels, piping, utilities, and other ancillary equipment;
Procedures for waste sampling and analysis for treatment, storage, and/or disposal determinations;
Radiological control procedures to be employed during decontamination and decommissioning;
Waste packaging and transportation procedures; and
procedures for verification sampling and analysis following D&D.

As soon as final engineering and implementation planning has been completed, procurement of items and services that have been identified as required for the D&D efforts should be initiated. This is especially critical in regards to long-lead procurement items.

REMEDIATION/D&D OPERATIONS

D&D operations include decontamination, dismantlement, disassembly, packaging, storage, and disposal of facility structures, components, systems, and equipment. These activities should be conducted in compliance with the Decommissioning Subproject Plan, which is established in the D&D Engineering and Implementation Planning project phase.

Decontamination activities are to be performed on individual pieces of equipment, such as tanks, gloveboxes, entire rooms, and systems which are no longer useful (in their present state) and have been transitioned for D&D. Materials generated from these D&D operations include asbestos; PCB's; concrete; steel; copper wire; glass; plastics; fabrics; and contaminated solutions. Some of these materials are uncontaminated and may be selected for reuse, recycle, disposal in a sanitary landfill, or free-release/reuse. Some of the equipment and materials will be decontaminated to low-level waste or low-level mixed waste for appropriate storage or disposal. Ultimate disposition of facilities depends upon whether or not a future use has been identified for the facility or equipment following decontamination, and the success or feasibility of performing the decontamination.

Past D&D projects at various DOE and U.S. Nuclear Regulatory Commission (NRC) regulated sites were evaluated to determine the critical elements of D&D field activity processes.

Major elements/typical activities to be considered for D&D operations which follow the issuance of the Readiness Review Report include:
1. Internal Activities
Decontamination Activities
- Chemical Cleaning
- Systems Flushing
- Non-Chemical Cleaning
- Partial Removal of Components
Removal of Radioactive Equipment
- Decontamination to the extent possible
- Isolation of the contaminated system and cutting pipe
- Sampling and characterization of radioactivity
- Sampling and characterization of RCRA/CERCLA contaminants
- Size/Volume Reduction
- Removal and packaging
- Onsite transportation
- Offsite transportation
Removal of Nonradioactive Equipment
- Segmentation and removal of large vessels and components
- Removal of other equipment
- Sampling and characterization of RCRA/CERCLA contaminants
- Verification of uncontaminated status
- Size/volume reduction
- Packaging and transportation
- Salvage or disposal
Major Systems Removal
- Segmentation and removal of large vessels and components
- Removal of other equipment
- Verification of uncontaminated status
- Size/volume reduction
- Packaging and transportation
- Salvage or disposal
Removal of Power and Control Systems
- Decontamination
- Operations interfaces
- Switch-over to construction/auxiliary power
- Verification of uncontaminated status
- Salvage or disposal
2. Structural Activities
Decontamination Activities
- Chemical Cleaning
- Systems Flushing
- Non-Chemical Cleaning
- Partial Removal of Components
Removal of Contaminated Concrete and Porous Materials
- Walls, floors, and structures
- Removal controls
- Removal methods
Demolition and Removal of Structures
- Removal of contamination
- Removal techniques
- Verification of uncontaminated status
- Salvage or disposal
3. Subgrade Activities
Treatment of Contaminated Soils
- Excavation
- Treatment processes (chemical/non-chemical treatment; flushing)
- Sampling and characterization of contamination
- Verification of clean status
- Final disposition
Waste handling activities are included in the requirements for Internal, Structural, and Subgrade activities. Considerations here include:
Liquid Radwaste Processing

- Present capacity and capability
- Additional services needed/required
- Solidification

  Solid Radwaste Handling, Packaging and Disposal

- Low specific activity (LSA) containers and casks
- Packaging and transportation
- Disposal
- Interim storage/staging

  Radioactive/Hazardous Mixed Waste

- Stabilization
- Packaging and transportation
- Disposal

  Nonradioactive Hazardous Waste Handling and Disposal

- Transportation
- Disposal or stabilization

  Clean Waste Handling and Disposal

- Identification of Laydown Areas
- Traffic management of waste site
- Local landfill site

POST-DECOMMISSIONING ACTIVITIES

After the completion of remedial/removal activities and related D&D operations, Project Closeout and Verification will be performed. This includes the preparation of a final radiological and chemical survey (or an independent verification survey) report.

Generic activities to be performed during the Closeout and Verification phase include:

  Closeout Documents

- Final Report
- Contracts
- Financial Records
- Safety Analysis Report
- Record of Completion
- Public Notices
- Project Data Package
- Long-Term Surveillance and Maintenance Plan

  Verification

- Independent Verification Documentation
- Approved Closeout

DEVELOPMENT OF THE GENERIC SCHEDULE

The generic activities discussed above were arranged into a time-phased integrated schedule/network of elements, after identification of dependency data and sequencing of events. The network, developed using PrimaVera Project Planner, is provided on the exhibited poster.

The D&D network starts with the issuance of an approved readiness review report and ends with the verified D&D closeout. Activities are time-phased into the major D&D elements/categories of Internal (equipment, utilities, etc.); Structural; Subgrade; and Closeout and Verification.

From this generic network, decisions may be made and facility-specific schedule networks can be prepared. The application of labor loading estimates for the activities in the facility-specific schedule network yields the projected costs to perform the field activities at the particular facility.

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LONG-TERM LYSIMETER FIELD TESTING OF LOW-LEVEL RADIOACTIVE WASTE FORMS

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ABSTRACT

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program is obtaining information on the performance of radioactive waste forms. Ion-exchange resins from a nuclear power station were solidified into waste forms using Portland cement and vinyl ester-styrene. These waste forms are being tested to obtain information on survivability of waste forms in a disposal environment and to develop a low-level waste data base. Radionuclide releases from those waste forms during the first 9 years of sampling have been evaluated. Also, upward migration of radionuclides was recently discovered. Lastly, lysimeter data are applied to a performance assessment source term model.

INTRODUCTION

The purpose of this paper is to present the experimental results of the two instrumented lysimeter arrays over 9 years of operation and to compare these soil leaching results to bench leaching results from similar waste forms and to predictions of the Disposal Unit Source Term (DUST) code. While results of this program have been presented at previous WM meetings, this paper gives an update of the study, which includes further discussion of the upward migration of cesium-137 and cesium-134 in one unit of the Oak Ridge National Laboratory array. Results of

analysis of sand cores from that lysimeter are presented and discussed. Also, cumulative radionuclide release from the waste forms to the leachate water passing through the soil columns is presented and compared to bench leach test cumulative fractional releases from similar waste forms.

Concern over the practices associated with the disposal of low-level radioactive waste has resulted in a very real need to obtain accurate data on the long-term field performance of these wastes. The U.S. Nuclear Regulatory Commission (NRC) has enacted regulations that link low-level radioactive waste acceptance criteria to the long-term satisfactory performance of the disposal facility. Under Code of Federal Regulations (CFR) 10, Part 61, "Licensing Requirements for Land Disposal of Radioactive Wastes" (1), commercially generated low-level radioactive waste is classified as Class A, B, or C. Wastes classified as either Class B or Class C must be stabilized for a minimum of 300 years.

To verify the 300-year stability of waste forms, the NRC specifies the use of short-term standardized tests with the intention that such tests would provide information relevant to near-surface disposal performance objectives. Those tests, which were initially published in the NRC Branch "Technical Position on Waste Form" (2), and have been revised in Revision 1 of the Technical Position (3), continue to undergo critical reviews to determine their applicability to the 300-year stability requirements.

A central requirement for disposing low-level radioactive waste is the need for a detailed understanding of the waste form behavior. That is necessary because the radionuclide source is the driving force behind the site performance. A major requirement in any site licensing is the site performance assessment, which is used to evaluate whether or not a proposed disposal site will meet performance objectives. Assumptions regarding the performance of the buried waste form have a direct bearing on the outcome of the performance assessment.

The objective of the Field Lysimeter Investigations: Low-Level Waste Data Base Development Program is to compare the results of short-term laboratory leach testing, performed earlier by the INEL, with actual leaching in the field. Also, the waste forms are being field-tested to develop a low-level radioactive waste leach rate data base. This program, funded by the NRC, has been operating lysimeters for over 9 years to obtain information on the performance of radioactive waste forms in a disposal environment and to investigate waste form stability per requirements of 10 CFR 61. The experiment measures the releases of radionuclides and chemical species from the waste forms and the subsequent transport through soil columns to sampling locations within the lysimeters. This study was developed to field test waste forms composed of solidified ion-exchange resin materials from EPICOR-II* prefilters used in the cleanup of Unit 2 of the Three Mile Island Nuclear Power Station (4). Wastes used in the study are significant because they have high loadings of radionuclides and are comprised on ion-exchange media of the type used by the nuclear industry.

EXPERIMENT DESCRIPTION

Wastes used in the experiment include a mixture of highly loaded, nuclear-grade, synthetic, organic ion-exchange resins from EPICOR-II prefilter PF-7 and a mixture of organic-exchange resins and an inorganic zeolite from prefilter PF-24. Solidification agents employed to produce the 4.8 x 7.6-cm cylindrical waste forms used in the study were Portland Type I-II cement and DOW vinyl ester-styrene (VES). Seven of the waste forms were stacked end-to-end and inserted into each lysimeter to provide a 1-L volume. The PF-7 waste contained 89% of the radionuclide activity as cesium-137, while PF-24 contained 94% cesium-137. The PF-7 waste also contained 5% strontium-90, and PF-24 contained 1% strontium-90. There were also measurable amounts of cesium-134, cobalt-60, and antimony-125 found in those wastes. Details on waste-form descriptions, formulations, and technical position testing are given in Refs. (4) and (5). A listing of lysimeter waste form and fill material types are given in Ref. (6).

Ten lysimeters were used in this study: five at Oak Ridge National Laboratory (ORNL) in Tennessee and five at Argonne National Laboratory-East (ANL-E) in Illinois. The lysimeters were designed to be self-contained units that will be disposed at the termination of the study. Each lysimeter is a 0.91 3.12-m right-circular cylinder divided into an upper compartment that contains fill material, waste forms, and instrumentation, and a lower compartment for collecting leachate (Fig. 1). Four lysimeters at each site are filled with soil; a fifth, used as a control, is filled

with inert silica oxide sand. The lysimeters at ANL-E contain soil indigenous to the site, while the ORNL lysimeters contain soil taken from Savannah River Laboratory in South Carolina. The soil columns are 2.21 m deep.
Fig. 1.
Instrumentation in each lysimeter includes moisture cup soil-water samplers and soil moisture/temperature probes. The probes are connected to an onsite data acquisition system (DAS), which also collects data from a field meteorological station located at each site. Porous cup soil-water samplers and the leachate collection compartment comprise the water sampling components of each lysimeter (Fig. 1). Incoming precipitation moves downward through the soil column to the waste form, then on to cups 3 and 1, and finally to the leachate collector at the bottom. Moisture entering the soil at the edge of the lysimeter encounters cups 5, 4, or 2 as it moves downward. Samples of moisture are withdrawn from the cups and the collector. Radial movement of waste form releases are detected in cups 5, 4, and 2, while vertical release is observed by cups 3 and 1. Lysimeter design, installation, instrumentation, operation, and data acquisition are explained in Ref. (6).
Monitoring of the lysimeters at ANL-E and ORNL began with the collection of liquid samples in September 1985 (3 months from the time of placement) and has continued with sample collection on approximately a quarterly basis thereafter. Samples of liquids were taken from locations near the waste forms and from the leachate collectors to track the migration of radionuclides, primarily cesium-137. The water samples were analyzed for strontium-90 and gamma-producing nuclides. Each month, data stored on a cassette tape in the DAS were retrieved and translated into an IBM PC-compatible disk file. Soil moisture and temperature at three elevations in each lysimeter, along with a complete weather history, were recorded on a continuing basis by the DAS. Testing results are presented in Reference (7) as well as in this paper.
RESULTS AND DISCUSSION
Weather and Soil Data
Precipitation, air temperature, wind speed, and relative humidity were recorded continuously by the ANL-E and ORNL DAS during the experiment. The cumulative volume of leachate from the lysimeters since the initiation of field work, and examples of the lysimeter soil temperature and moisture data obtained at ANL-E and ORNL can be found in Reference (7). Data recorded in FY-94 indicate that the lysimeter soil columns at both sites have remained moist during the last reporting period.
Radionuclide Data
Figures 2 and 3 show examples of data on the cumulative amounts of nuclides as determined in water samples obtained from ANL-E and ORNL leachate collectors, respectively. Other data show that not all nuclides consistently appeared in the water obtained from the moisture cups or the leachate collectors. The nuclide that appeared with the most regularity at both sites was strontium-90. Table I contains a comparison of the cumulative fractional release of strontium-90 and cesium-137 found in the moisture cups and leachate water. Consistent significant occurrences of strontium-90 have been observed in all the number 3 cups (22.4 cm below the waste form in the soil column) at both ANL-E and ORNL.
Fig. 2.
Fig. 3.
TABLE I
At both ORNL and ANL-E, recovery of strontium-90 in cups 3 and the leachate collectors indicates a varied waste form performance (Table I). Recovery of strontium-90 in the ORNL cups is comparable for those lysimeters containing the cement waste forms and one of the two containing VES waste forms. However, the cups at ANL-E are recovering much more strontium-90 from the VES waste forms compared to the cement waste forms. These data indicate that releases from the cement waste forms are generally larger than from VES waste forms.
Movement of the nuclide into the leachate collectors of the inert, sand-filled control lysimeter 5 is much greater than that of the other lysimeters and thus provides evidence of the moderating effect of soil (versus the inert sand) in those lysimeters. During the past several years, leachate collector water from the control lysimeters at each site has contained amounts of strontium-90 at least an order of magnitude larger than releases from the soil lysimeters (Figs. 2 and 3) (7). The total strontium-90 being measured in the leachate collector waters remains somewhat inconsistent between the two sites (Table I). It is suspected that this represents a

difference in how the environment at the two sites affects the movement of strontium-90 being released from the waste forms. The higher release of strontium-90 from the ORNL control lysimeter waste form reflects the nearly 50% higher rainfall experienced at that site over that seen at ANL-E.

Gamma-producing nuclides continue to occur with regularity at both sites (Table I). However, only waste forms at ORNL are releasing detectable amounts of cesium-137 to the leachate waters (Table I). It is not possible to make a comparison of cesium-137 releases from cement and VES waste forms at this time due to the small releases.

Table II is a comparison of cumulative fractional releases from field testing EPICOR-II waste forms in lysimeters to releases from bench-leach-testing similar waste forms as reported in Reference (5,8). Releases observed in the lysimeter is at least four orders of magnitude less for strontium-90 and at least five orders of magnitude less for cesium-137 in soil. It is interesting to note that release of strontium-90 in the sand-filled lysimeter is only one or two orders of magnitude less than bench test results. At the present rate of increase (see Figs. 2 and 3), these cumulative fractional releases will be of similar magnitude in a couple of years.

TABLE II

Upward Migration of Radionuclides at ORNL

During previous samplings, the presence of both cesium-137 and strontium-90 were discovered at the surface of lysimeter ORNL-5, which is the sand-filled control. Radionuclide activity was first detected during a routine gamma survey of the lysimeter's surface in 1991. At that time, more activity was found near the center than at the edges. Core samples were obtained from the center of the lysimeter at depths from 0 to 2.5 cm and from 2.5 to 5 cm for analysis of cesium and strontium-90. Scientists detected 1,760 pCi cesium-137, 10 pCi cesium-134, and 0.5 pCi strontium-90 per gram of sand in the 0 to 2.5-cm core, and 306 pCi cesium-137, 3 pCi cesium-134, and 0.1 pCi strontium-90 in the 2.5 to 5-cm core material. These data showed that more radionuclides were at the surface, suggesting some type of an active deposition mechanism. There remained a question, however, concerning the source of the radionuclides. In August of 1992, samples were again taken from the lysimeter and analyzed for cesium-137 and cesium-134. The results were similar to the previous sampling, with 1,533 pCi cesium-137 and 6 pCi cesium-134 being found per gram in the surface, and 574 pCi cesium-137 and 2.4 pCi cesium-134 per gram in the 2.5 to 5-cm sample. A comparison was made between the ratio of cesium-137 and cesium-134 in the surface material and the ratio in the buried waste form. It was concluded that the contamination of cesium came from the waste form.

On January 31, 1994, two cores of sand 80 cm long were collected from lysimeter 5. One core was taken from the side of the lysimeter near the wall and has not yet been analyzed. The other core was removed from the center of the lysimeter directly above the buried waste form (located approximately 100 cm below the sand surface). This sand core was sectioned into 5-cm segments. Radiocesium and strontium activity were measured for each segment.

The analyses show that cesium-137 is present throughout the length of the core (Table III). There are three peaks seen in the cesium content: one at 30 to 35 cm, a large peak at 45 to 50 cm, and a smaller peak at 70 to 75 cm. These peaks may be indicative of some sort of periodic movement of the cesium, but further laboratory study is necessary before the cause of this movement can be determined.

TABLE III

During the sectioning of the core, it was noticed that there was a fine plant root present throughout the depth of the core. The root material was extracted from each segment and counted (Table III). Cesium-137 activity is associated with the root, and the peaks in the root data occur at the same depths as do the peaks in the sand activity. It can be seen that there are higher concentrations of cesium-137 associated with the root than with the sand. Sand from the deepest three segments was analyzed: each whole segment was analyzed, and two subsamples of each segment were analyzed. Segment 2 (Table III) has a fairly wide range of activities between the whole segment and the two subsamples, suggesting that the activity in the sand is not evenly distributed. This could be a result of the root being involved in the transport process.

Strontium-90 analysis results show that there is significant strontium throughout the entire depth of the core (Table III). Peaks occur in the distribution at the same depths as for cesium in both the sand and roots. This suggests that the same

mechanism may be involved for transporting strontium upward as cesium. Strontium and cesium behave very differently chemically, suggesting that the process of migration is more physical than chemical, such as evapotranspiration enhanced by the presence of the root. The fact that the sand has a very low cation-exchange capacity is probably the reason that the physical aspect of migration is so evident. Further analysis of soil cores is planned. ORNL scientists performed gamma-radiation surveys of gravimetric soil cores collected annually, and determined that soil-filled lysimeters are not experiencing upward migration of radionuclides.

SOURCE TERM MODELING OF LYSIMETER RELEASES

The Disposal Unit Source Term (DUST) (9) code was used to model the release of the radionuclides cesium-137 and strontium-90 from the lysimeter waste forms. DUST is a one-dimensional code that accounts for container performance and waste form leaching (including diffusion-controlled release). Transport can be modeled through finite differences or by a multi-cell mixing cascade approach. The finite difference method was used in the simulations reported in this paper because it is more general than the mixing cell approach and permits modeling of dispersive transport. Use of these data in the DUST code was examined in detail in a paper presented at WM '93 (10).

The releases of cesium-137 and strontium-90 from Portland type I-II cement located in the inert, sand-filled lysimeters 5 at ORNL and ANL-E were chosen because releases from other lysimeters were substantially lower; therefore, the data were not yet sufficient to model. At ANL, lysimeter 5 contained resin waste from PF-7 solidified in Portland type I-II cement; at ORNL, lysimeter 5 contained resin waste from PF-24, which was also solidified in cement (Table I). Diffusion coefficient values measured in laboratory testing of these waste forms were 9.6E-10 cm2/s for strontium-90 in Portland cement (8) and 5E-11 cm2/s for cesium-137 in Portland cement (5). The Darcy velocities ranged from 2.59E-6 cm/s at ANL-E to 3.6E-6 cm/s at ORNL (7). The soil bulk density values were 1.55 g/cm3 at ANL-E and 1.60 g/cm3 at ORNL (6). Moisture content values were calculated using the effective soil porosity and the fraction of saturation values found in Reference (7). The dispersivity and retardation (Kd) coefficients have not been measured for strontium-90 or cesium-137 in this sand; therefore, they were estimated based on data in References (11) and (12) and by fitting the model predictions to the data. The cumulative activity collected in the lysimeter leachate water over the first 9 years of operation of the experiment, which was used to make comparisons to the DUST code predictions, represented cumulative fractional releases of about 0.0012 and 0.00017 of the strontium-90 in lysimeter 5 at ORNL and ANL-E, respectively (Table I).

As shown in Fig. 4, the actual data for strontium-90 from ORNL lysimeter 5 are compared with the DUST code predictions of releases in case 1 using zero dispersive flux BC, Kd = 24, and dispersivity = 8.5. Also shown are predicted releases of case 2 using zero concentration flux BC, Kd = 10, and dispersivity = 0.6. The measured waste form diffusion coefficient of 9.6E-10 cm2/s was used. The predicted releases show a very good fit to the actual data after initial stabilization of the test data. Case 2 releases less activity over 4 years than case 1; however, over 20 years, case 2 will have released 33% of the total strontium-90 inventory, whereas case 1 will have released 3.3% of the total strontium-90 inventory. The lack of measured dispersivity and retardation coefficient necessitates obtaining cumulative release data over a longer term.

Fig. 4.

CONCLUSIONS

The radionuclide that has appeared with most regularity at both sites is strontium-90, although cesium-137 is observed regularly in the leachate of all ORNL lysimeters. A comparison of total strontium-90 found in leachate of the control lysimeters shows that environmental effects have resulted in a much higher release at ORNL. The data indicate that Portland cement and VES waste forms have comparable releases of strontium-90.

Cesium-137, cesium-134, and strontium-90 are present throughout the upper 80 cm of the inert sand in ORNL lysimeter 5 directly above the waste form. The ratio of cesium-137/cesium-134 indicates that the radionuclides are from the buried waste form and not from an outside source and were transported vertically upward by some physical mechanism enhanced by the presence of a plant root.

DUST-predicted cumulative release of strontium-90 from ORNL lysimeter 5, which was plotted over the first 9 years of data collection, show a reasonable fit to the field data for Kd = 24. The accuracy of the DUST modeling study was limited,

however, by the lack of measured soil dispersivity and retardation coefficients. Further testing will provide larger cumulative releases to better model the release patterns from the lysimeters.
Data provided by these lysimeter experiments have been shown to be useful in computing many parameters used as input to performance assessment codes. The utility of this reliable source of data will be demonstrated through continued operation of the lysimeters with application of the results to source term models such as DUST. NRC planning recommends that this experiment be augmented by experiments containing solidified decontamination wastes and activated metals from commercial nuclear power stations.
ACKNOWLEDGMENTS
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PROCESS KNOWLEDGE EVALUATION AT LLNL*
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ABSTRACT
A system for evaluating the process knowledge provided by radioactive waste generators has recently been implemented at the Lawrence Livermore National Laboratory (LLNL). This paper describes the formal mechanism used to evaluate and track the processes that have the potential for generating low-level or transuranic waste and to determine whether the waste contains RCRA- or California-regulated hazardous components. The heart of the system is a set of standardized forms completed by the waste generator and an environmental analyst and reviewed by the waste certification official. The system allows all interested parties to look at an operation in detail, evaluate the input and output constituents, and determine if the waste stream meets the waste acceptance criteria. It also gives all parties a chance to evaluate waste minimization and waste segregation, as well as identify any additional training for the generator and any additional controls that may be

required for the specific process. The evaluation results in a documented list of pre-approved components for each waste stream that have already been evaluated for hazardous constituents and waste acceptance criteria, and thus in a more efficient final processing of the waste. The documentation for this evaluation system, the Process Knowledge Form, is kept on a 4th Dimension database and is reviewed and updated annually.

OBJECTIVE

In most cases the people who design and operate a process are the ones most knowledgeable about it and are the best source for evaluating whether it is being conducted safely and effectively. In the same light, these people are the ones best qualified to evaluate and characterize the materials that end up in the waste generated by the process. This is especially true with the small research and development projects that represent most of the activities that generate radioactive waste at LLNL The majority of low-level radioactive waste and over 90% of the transuranic waste is generated in the research laboratories of LLNL's Plutonium Facility. This facility is staffed by highly knowledgeable chemists, physicists, engineers, and technicians. It is only natural to rely on these experts as the primary information source when evaluating the waste stream.

This is not new. It is the way research has always been conducted. What is new, in today's climate, is a requirement to formalize this process in such a way as to assure regulators and ultimately the general public that the research is being conducted safely and in compliance with all established controls. In addition, this formalized process assures that oversight is documented in such a way that it can meet all of the requirements of a Quality Assurance program.

At LLNL we look at the control of waste generated by R & D work as a three-level process involving the following:

1. The researcher (waste generator)
2. An environmental analyst
3. The waste certification official.

The environmental analyst works closely with the researcher and is the one who has the specialized knowledge of hazardous materials as well as the environmental rules and regulations. The environmental analyst also is able to stand back from the work of the researcher and take an objective look at the process, evaluating input materials, material changes that may occur in the process, and what materials come out both as product and as waste.

The waste certification official is the one ultimately responsible for certifying for LLNL that the waste is properly characterized, meets all waste acceptance criteria, and is being disposed of safely.♀

PROCESS KNOWLEDGE EVALUATION FORM

Section I - Process Information Provided by the Researcher (i.e. Waste Generator)

Fig. 1.

Entries 1, 2, 3. Location identification - For transuranic (TRU) waste, this evaluation is done for each glovebox or work station. For low-level waste, it is done for each room or area as considered appropriate.

Entry 4. Identifies Decontamination and Decommissioning activities. These are one time procedures with a prescribed duration and a specific set of controls, distinct from on going research programs. Entries 5 through 13

Describes the process, any hazardous materials, and how they are used.

Entries 14 and 15. Lists the types of radionuclides used.

Entries 16,17,18 and 19. Cover the details of the waste including type, detailed component list, comments and sign off by the waste generator.

Section II - Environmental Review Conducted by an Environmental Analyst Familiar With the Process

Fig. 2.

In addition to completing entries 20 through 25, which include a thorough inspection of the area and review of written procedures, the environmental analyst reviews Section I with the waste generator to assure there is a complete understanding of the process and the anticipated waste stream. Entry 24 is where environmental analysts can write as much as they feel necessary on special problem areas and prescribe any controls they feel are appropriate for this area.

Section III - Waste Certification Official Sign Off.

Fig. 3a.

Fig. 3b.

After discussion and clarification with the generator and the environmental analyst, the waste certification official signs, signifying agreement with the information provided and that it is complete, accurate, and sufficient to define the waste stream.

WASTE STREAM EVALUATION FORM

Once the Process Knowledge Evaluation form has been completed and signed off, the waste generator and the environmental analyst prepare the Waste Stream Evaluation form. There are separate, slightly different, Waste Stream Evaluation forms for transuranic and low-level waste, but in both cases they provide the same information.

Entries 1, 2 and 3. Identify the waste form, describe the steps taken to prevent producing mixed waste whenever possible, and give a detailed description of required controls.

Entry 4. is a list of authorized components that have been pre-approved and may be placed in the waste stream without any additional review. This list is a subset of the waste generator's list under entry 17 on the Process Knowledge Evaluation form. The goal is to have all of the components pre-
approved but in practice this is not always possible. For example, used HEPA filters may contain fine particles above the allowable limit and can not be pre-approved. Components that do not appear on this authorized list must be reviewed and authorized by the environmental analyst and the waste certification official before being placed in a waste container.

When signed off by all three parties the Waste Stream Evaluation form becomes a controlled document and is posted in the laboratory. It specifies the controls applying to the waste stream from that area.

STANDARDIZED COMPONENTS

With the Process Knowledge Evaluation forms and the accompanying Waste Stream Evaluation forms, it is desirable to standardize the descriptions of waste components whenever possible. For example we use the description "paper wipe" instead of trade names such as Kleenex, Kimwipes, etc. With this procedure waste stream components are restricted to those that are on the authorized list and appear on the Waste Stream Evaluation form for that particular work station. If a waste generator wishes to dispose of an item not already authorized, the form must be updated and the change approved before that item can be discarded. With this in mind, the authorized components list should be as generic as possible while at the same time precise enough to not let unwanted substances enter the waste stream undetected. Choosing component descriptions that meet these criteria is critical.

DATABASE

The Process Knowledge Evaluation and Waste Stream Evaluation forms are stored on a 4th Dimension relational data base. The database makes it possible to easily standardize component descriptions and to keep track of those that have been pre-approved. Although not presented on the forms, the database also provides a convenient method of keeping track of the hazardous material in a waste component as well as quantities and other identifying information. The forms are reviewed and updated annually.

The 4th Dimension database and its report programs were originally developed by an administrative assistant over a period of a few months. It has since been revised and upgraded by a programmer working part time for six months- The database presently handles about 300 Process Knowledge Evaluation and Waste Stream Evaluation forms and can operate on a Macintosh system with 8 MB of RAM.

CONCLUSION

With this review process and the documentation created by the Process Knowledge Evaluation and Waste Stream Evaluation forms we have established a method of assuring that the process knowledge of the waste generator is sufficient to guarantee that waste entering the waste stream meets all waste acceptance criteria and that we have an auditable documented paper trail to support this guarantee.

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ABSTRACT
The Low-Level Radioactive Waste Management Office in conjunction with SENES Consultants Limited of Richmond Hill, Ontario, have developed a computer assisted surface gamma radiation survey system. The system has been designed to collect detailed, objective surface gamma radiation data in a cost effective and efficient manner. The use of this data collection system results in large amounts of data that are not easily handled manually. A number of statistical techniques have been developed to summarize these data and assist in the detection and delineation of radioactive contamination. These statistical methods and the large amount of data improve the ability to detect and delineate areas contaminated with low levels of radioactivity, and are routinely capable of detecting localized contamination containing 7400 Bq (0.2 mCi) Ra-226 (in equilibrium with its progeny) or less, and distributed contamination with concentrations of approximately 74 mBq/g (2 pCi/g) Ra-226 or less.
INTRODUCTION
The LLRWMO
The Low-Level Radioactive Waste Management Office (LLRWMO) was established in 1982 to carry out the responsibilities of the Canadian federal government for low-level radioactive waste (LLRW) management in Canada. The LLRWMO is operated by Atomic Energy of Canada Limited through a cost recovery agreement with Natural Resources Canada, the federal department which provides the funding and establishes national policy for LLRW management. Part of the mandate of the LLRWMO is to resolve historic radioactive waste problems that are a federal responsibility. Historic wastes are LLRW for which the original owner can no longer be held responsible and which are managed in a manner no longer considered acceptable. If they are wastes for which the federal government accepts responsibility, their management comes within the mandate of the LLRWMO. In fulfilling this mandate, the LLRWMO must frequently conduct outdoor, environmental-level gamma radiation surveys at sites across Canada for investigation, cleanup and compliance purposes.
Manual Gamma Radiation Surveys
Traditionally, these surveys have been conducted by field personnel using hand-held analog scintillometers. Gamma radiation readings are taken at evenly spaced points on a site, usually on a 3 by 3 metre grid. The area between the grid points are generally scanned by the surveyors, and any anomalous readings detected during the scan noted. All data collected are recorded manually on paper, and analysis of these paper records is used to determine the results of the survey.
This "manual" method of conducting surface gamma radiation surveys has several limitations. It is labor intensive, and large quantities of data (particularly for larger sites) must be managed by hand. As well, the scanning operation is quite subjective and can vary from surveyor to surveyor. More desirable is a system that increases objectivity in data collection and analysis, reduces manual data handling, and is more amenable to quality assurance procedures.
The Computer Assisted Large Area Surface Gamma Radiation Surveying System
In 1991 the LLRWMO contracted SENES Consultants Limited to assist in the development of a portable computer-assisted gamma radiation detection system capable of surveying large tracts of land in an efficient and reproducible manner. Using off-the-shelf gamma radiation detection instruments controlled by a portable computer with custom software, SENES and the LLRWMO were able to develop a system that i) increased the objectivity in data collection and analysis, ii) reduced manual data handling, and iii) provided a system more amenable to quality assurance procedures (1).
SENES Consultants Limited is an environmental consulting firm with experience ranging from solid waste management to risk analysis, and with extensive specific expertise related to occupational and environmental radiation and radioactivity, including low-level radioactive waste management, uranium mining and milling,

naturally occurring radioactive materials (NORM), radiological surveys, statistical analysis and interpretation, and especially the analysis of risk associated with exposure to radon and its progeny.
Four years of subsequent prototyping have lead to the development of a system consisting of two electronic radiation meters each using a two by two inch cylindrical sodium iodide scintillation detector. The meters are controlled by a portable computer, and a twelve volt battery is used to supply power to the instruments and computer. To provide portability in the field, these components are then mounted on a frame (initially a golf cart, which has lead to the system becoming known as the "cart"). Custom software collects and analyzes the data in real-time. The data analysis is aimed primarily at the detection of above-background radioactive material by identifying readings that are statistically high compared to other local readings. There are currently three configurations in use: a three-wheeled cart, a two-wheeled cart, and a backpack based system. The three-wheeled cart is shown in Fig. 1.
Fig. 1.
The main configuration (the three-wheeled cart) consists of the electronics mentioned above mounted on a large, three-wheeled frame. Its primary use is for surveying large, open areas such as parks, school yards and undeveloped areas. A two-wheeled configuration has been developed for smaller areas such as private residential properties. There is also a third configuration for use in difficult terrain, where the wheeled carts cannot be used. In this configuration, each probe is held to either side of the surveyor using customized "handles", one of which includes a simple control, and the remaining electronics are carried in a backpack on the surveyor's back.
The computer assisted gamma radiation data collection system produces larger amounts of more densely spaced data than the traditional manual surface gamma radiation surveying methods. Typically, the spatial density of this data for the computer-based system is two to four readings per square metre, but has been as high as eight readings per square metre in small area investigations. Statistical techniques have been developed and applied to assess objectively the presence and extent of radioactive contamination based on this large amount of data.
TYPES OF CONTAMINATION
Sources and Amounts of Radioactivity
The contaminated sites managed by the LLRWMO contain several types of radioactive materials that are in either disperse or discrete particle form. Some examples of disperse materials on these sites include: process wastes from uranium, radium, and rare earth refineries, incinerator wastes and soil contamination from air deposition. Discrete particle contamination includes pieces of uranium ore, dials and other materials painted with radium, and industrial equipment used in the uranium refinery industry.
The amount of radioactivity present in these contaminated materials varies greatly. For example, radium-226 concentrations range from soils with elevations of less than 37 mBq g-1 (1.0 pCi g-1) above the local background to refinery wastes exceeding 37 Bq g-1 (1000 pCi g-1). Radium-226 activity in discrete particles ranges from less than 7.4 kBq (0.1 mCi) to more than 7,400 kBq (100.0 mCi).
Most of the radioactive materials contain radionuclides that emit gamma radiation. Since the contaminated materials are often found at, or near, the surface, gamma radiation surveys have been used to locate contaminated areas. Detection of the contamination with gamma radiation surveys is relatively simple at those sites with high concentrations of disperse materials or discrete particles with high activity. However, the detection of contamination at sites that only contain materials with low levels of radioactivity requires extensive gamma radiation surveys and statistical analyses.
Spatial Distribution of Contamination
For the purposes of detection and delineation using gamma radiation survey equipment, contaminated areas can be loosely classified into two categories based on the spatial extent of the surface expression: localized and distributed. Localized contamination areas consist of a few (or possibly, one) discrete radioactive particles or a small volume of disperse radioactive material or both. Regardless, the contamination is present over spatial areas on the order of 30 by 30 cm or less. This definition of localized contamination arises because appreciable elevation in gamma radiation levels would be seen at only one, maybe two, measurement point with

the typical grid spacing used during the gamma radiation surveys. Distributed contamination areas consist of radioactive material, either discrete particles or disperse materials, spread out over larger areas. The extent of these areas ranges from the 30 by 30 cm localized area to a few square meters to hectares depending on the site.

These two types of contaminated areas have certain gamma radiation characteristics (or "signatures"). Both types of contaminated areas exhibit elevated gamma radiation levels; however, the elevation for localized contamination areas is expressed over a much smaller spatial extent. The incremental gamma radiation field from a single discrete particle, or a compact volume of disperse material, decreases rapidly with distance since the inverse-square law applies. For example, the incremental gamma radiation field at a distance of one metre from moderate activity particles, say, about 37 kBq (1.0 mCi) radium, is operationally indiscernible. Only a few, perhaps just one, gamma radiation measurement taken on a fairly dense grid, say 50 cm by 50 cm, would show elevated readings and, if the activity were small or a grid point did not coincide with the location of the discrete contamination, the elevation could be small compared to counting statistics or natural variability.

Areas of distributed contamination exhibit elevated gamma radiation levels at several points using a similar grid spacing since the spatial extent is larger than for the localized contamination areas. For most properties surveyed to data using the computer-based system, the radioactive contamination has not been uniformly distributed in areas of distributed contamination. Although gamma radiation levels were generally elevated at the grid points, there was variability in gamma radiation levels between grid-points due to the non-uniform spatial distribution of the contaminated materials.

SURVEY PROTOCOL AND DATA COLLECTION

All three of the survey system configurations make use of a similar surveying protocol. First, larger sites are broken down into smaller "blocks" (typically 30 by 30 meters) to make the field work more manageable. For smaller sites, the entire area to be surveyed can be defined as a single block.

Within each of these blocks, ropes and markers are used to help guide the surveyor. Data are continuously collected along specified lines and integrated readings stored every second. These lines are generally spaced either one half or one metre apart, depending on the detail of data required. The computer system determines spatial coordinates using the method of dead reckoning. This begins with the surveyor entering the starting coordinate of the line of data about to be collected. Knowing that the surveyor will travel at a constant rate (generally one metre per second), the computer can calculate the current location based on the time elapsed since the beginning of the line. To assist the surveyor in travelling at a constant speed, ropes or markers are placed at even intervals along the line, and the computer beeps when the surveyor should be passing these markers. As well, the computer constantly displays the current calculated position, which allows the surveyor to make continuous adjustments to his or her pace in order to stay synchronized with the computer. If a particular line of data is collected particularly poorly, that data can be erased and the line resurveyed. Figure 2 shows a surveyor collecting a line of data.

Fig. 2.

Data collection on a typical 30 by 30 metre block generally takes a single surveyor approximately one half hour to complete. Often, a second surveyor will assist with laying out ropes and markers, drawing site maps, and, when surveying residential properties, talking to home owners about the surveying process.

Once a survey is complete, the site record consists of the electronically recorded gamma radiation and spatial coordinate data along with a hand-drawn site map. The site map is used to record the locations of the site boundaries, areas on the site that were not surveyed (such as buildings), the type of terrain (such as grass, asphalt, bare earth, etc.), possible natural sources of anomalous readings (such as fieldstone or brick), and any other information of interest.

Use of this computer assisted surveying system results in the objective collection of a large number of relatively evenly spaced gamma radiation readings. Visualization and analysis of these large data sets requires the use of automated and statistical methods.

DATA ANALYSIS METHODS

Statistically based methods of data analysis have been developed to: i) consistently

analyze and summarize the large amounts of gamma radiation data collected; ii) improve the detection limits over other gamma radiation surveys; and, iii) accommodate variation in background gamma radiation levels.
A large amount of data is collected during the gamma radiation survey process. Upon completion of each 30 by 30 m survey block, up to 4,000 individual gamma radiation measurements are collected along with the northings and eastings (spatial co-ordinates) for each measurement. A consistent and relevant summary of these measurements is required for decision-making.
For some radium contaminated sites, the gamma radiation level at the clean-up criterion is within the range of natural background gamma radiation level. At one site, the soil criterion was 74 mBq g-1 radium when the local background was 37 mBqg-1; therefore, the incremental gamma radiation level from the contamination is 0.012 mSv h-1 (2.0 mR h-1) or lower. This increment is within the range of natural variability in gamma radiation levels from changes in soil moisture content and washout of radon decay products during precipitation events (2). The natural variability in gamma radiation exposure rate levels between survey locations is higher due to additional variability of other factors such as soil type. In fact, the background gamma radiation levels at one site were generally lower on the contaminated site than on the background properties located less than a few kilometers from the site. The effect of natural variability on assessment can be reduced by considering the variability in gamma radiation levels within the property instead of basing the assessment on absolute gamma radiation levels.
Localized and Distributed Contamination Statistics
Two statistics are calculated from the gamma radiation measurements on individual properties or survey blocks. First, the elevation in gamma radiation level above the local background is calculated for each measurement point and transformed to a statistically applicable measure. High values of this measure are indicative of discrete contamination. The detection limit for discrete particles depends on several factors including; depth of the discrete particle under the soil, local background gamma radiation level and the selection of acceptable false positive rate.
Figure 3 shows the probability of detection under normal conditions by activity and depth under the soil. Detection probabilities decrease rapidly for discrete particles buried deeper than 10 cm due to the combination of increased soil attenuation and increased distance between the particle and detector. Discrete particle detection limits do not change rapidly over the range of natural background gamma radiation levels.
Fig. 3.
The second statistic is calculated from the variability in gamma radiation levels over small areas, approximately 3 m2, on the property and, along with the localized contamination measure, is calculated for each measurement location. High values of this statistic are considered indicative of distributed contamination.
Discriminant Analyses
At many waste sites, locations with high values of either the localized or distributed contamination measures are immediately investigated and the appropriate remedial actions are completed. At other waste sites, the computerized gamma radiation survey has been used as a screening tool to identify properties, or areas, that have a high probability of containing contamination. Follow-up delineation on these properties is comprised of soil sampling, comprehensive hand-held gamma radiation scans, portable gamma spectroscopy or a combination of the methods. These follow-up investigations are costly; hence, an objective method for allocating follow-up effort, referred to as discriminant analysis, was developed.
Discriminant analysis has proven to be useful in the screening level applications. In this statistical technique, an empirical relationship between the quantitative variables and the classification of an object are established from a calibration set. This relationship is then used to predict the classification for previously unclassified objects from the corresponding quantitative variables (3). In this application the quantitative variables were the largest localized contamination measure and a summary of the distributed contamination measures on the individual properties. The classification variable was whether the property was contaminated or not.
The discriminant function was calculated during the first half of the survey program. The calibration set for background, or uncontaminated, properties were

randomly selected from local areas without contamination. The contaminated set where properties were previously known to contain localized or distributed or both types of contamination. The discriminant function suggested contamination on properties where either the localized or the distributed measure or both were relatively large. The appropriateness of the discriminant function was investigated by calculating the cross-validation error rate. This procedure involves removing, in turn, one calibration observation, recalculating the discriminant function, and comparing the predicted classification using the new discriminant function against the true classification of the removed observation. The cross validation error was small which indicates that the selected discriminant function was reasonable.

The discriminant function was applied to all properties and provided a single number summary of the gamma radiation measurements collected on each property. Discriminant scores for background properties were generally lower than scores for contaminated properties. The distribution for unclassified properties lies between the background and contaminated properties. This suggests that the unclassified properties are comprised of both background and contaminated properties. Previously unclassified properties with high discriminant scores were identified for more intensive investigations.

Geo-statistical and Visualization Analyses

Gamma radiation maps were constructed for some applications to aid in the interpretation and understanding of contamination on individual properties. Some of these maps showed choropleths of the gamma radiation levels on the property along with locations of highly localized and distributed contamination measures. Other maps showed only one statistic such as the distribution of highly localized contamination measures. These maps were useful in understanding the nature of contamination, in directing follow-up investigations and in communicating the gamma radiation survey findings to property owners.

Figure 4 is an example of a gamma choropleth map of a public school property. The areas occupied by buildings (one large and three small) are shown in white indicating that they were not surveyed. Although there are no indications of contamination on this property, there are other points of interest shown on this map. Note the lower areas (with gamma radiation exposure rates of approximately 3.0 to 4.0 microroentgen per hour (mR/h)) surrounding most of the main building and extending to the south-west corner of the map. These areas are a roadway, parking lot and playground, and are all covered with asphalt. The grassed areas to the north-west, along the north edge, and in two areas to the south are easily distinguished from the asphalt covered areas by their generally higher exposure rates (approximately 4.5 to 5.5 mR/h). Another interesting feature is the elevated area immediately surrounding the main building, particularly along the south face of the west wing. These exposure rates (up to 8 mR/h) are the highest found on site, and are due to the naturally occurring elevations found in the brick used in the construction of the school building.

Fig. 4.

Maps of over-all property discriminant score were also constructed during the site survey program. Spatial clustering of properties with high discriminant score was visually evident from these maps. This information facilitated the prioritizing of further efforts to ensure that those properties adjacent to those properties with contamination were adequately surveyed.

Geo-statistical techniques were also employed. The spatial structure of contamination at this site was investigated by constructing the semi-variogram of discriminant scores. Semi-variogram plots show the mean squared difference of the discriminant function at paired locations as a function of the distance between the paired locations. The semi-variogram showed spatial correlation up to about 40 m (or three properties). This information assist in the statistical sampling design for the properties surrounding the contaminated site.

Discriminant scores were predicted for unsurveyed properties using simple kriging (4). The patterns of high likelihood of contamination were similar to the subjective, or visual, assessments of the pattern on the maps. Both analyses provided estimates, during the survey program, of the number of follow-up surveys required and preliminary estimates of contaminated material volumes.

PROJECT EXPERIENCES

Survey of Properties in Scarborough, Ontario for the Investigation of Possible Ra-226 Contamination

In the fall of 1980, mildly contaminated soil was discovered in the residential community of Malvern in Scarborough, Ontario. The contaminant identified was Ra-226 from a radium recovery operation that took place in the 1940s on a farm site that was subsequently developed in the 1970s. At the recovery operation, waste materials from dial painting facilities (cardboard, rags, brushes, etc.) were reportedly incinerated to reduce the volume and provide a waste residue suitable for the chemical extraction of radium.

Traditional radiological investigations of the area identified approximately 3500 m3 of soil contaminated with above background concentrations of Ra-226. The areal extent and depth of the contamination were determined with the use of hand-held scintillometers. Gamma radiation surveys were conducted on approximately 130 residential lots in the area of the former farm site. A 3 by 3 metre grid was applied over each lot and gamma radiation exposure rates were recorded at each of the grid intersection points. As part of these gamma radiation surveys, the area between the grid points was scanned and any anomalous readings detected during the survey were recorded and investigated in detail to confirm the source of the elevated gamma radiation.

In addition to the bulk soil contamination, the results of the gamma radiation surveys and scans also identified discrete sources of radium-contaminated waste. These pieces of waste ranged in size from small particles of soil to pieces of 3mm diameter plastic tubing, approximately 1 to 5cm in length. Soil grading and spreading operations during the construction of the residential lots appear to have spread these small artifacts beyond the limits of the original farm site.

In the spring of 1990, a second accumulation of radium contamination was discovered within Malvern at a site located approximately one kilometer north of the old farm site. Investigations of this new site indicated the primary source of the radioactivity to be radium impregnated pieces of plastic tubing similar to the tubing found at the 1980 site. During the late summer of 1990 a remedial clean-up program was conducted and approximately 20,000 pieces of Ra-226 contaminated tubing were identified and recovered from 2500 m3 of soil.

The discovery of the second site of radium contamination prompted concern that additional radium waste may have been discarded at other locations within the Malvern community. To address this concern the LLRWMO contracted SENES Consultants Limited to assist in the development of a portable computer-assisted gamma radiation detection system. The design objectives were to provide a portable gamma radiation detection and recording system capable of surveying large tracts of land in an efficient and reproducible manner. The result was the first prototype cart system.

In 1992 and 1993 investigations were conducted using the cart system on 15 school properties and approximately fifty residential sites throughout the Malvern area, but not in the immediate vicinity of the old farm site. Although the expectation was that no historical radioactive contamination would be found, these surveys were to document surface gamma radiation levels on each site and to verify the absence of radium artifacts. No historic radioactive contamination was found during these surveys. However, a single piece of uranium ore (approximately 3 cm across) buried just below the surface of a schoolyard was detected and recovered.

In 1994, the cart system was used to complete an intensive survey program covering approximately 420 residential properties in the area of the original farm site. The purpose of these surveys was to document surface gamma radiation levels on the properties and to examine each for radium artifacts or radium contaminated soil that may have been deposited on the site as a result of historic soil moving operations. For each property surveyed, a discriminant function was applied to the data. The resulting "discriminant score" was used to select properties for further investigation. Follow-up investigations were performed on approximately 270 of the 420 properties. Hand-held instrument readings (both gamma spectroscopic and gross gamma) were collected over selected areas of each of the properties. In some cases, soil samples were taken from the surface and from shallow (less than 1.5 metre) boreholes for laboratory gamma spectroscopy analysis. These selected areas were determined using analysis techniques designed to detect both localized and distributed contamination, as described earlier under the "Data Analysis" section. Ultimately, it was determined that approximately 17 of the properties required remedial action. A review of the data was performed manually as a check on the automated data analysis methods, and to make final decisions in a few borderline cases.

It is interesting to note that annotated gamma radiation choropleth maps of individual properties will be used in this project to help inform home owners of the results of their surveys and status of their properties. With some explanation, these maps can help the layperson visualize the data much more effectively than a simple grid of numbers or a few summary statistics based on the collected data.
A similar survey program is planned for the summer of 1995, and will encompass approximately 1000 residential properties in the surrounding area. The same data collection and analysis techniques will be used. There is no expectation that any historic radioactive contamination will be found based on the results of the work performed in 1993 and 1994. However, there is a need to verify that the extent of contamination has been reached.
A thorough manual analysis of the data in the project would have been very expensive and time consuming. As well, the consistency of the collection and analysis of data could not have been maintained throughout the project without well defined data collection methods and objective, automated analysis techniques. By the end of 1995, approximately 1500 residential properties will have been surveyed, the results analyzed, and follow-up investigations performed. This work has proceeded in a cost effective and timely manner with the help of the computer assisted data collection and analysis system.
Survey of Industrial Sites in Fort McMurray, Alberta for Initial Characterization and
Post-Remedial Compliance
During an investigation of an historic uranium transportation network from Port Radium, Northwest Territories to Fort McMurray, Alberta, a number of sites were identified as having been contaminated by spillage of ore and ore concentrates. This transportation network served mines operating in the 1930s and 40s, and carried ore along a 1400 mile water transportation system, through the barge to railcar transfer station in Fort McMurray, Alberta, and on to Port Hope, Ontario for further refining.
In 1992/93 the LLRWMO performed cleanup operations at sites in the City of Fort McMurray, Alberta covering an area of approximately 120,000 m2 and has characterized further areas totalling approximately 150,000 m2. The bulk of the remedial work was in the immediate vicinity of the rail line, neighboring warehouses and loading/unloading areas. These area were contaminated primarily through the spillage of the radioactive materials. The movement of materials on these industrial sites in Fort McMurray, be it due to grading, tracking by people and vehicles or other activities, has distributed discrete fragments of uranium ore over much of the site areas.
As part of the characterization of suspect properties in Fort McMurray, a representative 10% of the surface areas were surveyed using the three-wheeled cart system. These data generated were analyzed using the localized anomaly detection method described earlier in the "Data Analysis" section. The areas identified by this method were each investigated with a comprehensive hand-held instrument scan. Where historic radioactive contamination was found, remedial action was initiated.
On completion of the 1993 remedial work, the LLRWMO conducted a verification program to ensure that the established clean-up criteria had been met. A 100% surface coverage of each remedial property using the computer-based survey system was intended. However, excavation on the sites contributed to the rough terrain, areas of the site were wooded, and other areas were scattered with debris. The three wheeled cart, though not originally intended to travel on such abusive terrain, fared well but could not be used on the roughest areas. The backpack system was developed to tackle these areas. It allowed data in many of the more uneven areas to be collected and analyzed by the same method as that collected with the three-wheeled survey cart. Some areas could not be surveyed using any of the survey system variants so were surveyed manually using a one metre grid pattern and electronic, integrating instruments. The manually collected data was interpreted using similar algorithms to those used on the data collected by the cart or backpack systems.
For the verification program each site was divided into 30 by 30 m areas to make the cart surveying more manageable. After each smaller area was completed, areas identified by the localized anomaly detection methods were investigated by technical staff and the sources of the anomalies recovered and identified, where possible. This project saw the first use of a discrete source density map. Mapping of the

locations of recovered sources of anomalous readings by material type depicted trends in the movement of materials on the site. That most of the anomalous readings were attributed to small pitchblende fragments in the northern end of a Fort McMurray site was characteristic of the loading and unloading of materials in that area. Towards the southern end of the same site the anomalous readings were more often attributed to a lower grade ore and signified the movement of surface materials of apparently little worth or concern. Materials such as gneiss and granite were also recovered from areas of the site. These rocks are native to the area itself and although they are recovered sources of gamma radiation reading anomalies, they are not a contaminant. Mapping of the recovered materials according to their type indicated trends in their past movement. Analysis of these trends pointed to areas of possible contamination which may not have otherwise been considered. Further investigation in these areas uncovered a volume of contaminated material that was attenuated by 50 to 75 cm of overburden.

SUMMARY AND CONCLUSIONS

The computer-assisted methods of data collection and analysis described in this paper are more powerful than the subjective analysis of more traditional manual surveys using integrated gamma radiation measurements at grid points and surface scans between grid points. Surveys using the new data collection and analysis techniques have identified the location of contamination on properties where previous methods were unable to locate such contamination.

These methods have proven especially useful where the incremental radiation from contamination is similar in magnitude to the variability in background radiation. Localized contamination containing 7400 Bq (0.2 mCi) Ra-226 or less, and distributed contamination with incremental concentrations of approximately 74 mBq/g (2 pCi/g) Ra-226 or less have been identified.

The computer-assisted gamma radiation detection system is capable of surveying large tracts of land in an efficient and reproducible manner with: i) increased objectivity in data collection and analysis; ii) reduced manual data handling and, iii) greater quality assurance.

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COST-EFFECTIVENESS EVALUATION OF SAMPLING PROGRAM TO SUPPORT PROCESS CONTROL, SAFETY, AND REGULATORY COMPLIANCE IN A MIXED WASTE EVAPORATOR SYSTEM AT THE HANFORD SITE

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ABSTRACT

A cost-effectiveness evaluation of waste characterization sampling and analysis programs for the 242-A Evaporator at the Hanford site is presented. Program selection is based on the greatest effectiveness-cost ratio. Cost is modeled in terms of sampling, analysis, and quality control costs. Effectiveness is defined as meeting all minimum requirements associated with the decision variables Process Control, Safety, and Regulatory Compliance expressed in terms of the statistical power of the test and confidence levels achievable.

INTRODUCTION

There is a need for rigorous, defensible, and easily implemented and understood cost-effectiveness approaches for high-level radioactive waste sampling and analysis programs. Often times, such programs are evaluated on cost merits without formal consideration to the effectiveness of the alternatives. A suitable tool for combining the program cost and the program effectiveness is cost-effectiveness analysis. Such an approach forces a complete and concise definition of the cost parameters and the effectiveness parameters. Based upon these definitions, the term

"cost-effectiveness" has meaning and is not vernacular.

STATEMENT OF THE PROBLEM

Several waste characterization sampling and analysis programs for the 242-A Evaporator needed to be evaluated for mid-FY-95 operations. Characterization of specific radionuclides and chemical constituents was required for selected evaporator process streams. Each program needed to provide sufficient analytical data to meet process control, operations safety, worker safety, and compliance with the Resource Conservation Recovery Act (RCRA) requirements.

The 242-A Evaporator, located at the Hanford Site in southeastern Washington State, uses a conventional forced circulation and vacuum evaporation system to concentrate mixed waste solutions. Candidate waste feed tanks must be characterized, pumped to the Evaporator feed tank and processed. The feed stream is separated into a slurry stream, a process condensate, and a gaseous process exhaust stream. Two non-hazardous effluent streams, steam condensate and cooling water, are produced and discharged to an evaporative pond.

The slurry stream consists of the concentrated bottoms and contain the majority of the radionuclides and inorganic constituents. The slurry stream is recycled until it is concentrated to target levels. It is then pumped to double shell tanks to be stored for further treatment. The condensed boiloff, or process condensate, contains primarily water, trace organic material, and a greatly reduced concentration of radionuclides. The process condensate is stored at the Liquid Effluent Retention Facility awaiting treatment capacity in the Effluent Treatment Facility. The process exhaust, or vessel vent stream, consists primarily of non-condensible gases drawn from the condenser system, is filtered and discharged through an exhaust stack.

TECHNICAL APPROACH

A cost-effectiveness approach was used to evaluate proposed waste characterization programs (1). This approach was chosen for several reasons. Evaporator operations management needed easily understood information to select among waste characterization programs for current and future operations. A major purpose of the 242-A Evaporator is to support delisting of various RCRA constituents. Sampling and analysis at the evaporator, therefore, needed to support similar requirements at the Effluent Treatment Facility which receives process condensate from the Evaporator. Finally, as retrieval and treatment operations mature at the Hanford Site, a historical database will be required to identify those waste characterization programs that offer the greatest cost-effectiveness in support of these operations.

Cost and effectiveness were modeled separately. The baseline waste characterization program was first defined. Alternative programs were then identified. The cost and the effectiveness of each alternative were determined and compared to the baseline program. Programs that were dominated, i.e., both the cost and the effectiveness parameters were less than at least one other program were eliminated from further evaluation. The remaining programs were then evaluated. The program with the greatest effectiveness-cost ratio was selected as the optimal program. Sensitivities to cost thresholds and effectiveness parameters were examined to determine the influence of these factors in the selection of the optimal alternative.

Baseline cost estimates were determined using sampling data, analytical services data, and quality control data from previous evaporator campaigns during FY-94 and early FY-95. Baseline effectiveness was estimated based on the statistical factors associated with the number of samples taken for the same campaigns. A Data Quality Objectives (DQO) process, including decision error tolerances, was used to identify mid-FY-95 sampling and analysis waste characterization program alternatives for the evaporator (2-4).

IDENTIFICATION OF WASTE CHARACTERIZATION PROGRAMS

Waste characterization program alternatives were identified as described below.

Baseline The Baseline case was the 94-1 evaporator campaign. The 94-1 campaign utilized an extensive sampling effort to determine the contents of certain feed tanks for processing. The sampling consisted of seven samples in the process streams. The baseline data was used as a historical estimate of constituent variability. A comprehensive suite of analyses was performed on the samples with extensive quality control checks.

Alternative 1 Alternative 1 was based on the waste characterization program used in evaporator Campaign 94-2. Five samples were taken in the process streams for this alternative. A comprehensive suite of analyses was also performed on the samples with equally extensive quality control checks.

Alternative 2 This Alternative has a reduced set of analytes, reduced number of samples, and fewer quality control checks. This Alternative was derived during the Evaporator DQO. One sample in each of the three process streams was identified as sufficient to meet process control, safety, and regulatory compliance requirements. Quality control checks were not as extensive as in the Baseline or Alternative 1.
Alternative 3 This Alternative also has a reduced set of analytes, reduced number of samples, and fewer quality control checks. This Alternative was also considered during the Evaporator DQO. Eight samples in each of the three process streams were identified as sufficient to meet process control, safety, and regulatory compliance requirements. Quality control checks were not as extensive as in the Baseline or Alternative 1.
Alternative 4 This Alternative is based on the implementation of DQO Process for Campaign 95-1 of the evaporator. Three samples taken in each of the three process streams were considered sufficient to meet process control, safety, and regulatory compliance requirements. Quality control checks were not as extensive as in the Baseline or Alternative 1.

COST MODEL

Baseline cost information was obtained from previous evaporator campaigns during FY-94 and FY-95. Originally, waste characterization program costs were to be modeled in terms of both direct costs and opportunity costs in discounted 1994 dollars. Direct costs were estimated based on sampling and analysis, sample preparation, and data package costs. Opportunity Costs were associated with schedule delays and impacts. In the course of cost data collection, we were unable to estimate opportunity costs with any degree of certainty. While these costs are integral to a complete cost estimate, we did not include such costs in our evaluation.
Costs associated with the Baseline and Alternative 1 are based on a comprehensive set of analyses and the maximum amount of quality control checks that would be accomplished by both the field samplers and analytical services personnel.
Costs associated with Alternatives 2, 3, and 4 are based on a reduced number of analytes and a reduced quality control checks. Use of the DQO Process helped evaporator operations personnel better define their needs and the value of the data requirements relating to process control, safety, and regulatory compliance.
Several methods were employed to arrive at cost figures when data was either unavailable or suspect. Cost estimating relationships were used to relate costs to operations variables. Specific analogy methods were used to estimate costs by comparison with known costs of similar items or services. Included in this cost factor are estimates for complexity factors or scaling laws. Expert assessment was used as a last resort when cost information was unavailable. Cost estimate uncertainty was determined to be + 10%. Table I summarizes the costs used for this investigation.

EFFECTIVENESS MODEL

Effectiveness is determined as a function of two factors associated with the results of any of the waste characterization program alternatives: (i) the confidence to correctly conclude the requirements are not met, and (ii) the risk associated with incorrectly concluding the requirements are not met (5). The requirements that must be met are process control, safety, and regulatory compliance. The minimum requirement is: the concentration of all analytes that represent a decision variable do not exceed a predefined threshold.
The waste characterization program alternatives were generated using a statistical hypothesis test of the mean value if one or more constituents for each of process control, safety, and regulatory compliance requirements. If the mean value exceeded a specific threshold, it was concluded that the requirements were not met. The confidence to correctly conclude the requirements are not met is commonly called the Power of the Test. The risk associated with incorrectly concluding the requirements are not met is called the Type I error, or the false negative rate. Therefore, the effectiveness measure is a function of the Power of the Test and the Type I error. We will use the convention of denoting the confidence to correctly conclude the requirements are not met as the power and the Type I error as the risk throughout the remainder of this paper. Therefore, the waste characterization alternative with the greatest power and smallest risk is considered the best alternative when both factors are equally weighted.
The general form of the effectiveness model is presented in Eq. 1.
Eq. (1)

where Ei is the effectiveness for waste characterization alternative i, w is the weight associated with the Power of the Test, and v = 1 - w, is the weight associated with the risk . Since there are usually several constituents that must be characterized for an alternative sampling program, the most conservative way to describe the effectiveness of any alternative is to use the minimum power, min (1-), the maximum risk, max (a), and weights such that w = v = 0.50.

For example, suppose three constituents are being examined in a specific waste characterization program. Let the power of the test for the three constituents be 0.90, 0.85, and 0.99, respectively. Let the risk for the three constituents be 0.20, 0.01, and 0.05, respectively. The minimum power is 0.85, and the maximum risk is 0.20. The conservative effectiveness value for the program is based on an 85% power to correctly conclude the requirements are not met, and a 20% risk associated with incorrectly concluding the requirements are not met. The optimistic effectiveness value for the program is based on an 99% power to correctly conclude the requirements are not met, a 1% risk associated with incorrectly concluding the requirements are not met. Both power and risk are equally weighted. The weights of the power and the risk are the basis for sensitivity studies of the effectiveness measure and subsequent trade-offs in the cost-effectiveness evaluation.

The power of the test, 1-, and the risk, a, for the Baseline and the remaining alternatives is presented in Table II. The values for the Baseline and Alternative 1 were computed based upon the number of samples taken and confirmed with evaporator personnel. The values for Alternatives 2, 3, and 4 were defined as part of the Evaporator DQO (2) to determine sampling requirements for process control, safety, and regulatory compliance.

EFFECTIVENESS-COST ANALYSIS

An effectiveness-cost ratio, ECi, was computed for each of the waste characterization sampling alternatives. For convenience, we will denote the effectiveness-cost ratio as the EC ratio. The alternative with the largest EC ratio was selected as the preferred alternative. The utility of this approach is that large values of cost, regardless of the effectiveness, will push the ratio close to zero, and small values of effectiveness, regardless of cost, will force the ratio to behave in a similar manner. The EC ratio is defined as:

Eq. (2)

Three activities were performed prior to the computation of the ECi value. First, we attempted to identify and assess any other decision factors that should be included in the effectiveness or the cost measures. Radiological worker safety was identified as a significant factor. The more samples taken, the greater the Radiological worker exposure. This meant that alternatives that had less samples were preferred to alternatives with more samples from a radiological worker safety perspective. We did not include this as a specific element in the effectiveness measure, but we did consider it in evaluation and selection of the preferred alternative.

Second, all alternatives were examined in terms of any one alternative dominating another alternative for all effectiveness or cost values. For example, suppose Alternative A had a cost of $100K, a maximum power of 90%, and a maximum risk of 10%. Suppose Alternative B had a cost of $200K, a maximum power of 60%, and a maximum risk of 20%. Suppose Alternative C had a cost of $100K, a maximum power of 90%, and a maximum risk of 20%. We would say that Alternative A dominated Alternative B. Alternative B would be eliminated from further evaluation since ECA > ECB in all cases. Since Alternative A and Alternative C tied in cost, we could not eliminate Alternative C although the maximum risk for Alternative C exceeded that of Alternative A. No one alternative dominated any other alternative.

Finally, we identified any minimum power required or any maximum risk allowed. It was determined that risk was the key concern. If any alternative had a risk that exceeded 0.10, the alternative would not be evaluated. This criteria resulted in Alternative 2 not being included in the evaluation since the maximum risk prescribed by Alternative 2 was 0.20.

Results of the waste characterization sampling program effectiveness-cost values are presented in Table III. The effectiveness value, the cost value, the EC ratio, and the marginal change between the baseline cost and the baseline effectiveness for each alternative is provided.

Using only the EC ratio, the preferred alterative is Alternative 4 (EC = 21.1). The next preferred alternative is Alternative 3 (EC = 10.0). Alternative 4 is preferred over all alternatives because it requires the least number of samples (three) and

displays the smallest cost.
Alternative 4 offers a relatively small marginal change over the Baseline (DE/DC = 2.0) when compared to Alternative 3, (DE/DC = 4.5). This clearly indicates that Alternative 4 is the preferred waste characterization alternative in the absolute sense, but Alternative 3 offers the greatest marginal return. Interpreted in terms of the effectiveness and cost measures, this implies that the alternative with the smallest risk is preferred given (i) equal weights for power and risk, (ii) if the power is the same for all alternatives, and (iii) if the cost model remains the same.
Sensitivities were performed by varying the weights from the base case of w = v = 0.50. Two cases were examined: w = 0.20 and v = 0.80, and w = 0.40 and v = 0.60. The ordering of the preferred alternatives did not change.

CONCLUSIONS

An easy to implement cost-effectiveness evaluation of waste characterization sampling and analysis programs for the 242-A Evaporator at the Hanford site has been presented. The utility of this approach is its relative simplicity and understandability. Data requirements include (i) estimates of cost in terms of sampling, analysis, and quality control costs, and (ii) statistical hypothesis measures in terms of the Power of the Test, the Type I error, and their associated weights.

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RECHARACTERIZATION OF BACKLOG MIXED WASTE AT ROCKY FLATS

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ABSTRACT

During 1994, an effort was undertaken by EG&G Rocky Flats, Inc. to review previous RCRA hazardous waste determinations and characterizations for almost 33,000 containers of backlog waste (waste generated before 1992) in storage at the Rocky Flats Environmental Technology Site (RFETS). The goals of this recharacterization project were as follows:

gather all existing process knowledge and analytical data concerning backlog waste generation and characterization into a central location,

build consensus between the waste generators and technical and regulatory experts on the correct waste characterization for all backlog waste,

verify the RCRA regulations regarding hazardous waste determination and characterization are applied consistently, and rectify any characterizations made in the past that were overly conservative or incorrect,

update all container labeling and computer records to reflect the new characterization,

publish a controlled document that concisely summarizes all backlog waste generation and characterization information gathered, and update the document as new information is obtained, and

build a basis for identifying future sampling and analysis requirements and storage, treatment, and disposal options.

The recharacterization effort was completed in late 1994 and has resulted in many changes to the characterization of the waste in storage at RFETS. Nearly 970 containers of waste previously considered to be nonhazardous waste were determined to be hazardous waste, and over 5000 containers previously considered hazardous waste were determined to be nonhazardous waste. All waste has been characterized consistently and container labeling and computer records are in agreement, resulting in a higher degree of RCRA compliance and a more accurate RCRA operating record. The information gathered during the recharacterization effort has better defined the extent of sampling and analysis required for groupings of backlog waste. This effort has been reviewed by and received praise from the facility owner, the Department of Energy, and the state regulating agency, the Colorado Department of Public Health and Environment.

HISTORY OF WASTE GENERATION AT ROCKY FLATS

Waste has been generated at Rocky Flats since the beginning of plant operations in the early 1950's. Until the late 1980's, the primary concern with waste generation was proper segregation based on type of material (glass, metal, combustibles, etc.) so that the nuclear material content could be properly measured. Each type of material was identified by a three digit Item Description Code (IDC) and a title; for example, a drum containing stainless steel pipe would be identified as IDC 480 - Light Metal, and lead shielding from a glovebox would be identified as IDC 321 - Lead. Based on the nuclear material content, the waste would then fall into one of four categories: nonradioactive waste, low level waste (waste containing less than 100 picocuries/gram), transuranic (TRU) waste (waste containing greater than 100 picocuries/gram and less the economic discard limit), and residue (material containing levels of special nuclear material deemed economical to recover).

In 1986, a compliance agreement was signed by the Department of Energy (DOE), the Environmental Protection Agency (EPA), and the Colorado Department of Public Health and Environment (CDPH&E), regarding the regulation of waste considered hazardous under the Resource Conservation and Recovery Act (RCRA). Initially, only nonradioactive hazardous and low level hazardous (mixed) waste was regulated. In 1988, further compliance agreements began regulating TRU mixed waste, and by 1991, various agreements and court orders began regulating mixed residue.

Since 1986, Rocky Flats has been working to bring hazardous and mixed waste into compliance with the RCRA regulations. It took until the 1991-1992 timeframe to implement a sitewide system that characterized and documented that characterization of all waste being generated at the Site, and to train all waste generators to properly use the system. All waste generated before that system was in place is called "backlog waste" and the accuracy of the characterization of these backlog wastes is extremely suspect.

Another problem that arose in the late 1980's was the cessation of offsite shipment of the majority of Rocky Flats waste. The Nevada Test Site stopped receiving low level and low level mixed waste in 1990 and the Idaho National Engineering Laboratories stopped receiving TRU and TRU mixed waste in 1989. Additionally, DOE stopped shipment of all nonradioactive waste because proper policies and procedures were not in place to determine what "nonradioactive" really meant. Finally, the shutdown of the Site's plutonium recovery facilities in late 1989 halted the reprocessing of residues. Rocky Flats was forced to store practically all waste generated after 1989.

THE NEED FOR RECHARACTERIZATION OF ROCKY FLATS BACKLOG WASTE

As previously mentioned, the accuracy of the characterization of backlog waste at Rocky Flats is suspect. Much of the backlog waste had been characterized by program support groups or by personnel storing the waste, not by the waste generator. Normally these "second-hand" characterizations were extremely conservative. The backlog waste had been characterized and recharacterized over the years by different groups of individuals having different levels of understanding of the RCRA regulations, resulting in poor consistency in how the regulations were applied. Most

times, the characterization of backlog waste was done by writing a letter, with no attempt made to verify that the waste containers actually got relabelled and the computer database that tracks the containers (the Waste and Environmental Management System - WEMS) was updated. Additionally, there was no systemized documentation of the rationale behind the characterizations of backlog wastes.

This approach in characterization of backlog waste had resulted in incorrect, overly conservative, inconsistent, and unreliable characterizations. The ability to properly store this waste in compliance with RCRA regulations was weakened by these problems. Long-term plans to treat and ultimately dispose of these wastes were severely compromised by the characterization problems. Additionally, the state regulating agency, CDPH&E, had little confidence in Rocky Flats' ability to properly characterize waste and regularly threatened enforcement action.

The "last straw" came when a drum of waste having one characterization recorded on the drum label and another characterization recorded in WEMS was almost treated at one of the Site's interim status treatment facilities. The facility did not have interim status for the treatment of one of the EPA waste codes recorded on the label. The question arose, which characterization is correct - the one on the label or the one in WEMS? It was decided that the only way to determine the correct characterization was to start at the beginning and review the generating process for this drum. The realization was then made that every backlog waste container needed to undergo this type of scrutiny before any of the backlog waste characterizations could be trusted. A project was initiated to "reassess" the hazardous waste determination and waste characterization of all backlog waste currently in storage at Rocky Flats.

THE REASSESSMENT PROCESS

Through review of the WEMS database, it was determined that approximately 33,000 containers fell into the "backlog waste" category. As previously discussed, after 1991-1992 a Sitewide system was in place which properly characterized and documented the characterization of the waste, so containers produced after this timeframe did not require reassessment. Fortunately, a majority of the 33,000 backlog waste containers were identified with an IDC which gave an initial clue as to what each container held. Containers having similar IDCs were grouped into waste forms. Table I lists the 44 waste forms into which all backlog wastes were grouped and the number of containers in each waste form. These waste forms were the starting point for the characterization reassessment.

In order to conduct the reassessment of the 33,000 backlog waste containers, a simple database separate from the very complex WEMS database was established. Thirteen pieces of important characterization-related information for each of the 33,000 containers were downloaded from WEMS into the dBASE IV Backlog Waste Reassessment (BWR) database. All queries, sorts, and groupings of containers for the reassessment project were conducted using the BWR database.

A technical person familiar with RCRA regulations and general waste processing was assigned as the lead "reassessor" for each waste form. The reassessor gathered all the information currently available for the waste form, including:

a listing of all the containers in that waste form, including container identification number, IDC, description of the waste, generation date, building and room in which the waste was generated, building and room in which the waste is currently stored, and the current characterization of the waste.

descriptions of the waste generating process, gathered from interviews with waste generators and technical experts or from review of operating procedures. Process information gathered included the process flow, chemicals used in the process, reactions that occurred in the process, materials of construction of the process equipment, and information regarding the handling and packaging of the waste produced.

any existing analytical data for the containers within the waste form, or similar containers of waste, or from any products or byproducts of the generating process. Due to the radioactive nature of the waste and the heterogeneity of many of the waste forms, analytical data was normally not available.

documentation of past characterizations and current characterizations of similar waste forms.

any relevant regulatory guidance.

Following a review of the information gathered, the reassessor determined the appropriate characterization for the containers included in the waste form.

Containers were divided into "subpopulations" of containers that had identical waste characterization. A subpopulation was defined as any group of containers that had the same IDC, EPA waste codes, compatibility code, Land Disposal Restriction status, and chemical constituent codes (a coding system used at Rocky Flats to track chemical usage). In some cases, thousands of containers would fit within the same subpopulation, and in other cases, a subpopulation would consist of only one container. Each container within the waste form was "tagged" with a population number and subpopulation letter to identify into which characterization subpopulation it fell.

The reassessor assembled all gathered information into a Backlog Baseline Book for that waste form. All Backlog Baseline Books followed an identical format which included an introduction, brief description of the waste form, detailed description of the generating process (many times including a process flow diagram), summary of any analytical data, discussion on the proposed hazardous waste determination and waste characterization, regulatory discussion, and a list of personnel contacted and reference materials used. Of greatest importance is the section discussing the proposed hazardous waste determination and waste characterization. This section begins with a table listing the subpopulations and the characterization of each subpopulation. Subsequent paragraphs explain the rationale behind the characterization of each subpopulation. Attached to each Book was a database listing of all the containers within that waste form, sorted by subpopulation.

The Backlog Baseline Book for each waste form was distributed to a group of reviewers including waste generators, technical experts on the generating process, personnel currently storing the waste (called "custodians" at Rocky Flats), regulatory experts, and quality assurance personnel. After allowing approximately a week for review, a meeting (normally 1 to 2 hours in length) was held to collect comments on the book and discuss questions. During the meeting, consensus was reached between all parties on the proper characterization of the containers within that waste form. Following comment incorporation, the final draft of the Backlog Baseline Book for each waste form was circulated for final review and concurrence signatures were gathered from generator representatives, custodians, regulatory experts, and the reassessor.

After each Book was finalized, it was issued to the current storage unit custodians so that any containers which changed characterization could be relabelled and information in the WEMS computer database could be updated. Each waste form Backlog Baseline Book was released separately over a 7 month period so that custodians would not be inundated with changes. All Books were released by December 8, 1994, and custodians are continuing to update container labels and WEMS. All 33,000 containers should be updated by September 1995.

All waste form Backlog Baseline Books (excluding the database listing of all the containers) have been combined into a single controlled document entitled the Backlog Waste Reassessment Baseline Book, which was distributed to affected Site personnel in mid-January 1995. This document represents the official characterization of all Rocky Flats backlog waste and will be updated as any new information becomes available.

BENEFITS DERIVED FROM THE BACKLOG WASTE REASSESSMENT EFFORTS

In total, the backlog waste reassessment project had taken over one year to complete at a cost of over $1.5 million, not including the relabelling and WEMS database updating efforts (which are considered part of the day-to-day requirements of maintaining compliant container storage units). However, the payoffs of the project will be great.

Correct, Consistent, and Reasonable Characterizations Leading to Improved RCRA Compliance

The majority of backlog containers had some change in characterization as a result of this project. Nearly 970 containers of waste previously considered to be nonhazardous waste were determined to be hazardous waste, and over 5000 containers previously considered hazardous waste were determined to be nonhazardous waste. Additionally, thousands of containers already considered hazardous or mixed waste had minor changes to EPA waste codes.

This project has provided consistency in characterization between containers of like wastes and throughout the entire population of waste inventory at Rocky Flats. Because of this project, the interpretation of how various EPA waste codes should be applied have been agreed upon between various groups at the Site and with DOE and

CDPH&E. The characterizations are appropriate, rather than the prior practice of being overly conservative. With the improved characterizations, a much higher degree of RCRA compliance can be reached in the container storage areas. This higher degree of compliance will earn a higher confidence level and better working relationship with CDPH&E and the EPA.

Characterizations are Well Documented and Defensible, and New Information Can Be Incorporated

The publication of the Backlog Waste Reassessment Baseline Book finally gives a single source that can be accessed to understand the generation processes and characterizations of the backlog waste. This Book will be used by the regulators, storage custodians, and waste personnel attempting to treat and dispose of backlog waste. If the characterization of a backlog waste is ever called into question, the information in the Book can be reviewed again and further debated. If new analytical data or process information becomes available, the information can be added to the Book, the waste recharacterized if necessary, and all affected containers can be identified, relabelled, and updated in WEMS.

Basis of Future Sampling and Analysis Efforts and Treatment and Disposal Options

This project compiled a wealth of information that will be useful for planning future sampling and analysis events. By grouping the containers in subpopulations, statistical methods can now be used to sample the waste form without sampling every container. The information gathered on each subpopulation is useful in determining which analytes must be requested when sampling, thereby eliminating costly testing for hazardous constituents that were never a part of the waste. Additionally, the information gathered during reassessment is extremely useful when planning for future treatment and disposal options. The form, volume, and characterization of the waste are all important considerations during this planning.

SUMMARY AND CONCLUSIONS

The historically poor characterization of Rocky Flats backlog waste resulted in a multitude of RCRA compliance problems and presented a barrier to future sampling and analysis events and the eventual treatment and disposal of the waste. The backlog reassessment project conducted in 1994 was successful in using available information to recharacterize 33,000 containers of waste on a subpopulation basis. The project brought together the organizations that could make the proper characterization of previously generated waste and then ensured that the container labels and WEMS database were properly updated consistent with the new characterization. The Backlog Waste Reassessment Baseline Book provides a concise description of the generating processes and characterizations of backlog waste and will be used in the future to determine required sampling and analysis, and for planning treatment and disposal options. The Book will also be updated as new analytical data and process information becomes available.

TABLE Ia

TABLE Ib

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CONSIDERATIONS IN THE DEVELOPMENT OF GUIDANCE ON THE APPLICATION OF PROCESS KNOWLEDGE TO THE RCRA CHARACTERIZATION OF DOE'S MIXED LOW-LEVEL WASTE

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ABSTRACT

The United States Department of Energy (DOE) is considering the development of a guidance document on the subject of using "process knowledge" in Resource Conservation and Recovery Act (RCRA) waste characterization. Based on a study of RCRA waste characterization requirements, objectives, and practices, the authors believe that such guidance should consider that:

Both process knowledge and sampling and analysis data are usually necessary to fully characterize waste for future treatment, storage, and disposal.

Process knowledge data are more suitable for meeting some RCRA waste characterization objectives than others.

The level of accuracy that process knowledge characterization should achieve depends on the subsequent management of the waste.
Given regulators' bias toward sampling and analysis, it may be important for DOE sites to develop a rationale for waste characterization determinations based on process knowledge.
Many different types of documentation can be used to substantiate process knowledge.
Tracking knowledge concerning waste matrices, underlying hazardous constituents, and non-RCRA waste characteristics is just as important as tracking EPA codes.

INTRODUCTION

Characterization is fundamental to the proper management (that is, the treatment, storage, and disposal) of all types of solid waste, including radioactive, hazardous, and mixed waste. Characterization of hazardous and mixed waste is required by regulations promulgated under RCRA and involves assembling the data necessary to treat, store, and dispose of waste in accordance with applicable regulatory requirements.
Under RCRA, there are two basic types of data used to characterize waste: 1) sampling and analysis, and 2) "process knowledge." The term "process knowledge" is short for "knowledge of the hazard characteristic of the waste in light of the materials or the processes used." Process knowledge data usually mean all information pertaining to a waste that can be used to characterize it, except for sampling and analysis data.
DOE facilities tend to rely heavily on the use of process knowledge in characterizing mixed low-level waste. Due to the radioactive nature of mixed waste, reliance on process knowledge can be justified based on the desire to:
control mixed waste sampling and analysis costs;
avoid the possibility of spreading radioactive contamination through sampling and analysis activities; and
reduce the personnel radiation exposure associated with sampling and analysis activities so as to comply with DOE and Atomic Energy Act (AEA) requirements to keep personnel exposures as low as reasonably achievable (known as the ALARA principle).
While the Environmental Protection Agency (EPA), state regulatory agencies, and other stakeholders are sympathetic to these goals, they remain concerned that:
waste classified by DOE as low-level (non-mixed) waste on basis of process knowledge may actually contain hazardous constituents; and
process knowledge may not have adequately identified the hazardous constituents or hazard codes of mixed waste needing treatment.
Characterization based on process knowledge engenders greater concern than characterization based on sampling and analysis because of the lack of guidance, procedures, or protocols defining the appropriate methods, level of accuracy to be achieved, quality assurance/quality control measures, or documentation/recordkeeping requirements that should be associated with the application of process knowledge to characterize waste.
DOE may address these concerns by developing guidance for DOE sites to follow when process knowledge is used to characterize waste in accordance with RCRA requirements.

CONSIDERATIONS

This paper describes considerations for DOE's guidance based on:
RCRA waste characterization requirements and objectives;
the current application of process knowledge to RCRA waste characterization processes at DOE facilities; and
EPA and State comments on DOE's Interim Mixed Waste Inventory Report, which were critical of DOE's reliance on process knowledge in RCRA waste characterization processes.

Relationship of Sampling and Analysis and Process Knowledge to Waste Characterization

There has been a tendency in regulations, regulatory preambles, guidance documents, and DOE's information-gathering efforts (such as the Interim Mixed Waste Inventory Report and the data compiled for the final Mixed Waste Inventory Report) to present sampling and analysis and application of process knowledge as different methods of waste characterization. Typically, however, both sampling and analysis and process knowledge are needed to fully characterize a waste for future management.
At one extreme, process knowledge alone may be sufficient to determine that a waste

could not be hazardous and therefore that other RCRA requirements, including land disposal restriction (LDR) standards, are not applicable. At the other extreme, nothing beyond what can be learned from physical examination may be known about a waste. In this case, sampling and analysis is needed to establish the basic parameters for further sampling and analysis.

For example, initial sampling and analysis may establish that an unknown waste contains metals. Additional sampling and analysis would be necessary to determine if RCRA metals were present at levels that would fail the Toxicity Characteristic Leaching Procedure (TCLP).

The most common situations, however, fall somewhere in the middle. For example, the choice of a sampling and analysis strategy depends on the amount and presumed accuracy of knowledge of the process(es) that produced the waste and the way in which the waste was containerized. This knowledge defines the waste requiring characterization and determines which sampling strategy is most appropriate to characterize it (e.g., random sampling, systematic sampling, stratified random sampling, composite sampling), the number of samples that must be taken, and the types of analyses that must be performed.

There are also situations where sampling and analysis data may be considered process knowledge. This occurs when sampling and analysis is used to characterize a particular waste stream or a surrogate waste, and the sampling and analysis data are subsequently used to characterize another similar waste stream.

Sampling and analysis of a waste stream that has already been fully characterized may be required when a process change could alter the RCRA characterization of the waste. Consequently, it is process knowledge that determines when changes to a process (and consequently to the waste) require both recharacterization of the waste and additional sampling and analysis.

The suitability of process knowledge or sampling and analysis to meet various characterization objectives is affected by the types of regulatory standards on which the characterization objectives are based. In the RCRA program, concentration-based standards favor sampling and analysis. Standards identifying wastes as hazardous on the basis of chemical uses or the processes used to generate waste (these types of standards identify "listed" wastes) require process knowledge to make a determination as to whether the waste is really hazardous or non-hazardous. Standards based on technologies used to treat waste favor process knowledge.

Table I illustrates the relationship between different types of RCRA regulatory standards, process knowledge, and sampling and analysis when waste is characterized for the purpose of meeting each of the two fundamental RCRA waste characterization objectives.

Table I. Relationship Between RCRA Regulatory Standards, Process Knowledge, and Sampling and Analysis

Standards promulgated in other regulatory programs have also influenced the ability of DOE sites to use process knowledge to characterize wastes. The promulgation of the Occupational Safety and Health Administration's Hazard Communication Rule in the mid-1980s significantly strengthened DOE's ability to use process knowledge. The Hazard Communication Rule requires that information about the materials used in the workplace be in the form of material safety data sheets (MSDSs) and be made available to employees. This has resulted in a great increase in the availability of MSDSs, which describe the chemicals and hazards present in commercially purchased chemical substances used in the workplace. MSDSs typically list the principal chemical ingredients in a material by percent and contain other data on chemical and physical properties and hazards.

The relationship between process knowledge and sampling and analysis will continue to evolve as new regulations are promulgated under RCRA and other laws. For example, the future promulgation of a hazardous waste identification rule allowing low-concentration listed wastes to exit the hazardous waste system may favor earlier sampling and analysis of listed wastes to determine whether they meet exit criteria.

Relationship of Process Knowledge to the Objective of Waste Characterization

Under RCRA, wastes are required to be characterized to meet a variety of objectives, including the following:

Generators of solid waste are required to determine whether the waste is hazardous (40 CFR Part 262).

Facilities treating, storing, or disposing of hazardous waste must obtain

sufficient information about the waste to treat, store, or dispose of it in accordance with applicable standards (40 CFR 264 and 40 CFR 265).

Generators and facilities treating, storing, or disposing of hazardous waste must characterize the waste to determine the applicability of RCRA LDR requirements (40 CFR 268).

DOE sites that generate mixed waste also typically store this waste for long periods of time. (Long-term storage of mixed waste is a consequence of the limited availability of treatment and disposal capacity.) Some of these sites also treat mixed waste or expect to do so in the future. In addition, some sites may dispose of mixed waste. For DOE sites, RCRA waste characterization may therefore serve a wide variety of different objectives, most of which fall into one of the following five categories:

identification of the waste as hazardous or non-hazardous;

determination of requirements for compliant/safe storage;

determination as to whether the waste meets LDR standards (either before or after treatment);

performance of a treatability evaluation; or

determination as to whether the waste meets the waste acceptance criteria (existing or proposed) of an existing or planned treatment or disposal facility.

Other important characterization objectives for mixed wastes (and nonhazardous radioactive wastes) that are not addressed in this paper involve assessing the radioactive nature of the waste to determine the applicable requirements for containment, radiation protection, and avoidance of criticality.

In general, the suitability of using process knowledge data to meet RCRA waste characterization requirements is heavily dependent on the waste characterization objective. In the past, process knowledge has been adequate to determine whether many DOE radioactive wastes should be stored as mixed wastes and to choose storage configurations that comply with the requirements of RCRA, the AEA, and other laws. As a result of the use of process knowledge to make these determinations, however, some radioactive wastes that contain hazardous constituents but are not hazardous (e.g., if tested, the waste would pass the TCLP), are managed as mixed wastes until better information is available.

While this strategy adds to the cost of waste management (because it is typically more expensive to manage mixed waste than nonhazardous low-level waste), the additional cost may be justified in some situations. The additional cost of managing a waste as a mixed waste must be balanced against the costs of sampling and analyzing mixed waste, considering the probability that sampling and analysis could demonstrate that the waste is nonhazardous. In general, the higher the probability that sampling and analysis results will allow wastes to exit the RCRA hazardous waste system, the more cost-effective it is to perform sampling and analysis. The following expression demonstrates this relationship:

Cost-Effectiveness = f[($HW - $SW - $CHAR) x (%Prob)]

Where cost-effectiveness is a function (f) of the following variables:

$HW	=	the future cost to manage the waste as a hazardous (or mixed) waste
$SW	=	the future cost to manage the waste as a solid (or nonhazardous radioactive) waste
$CHAR	=	the cost of additional characterization to determine if the waste is hazardous
%Prob	=	the probability that additional characterization will determine that the waste is nonhazardous

If cost effectiveness is less than or equal to 0, additional characterization is not cost-effective. The higher the value of calculated cost-effectiveness, the more cost-effective it is to perform additional characterization. Similarly, the lower the value of the calculated cost-effectiveness, the less cost-effective it is to perform additional characterization.

While process knowledge is often adequate to determine the regulatory status and the storage requirements for a mixed waste, sampling and analysis data are typically required to meet other waste characterization objectives. Sampling and analysis is

required more frequently to determine whether waste meets LDR concentration-based standards, to evaluate wastes for certain types of treatment, and to determine whether wastes meet the waste acceptance criteria and requirements of waste analysis plans of proposed treatment and disposal facilities.

In September 1994, DOE announced a policy of applying the Data Quality Objectives (DQO) Process, developed by the EPA, to all environmental projects where significant environmental data must be collected. EPA developed the DQO Process to minimize expenditures related to data collection by eliminating unnecessary, duplicative, or overly precise data, and to ensure that data collected are of sufficient quality and quantity to support defensible decisionmaking. In the area of waste management, DOE sites can apply the DQO Process to minimize the need for sampling and analysis when characterizing waste for storage and to meet the waste acceptance criteria and requirements of waste analysis plans of treatment and disposal facilities.

Defining the purpose of the waste characterization effort corresponds to performing Steps 1 (State the Problem) and 2 (Identify the Decision) of the DQO Process.

Different Types of Waste Characterization Errors Have Different Consequences

The level of accuracy that waste characterization based on process knowledge needs to attain should be based on the purpose of the characterization and the possible consequences of an error. Determining acceptable levels of accuracy corresponds to performing Step 6 in the DQO process, Specify Limits on Decision Errors.

For example, a process-knowledge-based determination that a waste is not hazardous should achieve a very high level of accuracy, comparable to the 95% confidence level prescribed by sampling and analysis. An error in this determination may be a RCRA compliance issue and may pose a threat to human health and the environment.

However, a process-knowledge-based determination that a waste is hazardous does not need to be accurate for the purpose of compliant storage. Managing a nonhazardous waste as a hazardous waste does not pose a threat to human health or the environment. In most cases, further characterization for treatment will allow inapplicable EPA codes to be eliminated.

Waste Characterization Rationale

Given regulators' bias toward sampling and analysis, it may be important for DOE sites to develop a rationale each time a waste is characterized to determine whether it is hazardous or to assign EPA codes to hazardous wastes. For example, a waste can be determined to be nonhazardous based on the following rationale:

   No hazardous constituents entered or could have been formed in the process producing the waste.

Sometimes many EPA codes are assigned to waste destined for robust treatment (e.g., incineration) based on process knowledge. These codes are assigned because there is a possibility that hazardous constituents could be present. If incineration is the best demonstrated available technology (BDAT), considering the waste matrix and the codes that are assigned, the waste characterization rationale could be:

   All EPA codes that could apply to the waste have been assigned. Further characterization of this waste will be necessary to determine whether it meets the waste acceptance criteria of the treatment facility. EPA codes will be verified and corrected, if necessary, when waste is characterized for treatment. Residuals from treatment will be tested to determine whether they meet LDR treatment standards.

Documentation of Process Knowledge

Many different kinds of documentation can support waste characterization rationales that rely on process knowledge. Such documentation may include testing protocols, work orders, process descriptions, MSDSs, training records (of employees who generate and characterize waste), and procedures for waste accumulation and management.

The documentation that is available may vary according to whether the documentation was produced contemporaneously with the waste, whether the waste was produced by routine or nonroutine processes, the types of operations producing the waste (e.g., manufacturing, laboratory operations, maintenance, environmental restoration, decontamination and decommissioning), and the purpose of the original documentation. For nonhazardous waste, it is particularly important that the documentation show that administrative and other controls have effectively segregated the waste from hazardous waste.

For DOE, other important considerations with respect to documentation of process knowledge include:

  The information in the documentation should be traceable.

The documentation should be readily retrievable and formatted/managed so it can be shared among various site contractors, sites, and with regulators.

The documentation should be formatted/managed so it can be kept for many years and transferred to successor contractors.

Contents of Documentation

RCRA waste characterization involves more than determining whether a waste is hazardous or nonhazardous and, if hazardous, assigning appropriate EPA codes. Thus, adequate documentation of waste characterization information also requires identifying matrix, underlying hazardous constituents, and various chemical and physical characteristics.

Different LDR standards (and thus treatments) are applicable to wastes with different matrices (wastewater, nonwastewater, debris). Matrix identification may depend on process knowledge (including visual inspection) and/or sampling and analysis.

Under EPA's September 1994 Phase II LDR rulemaking, new LDR standards were established that require the identification and appropriate treatment of underlying hazardous constituents in ignitable, corrosive, and TC organic wastes. The regulations allow identification of hazardous constituents standards that must be achieved using process knowledge. (Determining whether these standards have been achieved, however, often requires sampling and analysis.)

Waste movement/transportation, treatment, and disposal in accordance with RCRA standards may require information on many non-RCRA chemical and physical waste characteristics, including viscosity, specific gravity, heating value, organics content, free liquid and moisture content, and the presence of non-RCRA contaminants (radionuclides, asbestos, PCBs). Information on these chemical and physical characteristics may be developed using process knowledge, sampling and analysis, or a combination of the two.

Because determining and documenting waste matrix, underlying hazardous constituents, and non-RCRA waste characteristics are often necessary for proper management of waste, they are just as important as determining and documenting EPA codes. This information is usually developed based on a combination of process knowledge (including visual inspection) and sampling and analysis.

CONCLUSIONS

The authors believe that DOE's process knowledge guidance should recognize that:

1. Both sampling and analysis and process knowledge data are usually necessary to fully characterize waste for future treatment, storage, and disposal.
2. The use of process knowledge data is generally more suitable to some waste characterization objectives than to others.
3. The level of accuracy that waste characterization based on process knowledge needs to attain should be based on the objective of the characterization and the possible consequences of an error.
4. Given regulators' bias toward sampling and analysis, it may be important to develop a rationale for waste characterization decisions involving process knowledge.
5. Many different kinds of documentation can support waste characterization that relies on process knowledge.
6. Documenting and tracking knowledge about the waste matrix, underlying hazardous constituents, and non-RCRA waste characteristics are just as important as tracking and documenting EPA codes.

The authors invite comments on these considerations or on other considerations that should be applied to the development of guidance on the application of process knowledge to the RCRA characterization of mixed low-level waste.

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USING PROCESS KNOWLEDGE TO REMOVE PYROCHEMICAL SALTS FROM HAZARDOUS WASTE REGULATION AT THE ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE (RFETS)

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ABSTRACT
Five pyrochemical processes were used at Rocky Flats Plant (renamed to Rocky Flats Environmental Technology Site, RFETS). The pyrochemical processes produced a variety of pyrochemical salt residues. The five pyrochemical processes that produced these salts were: molten salt extraction, direct oxide reduction, electrorefining, salt scrub and pyroredox. Currently, there are approximately 3600 containers of pyrochemical salts in storage at RFETS. Most of these salts were conservatively characterized under the Resource Conservation and Recovery Act (RCRA) as having the RCRA characteristic of "reactivity" due to the presence of "reactive" metals such as magnesium, sodium, calcium and potassium. In addition, the salts were characterized as D007 (Chromium) based on conservative assumptions which considered that chromium could be present from incidental contact with stainless steel equipment and/or trace amount of chromium may have been present in the generating-process feedstock (site-returned plutonium). The regulated status of these salts is significant to the Department of Energy (DOE) due to the substantial costs associated with their storage, treatment and management as hazardous waste under RCRA.
Utilizing previously unreferenced chemical processing information, analytical tests on surrogate salts, newly developed objective criteria and regulatory analysis, RFETS determined that the definition of the RCRA "reactivity" characteristic does not apply to approximately 3200 of the 3600 containers of pyrochemical salts. The objective criteria resulting from this effort included information from operating experience and analytical data not collected under SW-846 (Environmental Protection Agency's waste analysis methodology) (1) and could serve as the basis for a standardized definition for RCRA "reactivity" for the Department of Energy complex. The D007 code was removed from all pyrochemical salts by the use of thermodynamic calculations. Process equipment was not constructed of stainless steel and the residue salts had no direct contact with stainless steel. Feedstock to the process contained only negligible quantities of chromium. The thermodynamic calculations conclude that, chromium could exist only in insignificant amounts in the residue salt after the pyrochemical reactions are concluded.
RFETS developed an economic model to estimate the potential savings resulting from the removal of these salts from RCRA regulation. The model estimates the potential cost savings across three different residue management pathways and time frames: a baseline new actinide separation facility pathway, a separate salt treatment facility pathway, and a salt repackaging pathway. Potential cost savings are due to reduced costs for storage, inspection, consolidation, permitting, treatment, and characterization. For each residue management pathway a comparison was made between the cost to manage the affected salts as RCRA regulated versus non-RCRA regulated. The model estimated a significant cost savings for managing the salts as non-regulated for all three of the residue management pathways. The potential savings ranged from $6.9 M to $20.1 M depending upon the management pathway assumed. Managing the salts as non-regulated also results in several additional benefits which include: reduced regulatory liability; opening up off-site treatment options; cost avoidance for other DOE sites; and regulatory good will.
INTRODUCTION
During the forty years of production at Rocky Flats various plutonium recovery processes produced spent pyrochemical salt waste streams. The pyrochemical salts were generated as by-products of several processes used to reduce plutonium oxide (PuO_2) to metal, refine Pu metal, and extract actinides from Pu metal. The specific processes utilized were molten salt extraction (MSE), electrorefining (ER), direct oxide reduction (DOR), pyroredox, and salt scrub.
The molten salt extraction process removes americium from plutonium by combining plutonium metal which has unacceptable levels of americium with a mixture of sodium chloride and potassium chloride with magnesium chloride. The mixture is heated until it melts. While the mixture is molten, the magnesium chloride oxidizes most of the americium along with some of the plutonium and the oxidized actinides report to the salt phase. When cooled, the salt and metal are separated. In 1989 a change was made to the MSE process by converting the use of magnesium chloride to dicesium hexchloroplutonate (DCHP).
Electrorefining purifies plutonium metal that does not meet foundry specifications.

Nonspecification plutonium metal is combined with magnesium chloride and an equimolar mixture of sodium chloride and potassium chloride. The mixture is heated until molten. A cathode and an anode/stirrer are lowered into the molten mixture and a current is applied to the anode/stirrer which flows through the molten mixture to the cathode. Pu ions migrate to the cathode and are reduced to purified metal. When the mixture is cooled, the salt, anode heel and purified plutonium metal are separated. Another reaction which occurs during the electrorefining process is that of the reduction of sodium chloride and potassium chloride to elemental sodium and potassium.

Direct Oxide Reduction reacts plutonium oxide with calcium in molten calcium chloride to produce a plutonium metal button by reducing the plutonium oxide to plutonium metal and oxidizing the calcium. The plutonium oxide is "calcined" by heating in a muffle furnace to remove moisture and drive off volatiles. The calcined plutonium oxide is then combined with calcium chloride and calcium metal and heated until molten. The molten material is stirred until the reduction is complete. When the material is cooled, the salt, calcium, and plutonium metal are separated.

Pyroredox is a two-step process for purifying impure plutonium metal. The impure metal is combined with sodium chloride, potassium chloride, and zinc chloride and heated until the mixture is molten. While the mixture is molten, the zinc chloride reacts with the plutonium metal to form plutonium chloride and zinc metal. The mixture is cooled and the salt and metal products are separated. The salt phase is then mixed with calcium metal and heated until molten. While molten, the calcium reduces the plutonium chloride to plutonium metal. The contents are cooled and the metal and salt separated.

Salt Scrub reduces and concentrates actinide ion metals present in spent MSE salt into a stable non-oxidizing metal alloy. The salts are combined with a reducing agent and an alloying agent such as magnesium and aluminum, calcium and gallium and calcium and cerium. The mixture is heated until molten. While in a molten state the plutonium and americium are reduced by the calcium or magnesium and the actinides are taken up in the metal alloy phase. When the mixture is cooled, separation of the salt and the alloy button occurs.

The pyrochemical salts contain recoverable amounts of plutonium and were therefore not considered to be subject to RCRA hazardous waste regulations until a Notice of Violation(NOV) was issued by the State of Colorado in 1989. The NOV identified the residues as mixed waste. DOE and the State of Colorado entered into a Settlement Agreement and Consent Order (2) establishing a responsible approach to identification, characterization and management of the mixed residues in storage. The 1992 Backlog Residue Hazardous Waste Determination Status Report (3) (a deliverable under the Order) identified all pyrochemical salts as RCRA hazardous and assigned the EPA codes D003 (reactivity) and D007 (chromium). The reactivity code was assigned because of the suspected presence of "reactive" metals (sodium, calcium, potassium and magnesium). The chromium code was assigned because it was suspected that chromium could be present in feedstocks to the pyrochemical processes or could be present in the spent salts due to incidental contact with stainless steel equipment.

REMOVAL OF D007 (CHROMIUM) CODE

Background

Two possible sources of chromium were elucidated in the Backlog Residue Hazardous Waste Determination Status Report (3), dated February 1992. The first source was thought to be incidental contact with stainless steel. The second was the possible presence of chromium in the feedstocks for the processes. Experimental analytical data from 1962 and 1963 on pyrochemical salts also indicated that chromium may be present. In April 1993, the State of Colorado requested reevaluation of the pyrochemical salt characterization for chromium.

Approach

Hazardous waste regulations state that the chromium present must be primarily hexavalent chromium in order for the waste to be hazardous for chromium. Process knowledge indicates that hexavalent chromium in the salt matrix could exist only in very small concentrations. Operating procedures indicate that the only sources of chromium are 1) chromium metal from the cell components or 2) chromium ions from the plutonium metal used as feed to the processing. If small quantities of chromium metal were present in the salt, they would not be oxidized to hexavalent chromium because the operating conditions did not allow for its formation. Utilizing standard

free energy of formation data and with the understanding that plutonium ions are always present in the salt and plutonium metal is always present in an immiscible phase, the free energy change can be calculated for the reaction:

PuCl3 + Cro = CrCl3 + Puo

The calculation results in a free energy change of +96.6 Kcal/mole. The positive result indicates that equilibrium favors the reactants over the product. Therefore, chromium in the metallic state will remain in that state and will alloy with plutonium metal.

The pyrochemical processes occur at extremely high temperatures (approximately 700 - 800oC) in an inert environment and, therefore, it is unlikely that significant oxidation of the stainless steel cell can occurred to form chromium oxide (CrO3). Hexavalent chromium compounds decompose at relatively low temperatures, and CrO3 will decompose at 250 oC to Cr2O3 and will then be reduced to chromium metal by the plutonium metal as indicated by the reaction:

4Cr2O3 + 6Puo = 6PuO2 + 8Cro

The standard free energy for the reaction is -105.7 Kcal/mole of Cr2O3 indicating that equilibrium favors to products rather than the reactants.

CONCLUSIONS

The thermodynamic calculations indicate that the pyrochemical salts do not exhibit the toxicity characteristic for chromium, and therefore, the D007 code was removed from the salts. The data was presented to the State of Colorado in May 1993 and the State concurred with the characterization.

REMOVAL OF THE D003 (REACTIVITY) CODE

Definition

The hazardous characteristic of reactivity is defined in the Code of Colorado Regulations 6CCR1007-6, Part 261.23 (4) as stated below:

A solid waste exhibits the characteristic of reactivity if a representative sample of the waste has any of the following properties:

1. It is unstable and readily undergoes violent change without detonating.
2. It reacts violently with water.
3. It forms potentially explosive mixtures with water.
4. When mixed with water, it generates toxic gasses, vapors, or fumes in a quantity sufficient to present a danger to human health or the environment.
5. It is a cyanide or sulfide bearing waste which, when exposed to pH conditions between 2 and 12.5 can generate toxic gases, vapors or fumes in a quantity sufficient to present a danger to human health or the environment.
6. It is capable of detonation or explosive reaction if it is subjected to a strong initiating source or if heated under confinement.
7. It is readily capable of detonation or explosive decomposition or reaction at standard temperature and pressure.
8. It is a forbidden explosive as defined in 49 CFR Sec. 173.51, or a Class A explosive as defined in 49 CFR Sec. 173.88.

Approach

Because the EPA definition is vague, objective criteria were developed to help Rocky Flats better define the conditions necessary for a waste to exhibit the characteristic of reactivity. Conditions that should be satisfied for determining the reactivity characteristic in pyrochemical salts included:

Are any of the containers bulging, signifying pressure buildup?

Is there evidence that any containers have ever exploded?

Are the containers vented? Does the design of the containers prevent gaseous buildup and gaseous mixing?

How long have the wastes been stored in the containers?

Have the wastes been exposed to water during their storage? What resulted?

Applying the objective criteria, knowledge of the composition of the pyrochemical salts gained from years of processing, handling, and storing them at Rocky Flats Environmental Technology Site and other sites within the DOE Complex with no "reactive" incidents demonstrates that pyrochemical salts do not meet the definition of reactivity as described by sections (1), (5), (6),(7), and (8). However, due to the possible presence of calcium, sodium, potassium, or magnesium metal which are known to generate hydrogen when mixed with water, sections (2), (3), and (4) were reviewed further.

The EPA definition of "water reactivity" is obscure, so efforts were made to clarify the meaning. EPA laboratories in Cincinnati and Las Vegas were contacted and it was

determined that the concern was twofold:
1. the amount and rate of hydrogen generation (i.e., whether the rate and volume of hydrogen generation is so large that it would blow off a barrel lid by pressure alone), and
2. the violence of the reaction (i.e., whether the reaction generates enough heat to cause an explosion).

Based upon this clarification, the salts were evaluated for water reactivity. Surrogate testing was performed using pure magnesium metal and salts which were produced by the molten salt extraction process without plutonium. Hydrogen gas bubbled slowly off the surface of the salt and the reaction was not "violent" and did not create enough heat to ignite the hydrogen that was produced. Operating experience was considered in addressing whether enough hydrogen would be generated to blow the lid off of the container. There are no documented cases at RFETS or other DOE sites of bulging drums or explosions relating to containers of any pyrochemical salts. D003 was removed from the Molten Salt Extraction (MSE) salts. The packaged salt would have to be immersed in water in order to generate sufficient quantities of hydrogen. For this to occur a pathway for the water to enter the container would have to be present and in such a case the drum would then be "vented" which would prevent it from becoming over-pressured. While the surrogate testing along with process knowledge removed the D003 code from the MSE salts, the remaining pyrochemical salts were considered to be reactive due to the presence of calcium and sodium metals which are known to react violently with water.

State of Colorado Clarification of Definition of Reactivity

The above information was presented to the State of Colorado in April 1994. During the presentation, the Colorado Department of Public Health and the Environment (CDPHE) provided additional clarification concerning the criteria used to determine whether a waste exhibits the characteristic of reactivity, as summarized below:

RCRA reactivity is a condition that requires the following three key elements:
1. a high reaction rate
2. containment of the reaction, and
3. an ignition source

and that there is a high probability of these elements occurring at the same time. Furthermore, reactive wastes are those that require aggressive management, such as being covered with oil or kerosene to shield the waste from contact with air or moisture, from the point of generation through storage and treatment.

Reevaluation

Additional objective criteria were developed to reevaluate the remaining pyrochemical salts which were considered hazardous for the characteristic of reactivity.The additional criteria included:
1. According to the Colorado Department of Public Health and the Environment, for a waste to be considered reactive, the reaction must be immediate. This presupposes that the reaction is vigorous and that a primary indicator of the vigorous nature of the reaction is a dramatic increase in temperature (e.g., a highly exothermic reaction) and, under contained conditions, a concomitant, significant rise in pressure. Do the containers during storage, or representative samples during testing, exhibit rapid temperature increase, indicating a highly exothermic reaction is occurring?
2. Are conditions which would increase the rate of reaction present, i.e. gas containment, temperature, ignition source, pressure increase, photosensitivity, presence of catalyst?
3. Since the residues and wastes are managed for the pyrophoricity of the Plutonium, what is the difference in the way these would be managed if they were removed from RCRA regulation?

Based upon these criteria and the clarification provided by CDPHE, the remaining salts were reevaluated for reactivity.

The Electrorefining salts (ER) and the pyroredox salts could contain sodium and potassium metal that was produced during the electrorefining process. However, there is a low probability that unreacted metal is still present in the salts. The CDPHE criteria that there be a high probability of a high reaction rate and a source of ignition occurring at the same time is not met because there is a low probability that any unreacted metal would be present. These salts would no longer be considered RCRA reactive. Subsequent sampling and analysis of the ER salts supported the process knowledge.

The DOR salts and the salt scrub salts remain reactive because of the presence of relatively large "buttons" of calcium metal or entrained cerium . These containers are "aggressively" managed in specially designed buildings, protecting the residues from the weather, water, and uncontrolled movement. Personnel access to the containers is strictly controlled. If containers of these salts were placed in an uncontrolled storage location, there is an increased probability that the three conditions of reactivity as defined by CDPHE would occur at the same time.

COST SAVINGS FROM THE RECHARACTERIZATION OF PYROCHEMICAL SALTS

The action to remove the D003 (reactivity) code resulted in the removal of 3169 containers from hazardous waste regulation. Utilizing an economic modeling technique developed by the Operations and Waste Management Organization at DOE's Rocky Flats Field Office, the cost savings/cost avoidance associated with the management of the pyrochemical salts as non-RCRA regulated were calculated.

It is estimated that the cost savings range between a minimum of $6.9 to a maximum $20.1 million over a 26 year period, depending upon which of the pyrochemical salt management options is used. Management options include construction of: 1) a baseline actinide separation treatment facility; 2) a salt repackaging facility; or 3) a separate salt processing facility. Savings were calculated based on reduced costs for storage, inspection, consolidation, permitting, treatment, and characterization for each management option.

Table I illustrates the results of the cost calculations for management of the pyrochemical salts over the 26 year period. The three management options for RCRA hazardous salts as well as the costs for management as non-hazardous are presented. Table II depicts the overall cost savings for managing the salts as non-RCRA hazardous as compared to the three options for RCRA hazardous salts.

FUTURE DIRECTION

First, Rocky Flats Environmental Technology Site now uses the objective criteria developed for the pyrochemical salts to define any "reactive" hazardous waste at RFETS. This provides a consistent basis for managing "reactive" hazardous waste at RFETS.

Second, RFETS issued a letter to the other DOE sites explaining the approach used and suggesting that the same objective criteria used to define a "reactive" waste at RFETS should be used throughout the DOE complex. The rationale for this is to minimize confusion and facilitate a consistent basis for managing these salts complex-wide.

Third, RFETS provided this information to a joint DOE-Headquarters/Environmental Protection Agency (EPA) Process Knowledge Project Team that is looking at developing nationwide guidance on the proper use and documentation of process knowledge to characterize waste throughout the DOE complex. Depending upon the scope of their guidance, RFETS has recommended that the objective criteria developed at RFETS be included in the guidance document to help the DOE complex achieve a consistent approach to managing its "reactive" hazardous wastes.

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A MODEL FOR PREDICTING CHARACTERIZATION REQUIREMENTS AND FUTURE ANALYTICAL LABORATORY LOADS FOR ALPHA-CONTAMINATED WASTE*

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ABSTRACT

Regulatory compliance and effective management of TRU waste in the United States requires a knowledge of the constituents present. With limited resources, the United States Department of Energy (DOE) needs a cost-effective characterization program.

In addition, the DOE needs a method for predicting the present and future analytical requirements for waste characterization so that it can provide facilities to meet the need. Thus, a model that can be used to predict the present and future waste characterization needs, using current knowledge of the TRU inventory, has been developed.

INTRODUCTION

Transuranic wastes have been generated, packaged, and stored at the United States Department of Energy (DOE) facilities, under a constantly changing climate of rules and regulations, for the past fifty years. Transuranic waste or TRU is defined as waste that contains more than 100 nCi/g of alpha-emitting, transuranic isotopes with half-lives greater than twenty years. Through the years, the rules and regulations in place at the time dictated the extent to which the wastes were characterized. Today, the DOE has about 300,000 m3 of transuranic waste stored at eleven facilities across the United States (1). This is equivalent to over one million 55-gallon (0.21 m3) drums of TRU waste. Using today's standards, only a small percentage of the DOE's transuranic waste inventory is believed to be adequately characterized. Continued storage, potential treatment, and ultimately disposal of transuranic waste will require a substantial characterization effort. Characterization is needed to safeguard worker health and safety and the environment, to protect and optimize potential treatment processes, to meet environmental regulations, and to satisfy the waste acceptance criteria for the disposal site. However, because TRU waste contains relatively high concentrations of alpha-emitting isotopes, characterization efforts must be conducted in specially designed facilities that limit worker exposure to alpha-contaminated waste.

The U.S. DOE is sponsoring the development of a model that can be used to predict the potential characterization requirements for TRU waste in the United States given a variety of possible management scenarios for the waste. The model has a three-fold value: first as a planning tool, second as a communication tool, and third as a decision tool. As a planning tool, the model will provide the DOE with information about the size, type, cost, and schedule required for transuranic waste characterization facilities given a variety of assumptions about the ultimate fate of the waste. As a communication tool, the model will provide the DOE with information about the consequences of any agreements concerning TRU waste characterization that may be reached with the regulatory agencies involved. As a decision tool, the model will provide the DOE with information that will help it decide between building new or retro-fitting old waste characterization facilities, or using commercial facilities.

OBJECTIVE

The model discussed in this paper is designed to provide an estimate of present and future analytical requirements for TRU waste characterization. It provides a defensible technical basis for predicting the amount of sampling and analysis that will be required for the TRU waste inventory in the United States over the next five to fifty years. The model is flexible and can account for the planned management scenario for the waste inventory and also can account for the current state of knowledge about the waste. In this context, "management" encompasses all of the steps involved from the time a material becomes a waste until it is placed in a permanent disposal site. "Knowledge" encompasses any analytical data pertaining to the waste and any information about the process that produced the waste.

TECHNICAL APPROACH

Sandia National Laboratories in Albuquerque, New Mexico has proposed a strategy for analysis of potential characterization needs for TRU waste that involves the following steps:

1. the processes involved in managing a TRU waste stream (waste management scenarios) are identified;
2. the analytical tests required to characterize the waste for the management processes identified are determined;
3. the number of samples required for each analytical test in the least-informed case, without the benefit of process knowledge or existing analytical data, is calculated;
4. the quality and quantity of the existing knowledge about the waste stream (both process knowledge and analytical data) is determined;
5. the number of samples required for each analytical test in the least-informed case is reduced to reflect the quality of existing data;

6. the number of samples calculated in step (5) is recorded in a matrix format, where the rows correspond to the analytical tests, and the columns correspond to media and sample radiation handling protection classes; and
7. the number of tests in each analytical test/media class category is translated into an equivalent amount of laboratory space.

These steps represent the major components of the theoretical model being developed as a tool to estimate future analytical requirements for the TRU waste inventory. The following discussion concerns an example calculation designed to demonstrate each of the steps for solidified process residues at Rocky Flats.

ANALYTICAL REQUIREMENTS FOR SOLIDIFIED PROCESS RESIDUES AT ROCKY FLATS

Transuranic waste has been generated at the Rocky Flats Plant (RFP) in Colorado in past years from routine production support operations for the U.S. nuclear weapons program. Production operations at RFP were suspended in 1990. The primary radioactive contaminant in the waste streams is plutonium. The primary hazardous contaminants are heavy metals, organic solvents, and pyrochemical salts. Prior to 1986, RFP TRU wastes were shipped out of Colorado for storage. In 1986, waste shipments to storage were terminated, and TRU waste has been accumulating at RFP since then.

Six RFP waste streams were chosen in order to demonstrate the theoretical model. The six waste streams are all solidified process residues from RFP (2). Altogether the six waste streams represent 1235 drums of transuranic waste, a total volume of 260.3m3. Of the 1235 waste containers, 27 were generated between 1970 and 1986 and 1208 were generated between 1986 and 1993 as shown in Table I.

The Waste Management Scenario

Management of a waste typically begins with its designation as either high-level, low-level, or transuranic waste as well as its designation as either hazardous or non-hazardous waste. Further, management includes a combination of storage, treatment, and transportation steps leading ultimately to disposal of the waste. The six waste streams chosen for this analysis have been designated by RFP personnel as TRU waste. Three of the waste streams are believed to contain both hazardous and radioactive components (ID Nos. RF-W010, RF-W013, RF-W038); three contain only radioactive components (ID Nos. RF-T010, RF-T013, RF-T038).

The DOE's current management strategy for these waste streams involves disposal at the Waste Isolation Pilot Plant (WIPP), a deep geologic repository in southern New Mexico. The repository is still under development, and until it opens, the waste will continue to be stored at the RFP site in Colorado. When WIPP opens, the waste will be shipped in containers designed specifically for transportation of transuranic wastes. It is believed that the six waste streams chosen for this analysis will not require any treatment before they can be shipped and placed in the repository.

The Analytical Tests Required

Depending on the management scenario selected, the knowledge required about a waste will vary. For example, the characterization activities required to support transportation of a waste may be different from those required to support a given treatment technology for the waste. As a result, the types of chemical or radiological analyses required will vary with the management scenario.

Given the DOE's current management strategy for TRU waste, none of the six waste streams chosen for this demonstration analysis will require treatment. Consequently, characterization activities will focus on gathering knowledge to support storage, transportation, and disposal of the wastes. The DOE has issued a tentative plan for characterization of these waste streams. In its plan (3), the analytical tests required are a function of a group number assigned to the waste stream. Group 1 includes those waste categories for which the original materials used are the predominant component of the final waste. Group 2 waste categories include waste that results from a specific process or a final treatment option. The Group 3 waste category includes mixtures of Group 1 and Group 2 waste categories. Group 1 waste includes combustibles and noncombustibles, graphite, filters, Benelex and Plexiglass, firebrick and ceramic crucibles, leaded rubber, metal, glass, and supercompacted waste. Group 2 waste includes inorganic waste water treatment sludge, organic liquid and sludge, solidified liquid, inorganic process solids and soil, cation/anion exchange resins, and pyrochemical salts. Group 3 waste includes both solid and solidified materials. The group numbers assigned for the six waste streams are shown in Table I. All of the waste streams chosen for this analysis have been

assigned to either group 2 or 3 for sampling and analysis. Table II identifies the suite of analytical tests that will be performed for wastes assigned to each of the groups.

The Number of Samples Required in the Least-Informed Case

Once the suite of analytical tests required to characterize a waste stream for the management scenario under consideration has been established, the number of samples needed for each analytical test in the least-informed case is calculated. The least-informed case refers to a circumstance where no analytical data and a minimal amount of process knowledge are available regarding the analyte of concern. The number of waste samples required for characterization of a waste stream in this case represents a maximum amount of sampling and analysis for the waste. The amount and quality of existing analytical data or process knowledge are not factors in determining the number of samples.

An algorithm for calculating the number of samples was developed for this application. The algorithm is a statistical function of the waste volume, the number of containers, the homogeneity of the waste form, the precision of the analytical method, and the proximity of the expected result to the threshold limit of concern. The first portion of the algorithm is based on standard statistical theory for confidence interval estimation to arrive at an initial value for the number of samples needed. To address other sources of uncertainty affecting waste characterization (like homogeneity of the waste form), a set of factors based on expert judgement were used. Computations for the maximum number of samples (least-informed case) for specific analytic test/waste matrix combinations were performed using the algorithm. Results for the six RFP waste streams are shown in Table III.

Evaluating Existing Data Quality

The least-informed case can be thought of as an upper bound on the number of analyses that need to be performed. If existing information is taken into account, the number of analyses required can be reduced. TRU waste inventories, both current and projected, have been described with unknown degrees of accuracy in several databases and reports (4), (5). Characterization to date has relied primarily on process knowledge, while a minimal amount of characterization has been based on sampling and analysis. Both process knowledge and sampling and analysis will be needed to adequately characterize the inventory of TRU waste.

Four reduction factors have been developed for this application. The reduction factors scale the number of analyses required down from the maximum (least-informed case), depending on the presumed quality of existing data for the waste stream and the analyte of concern. The reduction factors and brief descriptions of the quality levels are given in Table IV.

In general, there is a correlation between the amount and quality of information about a waste and the time period that the waste was generated. Records for TRU wastes generated before 1970 are very poor. As environmental standards changed through the years, the quality of information recorded for the wastes increased. Working with site personnel, four quality time periods for RFP wastes were established. Containers in the six RFP waste streams were assigned data quality levels based on the time period in which they were generated. Wastes generated prior to 1970 (Time Period 4) were assigned to quality level four. Wastes generated between 1970 and 1986 (Time Period 3) were assigned to quality level three. Wastes generated between 1986 and 1993 (Time Period 2) were assigned to quality level two, and wastes generated after 1993 (Time Period 1) were assigned to quality level one.

The Number of Samples Required

Once the quality of existing data has been determined, the number of samples required in the least-informed case is reduced to reflect the benefit that may result from using existing analytical data and/or process knowledge to guide the sampling and analysis program. The results for the six RFP waste streams are shown in Table V.

The Analytical Matrix

The number and types of analytical tests predicted for a waste stream are stored in matrix format as shown in Table VI. Rows in the analytical matrix correspond to analytes, categories of analytes, or test methods identified as important for characterization of the six RFP TRU waste streams. Columns in the analytical matrix are divided into media classes. The columns are further divided into sample radiation handling classes. For example, distinctions are made for samples that must

be handled in a hot cell or a glove box rather than on a laboratory bench with a hood.
A media class includes all physical forms that are amenable to laboratory preparation using the same processes. Here, the same processes means the same facilities and processing equipment. Minor changes in glassware, consumables, and/or reagents will not constitute a different process unless there is a significant effect on the required processing space, storage space, or equipment required for safe and effective handling.
The first media class is a group of waste matrices which can be prepared for analysis in a laboratory using the same facilities and equipment that are used for standard methods of analysis such as those specified in EPA SW-846 (6) or other consensus compendia, as applicable to the analyte. The second media class is a group of waste matrices that must be prepared for analysis using demonstrated methods which require facilities or equipment which are known but different from those required for the standard methods or adaptations falling in the first media class. The third media class is a group of waste matrices which cannot be prepared using standard or demonstrated methods. A methods development effort is likely to be required, and the facility and equipment needs must be assumed from waste matrix characteristics. Based on information given by the site, it is assumed that all of the samples taken from the six waste streams must be processed in a glove box. Further, all of the samples have been assigned to the second media class.
Also, a distinction has been made in Table VI between analyses for high concentrations of a compound and low concentrations of the compound. Equipment used to analyze samples containing high concentrations of a compound can become contaminated, making detection of low concentrations of the compound difficult. Therefore, separate, equivalent pieces of equipment are frequently used for analysis of high-concentration and low-concentration waste samples. Concomitantly, the laboratory space needed to process both high and low concentration samples will be greater than that required to process either high or low concentration samples. The high and low designations were made by examining the hazardous constituents listed for the waste streams by the site.
TABLE VI
The Laboratory Space Required
Summing the rows and columns in the analytical matrix can provide the DOE with valuable information about its intended characterization program for TRU waste. For example, for the six RFP waste streams described above, 600 waste samples are needed for characterization of the waste. This is equivalent to sampling 9% of the 1235 waste containers. Applying the model to the entire inventory of TRU waste will tell the DOE how much sampling and analysis is required for characterization of its TRU waste. Application of the model to the entire DOE inventory of TRU waste is scheduled for completion in October of 1995.
For the purposes of this paper, the project staff considered a very crude extrapolation of the results for the six RFP waste streams to the inventory of TRU waste intended for disposal at WIPP (approximately 800,000 drums or 168,000 m3). The extrapolation is made in order to demonstrate the types of results that are expected from this program. The reader is cautioned to remember that the six RFP waste streams on which the extrapolation is based are not particularly representative of the DOE TRU waste inventory. For example, none of the containers chosen were generated before 1970, and very few of the containers were generated before 1986. In addition, all of the containers chosen are contact-handled, and there is little or no fission product activity. In reality, the DOE has a large volume of TRU waste that was generated before 1986. It has remote-handled TRU waste, and at some of the DOE sites, fission products are present in the waste. Overall, extrapolation of the current results to the entire DOE TRU waste inventory is likely an underestimate of the DOE's characterization needs for TRU waste.
If 600 samples are required for 260.3 m3 of waste, an extrapolation indicates that 387,244 samples would be required for the 800,000 TRU waste containers intended for disposal at WIPP. Therefore, 387,244 samples would be sent to the laboratory for analysis. Of the 387,244 samples, 89,066 samples will be analyzed for volatile organic compounds, 89,066 samples will be analyzed for semi-volatile organic compounds, 46,469 samples will be analyzed for polychlorinated biphenyls, 85,194 samples will be analyzed for toxic elements, and 77,449 samples will be analyzed for mercury.

Depending on the time schedule imposed on the characterization program, the size of the laboratory facility required can be determined. For this example, it is assumed that the WIPP disposal site will open in 1998 and characterization will take place over a five year period. An estimate of the number of glove boxes and chemical analysis units for volatile and semi-volatile organic compounds is summarized here.
Analysis for volatile organic compounds by EPA SW-846 methods requires purge-and-trap extraction of the waste samples followed by measurement using a gas chromatograph (GC) or gas chromatograph/mass spectrometer (GC/MS). The extraction and analysis can be done at an average rate of about one sample every two hours. Based on working one 12 hour shift a day for 250 working days per year, sample through put is 6 samples a day, 1500 samples a year, or 7500 samples in five years. Twelve extraction/analysis units would be needed to perform 89,066 analyses in five years. Each unit should fit in a standard 4 ft. (1.22 m) by 8 ft. (2.44 m) glove box. From this information, the amount of laboratory space required for analysis of volatile organic compounds can be estimated.
Analysis for semi-volatile organic compounds by EPA SW-846 methods also requires extraction of the waste samples and analysis by GC or GC/MS. These procedures, however, permit the extraction and analysis to be done at different locations and different times. Extractions can be done at the rate of about 10 per day per glove box or about 2500 per 250-day working year or about 12,500 in five years. Eight glove boxes would be required to extract all 89,066 samples in five years. The GC or GC/MS analyses can be done at the rate of about 10 samples per 12-hour shift per GC or GC/MS unit. Based on one 12-hour shift per day and 250 working days per year, the number of GC or GC/MS units required is the same as the number of glove boxes for extractions.
The EPA SW-846 method for PCBs has requirements that are very similar to those for semi-volatile organic compounds. Thus, the same through put rates (12,500 samples in five years) can be assumed for semi-volatile organic compounds. The result is that four extraction glove boxes and four GC or GC/MS units will be required to perform the 46,469 analyses in five years working 250 days per year. Altogether, for the organic analyses, twenty-four glove boxes and twenty-four GC or GC/MS units would be required to complete the characterization program for organic compounds.
The toxic metals of interest are the RCRA TCLP metals (As, Ba, Cd, Cr, Hg, Pb, Ag, Se) and Ni and Tl. Samples are digested and the metals determined by Inductively Coupled Plasma (ICP) spectroscopy or Atomic Absorption (AA) spectroscopy. Mercury is analyzed by a different method than the other metals and will be considered separately below. It is estimated that samples can be digested by SW-846 methods at the rate of about 4000 samples per year per glovebox. If ICP is used for analysis, about 24,000 samples per year can be analyzed for the nine metals (not Hg) listed above. This assumes a simultaneous, multi-element ICP instrument is used. It will thus require five gloveboxes to digest 85,194 samples in five years and one ICP spectrometer to analyze the 85,194 digestates in five years.
If AA is used for analysis, each metal determination becomes a separate analysis. Thus each sample requires nine analyses to determine the nine metals of interest. Throughput for Ca, Cd, Cr, Pb, Ag, Ni, and Tl is estimated to be 24,000 analyses (3,430 samples) per year per AA. As and Se require either different or additional processing. Estimated throughput for these two elements is about 10,000 analyses (5000 samples for As and Se) per year per AA. Using these assumptions and estimates, it will require nine AA spectrometers to analyze 85,194 samples for these nine metals in five years.
Determination of mercury by SW-846 methods requires sample digestion followed by analysis by manual cold vapor atomic absorption. About 8000 samples can be digested per year per glovebox and about 8000 samples can be analyzed per year per AA spectrometer. based on these assumptions, it will requires four digestion glovebox and four AA spectrometers to complete analysis of 77,449 samples for Hg in five years.
DISCUSSION
Implementation of the model described above has a three-fold value for the DOE. First, the DOE is interested in a planning tool. By changing the assumptions input into the model, the DOE can see the impact of a variety of "what-if" scenarios. To demonstrate, the assumption about the quality level of existing data for the six RFP waste streams will be challenged in this discussion. The results presented above were calculated assuming two quality levels for data depending on the time period in

which the waste was generated. However, the quality of existing data, both process knowledge and analytical data, and the extent to which existing data can be used in lieu of further characterization is currently a subject of debate between the DOE and the regulatory agencies.
A total of 600 samples would be required for the six RFP waste streams assuming two quality levels that change with time. If the assumption about data quality in this example is in error, the number of samples required would change. For instance, if it is determined that the quality of data at RFP is Level 1 for all six waste streams, the number of samples required would drop to 141. On the other hand, if the regulatory agencies decide that all of the existing data for RFP is poor (Level 4), the number of samples required would increase to 1908. Extrapolating these numbers to the 800,000 drums of waste intended for disposal at WIPP would mean only six extraction/analysis units would be required in the best case (Quality Level 1) and seventy-seven extraction/analysis units would be required in the worst case (Quality Level 4).
This insight highlights the second value of the model which is communication. Estimates generated using the model can be used to communicate the consequences of proposed characterization programs to the regulatory agencies and the public. In this example, it is clear that the extent to which existing data can be used in lieu of characterization significantly impacts the ultimate cost of the DOE's characterization program. This impact can be communicated to the regulatory agencies during negotiations. The DOE is in a much better position to negotiate with the regulatory agencies if it knows the consequences of future programs.
Finally, as a decision tool, the model provides the DOE with information about the need for analytical laboratory facilities. Most of the major TRU waste storage sites in the U.S. have requested funding for additional analytical laboratory space for processing TRU waste samples. Using this model, the DOE can perform a comparative analysis of the need for laboratory space at the various sites. With this information, the DOE can decide how best to meet the need.
CONCLUSION
The model described in this paper provides the DOE with a method for predicting the present and future analytical requirements for waste characterization. It has been implemented in an integrated software package that is connected to a database of information about TRU waste in the U.S. Using this package, the DOE can set guidelines for TRU waste characterization at the sites and allocate resources to meet its waste characterization needs.
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A PROPOSED RISK-BASED RADIOLOGICAL CHARACTERIZATION METHODOLOGY FOR SOIL AND SEDIMENT INVESTIGATION DERIVED WASTE
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ABSTRACT
Environmental field investigations, accelerated cleanup actions, and construction activities result in the generation of residual materials that may be contaminated with chemical and/or radiological constituents. Residual materials/waste may include drilling mud, soil and sediment drill cuttings, water and soil treatment residues,

and excavation spoils, and may be referred to as environmental media. The management of such material must be protective of human health and the environment, and comply with all applicable regulations.

At the Rocky Flats Environmental Technology Site (RFETS), a Department of Energy facility managed and operated by EG&G Rocky Flats, Inc. in Colorado, approval was obtained from Region VIII Environmental Protection Agency and the Colorado Department of Public Health and Environment to characterize soil and sediment investigation derived materials/waste (IDW) for chemical contaminants using a risk-based determination. In order to ensure that a comprehensive waste characterization is conducted for all identified contaminants in the IDW, a determination must be made concerning the radiological constituents present. Presently at RFETS, radiological waste characterization is performed according to the No-Radioactivity-Added (NRA) Program, in accordance with the Performance Objective for Certification of Non-Radioactive Hazardous Waste as required by the Department of Energy Office of Waste Operations, Environmental Restoration and Waste Management (EM-30). The NRA Program requires the radiological waste characterizations be performed using process knowledge for non-liquid and/or non-homogeneous materials that are suspected of volume/bulk contamination. Using process knowledge as the sole method for the radiological characterization of IDW and similar environmental media precludes the use of a significant amount of analytical data which has been or may be collected in association with the waste materials.

An alternative methodology for the radiological characterization of environmental media has been proposed which takes advantage of best available technology and statistical analysis of sampling data populations. This approach is similar to the risk-based characterization for chemical contaminants that was developed and implemented for IDW at RFETS. The proposed method stresses the utilization of analytical data results to perform a radiological characterization of the waste material. The method incorporates a conservative sampling data screening technique which compares results against background screening levels for radioisotopes of concern. Sampling data results greater than the established background screening levels are compared to human health risk-based concentration values. Material which exhibit an unacceptable risk are characterized as radioactive. With the implementation of this proposed method, a significant reduction in the amount of low level and mixed waste resulting from environmental restoration and construction activities will be realized as compared to the more conservative methodology.

INTRODUCTION

RFETS is a DOE facility managed and operated by EG&G Rocky Flats, Inc. The primary mission of the Site, from its inception until the early 1990s, was to produce metal components for nuclear weapons. Both radioactive and non-radioactive wastes were generated during the research and production processes, and past waste handling procedures involved both on-site storage and disposal of wastes. In 1992, the primary mission of RFETS changed from weapons production to stabilization of the radioactive material remaining on the site and environmental restoration.

In the late 1980's, DOE/Rocky Flats Field Office (RFFO) began environmental investigation to determine the extent of contamination at the site. As a result of these investigations, by 1993 a backlog of over 4,000 drums of IDW had accumulated at the site. Originally the IDW generated during environmental investigations was to be stored near its point of generation, and disposed of in accordance with the Record of Decision (RDO) for each individual Operable Unit (OU). However, due to the lengthy investigation period leading to each ROD, alternative means of handling the IDW were required. The alternative selected was to package the IDW and manage the drums according to appropriate waste handling requirements.

Purpose

This paper presents a proposed methodology for the radiological waste characterization of soil and sediment IDW utilizing regulatory guidelines and risk-assessment methodologies. The proposed methodology includes a proceduralized technique that utilizes site historical information, i.e. "process knowledge", available sampling data, a background comparison, and risk-assessment equations to characterize the radiological component of the soil and sediment IDW. A risk-based methodology for the characterization of chemical nonhazardous and hazardous waste has previously been developed.

Background

An agreement was reached between EG&G, DOE/RFFO, and the Colorado Department of Public Health and Environment (CDPHE) in February 1994 concerning the waste compliance issues related to the IDW drums. Inclusive to this agreement was that a waste characterization of the drums would be performed for the radioactive constituents. EG&G indicated that the radiological waste characterization was to be performed according to the Performance Objective for Certification of Non-Radioactive Hazardous Waste (herein referred to as the Performance Objective), as required by the DOE Office of Waste Operations, Environmental Restoration and Waste Management (EM-30). The RFETS No-Radioactivity-Added (NRA) Waste Verification Program (herein referred to as the NRA Program) was developed in response to the Performance Objective. The NRA Program presently has limited approval for the application to surface contaminated and suspect bulk/volume liquid-homogeneous contaminated waste (DOE/EM-30 Memorandum to DOE/RFFO 1994). However, except for the use of process knowledge, the approval did not extend to media similar to the soil and sediment IDW, except for the use of process knowledge.

DOE Headquarters (EM-30) will allow RFETS to amend the approved NRA Program to include non-liquid and/or non-homogeneous materials that are suspected of volume/bulk contamination. Therefore, a proposed methodology which provides for a risk based radiological characterization of the soil and sediment IDW will be submitted to DOE/EM-30 as a revision to the NRA Program.

REGULATORY GUIDANCE

The scope of the Performance Objective includes those wastes originating in Radiologically Controlled Areas (RCAs) or areas where there is a potential for radiological contamination. Such areas have also been defined as Radioactive Material Management Areas (RMMAs). Soil and sediment IDW are subject to the Performance Objective in that historical information concerning the operation of RFETS indicates environmental contamination by radioactive materials. According to the DOE, the IDW drums must meet the requirements for unrestricted release in order to be considered for non-radioactive disposal. The Performance Objective defines unrestricted release as:

> A release of property (e.g., waste), based on a formal, documented decision reflecting risk-based standards and associated implementing procedures, that the property must be utilized, treated, or disposed of by any party without concern for radioactive content.

In order to classify the material as available for unrestricted release, a program must be established which clearly defines how the Performance Objective is to be met. The program must include process knowledge, radiological survey, sampling, and analysis, shipping and handling, quality assurance, training, and records retention requirements. As the RFETS NRA Program has been approved as meeting these requirements, the proposed risk-assessment methodology will be amended to the existing program. The focus of the proposed methodology regards sample analysis and data management, a background comparison, and the risk-based analysis in the characterization of the IDW.

Regulatory guidance available for the radiological waste characterization of environmental media and materials similar to the IDW directs the generator to 1) identify the bulk/volume radioactive contamination concentrations, and 2) perform a risk-based assessment that quantifies the contributing dose to the public and the environment due to the radioactive constituents. This is seen in DOE Orders, Environmental Protection Agency (EPA) remediation guidance, and proposed rulings by DOE and EPA for regulatory requirements. Although the Performance Objective does not specifically allow for the release of waste characterized by risk-based levels, DOE/EM-30 does recognize the ongoing effort to generate such release limits which have an insignificant impact to the public health and the environment (DOE Performance Objective, 1991).

The EPA has provided guidelines for the cleanup of radioactively contaminated sites in The Radiation Site Cleanup Regulation - Preliminary Draft (EPA, 1994) The preliminary draft of Title 40 Code of Federal Regulations (CFR) Part 196 provides detailed guidance concerning remediation of radioactively contaminated sites. The proposed rule sets standards that limit the radiation dose received by members of the public from a formerly contaminated site. The limit is an annual committed effective dose of 15 mrem/yr in excess of natural background radiation levels for 1000 years after completion of the cleanup. The annual committed effective dose of 15 mrem/yr corresponds to a lifetime excess cancer risk of less the 3 x 10-4 over a

thirty year exposure period (EPA, 1994). The proposed dose limit is intended for the protection of the reasonably maximally exposed (RME) individual in the population at risk. Through the application of conservative, multiple exposure pathways and land use scenarios, a worst case contributing dose can be derived. The derivation of the contributing dose demonstrates the application of the As Low As Reasonably Achievable (ALARA) concept, which becomes inherent to the site cleanup process. The proposed ruling applies to radiologically contaminated environmental restoration clean-up actions. As such, it would be appropriate to apply the same requirements to the development of release limits for bulk/volume materials containing residual radioactive materials, the two subjects being essentially the same.

The Department of Energy Order 5400.5, Radiation Protection of the Public and the Environment, contains guidance concerning the generation of risk-based soil guidelines. The Order specifies the use of the Residual Radioactive Material (RESRAD) computer code to develop the risk-based soil guidelines. The code applies conservative multiple pathways with a long term exposure, residential use scenario. Derived soil guidelines are to be based upon a contributing dose of 100 mrem/yr, excluding background and medical contributions, with the application of ALARA. DOE's proposed rule 10 CFR 834, Radiation Protection of the Public and the Environment, Subpart D, Release of Property Having Residual Radioactive Material, continues the application of authorized limits that are arrived at through dose assessment techniques. The primary receptors for the residual radioactive material are isolated to the most probable receptors (the general public) and doses are not to exceed primary or supplemental limits. The application of worst plausible use and actual/likely use scenarios are also required. The dose limit is initially set at 100 mrem/yr with the application of ALARA. The proposed ruling stresses a reduction in dose to less than/or equal to 15 mrem/yr.

The EPA provides further guidance in the development of Preliminary Remediation Goals (PRGs) in the guidance for conducting baseline risk-assessment for Superfund sites (EPA, 1991). The risk-based PRGs are calculated for the identified contaminants of concerns that are suspected or known to be present at a site. These may be based on multiple use scenarios, with the site specific information supplementing the calculations. Exposure assumptions, toxicity information, and the development process of the calculations must be clearly stated. The methodology of the EPA guidance on the calculation of PRGs is very similar to DOE guidance in that worst case exposure scenarios are clearly established, with exposure pathways and parameters defined within conservative assumptions.

IDW RADIOLOGICAL CHARACTERIZATION PROCESS

The remainder of this paper provides a descriptive of the method for performing the radiological risk-assessment of the soil and sediment IDW. Using this method, radiological concentrations in soil and sediment IDW may be characterized either by analytical results from corresponding environmental investigation samples or by drum-specific samples. The IDW drums covered by the proposed process have previously been chemically characterized to determine if they were RCRA hazardous, or nonhazardous. The results of the IDW Radiological Characterization will be combined with this information to complete a comprehensive waste characterization.

Radionuclide Information Determination

Prior to the background comparison or the risk-assessment, a determination of the available radioanalytical data will be made. A decision will be made as to the suitability of the available data, or lack thereof, to perform the radiological characterization of the soil and sediment IDW. Information will be considered sufficient when each container has at least one validated analysis for each of the selected radionuclides or when it is determined that any missing data do not impact the characterization. It is at this point that process knowledge will play a key role in the suitability determination. As an example, if historical information does not indicate that the location from which the IDW originated is in a source area (i.e. an area not considered to be radiologically contaminated), the material will be considered "clean". However, consideration must be lent to determining the appropriateness of the assumptions made regarding process knowledge. It is essential to verify that the process knowledge is complete and reliable, as any misinformation could lead to a fallacy concerning the radiological characterization.

If sample and IDW drum associations are available, laboratory analysis results will be obtained electronically from the Rocky Flats Environmental Data System (RFEDS). For IDW drums with only one analytical result, that value will be used to

characterize the drum. If there is more than one reported analytical result for a particular radionuclide, the average concentration will be calculated. The average concentration is calculated by summing the results (both negative and positive) for the radionuclide and dividing by the number of results.

Background Comparison

In order to determine if a material may be suspect of contamination, a comparison will be made to an established background concentration for each radionuclide. This effort will determine if the soil and sediment IDW exhibits the potential to be radioactive waste. The Performance Objective defines a radioactive waste as:

> Contains a measurable increase in radioactivity (at a statistically defined confidence interval) above background in volume or bulk resulting from DOE operations except for waste specifically exempted by the EPA, DOE, or NRC regulations.

and,

> All procedures and criteria must be aimed at determining is there is any measurable increase in radioactivity (at a statistically defined interval) above background from DOE operations.

The Performance Objective provides criteria for determining the radioactivity of materials similar to IDW as follows:

> For potentially radioactive soils, comparing analytical results for the suspect soil to analytical results for native soils from areas well outside suspected contamination areas.

Background radiological concentrations for the soil and sediment IDW have been determined for the Rocky Mountain Region surface soils. Background surface soil samples were collected along the Front Range and in the vicinity of the Rock Creek Drainage area. RFETS is located to the east of the Front Range. The Rock Creek Drainage area is located in the northwest corner of Site, in the Buffer Zone. The prevailing winds for RFETS are from the northwest, inferring a nominal or imperceptible impact to the sampling locations due to their location upwind from RFETS. This infers a limited redistribution of contaminants from Site operations. The soil structure and the texture of the selected sampling sites are similar to that found in many locations throughout RFETS. This serves to duplicate the presence and/or mobility of analytes that would be affected by soil and sediment IDW characteristics.

The Rocky Flats Method (RFM) was used to collect the surface soil samples. The RFM involves the collection and compositing of ten discrete soil samples collected within a polygon area, using a sample jig and scoop. The jig collects with a sample configuration of 10 cm x 10 cm x 5 cm deep. This method allows for the collection of undisturbed soils, preserving the surface soil matrix. This allows for the actual fate and transport mechanisms of the contaminants (actinides, naturally occurring radioactive material, and fission fragments) to be preserved.

Samples collected as part of the background surface soil sampling effort were submitted to procured Vendors for radiochemical analysis. Laboratories (i.e., Vendors) providing analytical services to RFETS are required to meet specific minimal detection criteria and laboratory protocol (GRRASP, 1991).

The Performance Objective requires that defined confidence intervals be established for background. The application needs of the proposed method are to establish radionuclide specific concentration values which can be stated to contain the population mean with a high degree of confidence. The purpose of the background comparison is not to predict the upper limit of the event, i.e. the radionuclide specific concentration, but to establish what is representative of the population with some level of confidence. A 95% confidence level is a reasonable representative of a normal population, and is expressed as the mean of the population of sample size, n, plus two standard deviations, s.

The background surface soil sample data is presented Table I, Native Background Surface Soil Values for IDW Radiological Characterization. The radionuclide, sample mean, standard deviation, sample number, and upper 95% confidence level values are shown.

Risk Assessment Methodology

A conservative risk assessment methodology is proposed that assumes a long-term residential, multiple pathway exposure to the receptor population. Radionuclide specific Risk Based Concentrations (RBCs) are calculated for direct ingestion of soil, dermal absorption, inhalation of suspended soil, external gamma exposure, and

ingestion of home grown food. A target risk level of 1x10-4 is used for an exposure duration of 30 years (6 yr. childhood and 24 yr. adulthood). Unlike the chemical-specific RBCs, the radionuclide-specific RBCs do not use body weight or averaging time. The radionuclide slope factors used for the risk equations do not require corrections for intestinal absorption and lung transfer efficiency.
Most of the intake parameters are standard default values prescribed by the EPA for long-term residential exposure scenarios. For the chemical risk-assessment, the State of Colorado required that time-weighted averages be used for all pathways in order to account for the subpopulation group of children under the age of 6. This is applied in the factors selected for the radiological risk-assessment. The general exposure factors that are applied to the process are:

- Exposure Duration, child (1-6 yr) (EDc) = 6 yr,
- Exposure Duration, adult (7 yr and up) (EDa) = 24 yr, and
- Exposure Frequency (EF) = 350 day/yr.

The exposure factors for soil ingestion are:

- Oral Intake Factor, soil ingestion (IFo) = 1.26x103 g,
- Soil Ingestion Rate, child (1-6 yr) (IRc) = 0.2 g/day, and
- Soil Ingestion Rate, adult (7 yr and up) = 0.1 g/day.

The exposure factors for dermal absorption of soil; for radionuclides only applies to Tritium, all other listed isotopes are metals and therefore not a dermal pathway; are:

- Dermal Intake Factor, absorption from soil (IFd) = 3.47x104 g
- Skin Area (exposed), child (1-6 yr) (SAc) = 4.6x103 cm2,
- Skin Area (exposed), adult (7 yr and up) (SAa) = 7.10x103 cm2,
- Absorption Factor, soil (AB) = 0.5 (unitless),
- Adherence Factor, soil (AF) = 1.00 mg/cm2/event, and
- Exposure Frequency (EF) = 350 events/yr.

The exposure factors for the inhalation of soil particilates are:

- Inhalation Intake Factor, soil (IFi) = 4.41 x 10-2 g,
- Inhalation Rate, child (1-6 yr) (INc) = 0.73 m3/hr,
- Inhalation Rate, adult (7yr and up) (INa) = 0.83 m3/hr,
- Exposure Time (ET) = 24 hr/day, and
- Particle Emission Factor, soil (PEF) = 4.63 x 103 m3/mg.

The exposure factors for the ingestion of homegrown food (vegetables) are:

- Oral Intake Factor, homegrown food (IFf) = TC*461,160 g,
- Transfer Coefficient, soil-to plant (TC) = radionuclide specific, use Bv or Br,
- Ingestion Rate, homegrown food (IRf) - 1.22 x 105 mg/day, and
- Fraction Ingested, contaminated source (FI) = 0.36 (unitless).

And, the exposure factors for external radiation exposure from soil are:

- External Radiation Intake Factor, (IFe) = 24 yrs,
- Gamma Shielding Factor (Se) = 0.2 (unitless), and
- Gamma Exposure Factor (Te) = 1 (unitless).

Radionuclide specific slope factors that quantify the risk per pCi were selected from the EPAs Health Effects and Summary Tables (EPA 1994). The units for the slope factors are risk/pCi. The more restrictive slope factors for 240Pu, 90Sr, and 233U are used in the risk calculations. Slope factor for 137Cs, 226Ra, and 238U include the contributions from the decay progeny. Radionuclide specific slope factors are shown in Table II, Radionuclide Specific Slope Factors.
The general risk equations are:
(1) Risk = Intake * Slope Factor (SF), with
(2) Intake = Intake Factor (IF) * RBC.

To calculate the target excess individual lifetime risk of cancer (TR):
(4) TR = Intake * SF = (IF*RBC) * SF.
For the following RBC's, the target risk is set equal to 10-4. The Intake Factor (IF) and the Risk Based Concentration (RBC) equations have been developed for the five pathways. The equation for the soil ingestion pathway RBC is:
(5) Soil Ingestion (IFo) = [(IRc)(EDc) + (IRa)(EDa)](EF),
with the RBC calculated as:
(6) RBC Soil Ingestion (RBCs) = (TR)/{[IRc)(EDc) + (IRa)(EDa)](EF)SFo)}.
The equation for the dermal absorption pathway RBC is:
(7) Dermal Absorption (IFd) = [(SAc)(EDc) + (SAa)(EDa)(AB)(AF)(EF)(CF)(Sfo).
The equation for the external radiation exposure pathway is:
(9) External Radiation (IFe) = (ED)[1-Se)](Te)
with the RBC calculated as:
(10) External Radiation (RBCs) = TR/{[(EDc+EDa)(1-Se)(Te)](SFe)}
The equation for the inhalation of soil exposure pathway is:
(11) Inhalation of Soil Particles (IFi) = [(INc)(EDc) +
(INa)(EDa)](CF)(ET)(EF)(SFi)}
with the RBC calculated as:
(12) Inhalation of Soil Particles (RBCi) = [(TR)(PEF)]/{[(INc)(EDc)+
(INa)(EDa)](CF)(EF)(Sfi)}
The equation for the ingestion of homegrown vegetable exposure pathway is:
(13) Ingestion of Homegrown Vegetables (IFh) = [(EDc) + (EDa)](IRf)(FI)(EF)(CF)(TC),
with the RBC calculated as:
(14) Ingestion of Homegrown Vegetables (RBCh) = (TR)/{[(EDc) +
(EDa)](IRf)(FI)(EF)(CF)(TC)(SFo)]
The equation for the total of the five pathways combined RBC is:
(15) RBCt = (TR)/[(IFo + IFd + IFf) x SFo) + (IFi x SFi) + (IFe x SFe)].
The equations are based on the stochastic, carcinogenic effects of ionizing radiation. Stochastic effects (such as cancer or genetic mutation) are an "all or none" effect for an individual. Increasing the dose only increases the probability of the effect, not the severity. Alternatively, nonstochastic effects would be associated with health effects such as cataracts, skin burns (fibrosis), etc. Considerably large doses of radiation are required to cause nonstochastic effects; doses which are highly unlikely to be encountered when handling the IDW or the environmental media.
The risk of cancer is factored in by using the most recent Slope Factors available from the EPA (EPA, 1994). The ingestion and inhalation slope factors are "best estimates of the age-averaged, lifetime excess cancer incidence (fatal and nonfatal cancer) risk per unit of activity inhaled or ingested, expressed as Risk/pCi." External exposure slope factors are "best estimates of the lifetime excess cancer incidence risk for each year of exposure to external radiation from photon-emitting radionuclides distributed uniformity in a thick layer of soil." The risks from ingestion and inhalation are based on an estimate of fatal and nonfatal cancer incidence, while external radiation risk is based on an incidence of cancer risk. RBCs for gross alpha and gross beta/gamma are not calculated, as risk-based analysis is conducted by radioisotope specific information. The total RBC for all five pathways is presented in Table III, Total Risk Based Concentration.
PERFORMANCE OF THE IDW RADIOLOGICAL CHARACTERIZATION
The basic steps for performing the proposed IDW Radiological Characterization can be clearly stated as follows:
Step 1:
Determine which samples can be used to evaluate each drum, or determine if the drum did not originate from a source area. As stated previously, this may require no further characterization effort due to process knowledge.
Step 2:
Perform evaluation of the analytical data including data usability, flagging and discarding any non-qualified values. Determine if single sample points or multiple samples exist which may be utilized to characterize the IDW drum. If multiple samples exist, calculate that average concentration of the samples by summing the values (negative and positive) and dividing by the total number of samples.
Step 3:
Calculate the RBCs for each of the selected radionuclides using the equations with the given intake parameters, exposure pathways, and slope factors. It is important

to calculate each individual RBC so that contributing factors may be identified and evaluated. Calculate the total RBCs for all five pathways.
Step 4:
Compare the mean, or the single data point, of the radionuclide specific analyte to the corresponding background surface soil level. If any single analyte exceeds the background level, then perform a risk-assessment of the material. If no analytes exceed the background levels, then the IDW drum requires no further radiological characterization.
Step 5:
Calculate the radiological risk ratio for the IDW drums which exceed background for any of the selected radionuclides. This is accomplished by initially subtracting the background value from the corresponding analyte of concern, to quantify the material which is the contributing dose. This value is than divided by the radionuclide specific total RBC for all pathways. The ratios are then summed, with any drum exceeding a value of one being considered unavailable for unrestricted release, i.e. radioactive waste.
After completing the steps, a final decision will be made regarding the IDW drum contents. The results of the Radiological Risk Characterization will be combined with those from the previous chemical characterization effort. Based upon the results, it is assumed that the IDW drum will be characterized as one of the following:
Case No. 1:
If the soil and sediment IDW was previously determined to exhibit the characteristics of corrosivity, ignitability or reactivity or exceeded the adjusted Toxicity Characteristic Leaching Procedure (TCLP) regulatory level, and/or the RCRA risk analysis criteria, but did not exceed the radiological risk analysis criteria, then the IDW will be classified as a RCRA hazardous waste and must be managed and disposed of accordingly.
Case No. 2:
If the soil and sediment IDW was previously determined to exhibit the characteristics of corrosivity, ignitability, or reactivity, or exceeded the adjusted TCLP regulatory level and/or the RCRA risk-analysis criteria, and exceed the radiological risk-analysis criteria, then the IDW will be classified as a RCRA mixed (RCRA hazardous and low-level radioactive) waste and must be managed and disposed of accordingly.
Case No. 3:
If the soil and sediment IDW was previously determined not to exhibit the RCRA characteristics as discussed for Case No. 1 and Case No. 2, but did exceed the radiological risk-analysis criteria, then the IDW will be classified as a low-level radioactive waste and must be managed and disposed of accordingly.
Case No. 4:
If the soil and sediment IDW was previously determined not to exhibit the RCRA characteristics as discussed for Case No. 1 and Case No. 2, and did not exceed the radiological risk-analysis criteria, the then IDW will be classified as chemically and radiologically nonhazardous and may be disposed of as an unrestricted release.
SUMMARY
RFETS anticipates that the proposed Risk-Based Radiological Characterization method for the soil and sediment IDW will be submitted to DOE Headquarters in the Spring of 1995. If the method is approved, it will be applied to the IDW drums using exiting sampling and analysis data. It is essential that the method reflect current federal and state requirements regarding the release of bulk/volume materials that may contain residual radioactivity. However, the ultimate goal of the program will be to meet appropriately approved guidelines while protective human health and the environment.

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LESSONS LEARNED IN COMPILING A COMPREHENSIVE INVENTORYOF RADIOLOGICAL AND NONRADIOLOGICAL CONTAMINANTS IN WASTE BURIED IN THE SUBSURFACE DISPOSAL AREA OF THE INEL
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ABSTRACT
The Subsurface Disposal Area (SDA) at the Department of Energy's (DOE's) Idaho National Engineering Laboratory is undergoing environmental remediation. A baseline risk assessment of the SDA is planned. To support the risk assessment, a comprehensive inventory has been compiled of the radiological and nonradiological contaminants in the disposed waste. The inventory was compiled by a new approach that stressed obtaining process knowledgeinformation about the processes that generated the waste. The paper reports numerous lessons learned, in compiling the inventory, that may be useful to other disposal sites undergoing remediation. Included are insights on the reliability of shipping records, creative approaches to obtain missing information, and methods for checking the reliability of the new inventory.
BACKGROUND
The DOE's Idaho National Engineering Laboratory (INEL) is on the National Priorities List of Superfund sites under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). A Federal Facility Agreement and Consent Order (FFA/CO) governs activities conducted to investigate and, as necessary, remediate actual and potential release sites at the INEL.
One waste area group (WAG) under the FFA/CO is the Radioactive Waste Management Complex (RWMC). The RWMC SDA has been the INEL's primary disposal site for solid radioactive waste since opening in 1952. Transuranic and low-level wastes, including mixed wastes, have been disposed of in more than 90 pits, trenches, and rows of soil vaults, covering 88 acres.
A CERCLA baseline risk assessment (BRA) is planned under the FFA/CO for the pits, trenches, and soil vaults in the SDA. Knowledge of the identity and quantities of radiological and nonradiological contaminants (including uncertainties) in the buried waste is essential for preparing the BRA.
It is not feasible to quantify directly the contaminants in a large, highly heterogeneous disposal site such as the SDA by conventional means, i.e., drilling, sampling, and performing laboratory analyses. Therefore, an inventory of the radiological and nonradiological contaminants in the waste disposed of in the SDA was compiled (1) from many information sources for use in the BRA. The inventory includes information on the physical and chemical properties of the contaminants to help postulate release mechanisms and rates. The inventory information will also be used in engineering and safety studies of remediation alternatives.
PURPOSE OF PAPER
This paper describes the new, comprehensive method that was used to estimate the inventory of radiological and nonradiological contaminants in the buried waste. The results of the inventory compilation are discussed briefly. The paper then lists the lessons learned in performing this complex inventory task. The lessons may prove useful for personnel involved in similar inventory tasks at other environmental remediation sites.
SHIPPING RECORDS
Historically, shipping records prepared at the time of disposal have been the primary source of inventory information on disposed waste. The shipping records for the waste disposed of in the SDA have been compiled into a database known as the Radioactive Waste Management Information System (RWMIS). Prior to the development of the new inventory reported in this paper, RWMIS was the key source of information for the waste in the SDA.
When waste disposal began at the SDA 43 years ago, requirements and practices were much less rigorous than the current requirements for waste characterization. Consequently, complete information about the waste was not recorded on shipping records when it was generated. Many of the shipping records contain only minimal or incomplete information. For example:
- Little information was included on nonradiological contaminants.
- Many shipping records listed only generic radioactivity terms (e.g., mixed fission

products).
Many shipping records listed only some of the radionuclides that are present. Often, only the one or two radionuclides that contributed most to the radiation level during shipment were listed (e.g., Co-60, Cs-137). The list seldom included low-radiation, long-lived beta/gamma-emitting radionuclides (e.g., Sr-90, C-14, and Tc-99) that could be the primary contributors to the post-disposal risk via environmental transport pathways.
Many previous studies had addressed portions of the contaminant inventory disposed of in the SDA. The studies drew their information mostly from RWMIS and from the shipping records themselves. However, no existing compilation of the inventory was considered adequate for use in the BRA.

THE NEW APPROACH

A new approach was developed that built upon previous studies and provided the information necessary to complete the BRA. The new approach emphasized obtaining process knowledgeinformation about the processes that produced the wasterather than simply re-reviewing the shipping records.
The first step was to divide the INEL and non-INEL facilities that generated the waste into seven groups. Seven corresponding lead data-gatherers were appointed to direct the compilation of the information. In nearly every case, each lead data-gatherer had previously worked at the waste generator location whose information he was assigned to compile, and each was familiar with the operational activities that had generated the waste.
Figure 1 depicts the flow of information in this approach. The data-gatherers compiled information from each generator, using process knowledge, plant operating records, hundreds of technical and programmatic letters and reports (including previously classified information), interviews with present and former operations personnel, and nuclear physics calculations, in addition to the traditionally used information sources of shipping records and RWMIS.
To simplify the collection and use of all this information, the waste from each generator was subdivided conceptually into several waste streams. Basically, a waste stream was defined so as to reduce the nonhomogeneity within a portion of the waste. For example, one stream consisted of all of the beryllium reflectors from the Test Reactor Area at the INEL. Another stream consisted of all of the organic sludges received from the Rocky Flats Plant.
For each waste stream, a detailed data form was completed. The data form covered general information about the stream (generator, building number, years generated, volume, etc.), the physical and chemical form of the stream, the quantities (including uncertainties) and the physical and chemical forms of the radiological and nonradiological contaminants in the stream, and the sources and reliability of the information.
The completed data forms were reviewed, then entered into a new database called the Contaminant Inventory Database for Risk Assessment, or CIDRA. The CIDRA offers a wide range of search capabilities for presenting various rollups of the information.
The inventory information was collected and evaluated in two phases. The first phase covered the waste disposed of from 1952 through 1983 and comprises 232 waste streams. The report documenting that inventory (1) is available upon request. The second phase covered the waste disposed of from 1984 through 1993, as well as the waste projected to be disposed of from 1994 through 2003. The total number of streams is 99. The report summarizing this waste is in preparation. The data from both phases, totaling about 3,000 pages, have been entered into CIDRA.

THE INVENTORY RESULTING FROM THE NEW APPROACH

For the waste disposed of from 1952 through 1983, a total of 100 radiological contaminants and 77 nonradiological contaminants was identified. The radiological contaminants of greatest activity were Co-60, H-3, Fe-55, Cr-51, Ni-63, Cs-137, Sr-90, Pu-241, Ce-144, Mn-54, Co-58, and Am-241. The nonradiological contaminants of greatest mass were sodium nitrate, lead, potassium nitrate, aluminum nitrate nonahydrate, carbon tetrachloride, 1,1,1-trichloroethane, trichloroethylene, and nitric acid.
The new inventory exhibits many changes from the previous estimates of contaminant quantities. Some of the most notable changes are greatly increased quantities of Pu and Am; tritium; and C-14, Tc-99, and I-129.
The quantities of plutonium and americium increased about 10 times over those listed in RWMIS. The new information was based on previously classified, plant-wide

mass balances maintained at the principal generating site for the plutonium and americium in the SDA, the Rocky Flats Plant in Colorado.

The tritium activity is more than 20 times larger than previously estimated, due primarily to identifying a new waste stream containing approximately 1 million Ci of tritium entrapped in the beryllium reflectors from test reactors.

Increases of more than a factor of 1,000 were identified for some long-lived fission products and activation products (e.g., C-14, Tc-99, and I-129). The new results reflect the use of nuclear physics calculations to estimate the activities of radionuclides that are difficult to measure and generally were not reported on the shipping records.

The guidance that implements the CERCLA regulations requires the estimation of bounds on the contaminant quantities. The upper and lower bounds were developed by examining in detail the sources of uncertainty for each contaminant in each stream, estimating these uncertainties, and then combining and propagating them to obtain an overall uncertainty for the estimated total quantity of each contaminant, using standard statistical techniques. The bounding values are analogous to 95% upper and lower confidence limits on the estimated total quantity of each contaminant.

As a completeness check, the resulting inventory of contaminants was compared in detail against inventories found in RWMIS and in previous studies. As expected, the new inventory was much more extensive than previous compilations, particularly for nonradiological contaminants. The new inventory was also compared against the results obtained from sampling and analysis in environmental monitoring conducted at the RWMC, to determine if contaminants detected at the RWMC are also in the inventory. Reasons were determined for differences between the new inventory and other inventories.

CONTRIBUTIONS MADE AND LESSONS LEARNED

Far from being a routine process, the compilation of the inventory resulted in many new contributions to technical knowledge about the waste and many lessons learned. The lessons may benefit other organizations faced with compiling inventories at large burial sites in a complex environment of waste information records (or lack thereof).

The principal contributions and lessons learned are discussed below. The sequence in which they are listed is that in which the topics might be encountered during the compilation of an inventory of this type. The lessons include programmatic considerations, data management considerations, and technical considerations. Some of the lessons may seem obvious, but the subtleties involved became apparent as the information was being gathered.

1. The scope of the inventory to be compiled needs to be defined very precisely in terms of the time period covered, the disposal locations included, and the types of waste addressed.

2. The exact types of information to be gathered need careful definition and thorough documentation. All of the data-gatherers (no matter how experienced and highly educated) were required to complete a training course on exactly what information was to be collected and how the information was to be entered onto the data form. This approach minimized inconsistencies in data reporting from one generator to another and made possible meaningful comparisons and rollups of the data across multiple generators.

3. It was important that the data-collection approach accommodate major differences among waste generators in the types of records available. For some generators, the shipping records were virtually the only source of information. For other generators, the shipping records were of little value and other sources were used almost exclusively. In retrospect, it would have been a serious mistake to require that all data-gatherers follow exactly the same process. The selection of data sources was left to the judgment of the individual lead data-gatherers.

4. The waste generators represented several DOE contractors at the INEL (reporting to three Operations Offices), several other DOE sites, and non-DOE organizations. The data-gatherers represented several contractors. Therefore, special management coordination was necessary to deal with the differing work priorities and different recordkeeping systems of the various organizations. It was important that the time of the data gatherers be committed to completing the task, and that their management support the commitment. It was also important to advise each generator that they would be expected to review and approve the results of the task. In this way, all parties, including regulatory agencies, can place confidence in the results.

5. Numerous instances of inaccurate shipping records were found. One extreme example was a 1965 shipment that was recorded as containing 700 Ci of U-235 (constituting a total mass of 327,000 kg and a definite criticality issue if the quantity were correct!). Investigation of the operating records revealed that the shipping record should have read 700 Ci of mixed fission products with a trace of U-235. "Sanity checks" of shipping recordscomparing the listed radioactivity against the radiation level, the volume and weight of the shipment, etc.are strongly recommended.
6. Many types of process knowledge provided information on the waste. Incident reports, annual reports, health physics logs, in-process chemical inventories, and other types of documents were useful.
7. Important information about the waste from two generators had been classified for security reasons. In both cases, the data-gatherers worked successfully with generator personnel to produce nonsensitive summaries of the classified information and get those summaries declassified for use in the inventory report.
8. Interviews with former plant operators were helpful, but the results were used with caution. Operators' memories of events that occurred long ago tend to fade with time and can become distorted. Some former operators may even see an interview as an opportunity to "grind an ax" and put forth their personal view of what is really in the waste.
9. Nuclear physics calculations were essential to identify radionuclides not listed on the shipping records but expected to be present based on other reported radionuclides. (For example, the shipping record might list only U-235, but U-234 and U-238 also are almost always present, in ratios that depend on the enrichment of the uranium.) Varying amounts of information on the nuclear core operating histories and the materials of construction were available. This allowed the use of a wide range of nuclear physics evaluations to arrive at a more complete and accurate inventory of the radionuclides in the waste.
10. The waste generators' projections of future waste disposals were found to be systematically biased. Statistical evaluation of several recent years' projections versus actual disposals indicated an average upward bias of a factor of four in the radioactivity. Accordingly, current projections of the radioactivity in future disposals were reduced by a factor of four.
11. A method was devised to deal with situations in which a contaminant was known to be present but the quantity was not known. Such entries were dealt with outside CIDRA, separate from the entries with defendable estimates of the quantity. For the unknown entries, upper-limit estimates were made to allow comparison with the totals for the known quantities. This approach allowed a determination on whether the unknown quantities were important uncertainties.
12. Substantial effort was devoted to converting all generic radioactivity terms (e.g., mixed fission products) into distributions of specific radionuclides for use in the BRA. For such entries, the approach used a different radionuclide distribution for each waste generator, based on nuclear analyses of the waste or of the reactor operations.
13. Systematic corrections were needed to many of the original estimates of the radioactivity in containers of waste. Some of the instruments and procedures used to make those estimates, particularly in the early days, produced data that were biased. A simplified procedure, commonly used at some INEL facilities to estimate the radioactivity in waste based on radiation readings outside the container, had been found in laboratory tests to be biased (2-4). The amount and direction of the bias depend on (a) the geometric relationship of the radiation source and the detection instrument, (b) the density of the fill material inside the container, and (c) the particular radionuclides present compared with those for which the technique was calibrated. Data in the three cited references (2-4) were reviewed. It was concluded that the original estimates of radioactivity that were made using this method should be reduced by a factor of two.
14. Because the BRA calculations are based on the upper bounds of the contaminant quantities, the evaluation of the uncertainties resulting in those bounds is important. A considerable amount of data evaluation and statistical analysis was devoted to identifying and quantifying the sources of uncertainty in the contaminant quantities.
15. If at all possible, it is useful to identify whether each waste stream is RCRA-listed. The information may be important during CERCLA Feasibility Study evaluations and the associated identification of ARARs (Applicable or Relevant and

Appropriate Requirements).
16. A very thorough process of quality assurance, review by waste generator personnel, independent technical review, checking against other inventories, and "sanity checking" of the inventory information was essential. To confirm the completeness of the inventory, one particularly useful check was a comparison against the list of contaminants detected in environmental monitoring conducted at the disposal site.
17. Compiling a defensible inventory of the contaminants in a large waste disposal site that has been used for decades was a lengthy and expensive task. Unplanned-for issues arose frequently, and they often impacted the schedule and cost of the task. Most of the above lessons were learned during the first phase of the effort. The knowledge gained thereby was valuable for conducting the second phase of the inventory compilation.

CONCLUSIONS

For large waste-disposal units undergoing the CERCLA process, a sizeable effort may be needed to compile a defendable contaminant inventory upon which to base risk assessments. Creative approaches based on using a broad array of process knowledge can produce more complete and accurate inventory information than may be obtained from shipping records only. Careful planning and execution of the effort, factoring in the lessons learned at other remediation sites, can result in avoiding many of the pitfalls in compiling the inventory.

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A PC-BASED SOFTWARE PACKAGE FOR MODELING DOE MIXED-WASTE MANAGEMENT OPTIONS

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ABSTRACT

The U.S. Department of Energy (DOE) Headquarters and associated contractors have developed an IBM PC-based software package that estimates costs, schedules, and public and occupational health risks for a range of mixed-waste management options. A key application of the software package is the comparison of various waste-treatment options documented in the draft Site Treatment Plans prepared in accordance with the requirements of the Federal Facility Compliance Act of 1992. This automated Systems Analysis Methodology consists of a user interface for configuring complexwide or site-specific waste-management options; calculational algorithms for cost, schedule and risk; and user-selected graphical or tabular

output of results. The mixed-waste management activities modeled in the automated Systems Analysis Methodology include waste storage, characterization, handling, transportation, treatment, and disposal. Analyses of treatment options identified in the draft Site Treatment Plans suggest potential cost and schedule savings from consolidation of proposed treatment facilities. This paper presents an overview of the automated Systems Analysis Methodology.

INTRODUCTION

Activities mandated by the Federal Facility Compliance Act of 1992 and pressures to reduce budget expenditures led the U.S. Department of Energy's (DOE) Office of Program Integration (EM-33) to develop a model that simulates mixed-waste management activities for the entire DOE Complex. This model is based on the premise that waste management is an interdependent system and should be viewed as a complete system. Decisions on how to manage one part of that system (e.g., treatment at a particular site) must consider potential impacts on other parts of that same system (e.g., disposal at a different site).

The Mixed-Waste Systems Analysis Model (the Model) simulates DOE mixed waste-management as a complete system. It addresses several key requirements that are needed for informed decision making. The Model provides a consistent approach to the analysis of mixed-waste management options, and does not rely upon calculated results that have been developed using widely differing inputs and assumptions. It provides rapid analysis of a large number of options. Average turnaround time for an option analysis ranges from a few hours to a few days, depending upon the complexity of the option. The Model uses enhanced versions of DOE mixed-waste inventory and facility information, including the Mixed Waste Inventory Report and the Site Treatment Plan databases. It provides results in a variety of camera-ready graphical or tabular forms, depending on the requirements of the analysis.

This paper reviews the process used to develop the Model, describes its capabilities, provides examples of its output, and summarizes planned enhancements to the Model.

BACKGROUND

The Mixed-Waste Systems Analysis Model is the automated version of the Mixed-Waste Systems Analysis Methodology that has been under development for the last three years. The Methodology mapped all components of mixed low-level waste management, compiled data on these components needed for options analysis, and provided results of analyses that were performed prior to automation of the Methodology. Components of mixed-waste management that are included in the original Methodology and the automated Model are as follows:

- Storage (both before and after treatment)
- Characterization (both before and after treatment)
- Waste sorting (prior to treatment)
- Certification and preshipment activities
- Transportation
- Receipt and inspection (after any movement of waste)
- Treatment (including any additional pretreatment)
- Solidification/stabilization (produces final waste form)
- Disposal

These waste-management activities are evaluated relative to several criteria. All activities are evaluated in terms of cost, which includes technology development (where appropriate), facility design, permitting, construction and start-up, operations, maintenance, and decommissioning. Each cost component also can be scheduled, with costs distributed over that schedule. Public and occupational health risks associated with each waste-management activity are calculated so that the user can compare costs to risks as a form of cost-benefit analysis. Given that decisions on treatment will be influenced by disposal requirements, long-term disposal is modeled using the Pathways Analysis Model. Options are rejected when releases from the treated final waste form are found to exceed release standards. Finally, options are evaluated qualitatively for potential regulatory compliance issues.

Automation of the Methodology began in May 1994 and remains a work in progress. As of late February 1995, cost and schedule have been fully automated. Given the extensive base of information available when the automation effort began, the level of complexity coded into the Model, and the accelerated pace at which the programming effort has proceeded, the five-to- seven-person team used in automating the Methodology should be assumed to be the bare minimum required for programming

efforts such as this.

DESCRIPTION OF THE MODEL

The Mixed-Waste Systems Analysis Model is a PC-based program built around the Paradox database software package. The Model consists of three major componentsthe option-configuration module, the calculational module, and the reports module. The option-configuration module includes a series of menus that prompt the user in configuring a mixed-waste management option. These menus allow the user to group wastes with a common management "train" and to specify all of the waste-management activities to be included in that train. For example, a typical management train may begin with storage at the generator site, followed by characterization/sorting/certification prior to shipment off-site for treatment, then transportation to the treatment site, followed by treatment, followed by storage then final solidification/stabilization, followed by additional long-term storage, then transportation to the disposal site, and finally disposal. In a different option, the user may choose to configure a completely different management "train" for the same group of wastes. For each activity, the Model uses the volumes and masses of the waste streams involved to determine which existing or planned facilities are viable candidates and displays this information to the user. In the case of treatment and solidification/stabilization, wastes and facilities are matched based on "treatability group" compatibility. A "treatability group" is defined as a combination of radiological-component characteristics, waste matrix, and hazardous-component characteristics. At the completion of the option-configuration module, the option configuration is saved for future reference. Configuration of an option using the Model will take anywhere from a few hours to a few days to complete, depending on the complexity of the option.

Once all mixed-wastes have been accounted for, the user can instruct the code to calculate costs and develop schedules using the calculational module. No additional user input is required to run this module. The code automatically sums all wastes to be processed at each facility, sizes the facility based on the inventory work-off period specified by the user, calculates costs and develops schedules for that facility based on the treatment technology specified, and stores results in a detailed form to maximize reporting capabilities. Computer run times for these calculations range from 10 to 50 minutes, depending upon the complexity of the configured option.

The reports module provides the user with options for reporting cost/schedule results in either a graphical or tabular form. Results can be presented in any combination of the following cost/schedule accumulators:

- Waste-management activity
- Waste operation (preoperations, construction, operations and maintenance, and decontamination and decommissioning)
- Waste matrix (e.g., all costs for treating aqueous wastes)
- Site where activity will occur
- Generator site (allows costs for all activities required to manage a group of waste to be assigned back to the original "caretaker" of the waste)
- Treatment facility
- Year activity will occur.

The reports module provides a menu for the user to select the structure of the desired report. This menu gives the user the capability to specify up to 10 configured options on any report. Computer run time for each report is 1 to 2 minutes, once all configured options to be included in the report have been run through the calculational module.

EXAMPLES OF ANALYSES PERFORMED

The Mixed-Waste Systems Analysis Model has been applied to a variety of complexwide and site-specific applications. An example of the graphical presentation of results is provided in Fig. 1. An example of the tabular presentation of results is provided in Table I.

Fig. 1.

TABLE I

ENHANCEMENTS TO THE MODEL

Enhancements for the Mixed-Waste Systems Analysis Model could include addition of selected modeling of mixed-transuranic and high-level waste management and the automation of public/occupational health risk calculations and disposal-facility long-term performance. The cost and schedule modules of the Model will be refined

and further integrated as required by Model users. There are plans for a controlled distribution of the Model and for a user's network to share results of options analyses. Longer term, there are plans to include uncertainty analysis for cost and risk. There are also plans to expand mass/energy balance calculations that support the cost calculations for treatment.
In terms of the databases used by the Model, there is a continuous effort to expand and refine these databases based on information that becomes available. Updated databases will be included in each new version of the Model.

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A SYSTEMATIC APPROACH FOR FUTURE SOLID WASTE ACTIVITIES AT THE HANFORD SITE
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ABSTRACT
The Hanford Site, an area of approximately 560 square miles in south central Washington State, was a major producer of plutonium beginning in the 1940's and lasting for nearly half a century. The production of plutonium generated large volumes of radioactive and hazardous wastes in both solid and liquid forms. Today, the U.S. Department of Energy (DOE) is in the process of restoring the Hanford Site. As defined in the Hanford Mission Plan (DOE 1993), "The primary Hanford mission is to clean up the Hanford Site, eliminate potential risks to the public and our workers, and serve as the DOE model in environmental restoration."
This paper describes the systematic approach to the treatment, storage, and disposal system (TSD) planning and management that has been developed and implemented by Hanford's Solid Waste Program.
The systematic approach includes:
- collecting the forecast and waste inventory data
- defining Hanford's TSD system
- studying and refining the TSD system using analysis tools
- documenting analysis results.

The customers responsible for planning, funding, and managing future solid waste activities have driven the evolution of the solid waste system.
Currently, all treatment facilities are several years from operating. As these facilities become closer to reality, more detailed systems analysis and modeling will be necessary to successfully remediate solid waste at the Site. The tools will continue to be developed in detail to address the complexities of the system as they become better defined. The tools will help determine which facility lay-outs are most optimal, will help determine what types of equipment should be used to optimize the transport of materials to and from each TSD facility, and will be used for performing life-cycle analysis. It is envisioned that in addition to developing the tools to be adapted to the more specific facility design issues, this approach will also be used as an example for other waste installations across the DOE complex.
INTRODUCTION
The Hanford Site, located in south central Washington State, was a major producer of plutonium beginning in the 1940's and lasting for nearly half a century. The production of plutonium generated large volumes of radioactive and hazardous wastes in both solid and liquid forms. Today, the U.S. Department of Energy (DOE) is in the process of restoring the Hanford Site. As defined in the Hanford Mission Plan, "The primary Hanford mission is to clean up the Hanford Site, eliminate potential risks to the public and our workers, and serve as the DOE model in environmental restoration" (DOE 1993).
As part of the restoration mission, the Westinghouse Hanford Company (WHC) Solid Waste Program is responsible for the management and final disposition of the solid wastes generated or received at Hanford.* The purpose of this paper is to present an overview of the Program's approach to analyzing and planning for future waste management activities.
As described in this paper, there are three major components to the Solid Waste Program planning approach: data collection; a treatment, storage, and disposal (TSD) system; and analysis tools. The results from development and use of these components are used by "customers" and Westinghouse Hanford Company (WHC), which operates the

Program, to successfully manage solid waste at Hanford and integrate system planning with other DOE installations.

As part of this overview, Section 2.0 outlines the types of customers that use this information and describes how these customers have influenced the analysis process. Section 3.0 provides an overview of the solid waste planning and management concept, including:

- collecting the forecast and waste inventory data
- defining Hanford's TSD system
- studying and refining the TSD system using analysis tools
- documenting analysis results.

Section 4.0 discusses the plans for improving the understanding of the solid waste management system.

CUSTOMERS OF THE DATA

The customers responsible for planning, funding, and managing future solid waste activities have driven the evolution of the solid waste planning and management concept. The current customers represent a wide array of individuals from different programs, ranging from representatives of the DOE-Headquarters to specific Hanford cognizant engineers who are documenting the design requirements for the future solid waste TSD facilities. As new customers are identified and additional requirements are recognized, new analysis techniques and tools will be applied to meet customer requirements. A summary of the main customers and their data requirements follows:

DOE-Headquarters and DOE-Richland Operations Office. DOE program managers request information from each DOE installation on the types of radioactive waste that is currently stored at the Site and how much is anticipated to be generated at, and/or shipped to, the Site in future years. This information is requested to determine how large the DOE waste management system is and the magnitude of waste volumes being generated at each installation. The information is transmitted through formal data requests, personal visits, and on an ad hoc basis.

Facility Planners. Facility planners are outlining the requirements of future TSD facilities. To perform this work, these planners require insights into how much waste is currently at the Site and how much will be generated in the future. In addition, these planners need specific waste characteristic information to help determine the types of examination equipment, treatment technologies, and disposal grounds that will be necessary for waste characterization and final disposition.

System Analysts and Solid Waste Program Managers. Hanford solid waste system analysts use the data to support system-wide TSD planning and to ensure all waste forms have a defined waste management path, eventually reaching onsite or offsite disposal. These data are also used to test the system sensitivity to alternative planning options. By understanding future TSD requirements, the information is also used by the Program to justify funding for facilities that are required to safely and expeditiously manage solid waste.

Financial Analysts. The financial system uses the data to set billing rates for onsite and offsite waste generators who will be sending waste to Hanford's Solid Waste Management System over the next year.

A SOLID WASTE PLANNING AND MANAGEMENT CONCEPT

Management of Hanford's solid waste is a complex mission. Many factors impact the success of this mission: complying with all laws and regulations; identifying technologies that are feasible and available; identifying future waste types and volumes that will be managed at the Site; ensuring worker and public health and safety; ensuring funding is available to carryout the mission; and identifying and managing uncertainties associated with future program missions, waste volumes, and funding appropriations. This section describes the systematic approach that has been implemented to address design and planning issues associated with Hanford's complex solid waste TSD system. Figure 1 illustrates the primary system components that constitute the solid waste management approach. Each of these components is described in the following subsections.

Fig. 1.

Waste Volumes

Each year information is collected on how much solid waste will be generated by and sent to Hanford in the future. This information is requested from the specific waste generating sites and is maintained in an evolving electronic database. Hanford also has large volumes of solid waste that were generated in the past and are currently being stored in trenches, caissons, and buildings. Waste characteristic information

for the waste currently stored at the Site has been maintained in a separate electronic database.
The information that is collected and maintained for both the forecasted waste and the stored waste is driven by the needs of the data users. The types of information currently available include the generating source, level of radioactivity, date of generation or shipment to the Site, volume of waste, the container types used for shipping and storing the waste, the physical matrix of the waste contents, and the hazardous constituents within each container.
The waste characteristic information serves several purposes. The information aids in determining waste management handling requirements. For example, large odd-shaped containers may require special devices to load the material to and from storage, disposal, and treatment. The specific container types also allow a planner to determine how many packages can be placed in a building and how the material can be stacked and organized. Containers that contain specific waste characteristics may need to be stored in special designated areas within a building. The radioactivity level, physical matrix, and hazardous constituents present within a waste package will also dictate the treatment and disposal requirements. In some cases, the waste may need to be shipped offsite for final disposition.
Treatment, Storage, and Disposal (TSD) System Description
A written description of Hanford's TSD system has been developed to narratively capture the future plans for TSD. This information is obtained from technical documentation, facility planners, program managers, and system analysts working within the Solid Waste Program. Consensus meetings are held to ensure that the system description adequately reflects all participants' points of view and documents a technical baseline from which sensitivity analyses can be performed. The latest system description contains multiple storage buildings, several treatment facilities, and both onsite and offsite disposal sites. Figure 2 illustrates the current state of the planned system, while Fig. 3 portrays Hanford's vision for the solid waste management system in the future. The vertical columns in the figures identify the waste sources, storage areas, characterization and treatment facilities, and disposal sites. Many of these components of the TSD system do not yet exist but will be constructed in phases, with the first major treatment facility (WRAP Module 1) planned to be on-line in 1997 and the last treatment facility scheduled for operations by 2013.
The waste sources shown in Fig. 3 include 22 offsite and 68 onsite waste generators who will be sending waste to Hanford over the next 30 years. Several storage areas are planned to store the waste arriving from these waste generators as well as the work-in-process waste that is awaiting additional treatment or shipment to disposal. Six treatment facilities are planned to treat the waste, and several disposal sites have been identified to receive the waste for final disposal. Each TSD component is subject to environmental regulations and treatment requirements that dictate what type of activities and functions will be performed.
Fig. 2.
Fig. 3.
Once this system has been described to the level of detail required by the customer, the information is documented in flow diagrams. An example of a flow diagram is shown in Fig. 4. These flow diagrams depict the system description in a more structured, analytical method, tracking each waste stream from point of generation to its final disposition. These diagrams provide a flow diagram model of the solid waste management baseline. The flow diagram model is imported into a computer model so that the system can be analyzed and sensitivity studies can be performed.
Fig. 4.
Analysis Tools
Several analysis tools are used to understand and refine Hanford's solid waste management system. Because of the large quantities of data that are collected, databases have been used to maintain and store the forecasted waste information and waste currently in inventory. The databases are updated as volume information is collected. Because this information is maintained electronically, specific information can be extracted to respond to data requests, to provide information to the financial department, and to create an electronic file to be used during modeling activities.
A second tool used at the Hanford Site is a simulation model that is based upon Hanford's TSD system description. The TSD functions are coded within the simulation

model, and the input data from the forecast and stored waste databases are fed into the model to analyze the baseline system and alternative operating strategies. A baseline is established to gain an understanding of how the system may operate in the future.

Once the baseline is understood, alternatives can be evaluated using the simulation model to determine if other operating strategies are more optimal than the baseline. For example, the baseline system may indicate that short falls will exist by the year 2000 in planned storage capacity; therefore, one may want to evaluate alternatives that could alleviate this problem. Model runs could be performed delaying the timing of the waste receipts so that less volume requires storage within a given time period, or increasing the capacity of one of the treatment facilities to process waste volumes waiting in storage more rapidly.

Another reason for investigating alternatives to the baseline system description is to evaluate the system's sensitivity to events not represented in the baseline that have a probability of occurring. For instance, one may want to determine the impacts to the system if the waste volumes received at the Site have the potential to be twice the amount forecasted over the next 10 years. Several alternatives could be analyzed to assess the optimal sizes and startup dates for the TSD facilities. This tool provides the flexibility to easily analyze and study alternatives before designing a full-scale system.

Data obtained from the simulation model are stored in a database that allows a user quick access to the end results. The database allows for model output results of different alternatives to be compared and viewed graphically. Each time an alternative is generated, the new results can be compared to previous results to determine how the TSD system components are impacted.

Results

Several types of formal results from the analysis work are generated and provided to the customers, including results obtained from the collection and organization of the forecast and inventory data and results obtained from the system description and simulation model.

Results From the Forecast and Inventory Data

Due to the large volume of forecast and inventory data currently maintained and the diversity of the customers requesting information, several reports and summaries are issued. The following list identifies a sample of these reports and summaries, and provides a brief description of the information contained in each document.

Solid Waste Forecast Volume Summary Report. The objective of the document is to provide a base line volume forecast that can be used for standard reporting, preparing input files for modeling Hanford's waste management facilities, establishing a basis for billing rates, and responding to special data requests from solid waste facility planners and system analysts.

Treatability Group Summary Report. This document describes the physical waste forms and hazardous constituents for the waste that will be shipped to Hanford's Solid Waste Complex and how these characteristics enable the mixed waste to be identified by nationally established treatability groups. This information provides insight into the treatments and capacities that will be required at the Site.

Container Volume Summary Report. This document describes the containers that will be used to ship waste to the Hanford Site and the volume of waste associated with each container type. This report is used primarily for planning the handling, storage, and disposal requirements at Hanford's solid waste complex.

Stored Waste Data Analysis Summary. This report analyzes the use of stored waste data to determine if past waste receipts provide insight into future waste shipments. It compares past waste volumes with the forecast volumes to identify trends among onsite and offsite waste generators for each waste generator from year to year and for each waste category. The report includes both the forecast data and stored waste data.

Additional Solid Waste Reports. The specific data requirements of the TSD planners often vary from year to year. For this reason, unscheduled reports are frequently generated to supply these data needs. These reports differ in subject matter and level of detail; however, the objective is to provide a documented data set that may be used consistently for short- and long-term planning.

Integrated Database Support. Every year Hanford is required to provide input to the DOE Integrated Database (IDB). A portion of the required information is extracted from the solid waste forecast database.

Results From the System Description and Simulation Model

Reports are developed that document the baseline system description and the baseline model results. Once the baseline assumptions and results are established, the system is analyzed with the simulation model by investigating "what-if" scenarios. The following list identifies the products delivered to the customers as a result of the system description and modeling activities.

TSD (Baseline) System Description. This document describes a cradle to grave approach for achieving the mission to manage Hanford's solid waste. The report includes detailed definitions about each treatment facility, each storage building, and each onsite or offsite disposal site. In addition, the report identifies the facilities' functional requirements, the system's and facilities' interfaces with other systems or facilities, and the operational constraints of the system.

Baseline Model Results and Systems Analysis. Once the baseline system description is established, the system is modeled using a simulation package to indicate how the current system plan meets the projected waste volume demands over the next 30 years. A report is issued that summarizes the results of this simulation. The results that are summarized include facility throughputs, storage requirements, disposal accumulations, and transporter movements. This information is presented in written descriptions, tables, and interim reports as well as through presentations to provide feedback to management and facility planners on the baseline system results. The analysis of the end results may introduce alternatives to be studied.

Alternative Operating Strategies. Feedback obtained from WHC management and facility planners allows for several alternative studies to be conducted to show how Hanford's system could be impacted and/or improved by varying the system components or input data. Such alternative studies have included increasing/decreasing a treatment facility's capacity, adding waste volumes to the system that have the potential of requiring treatment at Hanford, re-routing waste volumes through other treatment facilities, and never opening a disposal site. The information provided for these studies is specific to the study performed but primarily includes treatment facility throughputs and storage requirements.

Integrated Database Support. The simulation model has also been used to respond to the IDB in the past, as defined in Section 3.4.1.

CONCLUSION AND FUTURE VISION

Hanford has developed a systematic approach to accomplishing its mission to remediate solid waste. The four major activities performed toward achieving this mission are:

- collecting the forecast and waste inventory data
- defining Hanford's TSD system
- studying and refining the TSD system using analysis tools
- documenting analysis results and communicating these results to the customers.

Over the next year, Hanford will continue to analyze the solid waste management system by performing the above activities. These activities illustrate to system designers which areas need improvement or attention. Also, these activities assist Hanford in quantifying the complex pieces of the solid waste management system and in identifying those components of the system that can greatly impact the system. Currently, all treatment facilities are several years from operating. As these facilities become closer to reality, more detailed systems analysis and modeling will be necessary to support successful planning of solid waste remediation. The tools will continue to be developed in detail to address the complexities of the system as they become better defined. The tools will help determine which facility lay-outs are most feasible, will help determine what types of equipment should be used to optimize the transport of materials to and from each TSD facility, and will be used for performing life-cycle analysis. It is envisioned that in addition to developing the tools to be adapted to the more specific facility design issues, this approach will also be used as an example for other waste installations across the DOE complex.

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UPDATED VOLUME PROJECTIONS FOR GREATER-THAN CLASS C LOW-LEVEL RADIOACTIVE WASTE

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ABSTRACT

The Department of Energy's (DOE's) planning for the disposal of greater-than-Class C low-level radioactive waste (GTCC LLW) requires characterization of the waste. This paper estimates future volumes, radionuclide activities, and waste forms of GTCC LLW. It groups the waste into four categories, representative of the type of generator or holder of the waste: Nuclear Utilities, Sealed Sources, DOE-Held, and Other Generator. In the future, Nuclear utilities will probably generate the largest volume of GTCC LLW with 65-83% of the total volume. The Other Generators will generate 17-23% of the waste volume, while GTCC Sealed Sources are expected to contribute 1-12%. A preliminary review of current DOE-Held wastes indicates that they will not require management as GTCC LLW because of the contractual circumstances under which they were accepted for storage. In addition, this paper discusses the uncertainties that may affect the predictions of this kind. This paper concludes that the volume of GTCC LLW should not pose a significant management problem from a scientific or technical standpoint. The projected volume is small enough to indicate that a dedicated GTCC LLW disposal facility may not be justified. Instead, co-disposal with other waste types is being considered as an option.

INTRODUCTION

This paper estimates future greater-than-Class C low-level radioactive waste (GTCC LLW) volumes, radionuclide activities, and waste forms, and identifies potential waste generators. Information for this paper is taken from the report Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics(1). This report was originally published in 1991. Since 1991 there have been several additional studies conducted to further identify and quantify projected volumes and activities of GTCC LLW. The report was revised in 1994 to include new information affecting the estimated volume and activity projections for GTCC LLW. These estimates will provide information to the U.S. Department of Energy (DOE), the agency responsible for ensuring the safe disposal of GTCC LLW, to assist in developing long-term strategic planning for storage, treatment, fee assessment, and final disposal of GTCC LLW. The estimates in this paper represent the best available data for DOE planning.

The DOE began the GTCC LLW Management Program in response to a congressionally mandated responsibility, Public Law (PL) 99-240, the Low-Level Radioactive Waste Policy Amendments Act of 1985. Low-level radioactive waste (LLW) is defined in PL 99-240 as radioactive waste that is neither high-level radioactive waste (i.e., spent fuel) nor by-product material (i.e., mill tailings). In the absence of a concentration-based definition for high-level waste, there is currently no upper limit for the concentration of radionuclides in LLW.

The U.S. Code of Federal Regulations, 10 CFR 61, regulates the disposal requirements for three classes of LLW considered generally suitable for near-surface disposal: A, B, and C, with Class C waste requiring the most rigorous disposal specifications. The radionuclide concentration limits and calculation methods for determining Classes A, B, and C LLW are found in 10 CFR 61.55, Tables 1 and 2. Waste with radionuclide concentrations above Class C limits is identified as GTCC LLW. GTCC LLW is not suitable for near-surface disposal, except on a case-by-case evaluation and the U.S. Nuclear Regulatory Commission (NRC) approval of the proposed disposal method.

The waste generators described in this paper are licensed either by the NRC or a State agency. Commercial nuclear power plants are licensed by the NRC, regardless of the State in which they reside. Other radioactive waste generators can be licensed by the NRC, or a State regulatory agency that is accredited by the NRC. States that issue licenses in this manner are known as Agreement States. PL 99-240 assigns to the States the responsibility for providing disposal facilities for Classes A, B, and C LLW. The same law makes the Federal government (specifically, DOE) responsible for disposing of GTCC LLW.

This paper addresses GTCC LLW as waste that meets the following criteria:

- Exceeds the Class C limits as defined in 10 CFR 61.55, Tables I and Table II
- Is generated by radioactive waste generators licensed by the NRC or an Agreement State

Is not "owned or generated" by DOE, or generated as a result of nuclear weapons activities or decommissioning of nuclear powered naval vessels
Requires management or disposal under PL 99-240.

In this paper, GTCC LLW is divided into the following four categories, each of which is determined by the major industry that produces the waste or by the agency holding it:

Nuclear Utility
Sealed Sources
DOE-Held
Other Generators.

Nuclear Utility waste includes activated metals and process wastes generated by commercial nuclear power plants. Sealed Sources consist of radioactive materials contained in small, metallic capsules and used in devices for measurement or calibration. DOE-Held waste consists of commercially generated wastes that a) exceed Class C limits, and b) DOE has accepted for storage. Other Generators waste consists of sludge, activated metals, and other wastes that are generated by radionuclide manufacturers, commercial research, sealed source manufacturers, and similar types of operations.

In calculating volumes and activities for these GTCC LLW categories, this paper uses the following definitions:

Concentration averaging: 10 CFR 61.55 allows the radionuclide concentration to be calculated over the volume or mass of the waste. In practice, similar wastes of varying radionuclide content are packaged together for shipment to a LLW disposal facility. The total radionuclide content in the package is averaged over the total waste volume or mass in order to calculate the average radionuclide concentration per unit of volume or mass for waste classification. This method of calculation is known in the industry as "concentration averaging."

Unpackaged volume: This is the volume of waste before packaging for disposal. The unpackaged volume is often used as the foundation for applying the packaging scenarios that calculate the packaged waste volumes presented in this paper. For the Nuclear Utility category, estimates of unpackaged volume do not reflect the impact of concentration averaging. All other categories account for concentration averaging in the unpackaged volume estimates.

After-concentration averaged (ACA) packaged volume: This is the packaged volume of waste that is classified as GTCC LLW, after all other waste has been classified as Class A, B, or C LLW using concentration averaging practices.

Low, Base, and High Cases: Three cases are calculated for the volumes described above. These values are defined as the low, base, and high cases. The base case is the most reasonable volume and activity estimate, and is based on inventories, surveys, and industry production rates. High and low cases are considered upper and lower limits of the base case. Assumptions used to calculate the low, base, and high cases may be different for each category of waste. Specific definitions and assumptions used to make these calculations are described for each category in the appropriate sections.

NUCLEAR UTILITY GTCC LLW

The information on Nuclear Utility GTCC LLW was developed in order to project volumes and activities to 2055 for current-technology reactors. This projection will account for waste generated by some nuclear reactors that may apply for and receive a 20-year license extension.

Nuclear Utility data are divided into pressurized water reactor (PWR) and boiling water reactor (BWR) types. BWRs are designed and manufactured by General Electric (GE). PWRs are designed and built by three manufacturers: Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse (W).

Waste materials, which are unique to each of the reactor designs, are generated during normal plant operations and at decommissioning. Operations wastes are routinely generated during the normal operation of a reactor. Decommissioning wastes result from the closure and cleanup of a nuclear reactor, a one-time event that accounts for the major volume of GTCC LLW generated.

Activated metal GTCC LLW, from both operations and decommissioning activities, consists of irradiated metal components from nuclear reactors. Hardware internal to the pressure vessel absorbs neutrons during reactor operation and becomes radioactive. If the hardware is exposed to a sufficiently high neutron flux, the radioactive constituents could reach concentrations above Class C limits.

Process wastes consist of the waste generated from the cleanup of liquids containing soluble and insoluble radioactive constituents. Those wastes are in the form of ion-exchange resins and cartridge filters.
The volume and activity of GTCC LLW from nuclear utilities depend on several factors:
- Life cycle for the operations and shutdown of commercial nuclear power stations
- Packaging factors used to calculate packaged volumes
- Extent to which waste can be packaged for disposal in a near-surface disposal facility by using concentration averaging.

Three possibilities exist to predict the time when operations and decommissioning GTCC LLW will be generated from commercial nuclear utilities: early shutdown of a reactor, operation for license duration, and license extension. In 1993, 109 commercial nuclear power plants (37 BWRs and 72 PWRs) were operating in the United States. Forty-six of those units will reach the end of their 40-year operating licenses by 2015. Because many of these plants have a relatively low capacity for generating electrical power, it may not be economically feasible to extend their lives past the end of their operating licenses. Several nuclear plants have recently shut down permanently before the end of their 40-year licenses. However, nuclear plants might choose to apply for license extensions to allow them to operate beyond 40 years. The following general assumptions were used concerning timing issues related to nuclear utility plant operations and decommissioning:

High Case: The high case uses DOE's 1991 National Energy Strategy(2) to predict the percentage of plants that will have a license extension, which could move shutdown and decommissioning dates 20 years into the future for many reactors.

Base Case: The base case assumptions and calculations represent the most probable waste generation timing using the most current projections in the DOE's Annual Energy Outlook 1993(3). Extension of licenses occurs less frequently in the base case than in the high case, and early reactor shutdown is assumed for some reactors in order to account for regulatory, political, and economic uncertainties.

Low Case: The low case establishes a lower bound for waste volume calculations. The low case, which also includes the possibility of early shutdown to account for regulatory, political, and economic uncertainties, presents a conservatively short projection for nuclear plant life. No plants will receive license extensions in the low case.

GTCC LLW will require packaging before disposal. For Nuclear Utility waste, this study uses a packaging factor (PF) to convert the unpackaged volume to a packaged volume. The PF is defined as the volume of the GTCC LLW disposal container divided by the unpackaged volume of the waste loaded into the container. Packaged volumes were determined for the low, base, and high cases. The base case is the most realistic value; the high case indicates a higher ratio of disposal container volume to waste volume; and the low case indicates a lower ratio.
This paper uses the following assumptions relating to the concentration averaging of nuclear utility waste:

High Case: The high case assumption is that no concentration averaging is allowed. This yields the maximum packaged volume estimate.

Base Case: The base case assumption is that the 1993 NRC draft Branch Technical Position (BTP) on concentration averaging will be the standard adopted by most LLW generators and disposal site operators. The base case assumes that concentration averaging will continue in the future. This volume is the base case ACA packaged volume.

Low Case: The low case assumption is that an additional volume reduction from the base case GTCC LLW will result because of the alternatives provision found in the 1993 NRC draft BTP on concentration averaging. The alternatives provision clarified the concentration averaging guidelines by stating that certain copackaging practices would be allowed on a case-by-case basis.

The low, base, and high case concentration averaging assumptions are used to determine concentration averaging factors for each of the Nuclear Utility components that exceeds Class C limits. The concentration averaging factor is used to calculate the volume of waste components that will be classified as GTCC LLW after all other LLW has been disposed of as Class A, B, or C. The packaged GTCC LLW volume remaining after concentration averaging is referred to in this paper as the ACA packaged volume.
In all three cases, the predicted unpackaged volume of Nuclear Utility GTCC LLW is

the foundation for estimates of ACA packaged volume.
The timing, packaging, and concentration averaging assumptions were used to project nuclear utility GTCC LLW to 2055. Nuclear Utility GTCC LLW projections are base case unpackaged volume of 1,144 m3 and base case ACA packaged volume of 1,347 m3. The core shrouds/baffles and lower core barrels are the major volume contributors. Table I shows the total Nuclear Utility GTCC LLW volumes in 2055. The base case activity of the Nuclear Utility GTCC LLW is estimated to be 88,400,000 Ci in 2055. The major portion of the activity will be from short-half-life radionuclides. The highest activity components result from the PWR core shrouds and baffles at decommissioning.

SEALED SOURCES GTCC LLW

Sealed sources are radioactive materials encapsulated in a metallic container. While the sources are small in volume, they are often highly radioactive. Sealed sources of concern to DOE and the GTCC LLW Management Program are those that exceed the Class C limits for waste classification and are not useful for their designed purposes.
A study of sealed sources was conducted to gather information concerning the types of sealed sources, the total number of licensed holders of sealed sources, the radionuclide activities, and to determine how many sources are in use.(4) This information was compiled in a database that was used to estimate the inventory of sealed sources in the year 2035. This information was collected and calculated for sealed sources held by both general and specific licensees. It is important to note that a high degree of uncertainty is associated with the data used to estimate the current inventory and future projections for sealed sources held by general licensees. For this reason, no attempt was made to add the totals for general and specific licensees.
In the base case, this study estimates that in 2035, 43,920 GTCC LLW sealed sources will be held by general licensees. These sources could have an unpackaged volume of 0.12 m3 and an ACA packaged disposal volume of 8 m3.
In 2035, a base case total of 54,278 GTCC LLW sources will be held by specific licensees. The estimated unpackaged volume will be 0.87 m3, and the ACA packaged volume for disposal will be 234 m3. Table II shows the estimated 2035 inventory of sealed sources.

DOE-HELD GTCC LLW

A number of commercial facilities have generated waste with radionuclide concentrations above Class C limits, and through contractual arrangements with DOE or for health and safety reasons, have stored these wastes at DOE facilities. This waste was identified as DOE-Held potential GTCC LLW. The GTCC LLW Management Program has investigated these wastes to determine the legal and contractual requirements for managing and disposing of the wastes.(5)
A decision process was developed to evaluate the legal and contractual arrangements by which DOE accepted waste for storage. That process was developed to determine which of the potential GTCC LLW wastes held by DOE would be classified and managed as GTCC LLW under PL 99-240. That decision process includes the following questions:
Does the waste exceed the Class C limits as defined in 10 CFR 61.55, Tables 1 and 2?
Is the waste generated by a radioactive waste generator licensed by the NRC or an Agreement State?
Is the waste "owned or generated" by DOE under PL 99-240 or generated as a result of nuclear weapons activities or decommissioning of nuclear powered naval vessels?
Is management or disposal of the waste required under PL 99-240?
In 1991 it was estimated there was 1,080 m3 of DOE-Held potential GTCC LLW. From 1991 to 1993, the GTCC LLW Management Program identified an additional 1,869 m3 of DOE-Held potential GTCC LLW for a total volume of 2,949 m3.(3) The decision process indicates that, because of contractual agreements and circumstances under which the wastes were generated, none of the waste identified as DOE-Held potential GTCC LLW falls under the requirements of PL 99-240. Therefore, these wastes are not included as GTCC LLW in this paper.
All current DOE-Held wastes will be classified as DOE Special-Case Waste (SCW). SCW is waste that is above Class C limits and is owned or generated by DOE for various reasons. For example, it was generated by DOE or by a subcontractor performing work for DOE, or DOE has taken ownership of it under provisions other than PL 99-240. The management and disposal issues for DOE SCW are being studied under another program.

OTHER GENERATORS GTCC LLW

The Other Generators category includes all GTCC LLW that is not generated by nuclear utilities or sealed source licensees, or held at DOE facilities. Generators include

- Manufacturers and users of 14C
- Industrial research and development firms
- Fuel fabrication and irradiation research (burnup) labs
- Academic nuclear research reactors
- Sealed source manufacturers
- Non-medical academic institutions.

Generator operating histories and estimates by the generators were used to determine annual production rates for each generator. The annual production rates were used to project the total volume and activity inventory to 2035. Table III shows the estimated 2035 inventory of Other Generators GTCC LLW.(1)

DATA SUMMARY OF ALL CATEGORIES

The following volumes of GTCC LLW are projected from all categories: (numbers presented here are rounded off to the nearest cubic meter from the calculated values presented in Table IV).

- A base case unpackaged volume of 1,380 m3
- A base case ACA packaged volume of 2,054 m3.

Table IV shows the low, base, and high case summary of all categories for future unpackaged and ACA packaged volumes. Table V shows the future total activity for each category.

AREAS OF UNCERTAINTY

This paper contains the best available data and information for projecting GTCC LLW volumes and activities for each of the categories. However, there are areas of uncertainty that must be considered when applying the volume projections presented in this paper. The following sections describe the uncertainties associated with each category of GTCC LLW.

Uncertainties in Nuclear Utility Areas of GTCC LLW Volume Projections

The actual volumes of GTCC LLW generated by nuclear utilities in the future can be affected by the following five areas of uncertainty:

- Concentration averaging practices
- The interpretation of the Standard Contract (10 CFR 961), that identifies the wastes which meet the definition of spent fuel
- Implementation of procedural controls to reduce the volumes of operations wastes generated
- The assumed concentration of cobalt, nickel, and niobium in the stainless steel making up the core components
- The construction of advanced-technology power reactors to supply future demand for electrical energy.

The following paragraphs discuss these five factors.

The first factor that can affect the total volume of Nuclear Utility GTCC LLW is future concentration averaging practices. The volume of potential GTCC LLW that can be disposed of at a commercial LLW disposal site is determined by the NRC or State regulatory agency (monitored by the NRC) who issues the license to the site. The LLW sites' licenses regulate the waste acceptance criteria, which define the concentration averaging practices that will be allowed. The only commercial LLW sites currently accepting radioactive waste are located at Barnwell, South Carolina, and Hanford, Washington. The Barnwell site has closed to all States except those in the Southeast compact region. The Hanford site is closed to all States except those in the Northwest and Rocky Mountain compact regions. The waste acceptance criteria for new LLW sites have not yet been developed or approved. Therefore, it is unclear what concentration averaging practices will be allowed in the future.

Concentration averaging practices that are acceptable to the NRC are described in the draft 1993 BTP on concentration averaging. Since one purpose of this BTP is to promote consistent concentration averaging criteria, the base case ACA packaged volume for nuclear utilities is calculated with factors based on the guidance provided by the BTP. The high case assumes that no concentration averaging will be performed. The low case assumes a further GTCC LLW volume reduction for some waste forms based on special exceptions that might be allowed under the alternatives provision in the BTP. The actual concentration averaging practices that will be adopted by the Agreement State licensees are unknown at this time and could be more restrictive than the NRC guidance.

The second factor that can affect the total volume of Nuclear Utility GTCC LLW is

the interpretation of the Standard Contract (10 CFR 961), which identifies the wastes that meet the definition of spent fuel. Studies are underway that may result in some fuel-related components being classified as high-level waste and eligible for disposal in a spent fuel repository when it opens. This reclassification may result in a decrease of the volumes reported in this document.

The third factor that can affect the total volume of Nuclear Utility GTCC LLW is procedural control implemented by the power plants. For example, a plant can change out filters, control rod blades, and instrument strings more often so that they will not be irradiated above Class C limits. Procedural controls can result in an overall decrease in operations GTCC LLW from nuclear utilities.

The fourth factor that can affect the total volume of Nuclear Utility GTCC LLW is the assumed trace concentration of cobalt, nickel, and niobium in stainless steel. The 10 CFR 61.55 classification of activated metals is driven by nickel and niobium concentrations. Standard waste classification techniques are based on assumed values for cobalt, nickel, and niobium in stainless steel. The assumed cobalt (Co) concentration is used together with the measured radiation readings to estimate the 60Co activity. Scaling factors are used to estimate the nickel and niobium concentration from the 60Co value. These assumed values are taken from Long-Lived Activation Products in Reactor Materials.(6) In type 304 stainless steel, nickel is a major constituent; its content can vary by +15% from the value reported in Reference 6. The uncertainty associated with this variance is not significant. This is not the case for niobium, which is also a trace metal in type 304 stainless steel. The actual trace quantity of niobium can vary by as much as +50% from the Reference 6 values. If the assumed value for cobalt, nickel, or niobium changes, the waste classification of the activated metals components could also change.

The fifth factor that can affect the total volume of GTCC LLW generated by nuclear utilities is the possible construction of advanced-technology reactors and the volumes generated from their operation of these reactors. The possible estimated volumes from operating these reactors are addressed in Reference 1 but are not included in the Nuclear Utility totals or in the GTCC-LLW totals from all categories presented in this paper. These volumes have not been included at this time because the future construction of these reactors is so uncertain. If advanced-technology plants are built and begin operation, the total volume of GTCC LLW from nuclear utilities could increase.

Uncertainties in the Sealed Sources GTCC LLW Volume Projections

The Sealed Source areas of uncertainty include possible changes in future concentration averaging practices and variances in the future production and recycling rates from the assumed rates used in making the volume projections for both general and specific licensed sealed sources.

Uncertainties in DOE-Held GTCC LLW Volume Projections

Future acceptance of GTCC LLW by DOE will be on a case-by-case basis until a final disposal option for GTCC LLW is identified. Any future wastes accepted into this category will come from one of the other categories. Therefore, the volume in this category may increase in the future, but that will not increase the total volume of GTCC LLW. A change in the legal interpretation of legislative requirements could also impact the estimated volume of GTCC LLW in this category.

Uncertainties in Other Generators GTCC LLW Volume Projections

Two factors can affect the volume of Other Generators GTCC LLW:

- Concentration averaging practices
- A major increase or decrease in waste generation.

Other generators of GTCC LLW have already performed concentration averaging as much as possible for their waste. Future projections assume that current practices will continue. Therefore, an area of uncertainty for the Other Generators GTCC LLW is the effect of concentration averaging. The reasons for, and effects of, this are the same as explained for the nuclear utilities. Until concentration averaging guidelines are better defined by the NRC and the Agreement States that will license future LLW disposal sites, the base case ACA packaged volumes reported are the best data available.

Another factor that could impact these volumes is a major increase or decrease in GTCC LLW generation rates in the future. No one can accurately predict the economic and political forces that could affect their production. The generation rates used in this paper are based on the best data currently available. In addition, reasonable assumptions are made about the future production rates and number of new

generators that will arise in the future. If the generation rates change, or the anticipated number of new generators differs from these estimates, the actual GTCC LLW volume from Other Generators could vary.

CONCLUSIONS

This study estimates a base case, unpackaged future GTCC LLW volume from all categories of 1,380 m3. If concentration averaging is performed, the base case ACA packaged volume of GTCC LLW is estimated to be 2,054 m3. Table VI summarizes the estimated future GTCC LLW from all categories.

The information and data in this study leads to the conclusion that the volume of GTCC LLW should not pose a significant management problem from a scientific or technical standpoint. The projected volume of GTCC LLW is small enough to indicate that a dedicated disposal facility for this waste may not be justified. Instead, co-disposal with other waste types (i.e. high level waste or DOE special case waste) is being considered as an option.

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HAZARDOUS WASTE TRANSPORTATION RISK ASSESSMENT: BENEFITS OF A COMBINED DETERMINISTIC AND PROBABILISTIC MONTE CARLO APPROACH IN EXPRESSING RISK UNCERTAINTY*

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ABSTRACT

This paper presents a combined deterministic and probabilistic methodology for modeling hazardous waste transportation risk and expressing the uncertainty in that risk. Both the deterministic and probabilistic methodologies are aimed at providing tools useful in the evaluation of alternative management scenarios for U.S. Department of Energy (DOE) hazardous waste treatment, storage, and disposal (TSD). The probabilistic methodology can be used to provide perspective on and quantify uncertainties in deterministic predictions. The methodology developed has been applied to 63 DOE shipments made in fiscal year 1992, which contained poison by inhalation chemicals. Models have been applied to simulate shipment routes, truck accident rates, chemical spill probabilities, spill/release rates, atmospheric dispersion, population exposure, and health consequences. The simulation presented in this paper is specific to trucks traveling from DOE sites to their commercial TSD facilities, but the methodology is more general. Health consequences are presented

as the number of people with potentially life-threatening health effects. Probabilistic distributions were employed for several input parameters; namely, accident time (hour and season), amount(s) of hazardous material released and spill characteristics. The meteorological conditions for the location and time of each modeled accident are obtained probabilistically from a preprocessed meteorological database consisting of hourly records for 61 cities in the contiguous United States. The results show an approximate 98.6% probability that during 20 years of shipments (assuming the 63 DOE shipments of 1992 are made each year), no people would suffer potentially life threatening health effects. However, the last 1.4% of probability includes accidents that can have significant effects, although of very low probability of occurrence. For instance, over a 20-year period, there is approximately a 1 in 86,000 chance totaling 100 or more people and a 1 in 2,000,000 chance of accidents totaling 1,000 or more people suffering potentially life-threatening health effects. The deterministic prediction of risk for the 20-year period, 0.00348 people suffering potentially life-threatening health effects, was found to be at the 99.5th percentile of the probabilistic distribution, which is not surprising given the very skewed shape of the distribution. Expressing risk with a combined deterministic and probabilistic method is helpful to decision makers because a perspective of the uncertainty in the deterministic risk value is given when placed on the cumulative risk probability distribution.

INTRODUCTION

The U.S. Department of Energy (DOE) has approximately 45 sites in the United States that produce chemical hazardous waste (HW) from research and production processes. Much of this waste must be transported on public highways to commercial facilities that are permitted for treatment, storage, and disposal (TSD). During transit, a truck may become involved in a accident. Though rare, such accidents may lead to a hazardous material release which may result in potentially life-threatening health effects, increased probability of carcinogenic effects, and/or other adverse human health effects. As the generator of the waste, DOE must assume partial responsibility for the consequences of such an accident.

The DOE is considering various options (alternatives) for the management of this HW. These options are being evaluated and compared in terms of the relative risk associated with the three health endpoints mentioned above, as a function of the management scenario under consideration. The alternative scenarios differ mainly as to where the 45 DOE sites would transport their chemical wastes for TSD. Portions of the waste might be handled on-site or at regionalized DOE locations and the rest handled off-site at commercial facilities. Determining how to distribute the waste between on-site treatment and off-site TSD facilities leads to a comparison of multiple alternatives requiring the evaluation of human health risks, costs of treatment and transportation, and ecological, land-use, and socioeconomic impacts. The current cases under consideration by DOE include baseline or no action, decentralization, and 5-region and 2-region regionalization.

The need for quantifying the risk in these alternatives and the exigency for providing decision makers with a means for proper interpretation and understanding of these risks gives rise to the necessity to determine and communicate the uncertainty behind the risk estimates. Typically, risk assessments are conducted with deterministic methods that use single values or "point estimates" of parameters pertinent to the risk calculation. However, risk parameter values are often quite uncertain and/or variable, and may in fact fall within a range of possible values. Because deterministic methods do not account for this error, they have limited usefulness in providing a perspective on risk estimates for informed policy decisions. Therefore, an approach has been developed and applied that specifically accounts for key parameter variability and uncertainty. This approach is not intended to replace deterministic methods, but rather to supplement when appropriate information is available to develop necessary parameter data distributions.

This paper presents a state-of-the-art stochastic methodology for quantitatively expressing the distribution of HW transportation risks. When combined with a deterministic risk computation, an expression of uncertainty in the typical deterministic risk estimates can be obtained. The paper also presents results from applying this methodology to assess the risks of transporting HW generated by DOE in fiscal year 1992. The risk assessment methodology was applied only to DOE-generated HW that was transported under the baseline alternative, with the risk measure being the number of people with potentially life-threatening health effects. The candidate

hazardous chemicals in the DOE waste were identified for evaluation with the help of the poison by inhalation hazard (PIH) list from the U.S. Department of Transportation (DOT) (1). The risk assessment methodology developed can be routinely applied to the other health endpoints and waste management alternatives using the computer models developed as a result of the methodology provided in this paper.

HAZARDOUS WASTE DATABASE AND ROUTE PREDICTIONS

To analyze risk from off-site transportation accidents, the Hazardous Waste Risk Assessment Modeling Database (HaWRAM) has been developed at Argonne National Laboratory (2). This database is structured to manage information related primarily to DOE-generated HW that was transported to commercial facilities for treatment and/or disposal during fiscal year 1992. The 63 shipments with PIH chemicals to which the combined deterministic and probabilistic methodologies was applied have been identified by their chemical names, and these names have been entered into the records of the HaWRAM database.

A computer routing prediction model was used to determine routes traveled for each alternative. Routing information was predicted with the HIGHWAY 3.1 Enhanced Highway Routing Model (3) developed at Oak Ridge National Laboratory. This model takes as input the origin and destination cities (generator location and designated facility location) and supplies a detailed output of population densities at locations defined by latitude and longitude coordinates, mileage between successive locations, and interstate and local roads traveled. Several of the actual routes traveled for the baseline alternative were obtained from carriers and compared against routes predicted by HIGHWAY 3.1 to determine accuracy. Each of the predicted routes served as a good estimate of the actual route taken.

DETERMINISTIC RISK METHODOLOGY

Risk due to any one mile of transporting HW is defined as the product of the probability of a release of this waste in transit multiplied by the consequence of the release. The total shipment risk for any one shipment is calculated by summing each of the individual one mile risks along the entire route. The total risk for an entire scenario (management option) is computed by summing each of the shipment risks into one final risk number. Equation 1 defines the formula for computing risk over an entire hazardous waste management scenario:

Eq. (1)

where

S = number of shipments;

L = one-mile locations along the route traveled for the current shipment;

ARL = accident rate or probability of an accident in any one mile of travel (value depends upon the demographic region of the current location);

PRS = probability of a release given the occurrence of an accident (value depends upon shipment packaging, i.e., bulk or package freight);

EAS = exposed area in which the accident plume concentration is greater than the health criterion (exposed area depends upon the chemical properties of the HW released, meteorological conditions, release rates, health criterion, amount of liquid released that flashes or vaporizes directly to the atmosphere and liquid evaporation rates); and

PDL = average population density at the current one-mile location.

In Eq. 1, the product of ARL and PRS gives the probability of a release and the product of EAS and PDL gives the consequence of that release. The remainder of this section provides additional details about the probability of a release and the consequence.

The probability of a release is derived from the product of the probability of an accident in any one mile multiplied by the probability of a release given an accident. The probability of an accident in any one mile is a function of the demographic region (rural, suburban, or urban), and the probability of a release given the occurrence of an accident is a function of container type (package freight or bulk) and physical state of the waste (liquid or gas). The actual probabilities are given in Table I. Accident probabilities are based on California state truck

accident data, and conditional release probabilities are based upon Federal Highway Administration truck accident data. For further details, the reader is referred to Chapter 4 of Present Practices of Highway Transportation of Hazardous Materials (4).
Consequence is defined as the number of people impacted by the particular health endpoint considered. In this paper, the consequence is measured as the number of people with potentially life-threatening health effects. In future applications of the risk assessment methodology, consequence could be measured in terms of the number of people with carcinogenic or other adverse health effects. The number of people affected is computed as the exposed area in which the predicted accident concentration is above the health criterion (for that human health endpoint) multiplied by the average population density within that exposed area.
The average population densities for one of three population zones rural, suburban, and urban were determined by the HIGHWAY 3.1 routing model (3). In this analysis, all rural areas were assumed to have the same population density, and a similar assumption was made for urban and suburban areas.
The distribution for the areas covered by HW concentrations above the health criteria was computed using the Chemical Accident Stochastic Risk Assessment Model (CASRAM). CASRAM has two principal model components that are used to simulate each hypothetical accident: a source-term model and a dispersion model. By considering thousands of accidents for each shipment, a wide distribution of release scenarios is considered. For a deterministic model, each variable is treated as fixed; for the probabilistic version, there is a distribution of release rates and pool sizes (if a pool does form). If the container includes a gas, the release rate and duration of the gas release is predicted instead. The probabilistic method computes a probability distribution of such rates.
The CASRAM source model requires several input parameters such as the chemical(s) released, the total HW quantity shipped and container type for each chemical released, and ambient meteorological conditions. The output includes the release rate and duration from the container and, if applicable, an evaporation rate. These release rates serve as input to the dispersion model.
Prior to determining the release rate with the deterministic method, the source model quantifies the release amount of each hazardous material in the shipment by using average release fractions as a function of container type. These release fractions, shown in Table II, are based upon thousands of HW releases resulting from truck transportation accidents that occurred between 1985 and 1992. The accident data were supplied by the Hazardous Materials Incident Reporting System Database maintained by DOT's Research and Special Programs Administration (6). The average release fractions are small because most releases occur as a result of a damaged valve or receptacle, which leads to a small release.
An additional input parameter required for the dispersion model is the human health criteria. The health criteria refer to the air concentration (and duration) that, when exceeded, can lead to adverse consequences under the health endpoint considered. The health criteria parameter is a function of the duration of inhalation. Three values have been supplied for each health endpoint, a 15-minute, 30-minute, and 60-minute inhalation duration. Only a few chemical wastes in the HaWRAM database have the potential to cause lethal, carcinogenic, or other adverse human health effects (other adverse effects include nonlethal and noncarcinogenic effects such as respiratory problems). An example of a chemical (hydrogen selenide) released in a transportation accident is given in Table III. The results show that the concentration resulting in adverse health effects decreases as the inhalation duration increases.
The actual fixed input parameter values used for the deterministic risk assessment, all of which are either average or conservative, are as follows: stability class D; ambient temperature 95F; wind speed 4 m/s; inversion layer none; cloud cover 10/10 (completely cloudy); and humidity 50%. Potentially life-threatening concentrations are assessed for specific HW chemical components designated as PIH by DOT (Code of Federal Regulations, Title 49, Parts 173.115 and 173.132-133). Liquids and gases that have high toxicity based upon animal 50% lethal concentrations (LC50), as well as liquids with medium to high volatility, have been placed on DOT's PIH list.
Two possible toxicity values that are often available in the literature for estimating potential human life-threatening health effects are the LC50 and the LCLO. The LC50 is defined as the concentration of gas or vapor that causes death in

half the animals tested when administered by continuous inhalation. Because LC50 values are obtained only from animal tests, the results must be extrapolated for application to humans. The LCLO is defined as the lowest concentration of gas or vapor that causes death in any exposed species. The LCLO values may be obtained from animal tests or from actual accidental human exposures. When obtained from human exposures, the lethal concentration measurement may not be accurate.

When several chemicals are transported as part of one shipment on the same truck, additive health effects are modeled to determine the exposed area. This determination is based on the following inequality:

Eq. (2)

where

Ci = the air concentration of the ith chemical, and

Ti = the concentration threshold (health criterion) for the ith chemical

The plume area with concentrations above a particular human health criterion is computed via numerical integration starting at the emission source. Distances closer than 100 ft from the accident scene are omitted because people do not live that close to the roadways. At each downwind integration step, a crosswind integration algorithm is used. As crosswind distances are incremented, Eq. 2 is employed to determine whether or not the current location is included in the exposed area. If there are multiple chemicals in the same accident shipment, then the resultant area is usually a bit larger than the largest area obtained from treating each chemical separately. This method accounts for the additional dosage contributed by each chemical in the mixture released without accounting for synergistic effects.

A PROBABILISTIC METHODOLOGY

Four major parameters or categories of parameters meteorological conditions (atmospheric turbulence characteristics, wind speed, temperature, etc.), temporal conditions (time of day and month of year), release quantities, and health criteria are required to determine the plume area with chemical concentrations above a particular health criterion. Using the deterministic method, these parameters are treated with fixed values or point estimates, although they are truly stochastic. Meteorological and temporal conditions are clearly stochastic parameters because they are constantly changing; release quantities vary with the severity of the accident.

Most HW releases resulting from truck accidents involve only a damaged receptacle or valve leading to small, slow releases. However, more severe accidents can lead to more catastrophic releases where the entire cargo contents is released almost instantaneously. Similarly, there is uncertainty in human health criteria because humans have varying sensitivities to chemical effects, and because health criteria are based upon limited animal studies. A probability distribution for health criteria defined by the "potentially life-threatening health effects" endpoint could be determined on the basis of laboratory data (if available for a given chemical) regarding the concentrations causing death in 10%, 50%, 90%, etc., of an animal population. An uncertainty factor (converting animal data to human data) would have to be included to adjust those animal concentrations. Once that is done, a probability distribution of concentrations can be developed to represent potentially life-threatening health effects in humans.

The probabilistic methodology presented in this paper has been developed to account for variation in the key parameters. Meteorological conditions, temporal conditions, and release rates have been treated probabilistically. The health criterion has been treated deterministically at this time because it was beyond the scope of the current work to define its variation, which depends on the chemical to some degree. Future improvements to the methodology presented may include a stochastic treatment of the health criteria as well.

The probabilistic method is based upon a Monte Carlo algorithm in which a distribution of consequences is derived. The model can be split into two steps. The first step is to derive a distribution of risk for each shipment under consideration (for this application, 63 shipments of DOE-generated PIH HW and subsequently 63 distributions of risk). The results of this first step can be used to examine individual shipments carefully. The second step is to determine a single combined distribution of risk for the entire scenario (waste management alternative) based upon the 63 individual shipment distributions of risk.

Distributions of Risk for Individual Shipments as Determined in CASRAM

The CASRAM model was employed to determine the risk distribution for each of the 63 shipments mentioned earlier. In this section, we outline the CASRAM model to provide the reader some perspective on the Monte Carlo approach as applied in this study. A flowchart illustrating the Monte Carlo approach considered here is shown in Fig. 1. The probabilistic components of the Monte Carlo process illustrated in Fig. 1 are marked by the "Random " notation. As initial data, the model requires the route traveled, the shipment contents, and the population densities along the route. The model begins by looking at the first mile of the route, selecting a random number 'r' between 0 and 1, and determining if an accident occurs. If 'r' is less than the appropriate probability of an accident, then an accident occurs; otherwise, the location is incremented to the next mile. Similarly, when an accident occurs, another random number 'r' is selected between 0 and 1 and tested to determine if a release occurs. Once an accident occurs, a release is modeled and the consequences are recorded. To obtain a smooth distribution, the model continues until 100,000 releases are modeled.

The probability of an accident and the probability of a release given an accident are functions of demographic region and container type, respectively, as described in the section on the deterministic method. Similarly, these values are exactly those used in the deterministic method.

Distributions of time of day and month of year, given an accident, have been developed on the basis of data collected from thousands of actual HW releases resulting from truck accidents. These data are contained in the HMIRS database. Tables IV and V show the distributions numerically.

Two types of meteorological data from 61 cities distributed across the continental United States (Fig. 2) were used for the probabilistic treatment of meteorological conditions. These data were 1) surface airways data to specify the surface turbulence characteristics data (organized through the National Solar Radiation Data Base and essentially reproduced directly from the original National Climatic Data Center [NCDC] 3280 database) and 2) upper air data (NCDC 6201 format). The data are used for specification of hourly atmospheric boundary layer parameters which, with the morning temperature profile, allows for determining daytime inversion heights. Inversion heights during stable conditions were determined through diagnostic equations that are incorporated directly into the dispersion scaling. In the probabilistic modeling, the meteorological data used are those appropriate to the mile in which the accident occurs.

Five cumulative probability distributions of release fractions were developed using the data in the HMIRS database. Each distribution is representative of a different container type (see Fig. 3). The five container types accounted for include 1) small drums (capacity 20 gallons), 2) large drums (capacity >20 gallons), 3) pressurized cylinders, 4) pressurized bulk containers, and 5) nonpressurized bulk containers. Only package freight (pressurized cylinders, small drums, and large drums) distributions are presented in this paper because bulk containers were not used in packaging any of the HW in the 63 shipments of PIH wastes under consideration.

Due to lack of acceptable techniques, a simplified method for estimating pool size was employed. By considering a few representative chemicals for which data were available, equilibrium pool thickness was estimated on a flat surface. These estimates typically yielded coverage areas near 1 m2/gal. Most often, spills do not occur on flat uniform surfaces. To account for the large variety of surfaces with varying porosity and slope, a probability distribution for unit spill coverage area was selected.

In addition to typical atmospheric conditions, the source model also requires a surface temperature profile for accurate
determination of evaporation rates from pools of liquid. The surface temperature profile used differs from the usual ground temperature in that the energy balance at the surface does not include evaporation or transpiration from plants. This modification greatly affects the local energy budget, leading to large variations in surface temperature and conductive heat fluxes compared with those observed on normal ground. For the application presented in this paper, the surface temperature profile was based on the assumption that all evaporating pools are formed on pavement. Although the model does not require this assumption, it is believed to be the most conservative assumption considering the type of surface on which pools may form. Temperatures on the pavement can exceed those of other surfaces, such as soil, by as much as 20F.

The source model uses the release fractions, chemical property data, and information supplied by the meteorological preprocessor to determine release rates and evaporation rates for liquid spills onto the ground. Two types of releases are considered in the model. For gases, liquefied gases or liquids whose boiling point is below the pavement surface temperature, the released quantities are assumed to be released instantaneously. For cases when the pavement temperature is between zero and ten degrees above the boiling point, some liquid will flash and the rest will form a pool. Due to lack of available data, all liquid that is not flashed is assumed to form a pool on the pavement surface. Undoubtedly, this does not always occur in actual accidents.

The dispersion model uses local meteorological data, chemical release rates, and evaporation rates (if applicable) to determine the area downwind in which a particular health criterion is exceeded. The concentration downwind of the source is determined with a similarity-based method that uses nondimensional relationships for the ground-level, crosswind-integrated concentration together with relations for the horizontal plume spread. This method is superior to the Pasquill-Gifford-Turner curves because it appropriately accounts for the atmospheric boundary layer's physical structure and continuously relates the meteorological parameters to the downwind concentration estimates. Furthermore, model predictions agree well with a wide variety of studies, most notably the Prairie Grass atmospheric dispersion experiments. Besides standard plume releases, the dispersion model also has the capability of treating puff releases. Such capability is required because gaseous and flashed liquid releases are instantaneous (forming a puff), whereas evaporation from a liquid pool is continuous (forming a plume). For a more thorough description of the dispersion modeling methodology, the reader is referred to Statistical Determination of Downwind Concentration Decay for the 1993 Emergency Response Guidebook (5).

Distribution of Risk Values for Multiple Shipments

To determine the distribution of risk values for the multiple shipments scenario, an additional Monte Carlo algorithm was implemented. For this application, each of the 63 individual shipment distributions of risk was sampled. The consequence values, one from each of the 63 individual shipment distributions, were summed and recorded. The recorded number represents the total number of people at risk for one trial (at risk of potentially life-threatening health effects for the application presented in this paper). Several hundred or thousand trials are repeated to produce the desired distribution. For this application, the process was repeated 100 million times to guarantee sufficient sampling from the upper tails of the individual shipment distributions of consequence.

RESULTS AND DISCUSSION

Figure 4 shows the cumulative probability distribution of risk from the combination of all 63 shipments of PIH wastes transported to commercial TSD facilities in fiscal year 1992. For this run, the health endpoint under consideration was the number of people with potentially life-threatening health effects.

The cumulative probability distribution presented in Fig. 4 is heavily skewed. Although the probability of a release in any one mile of travel is a function of the demographic region (rural, suburban or urban) and the container type (package freight or bulk), its value is always less than or equal to 1.88E-7. As a result, the probability of no releases occurring along the entire route traveled for all 63 shipments of PIH chemical HW is greater than 97%. Furthermore, most releases are small, often leading to zero risk even though a release does occur (due to the 100 ft offset distance). This fact leads to a probability of zero risk for the entire scenario in excess of 99%. However, in the extreme tail of the distribution (the last 1% of probability), the number of people with potentially life-threatening health effects can be large. The large probability of zero risk coupled with the rare high-risk accidents heavily skews the distribution.

The mean of the risk distribution for the baseline waste management alternative (see Fig. 4) is approximately 3.48E-4 and the deterministic risk value was computed to be 1.74E-4 (nearly half of the mean). Although the deterministic value is half the mean, these two values are at the 99.941th and the 99.947th percentiles, respectively, on the cumulative probability distribution.

Accidents leading to one person or more with potentially life-threatening health effects can occur. However, its probability is 3.6E-5 or 1 in 28,000 over a one-year period for these 63 shipments. Accidents can lead to 10 people or more with

potentially life-threatening health effects, with a probability of 5.4E-6 or 1 in 180,000 for a one-year period of transporting hazardous waste under this scenario. Also, 100 people or more may be subject to potentially life-threatening health effects in a one-year period, with a probability of 5.8E-7 or 1 in 1,700,000 for that year. In summary, very serious accidents with catastrophic impacts could occur with these 63 shipments, but the probability is extremely low. Such accidents would occur under the very unlikely combinations of worst-case release rates (i.e., a very serious traffic accident) along with near-worst case meteorological conditions in highly populated areas. The probability of all such worst-case components occurring at the same time is extremely low, and the Monte Carlo method was able to quantify that probability and impacts on the population. Even after a 20-year period, the probability that no people would suffer potentially life-threatening health effects is approximately 98.6%. Similarly, over a 20-year period, the probabilities that more than 100 people and more than 1,000 people would suffer potentially life-threatening health effects are 1 in 86,000 and 1 in 2,000,000, respectively.
It is also interesting to note that the risk of physical trauma fatalities resulting from a truck accident is higher than the risk presented by a release of a PIH substance as a result of a truck accident. This means that cargo-related risks appear to be smaller than vehicle-related risks. On the basis of data supplied by the U.S. Bureau of the Census and the Federal Highway Administration, the fatality rate for truck accidents is estimated to be 2.02E-8 fatalities per truck mile traveled, leading to a risk of 6.3E-4 for physical trauma fatalities resulting from the 31,250 miles that would be traveled in one year to transport the 63 PIH shipments. This risk level is greater than the cargo-related risk of about 1.7E-4 (deterministic risk value) or 3.4E-4 (mean of the cumulative probability distribution of consequence for all 63 shipments in the entire baseline scenario).
Finally, it is interesting to note that both the probabilistic and deterministic methods show that risk is not uniformly distributed among the 63 shipments. In particular, shipments 3, 12, and 23 of the 63 shipments represent the highest risk. These three shipments alone account for 73% of the deterministic risk and have the most potential for a catastrophic release in which more than 100 people could be affected with potentially life-threatening health effects. These three shipments consist entirely of PIH wastes transported in a gaseous state.
Decision makers will be evaluating transportation risk predictions for the baseline, decentralized, regionalized number 1, and regionalized number 2 alternatives. Presentation of only a deterministic value for each alternative might lead to relatively small differences among the risk predictions. It is possible that the uncertainty in the risk predictions might be so large that it envelopes all the deterministic predictions of the alternatives. Without any clear quantitative understanding of the magnitude of the uncertainty, the decision maker cannot determine whether the predictions are essentially the same or whether they represent perhaps the 10th, 50th, 75th or 90th percentile of the risk probability distribution.
The Monte Carlo approach used in this paper is able to quantify much of the uncertainty in the risk. Questions regarding the completeness of the database of PIH shipments; the accuracy of the dispersion codes; the effects of fire, rain and water reactivity (hazardous waste released into a body of water); and the uncertainty in the health criteria were not quantified in this version of the Monte Carlo model. However, we believe that the bulk of the uncertainty has been captured and that conclusions based on the combined deterministic/probabilistic methods will be more supportable than from the usual deterministic modeling that has been done in the past. The presence of uncertainty in many aspects of the risk calculation needs to be accounted for in a quantitative fashion to make the best use of deterministic predictions for the benefit of decision makers.

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IMPACT OF DISPOSAL COSTS ON THE SELECTION OF MIXED WASTE PROCESSING SYSTEMS

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ABSTRACT

The mixed low-level hazardous waste in the U.S. Department of Energy (DOE) complex consists of organic and inorganic solids and liquids comprising a wide variety of materials contaminated with radioactive substances. Treatment systems are needed to destroy the hazardous organic materials and to immobilize any hazardous inorganic materials, along with the treatment residues and radionuclides, in order to comply with the regulations promulgated by the Environmental Protection Agency (EPA), state agencies, and the DOE. This paper identifies the importance of final waste forms and the sensitivity of the cost of disposal to the waste treatment technology selected.

INTRODUCTION

The DOE's Environmental Management Office of Technology Development has commissioned an Integrated Thermal Treatment Systems (ITTS) Study to assess alternative systems for treating contact-handled, alpha and non-alpha, mixed low-level radioactive waste (MLLW). The purpose of the ITTS study is to perform a systematic engineering evaluation of a variety of integrated MLLW treatment systems. The emphasis in this study is on the total system, from the receiving dock through disposal of the treated wastes. A total system approach is critical since the selection of one treatment technology over another can have impacts elsewhere in a complete system, such as in waste sorting and sizing, immobilization of the residues, waste disposal volumes, and total life cycle costs.

A major cost component in the treatment of mixed low-level waste is the cost of disposal. In studies sponsored by DOE, total life cycle costs have been developed for a wide range of mixed waste treatment systems (1,2). These studies show that disposal costs can be between 10 to 40 percent of the total life cycle costs, depending upon the wastes being treated, the technologies used in waste treatment and residue stabilization, and the costs of disposal. This paper presents some of the results from the ITTS studies to demonstrate how these inter-related options can affect the choice of a treatment technology.

WASTE COMPOSITION

To identify appropriate waste treatments, knowledge of the physical and chemical makeup of the waste is needed. The waste composition used in this study was derived from a report (3) compiled for the DOE from inputs from 49 DOE sites. The data were known to be imprecise but represented the best possible inventory at the time. The waste in storage at the DOE sites comes from operations that span almost 40 years -- long before requirements for good record keeping, waste treatment, and disposal

existed. Consequently, the data on physical matrices generally do not have a significant level of detail. Furthermore, most of the waste is a very heterogeneous mixture of trash-like material. Little or no segregation of wastes into discrete types of physical matrices was performed. Consequently, the data normally used in the design of combustion or other chemical treatment systems do not exist. Nevertheless, for the purposes of this study, an effort had to be made to define the DOE MLLW in terms of physical matrices in order to get some estimate of the chemical composition of the wastes. The chemical composition can then be used to perform comparative mass and energy balances for each of the different treatment system concepts. The results of the study leading to the waste profile used for the ITTS study are documented in Ref. 4. The physical matrix data from each DOE site are summarized in Table I. These data represent an estimate of the "average" DOE waste arrived at by combining the data from all sites. After assigning a chemical composition to the physical matrices and combining the data, the final waste profile was developed for mass balance calculations; the input waste quantities are shown in Table II.

TREATMENT OPTIONS AND RESIDUE STABILIZATION

Over the last several years, numerous concepts for the treatment of DOE MLLW have surfaced from the DOE National Laboratories and private technology developers. The concepts typically focus on one aspect of a total treatment system -- usually the organic destruction operation or stabilization of the treatment residues. For a waste profile that is primarily organic, the volume reduction associated with thermal treatment is very large. For wastes consisting of inorganic material, volume reduction may be small. The ITTS studies were commissioned by DOE to examine the effectiveness of some of the technologies when combined into an integrated system for treatment of the "average" waste described above. The technologies and a brief summary of the key features of each system are shown in Table III. More details on the technologies are presented in a companion paper in this conference (5). The primary treatment technologies include conventional incineration as well as many non-conventional thermal technologies. For most incineration systems considered, the bottom ash, and usually the fly ash, is converted to a vitrified waste form for disposal. Other systems use polyethylene or grout for primary system residue stabilization. Some of the most innovative systems have the primary treatment combined with the primary stabilization.

In addition to the primary treatment systems, additional subsystems re required to treat the remainder of the waste (mercury, lead, and special waste; metal decontamination; and metal melting (Table II)). These subsystems, which are common to all systems in the study except those that provide the metal melting function in the primary system, also contribute to the total system waste residues.

STABILIZATION OPTIONS AND ADDITIVE REQUIREMENTS

As noted in Table III, the disposal waste forms include vitrified material, polymer stabilized material, and grouted material. For vitrification of the treatment residues, the requirement for glass forming reagents will vary depending upon the post-thermal treatment composition and the type of glass or glass-ceramic final waste form being produced. For this study, it was assumed that the vitrification would be conducted in a melter capable of operating near 1600C. The ash melts easily at this temperature and, depending on its composition, can become an acceptable waste form without any other additives (6). However, to ensure that the ash will be acceptable for disposal, additives are expected to be necessary. Soil is one such additive that has been considered because it can produce an excellent, leach resistant glass-ceramic when melted and cooled (7). Work on vitrification of surrogate waste residues has shown that for virtually any credible post-thermal treatment composition, a soil additive ratio of 40 to 60% (by mass, referenced to the post-combustion residue (ash)) produces a good final waste form (6,7). Thus, to establish an estimate of the disposal residues, a mass ratio of soil to waste residue of 0.5:1 was chosen for the ITTS mass balance calculations. If contaminated soil is used in the vitrification process, then additional benefits are achieved by treating two waste streams in one process.

For the ITTS systems that use polymer stabilization as the primary stabilization option, a mass ratio of post-thermal treatment residue to polyethylene of 1:1 was selected based on work done at the Brookhaven National Laboratory on polyethylene stabilization. As was the case for vitrification, sometimes a higher waste mass loading is possible with polyethylene but a conservative value was chosen for this

study.
For systems that use grout for stabilization, a mass ratio of cement to waste residue of 2:1 was selected. This mass ratio is believed to be suitable for ash waste streams that have minimal chloride residues. The ITTS air pollution control systems are configured to separate the flyash from the scrubber salt residues and should meet the low chloride requirement in the ash residues.
Systems that use polyethylene or grout as the primary stabilization agent do not require addition of soil to produce an acceptable waste form. However, polyethylene or cement must be added, which results in a greater mass and volume for disposal. For example, if a system produces 1000 lb of ash residue, 500 lb of soil would be necessary for vitrification, but 1000 lb of polymer or 2000 lb of cement would be necessary for polymer and grout stabilization, respectively.
All systems that produce a salt in the scrubber blowdown use polyethylene for salt stabilization. Systems producing calcium carbonate and calcium chloride also use polyethylene for stabilization of those salts.

CALCULATED DISPOSAL VOLUMES

Using the stabilization ratios discussed above, the mass balance and disposal volume generation rates for the ITTS systems were calculated for the waste composition and processing rates shown in Table II. The calculated disposal volume rates for the ITTS systems are shown in Fig. 1. The volume rates were generated using the ASPEN PLUS computer code for the primary system mass and energy balance (8).

DISPOSAL VOLUME SENSITIVITY

From Fig. 1, it is clear that disposal volumes vary considerably between the various systems. The total volume range between the systems studied is nearly a factor of four for the waste composition used. The range is attributable to both the choice of processing system and the choice of the stabilization media. For the baseline system (labeled A-1), a separate analysis (9) has been done in which process variations were selected to produce three different waste forms -- vitrified as in the baseline study, polyethylene stabilized, and cement grouted. This resulted in a waste volume generation rate for system A-1 of 13.6, 27.0, and 29.3 ft3/h for vitrified, polyethylene, and grout waste forms. The difference is attributed to the mass mixture ratios described above and the density of the final waste form.
The waste used in this study is mostly inorganic material, as shown in Table II, and only about 23% is combustible (these percentages exclude the soil used for vitrification). If the waste was higher in combustible material, the resulting disposal volumes from most of the systems would be lower.
Another assumption that affects the disposal volumes is the final disposition of melted or decontaminated metal. If the metal is separable from the slag or ash, then it can be melted into ingots and perhaps recycled. While the metal is likely to have some radioactive contamination , so that unrestricted release is not possible under DOE regulations, it probably can be used for fabricating waste storage containers. Currently, DOE is sponsoring several initiatives to investigate such metal recycle. In this study, it is assumed that the metal can be recycled and thus is not disposed. A total of about 1.3 ft3/h of ferrous metal is produced for recycle. If recycle is not viable, the waste disposal volume increases for most systems.

DISPOSAL COST

Disposal unit cost is one of the most uncertain parameters in this study. Very little data exist to estimate the unit cost of disposal on DOE sites because previous costs have been combined with various other operational activities and separating the disposal costs is difficult. Informal, undocumented estimates have suggested costs between $50 and $100/ft3. Direct inquires in FY 1994 for disposal of uranium oxide at the Nevada Test Site and Hanford, the only two sites accepting wastes other than their own, produced numbers of $10 and $58.70/ft3 for low-level waste and $36 and $168.68/ft3 for mixed waste (Reference 10). Commercial radioactive waste disposal rates can approach $300/ft3 and plans for new commercial disposal facilities often have rate projections that exceeded this amount. Consequently, for the ITTS study it was assumed that the disposal facility would be new and that it would meet Resource Conservation and Recovery Act regulations for mixed waste. Such a facility was costed at $243/ft3. The most recent waste disposal cost, which was provided by Envirocare, references the disposal cost in their contract with DOE (Reference 11). This rate was quoted as about $35/ft3.
Higher disposal unit costs provide greater incentive to minimize the disposal volume. Higher fractions of noncombustible residues increase the importance of

maximum waste residue densification. Fig. 2 shows the impact of the disposal unit cost on total life cycle cost for the baseline system producing grouted and vitrified waste forms. For this system, the grout-based waste form has a lower life cycle cost if disposal unit costs are less than $58/ft3. For disposal units costs above that amount, using a melter to vitrify the residues yields a lower life cycle cost.

For vitrification, system A-1 uses 498 lb. of contaminated soil per 1000 lb of waste to provide the proper quantity of glass formers as discussed earlier in this paper. There are large quantities of contaminated soil throughout the DOE system that are not listed in Table I. This soil equates to about 7.4 ft3/h of additional waste (at 67 lb/ft3) and produces an additional 2.7 ft3/h of glass-ceramic waste form (at 187 lb/ft3), which is included in the totals of Fig. 1. A cost credit can be claimed for the systems that treat this contaminated soil. The credit depends on the cost of treating and disposing of the soil using alternative means. An alternative treatment cost of $300/yd3 for contaminated unsolidified soil would produce a credit of $82/hr when the soil is used as a glass former in the ITTS system. Since the baseline system disposes of 13.6 ft3/h, the credit for treatment of the soil is estimated to be about $6/ft3 of vitrified waste. Thus, the break-even point for vitrified vs grouted waste disposal is reduced to about $52/ft3. If the cost of remediating the radioactively-contaminated waste is higher, the credit is proportionately higher.

CONCLUSIONS

From the material presented in this paper, it is clear that different systems for processing complex waste compositions can produce widely differing disposal volumes. Disposal volume is a function of the waste makeup and residue solidification method. The systems in this study had disposal volumes that varied by nearly a factor of four -- from 10.7 to 42.5ft3/h. The choice of a technology and final waste form can strongly affect the total life cycle cost. Disposal unit costs greater than about $58/ft3 appear to justify the cost of residue vitrification on economic grounds alone, without giving credit for the use of contaminated soil in the vitrification process. If credit for the use of contaminated soil is included, this break-even cost reduces to about $52/ft3 (using a $300/yd3 avoided cost). Vitrified wastes provide other intangible benefits that are not the subject of this paper.

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46-46
ASSESSMENT OF THAILAND INSTITUTIONAL
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ABSTRACT
A survey of radioactive sources and types of wastes arising from their usage in Thailand has been conducted, following the analogous study performed in the year 1990. The significant increase in the volume of wastes from medical application was found, while a small change occurred in the other sectors. The projection of radioactive wastes generation in the next 30 years was established including the accumulated waste volume to be disposed of at about 3500 m3 in the year of 2025. From the information obtained, the conceptual plans for treatment, conditioning, storage and disposal facilities are drawn and presented.
INTRODUCTION
Since the introduction of the nuclear energy program to Thailand in 1962, the applications of nuclear techniques has gradually increased. At present, nuclear technology is well known by most of the Thai scientists, academicians, medical doctors, industrial men, and etc. The principal usages of nuclear technology are those in the medical application i.e. therapy and nuclear medicine, research work, agriculture and industry. Currently, there are 245 licensed radioactive users(1). These licensees are a heterogenous mixture of individuals and institutions. Apart from net benefits, radioactive uses is inevitably accompanied by some negative aspects. One of them is the production of radioactive wastes.
It is a worldwide practice that radioactive wastes have to be kept under control and that their potential impact on man and his environment has to be acceptably low. Therefore, it is the policy of the Thai Atomic Energy Commission that the Office of Atomic Energy for Peace (OAEP) has to render the service of management of the radioactive wastes arising in Thailand. It is the duty of the Waste Management Division to fulfill this mission properly. All radioactive wastes generated by any user in Thailand are sent to this division for treatment and temporary storage. At present, the division is operating a waste management facility consisting of incoming waste storage tanks/rooms, a precipitation plant including ion-exchangers unit, a compactor, a small incinerator, a cement solidification unit, a crushing machine and temporary storage facility. Owing to the increasing applications of radionuclides which in turn results in the increase of the radioactive wastes generation, the present temporary storage at OAEP needs to be expanded or to be replaced by a new permanent disposal site. It is also necessary for OAEP to improve the treatment technology to serve all radioactive wastes generated in the future. In December 1989, the Thai government set a policy to establish the new Nuclear Research Center in a low population area outside Bangkok. Waste Management Facilities will also be attached to that site as a Centralized Waste Processing and Storage Facility (WPSF). At present, this Project is being processed. The new treatment technologies and interim storage facility for processed waste packages will be provided and put into operation within this decade. Additionally, the OAEP is also planning to construct the disposal facility in the near future.
To achieve in managing of radioactive waste, it is necessary to known about sources, significant portion, and characteristic of radioactive waste generated in this country. Therefore, in 1993, Chulalongkorn University in cooperation with the OAEP conducted a survey, via a mailed questionnaire and by personal contact, about sources, types, volumes, characteristics of radioactive wastes arising in Thailand. The information obtained from the survey were used and referred to for the WPSF's conceptual plan for treatment, storage and disposal facilities.
SOURCES AND TYPE OF RADIOACTIVE WASTES
The radioactive wastes generated in Thailand originate from 123 licensees at present. There are 34 medical schools, 24 hospitals, 30 academic and research institutions, and 35 industries. Type of wastes can be classified into two categories.
According to source : Data regarding the uses of radioactivity were used to categorize the wastestreams. The three resulting wastestreams were medical, academic and research, and industrial. The distribution of type of Institution by wastestream

is presented in Table I.
According to waste form : this category can be divided into
- liquid waste form
- solid waste form
The waste compositions are shown in Table II. The produced wastes were sent to the OAEP which is responsible for controlling and managing all radioactive wastes produced in Thailand.

THE PROJECTION OF WASTE FOR THE NEXT 30 YEAR

For estimating the amount of waste production in the next 30 year, the personal communication and information obtained from the survey are used along with the following assumptions.
the projection plan starts from 1995 and ends at 2025
300 m3 of treated wastes are accumulated at the OAEP before starting the plan
From 1995-2005, the average annual treated waste will be 70 m3 based on the survey data.
From 2006-2025, the average annual treated waste will increase to 100 m3 according to the trend in increasing applications of radioisotope uses after the New Nuclear Research Center to be established.
500 m3 of waste volumes from abnormal operation will be reserved.
Therefore, approximately, the total accumulated waste generation to be disposed of at the end of the year 2025 will be 3500 m3. This amount of estimated wastes excludes the LLW and ILW from Nuclear Power Plant which may be operated at about the middle of the projection plan period. The example of the estimated waste volumes for 10 years period (1995-2005) is shown in Table III.

WASTE PROCESSING AND PACKAGING TECHNOLOGIES

In order to treat for all annual generated wastes, the processing methods, treatment technologies available now and in the future at the WPSF can be categorized and summarized as follows :

Liquid Waste Treatment

Chemical precipitation process accompanied with sludge separation process and an ion-exchanger have been used for treatment aqueous liquid wastes. For organic liquid waste, the incineration or immobilization with cement may be applied. The suggested capacity of precipitation/separation process would be 5 m3/hr.

Solid Waste Treatment

There are two suggested treatment processes for solid wastes :
Incineration: combustible materials such as papers, rubber, protective clothes, plastics and others are put into an incinerator which is associated with the off-gas treatment system. The suggested capacity of the incinerator would be 20 kg/hr of dry solid waste.
Compaction: non-combustible materials but compactable such as metal cans, glasswares, etc. are pressed by hydraulic compactor. The volume reduction factors would be in the range of 5-10.

Waste Packaging

The treated wastes shall be immobilized by cementation and cement encapsulation in 200 liter steel drum.
The flow charts of conceptual plan for processing methods are shown in Fig. 1.

INTERIM STORAGE FACILITY

The processed waste packages are temporarily stored pending final disposal. The conceptual design for storage facility would be based nominally at 10 years of the expected annual waste quantity after treatment and immobilization. (The expected quantity would be about 350 drums/year in the first 10 year plan.) Stacking the drums should be limited to three high, the storage area design should be expandable in simple modules.

DISPOSAL TECHNOLOGIES

The preliminary study for site screening has been done(2). The potentially suitable areas for waste disposal (unshaded areas) are shown in Fig. 2. From the topography and climatic considerations, three types of disposal design are suggested. There are Shallow Land Burial (SLB) with concrete floor, Above Ground Vault (AGV), and Below Ground Vault (BGV). The design basis for SLB, AGV and BGV is described as follows :

Site Capacity:

The capacity of all conceptual designs considered in this study is 3,500 m3 of waste at the time of disposal according to the waste volume estimated above.

Disposal Unit Capacity:

Each disposal facility is composed of four disposal units, whose capacity vary from facility to facility. Each disposal unit is divided into several cells. Each disposal cell is sized so that it contains at least one year waste generation.

Segregation of Waste by Waste Class:

In all disposal facilities, it has been required that all waste be in a stable form. ILW will be placed at the bottom of disposal cell. Void spaces between waste containers are be backfilled with earth material in all conceptual disposal facilities.

Support Facilities:

Common support facilities are provided for all disposal technologies. These include components such as the administrative, health physics, security, waste storage and equipment storage/maintenance building, construction warehouse, holding and evaporation ponds, and truck washdown facility. Waste storage facility is provided with a capacity sufficient to accommodate deliveries of wastes for three bad weather months (the wet season) or about one-fourth of the annual waste volume to be disposed.

Buffer Zone:

For all disposal technologies, a buffer zone of 100 meters is provided between the restricted and the unrestricted land areas. The administrative facilities are outside the restricted areas, but within the buffer zone.

The layout and section drawing of BGV and AGV disposal facilities are shown in Fig. 3 and Fig. 4, respectively.

CONCLUSIONS

The relatively static population of medical institutions produce a significant fraction of the total waste volume, if the liquid wastes from research reactor is excluded. The radioactive wastes generated from academic and research institutions depend upon the research projects they received in those years.

Most of the produced radioactive waste were LLW and ILW. The main radionuclides contained in the wastes are : H-3, C-14, P-32, S-35, Ca-45, Cr-51, Tc-99m, I-125 and I-131 and the average activity was less than 109 Bq/m3.

In the year of 2025, the estimated accumulated volumes to be disposed of would be 3500 m3. The four treatment facilities : incineration, compaction, precipitation and solidification with enough capacity would be used for all wastes generated in the future.

For selecting the suitable disposal technology, a more detailed assessment on a smaller scale using more detailed topography and other maps should be studied.

In the future plan, if the final disposal site would be established near the WPSF which is located in humid region and has a low depth of ground water table (<3m), the above ground vault technology would be the most appropriate.

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46-47

MANAGEMENT OF EXCESS HIGH EXPLOSIVES AT PANTEX PLANT

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ABSTRACT

The Department of Energy's (DOE) Pantex Plant demilitarizes about 45,000 kg of high explosives each year by burning them outdoors (commonly called "open burning)." Half of this material comes from nuclear weapons that have been removed from the nation's stockpile and dismantled. Pantex Plant is the only DOE site in the nation where extensive dismantlement is done.

Burning the explosives demilitarizes them and removes all classified characteristics at the same time. Doing this outdoors is the safest and most economical way. It is a method approved by the Environmental Protection Agency (EPA) because it is

considered the best demonstrated, available technology for treating explosives. Such burning is conducted at the Pantex Plant at a special site according to strict procedures that ensure safety for personnel and the environment. Even so, alternatives to burning of these explosives are being explored.

The best alternatives to open burning are to reuse the material or sell it to commercial firms. Such sales would yield considerable savings. The major problem is that the material must first be cut up to destroy its classified nature, and such processing is inherently dangerous.

The most promising non-thermal alternative is a process known as "base hydrolysis." In this process, bases such as sodium hydroxide attack heterocyclic explosives such as 1,3,5,7-tetranitro-1,3,5,7-tetrazacyclooctane (HMX) or 1,3,5-trinitro-1,3,5-triazacyclohexane (RDX). The primary products are nitrogen oxides and sodium formate, a salt.

INTRODUCTION

The U.S. military and the Department of Energy burn, detonate, or incinerate thousands of tons of explosives each year. The military's munitions are destroyed when it is determined that they are no longer necessary for the nation's stockpile. The DOE, as part of its international commitment to remove nuclear weapons from the stockpile and dismantle them, demilitarizes and removes classified information from explosives and explosive-contaminated parts, as well as treats explosive waste by burning. Most of the dismantlement work for the DOE is done at the Pantex Plant near Amarillo, Texas. The Plant is operated for the DOE by Mason & Hanger - Silas Mason Co., Inc. Three categories of materials are burned there: 1) explosives that were the main charges in the weapons, 2) small components that contain only a few grams of explosives, and 3) wastes consisting of explosive-contaminated waste materials and explosive waste.

OUTDOOR BURNING

Background

The Pantex Plant processes about 45,000 kg of explosives and associated material each year by burning outdoors (also called "open burning.") Of the material that is strictly high explosives, half is main-charges from nuclear weapons and half is scrap from manufacturing activities on the plant site. The amounts of explosive components, explosive-contaminated parts and explosive-contaminated waste are highly variable.

Most of the bulk explosives are plastic-bonded materials made from either HMX or RDX and a plastic like Viton. The molecular structures of these compounds are depicted in Fig. 1.

Burning these explosives out-of-doors is the safest way to treat them. Open burning is a method that has been approved by the EPA for treating explosive wastes, because it is considered the best demonstrated, available technology. For the explosive components or explosive-contaminated components, such burning renders them unsuitable for military use and removes all classified characteristics at the same time. Moreover, it is economical.

Special Handling

The remainder of this section on open burning deals with the treatment of main-charges and scrap explosives. Explosive components and wastes are handled separately by a different procedure, and are not discussed.

Before the high explosives leave the weapon-disassembly buildings, they are packed in special protective containers and transported by truck to special buildings for temporary storage. The buildings are bunkers like the one depicted in Fig. 2. The bunkers' walls are made of reinforced concrete, and the buildings are flanked by hills of dirt that would deflect upwards the force of a blast in the unlikely event that the explosives stored there were to detonate.

From storage, the explosives are reloaded aboard specially-designed trucks and carefully transported to the "burning ground" for final treatment.

At each step in the process, the explosives are handled according to strict, written procedures by trained personnel who work in pairs under the guidance of a supervisor. Such care ensures the safety of the workers and guarantees that the facility remains in compliance with all rules and regulations.

The "Burning Ground"

Burning is done at a remote, 0.24-km2 site on the Pantex Plant reservation. The reservation itself is in an agricultural, sparsely-populated area. It occupies about 64 km2 of prairie in the Panhandle of Texas. The nearest town has a population of

2,300 and is 16 km to the east. The nearest major city is Amarillo (population 165,000) about 27 km to the southwest.
Access to the reservation is restricted, of course, but the burning ground is further secured against entry by unauthorized personnel with a gate and an orange barrier across the road leading to it. Warning signs are posted at the entrance.
Nine "burning trays" are regularly used. As shown in Fig. 3, the trays are steel tables measuring about 1.2 m by 6.1 m, having low walls lined with firebrick. A layer of sand covers the steel tray and insulates it against the heat. The roof-like structure on wheels at the end of the tray is a cover that is rolled over the tray at the end of the day to protect the trays and the ashes from rain.
The trays are widely separated from one another, and they are surrounded by a strip of plowed earth that guards against fire spreading beyond the immediate area should burning materials fall off them.
Strict Procedures
Once at the burning ground, the containers of explosives are unloaded from the truck either by hand or by means of a forklift truck, depending on the size of the containers. The containers are placed beside the burning trays; then the containers are opened one at a time, and the explosives are carefully removed by hand. They are placed on the trays and arranged over the surface. A paper napkin is wet with diesel fuel and taped to a piece of explosive. A squib is then placed in the napkin.
The workers ignite the fire from the safety of a bunker by electrically actuating the squib. The flame propagates from one piece of explosive to the next without causing any to detonate.
The fire burns for only 100 to 300 seconds, but the workers wait 30 minutes before going to visually inspect the tray for any unburned explosives. Up to 680 kg of explosives can be safely burned at a time.
Air Emissions
The burning ground currently operates under a written grant of authority from the Texas Natural Resources Conservation Commission (TNRCC). For the thermal treatment of waste, it operates under Resource Conservation and Recovery Act (RCRA) Interim Status. A permit modification for the burning ground is pending.
Although high explosives burn rapidly and cleanly, some explosives contain fluorine because some of the plastic binders contain fluorine. As shown in Table I, the allowable concentration of hydrogen fluoride is the most restrictive of the pollutants that might be found in the smoke. For this reason, the amount of fluorine-containing explosive that is burned at any time is carefully regulated.
The combustion products from a typical fluorine-containing explosive are listed in Table II. Although carbon monoxide and oxides of nitrogen are largely absent, the amount of hydrogen fluoride is 2% of the total. Because the allowable concentration of hydrogen fluoride is only 4.9 mg/m3 (per Table I), this is the limiting factor when such explosives are burned. Accordingly, fluorine-containing explosives are burned only once every three hours, and the amounts that are burned are carefully limited. By contrast, the explosives that contain no fluorine can be burned in almost any amount, subject only to the self-imposed limitations that have been established because of safety considerations.
Planned Improvements
Because no acceptable alternatives to open burning are yet available, the present facility is to be modified and improved. For example, the "burning trays" on which bulk explosives are burned will be equipped with natural-gas burners and will have a new, remote control system for ignition. Other improvements being considered include a gas-fired chamber to treat equipment that has been contaminated with explosives. Although open to the atmosphere, the chamber would have a way for gaseous emissions to be sampled, and rainwater runoff would be controlled. The improvements, costing about $1 million, are provided for and subject to the requirements of the proposed Hazardous and Solid Waste Permit modification to be issued by the TNRCC.
ALTERNATIVES TO OUTDOOR BURNING
Introduction
Although open burning is the best demonstrated, available technology for treating explosives, Mason & Hanger's researchers are seeking alternative methods of dealing with these materials. No single technology will serve all needs, because there are several, diverse process streams involved. Not only must main-charge explosives be treated, but there are numerous mechanical components that contain small amounts of energetic material that must be demilitarized and sanitized, too. Explosive waste

and explosive-contaminated wastes must also be treated. In general, a different technique is required for each kind of component or waste material.

Recycle

The best alternative to burning the main-charge explosives is to recycle by grinding the explosive to powder, then pressing it to form new charges. This option is especially attractive for the newer, insensitive explosives such as 1,3,5-triamino-2,4,6-trinitrobenzene, known as "TATB." (TATB is one of the best high explosives for making weapon components, because it is so insensitive that it is very safe to handle.) If TATB could be recycled, the current inventory of this material would be extended indefinitely. Recycling the old types of explosives is not desirable, because they are no longer being used much in new weapons.

Although grinding TATB to powder and then pressing it into a new part is easy to do, it is not known whether some important property will be adversely affected. Experiments completed so far show that, after three cycles of grinding it to powder and then pressing the powder to shape, the density of the charge has been satisfactory; and no defects, such as cracks, have been observed. Particle size measurements show the particles are becoming smaller, which is desirable.

Reprocess, then Sell Commercially

The second best alternative to open burning is to sell the excess material to commercial firms. Such sales would yield considerable savings in the costs associated with warehousing and with outdoor burning. However, many potential customers want pure explosives, not material containing plastic binders.

To purify the material requires that the plastic-bonded explosive be dissolved in a solvent, and the purified HMX or RDX be recovered by recrystallization. However, only a few solvents are suitable for this. Reprocessing is more complicated than simply recycling the material, because the particle morphology may be changed, yielding a much more sensitive, and more hazardous, material. Then, too, the plastic binders and dyes found in many of the explosives present problems and they must be cleanly removed. Fortunately, a Mason & Hanger researcher has developed a recovery process and is seeking a patent for it.

Even when the explosives from weapons can be sold without removing the binders, the main-charges must be processed to remove classified information. Such handling is inherently dangerous. As an alternative to cutting the explosives, Mason & Hanger researchers are investigating the feasibility of pulverizing the explosives by first freezing them in liquid nitrogen, then crushing them while they are still brittle.

New, Non-Thermal Methods

It is unlikely that all explosive material can be recycled or sold, so some will have to be managed in some other manner. As a means of demilitarizing and sanitizing large pieces of explosives, workers at the Pantex Plant have collaborated with those at Los Alamos National Laboratory to develop a process known as "base hydrolysis." In this process, a base attacks the heterocyclic compounds to produce, primarily, nitrogen oxides, nitrite, and formate. Acetate and other minor products are also formed. Mason & Hanger has perfected this base hydrolysis process and is seeking patent protection for the improvements.

Small Components That Contain Explosives

Besides main-charge explosives, the Pantex Plant has components and other items that contain small amounts of explosives. Detonating fuses are especially troublesome items. As depicted in Fig. 4, these fuses are sheaths or tubes of lead filled with an explosive. These fuses cannot be treated by open detonation, because lead particles would be thrown about. To solve this problem, a special chamber has been designed at the Pantex Plant in which the lead is melted and recovered, while the explosive burns harmlessly away.

Other small components from dismantled weapons can be rendered harmless by discharging them inside a chamber, but this process is very labor-intensive. Another approach being investigated is to pyrolyze the explosive in a plasma torch. Such torches are commercially available and, when operated under reducing conditions, they have not been defined as incinerators by the EPA and several state agencies. Preliminary tests by Plasma Energy Applied Technology in Huntsville, Alabama, show that although the components may occasionally explode, such explosions are small enough that the equipment is not damaged, nor are fugitive emissions released to the atmosphere.

CONCLUSION

Mason & Hanger is in the forefront of managing excess high explosives. Although open

burning is still the best method for treating such explosives, base hydrolysis and other alternatives are being developed for use with some of the explosive materials.

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SECONDARY WASTES AND HIGH EXPLOSIVE RESIDUES GENERATED DURING PRODUCTION OF MAIN HIGH EXPLOSIVE CHARGES FOR NUCLEAR WEAPONS

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ABSTRACT
This study identifies the sources of high-explosive (HE) residues and hazardous and nonhazardous wastes generated during the production of the main HE charges for nuclear weapons, and estimates their quantities and characteristics. The results can be used as a basis for design of future handling and treatment systems for solid and liquid HE residues and wastes at any proposed new HE production facilities.
This paper outlines a general methodology for documenting and estimating the volumes and characteristics of the solid and liquid HE residues and hazardous and nonhazardous wastes. We prepared volume estimates by applying this method to actual past Pantex plant HE production operations. To facilitate the estimating, we separated the HE main-charge production process into ten discrete unit operations and four support operations, and identified the corresponding solid and liquid HE residues and waste quantities. Four different annual HE main-charge production rates of 100, 500, 1000, and 2000 HE units/yr were assumed to develop the volume estimates and to establish the sensitivity of the estimates to HE production rates.
The total solids (HE residues and hazardous and nonhazardous wastes) estimated range from 800 to 2800 ft3/yr and vary uniformly with the assumed HE production rate. The total liquids estimated range from 73,000 to 1,448,000 gal/yr and also vary uniformly with the assumed production rate.
Of the estimated solids, the hazardous wastes (e.g., electrical vehicle batteries and light tubes) were about 2% of the total volumes. The nonhazardous solid wastes were not very sensitive to the HE production rates because the air filter change-outs from the building air handling units did not depend on HE throughputs. The generation of solid HE residues varied uniformly with the HE production rates and ranged from about 20% of the total solids volume for the 100 HE units/yr case to about 60% for the 2000 units/yr case. The HE matching operations generated 60 to 80% of the total solid HE residues, depending on the assumed production rate, and were also the sources of the most concentrated HE residues.
Of the total estimated liquids, the nonhazardous wastes were dominant, with 99% of the total liquid volume being generated from the use of a once-through (i.e., no-recycle) HE machining cooling-water system. Future plants will most certainly use recycle systems and, therefore, not generate such large nonhazardous waste quantities. Thus, more meaningful comparisons of these volume estimates can be derived for design purposes by removing the once-through machining cooling water quantities and then making a new set of comparisons. The volumes of hazardous liquids are relatively small, ranging from 30 to 600 gal/yr depending on the HE production rates. The hazardous liquids are generated by x-ray film processing and analytical chemistry laboratory operations, with the former accounting for about 80% of the total hazardous liquid wastes. The liquid HE residues are generated primarily from the wet mopping of floors and contain low concentrations of HE residues. The maintenance of the wet scrubber system in the HE pressing area generates the same quantity of HE residues for all HE production rates and is only 10 to 30% of the total liquid HE residues.
We conclude that the total quantities of solid and liquid HE residues and hazardous and nonhazardous wastes generated from the production of main HE charges are not very large compared with those generated by many other types of production processes. Nevertheless, these solids and liquids must be addressed in the design and operation of future plants. This conclusion is particularly valid if future production plants eliminate the use of once-through machining cooling water systems, which generate large quantities of nonhazardous aqueous liquids.
Additional analyses of these results and the information developed in this study can provide insights into the specific HE production process operations for which wastes

and HE residues can be minimized, eliminated, or collected for recycle. The methodology can be extended to other parts of the HE production processes (synthesis, formulation, and nonmain-charge HE components) and nuclear-weapons dismantlement operations.

INTRODUCTION

As a participant in the DOE Weapons Complex Reconfiguration Project - Complex 21, Lawrence Livermore National Laboratory has been assigned the lead laboratory role for the HE production operations. This assignment includes preparing design criteria for use by DOE and the architect engineer (AE) in developing new high-explosive production facility conceptual designs for the next-generation weapons complex. A key part of developing new facility designs is providing the necessary waste-treatment facilities and equipment for processing all solid, liquid, and gaseous wastes and HE residues resulting from high-explosive production operations. Design of waste-treatment facilities and equipment requires that estimates of the amounts of HE residues and wastes and their characteristics be available.

This study identified the major sources of solid, liquid, and gaseous secondary wastes and HE residues generated during actual past production of HE main charges at the Pantex facility. We estimated the amounts and characteristics of these wastes and residues on the basis of the currently constructed Pantex facility. These estimates were developed by establishing a reference HE production flowsheet, by interviewing Pantex staff experienced in actual HE production operations, by walking through the Pantex facility production lines, and by using engineering judgment. To facilitate the systematic development of these estimates, we defined and used a three-step estimating methodology.

This report summarizes the results of the study and provides the methodology and major assumptions used to estimate the secondary wastes and HE residues. The estimates can be used both to prepare design criteria and identify candidate process areas for waste minimization. In addition, since a primary goal of the lead laboratory effort is to treat all HE residual materials as an asset, assessments must be made to consider recycle and reuse of HE residues, even though some residues must be destroyed. With this information, the design bases for a new HE main-charge production facility can be prepared.

METHODOLOGY FOR DEVELOPING ESTIMATES OF WASTES AND HE RESIDUES

We used a three-step approach to prepare the estimates. In step one, a reference process flowsheet and facility layout were defined on the basis of the Pantex facility HE main-charge production processes. We then divided the production process and facility layout into ten discrete, major, unit-operations areas. Figure 1 summarizes these ten areas. In addition, we defined four areas for support or ancillary operations, which are shown at the bottom of Fig. 1. Each area was established on the basis of past HE production process operations or current Pantex facility layouts. All areas were selected so as to be correlated with discrete groups of process operations known to create major quantities of secondary wastes and HE residues.

In step two, we studied each area in sufficient detail to identify the major equipment and operations required to carry out the functional operations. We used this information to develop process flow diagrams that show the detailed subactivities for each of the blocks in Fig 2. Figure 2 shows an example of these diagrams. We used these levels for greater detail to estimate the actual secondary wastes and HE residues that required collection and treatment in all facility designs.

In step three, we examined all activities or operations individually in order to develop estimates of solid, liquid, or gaseous waste characteristics and quantities. Where insufficient Pantex production staff experience existed, or where Pantex production records of waste and HE generation were incomplete, we used engineering judgment to develop the estimates. We then estimated the total volumes resulting from the HE main-charge production processes by tabulating the results from each of the ten discrete areas and four support areas for each waste and HE residue type.

Reference HE Production Facility and Operations

The Pantex production site is located about 18 miles northeast of Amarillo in the panhandle region of northwest Texas. The site is located on U.S. DOE property and has rail and truck access. The feed materials required for the HE production process are received and stored in storage igloos at the on-site receiving and storage area.

The main-charge HE production facilities at the Pantex site are housed in four buildings. All of these facilities are assessed in this study. They are 1) the HE and materials staging facility, 2) the main He pressing areas, 3) the radiography/x-ray facility, and 4) the HE machining areas. The building that houses the main HE pressing areas also contains the isostatic press control room, the rotoclone ventilation scrubber system, and the isostatic press fluid equipment room. The major activities performed in the building that houses the HE and materials staging facility include the receipt and staging of HE process feed materials from the storage igloos and from other staging operations, the staging of intermediate products, and the staging of HE residues from the production processes. HE process feed materials are stored in one of seven magazines until required by the production process. Intermediate HE production process products are also stored in sealed cans containing a desiccant and are staged during various phases of the HE production process. Production wastes, HE residues, and other materials are placed in cans until final disposition is determined.

In examining the HE main-charge pressing operations, we looked at both the pressing operations and associated materials-handling operations. The HE main-charge pressing operations are conduced as follows. HE materials are prepared and then sent to undergo heat treatment and subsequent materials loading into the HE mandrels/liners. The main HE charges are then isostatically pressed. The pressed HE mold breakout and initial shadowgraph product inspections are then carried out. HE products passing the shadowgraph inspections are sent in sealed cans to the radiography/x-ray building to be x-rayed or to the HE and materials staging facility for staging prior to x-ray. Any shadowgraph rejects are sent in cans to the HE and materials staging facility.

HE specimens that pass the x-ray inspections are sent to the building that houses the main HE pressing area. The specimens then undergo the lathe machining operations that are required for fabricating the rough outer, final inner, and final outer HE shapes. Although the specific order varies for different weapons components, it is assumed that the HE products from the rough outer shaping are then transported elsewhere in the building in sealed cans for density measurement and dye penetrant crack measurements. After these measurements, they are transported back to the machining area for the final inner and outer shaping operations.

The shaped HE products are then transported in sealed cans to an area where the milling machine operations required to fabricate all special shape features into the primary HE charges are performed. The machined HE products are then transported in sealed cans for dedicated gauging measurements or for gauging with a coordinated measuring machine. HE products meeting specifications are transported in sealed cans to the HE and materials staging facility for staging before being sent to subassembly. Any HE products not meeting the inspection requirements are transported in sealed cans to the HE and materials staging facility for interim product storage.

To develop the waste estimates, we assumed the specific process flow associated with the activities described above, including all the necessary HE main-charge production operations that can generate wastes and HE residues. The specific process flow of the inspection operations, gauging operations, lathe operations, and milling operations can very depending on a specific HE charge component design. In addition to the parts of the building housing the HE machining areas, we examined two associated support equipment facilities in other buildings, required by these operations for generation of waste and HE residues. One facility houses a waste-water treatment system and the other houses a central vacuum system required by the HE machining operations.

The radiography/x-ray operations generate wastes during the setting up and taking down of HE main charges from the film racks. However, the development and processing of the film generates the most significant waste quantities. These wastes, many of which are hazardous, include developer and fixer liquid solutions, drying solid materials from film processing, film processing rollers, lead cassettes, and lead film filters.

In addition to the main production process operations, there are four ancillary support operations (Fig. 1). These include 1) preventative maintenance, 2) analytical laboratory, 3) mechanical tests (mechanical and physical HE testing) and 4) HE test firing. The analytical and test facilities provide support services for HE lot acceptance, determinations from suppliers, and HE product certifications processes. Preventative maintenance includes major maintenance operations and

buildings (e.g., ventilation systems and lighting), and equipment used for the main HE production processes (e.g., forklifts and carts).
The mechanical and physical testing of HE is conducted in the same building as the one used for radiography. The operations include conducting tensile (dog bone) tests, compression (nonfailure) tests, thermal (grain growth) tests, and test specimen gauging measurements. Specimens requiring testing are received in sealed cans from either the HE and materials staging facility or the radiography/x-ray facility. The tests do not produce significant fracture particulates from the HE specimen. In fact, all specimen pieces are placed after testing in sealed cans as HE residues and are returned to the building housing the HE and materials staging facility for final disposition. Small quantities of trash (wipes and gloves) are generated during the testing.
The HE test firing operations are conducted near the on-site receiving and storage area. The preparation and setup for the test firings, as well as the actual test firings, generate wastes and HE residues. These test firing operations include measurements of HE hydrodynamics, detonation velocities, gap sensitivities, divergence, and snowball/onion skin tests. These operations generate wastes that include test cables, plastic tubes, styrofoam, wood, glass mirrors, and metals. During the test firings, the HE is consumed; however, the test remnants are currently classified as containing HE residues. The test setup operations generate primarily nonhazardous trash.
The analytical chemistry laboratory measurements are conducted in a separate building. These operations include both wet and dry sieving particle size measurements, numerous sample weighings, bulk density measurements, gelchromatograph analyses, solid (TATB and KEL-F) digestions, titrations, volatility measurements, infrared spectroscopy, gas chromatograph analyses, and burned ash content determinations. Many tests generate wastes classified as hazardous because of the organic contents. Typical wastes include gloves, wipes, reagent bottles, glassware, analytical residues, and spent test solvents.
The preventative maintenance operations that generate wastes and apply to multiple HE production process steps include maintenance for electric forklifts, electric carts, building lighting, and building ventilation filters. Other major preventative maintenance operations are discussed in the specific HE production process steps. Hazardous wastes generated consist of spent lead-acid batteries and mercury-contaminated, crushed light tubes/bulbs. The routine scheduled change-outs of the 2 X 2 ft X 2 in. filters in the numerous building air handling units generate a large quantity of wastes readily amendable to volume reduction processes.
ASSUMPTIONS AND QUALIFICATIONS
We developed flow diagrams for the blocks shown in Fig. 1 in order to identify and quantify the major process operations and equipment that generate the residues and wastes. The ten major areas are:
1. Receipt and storage on-site of HE feed materials.
2. Staging of HE feed materials and production products.
3. Inspection and weighing of feed materials.
4. Heating and HE mold preparation.
5. Isostatic HE pressing.
6. HE mold breakout and inspections.
7. HE x-ray inspections.
8. HE shape machining.
9. HE product gauging.
10. Inspection of final HE products.
The four ancillary support areas are:
11. Preventative maintenance facilities.
12. Analytical laboratory.
13. Mechanical tests.
14. Test firing.
The ten discrete areas and four support areas can be correlated directly with the reference facility and process operations described above. We examined each area one at a time by interviewing Pantex production staff experienced in the actual HE production processes to identify specific activities and equipment associated with the process operations. The activities were systematically reviewed and all major secondary HE residues and waste streams identified, estimated, and recorded. Note that no estimates for general support facilities are provided in this initial study.

To facilitate the estimating process, all activities were assigned to one of four waste categories: 1) normal operations 2) housekeeping operations, 3) preventative maintenance operations, and 4) decontamination operations.
Normal operations are defined as daily activities performed to operate the HE production processes. Normal operations include HE materials process feed preparations, HE pressing, HE product inspections, HE machining, HE materials transfers, and HE material storage or staging.
Housekeeping operations are defined as routine activities performed to keep the HE production facilities free of trash and dirt. The solid-trash floor sweepings and mop water generated in areas that contain HE materials are generally assigned to this category. In some cases, wastes in this category were assigned to the normal operation category. If these wastes were generated in an area with HE materials present, they would require special handling because they potentially contain HE residues.
Preventative maintenance activities are defined as activities performed to keep and maintain the equipment operating at the assumed production rates. These activities include liquid-system filter replacements, building ventilation filter replacements, and routine maintenance of process equipment scheduled at regular intervals.
Decontamination activities are defined as those in which HE materials are specifically removed from equipment, molds, and machine tools.
Using these four waste categories and the process flow diagrams that we developed for the blocks in Fig. 1, we tabulated the solid and liquid HE residues and waste streams for each subactivity. Only solid and liquid wastes were identified. Gaseous wastes were assumed to be insignificant and were not tabulated.
We estimated rates of waste generation by either quantity per HE unit produced or by time interval (e.g., per shift, per month). For filter changouts and scheduled maintenance events, waste quantities were tabulated per unit of time. In most cases, Pantex staff provided rates during the interviews based on past production experience. Otherwise, estimates were based on engineering judgments. For each of the ten areas and corresponding process flow diagrams, a summary datasheet was developed and totaled. Datasheets were also developed for the four ancillary support operations.
The annual secondary waste quantities generally depend on the number of HE main charge units produced per year and the number of shifts per year. Therefore, we examined four different HE production annual rates of 100, 500, 1000, and 2000 units to evaluate the sensitivity of the wastes generated for various production rates. For all production rates, we assumed only one shift of 250 working days/yr, and a production rate of 10 HE units/day. The sensitivity of waste generation to multiple shifts per day was therefore not examined. Estimates of wastes generated when no HE production takes place in a year because of excess HE production capacity were also developed.
We assumed that all HE formulation and synthesis operations were performed by an off-site supplier. Those associated secondary wastes were not in the scope of this initial study. Such required HE production operations will need a similar quantification of associated secondary wastes for a new facility design basis. It is assumed that the previously formulated and synthesized HE process feed materials from off-site suppliers were received by truck shipments. HE materials production lot sizes of 30,000 lb and single truck shipments of 10,000 lb in separate 50-lb boxes were assumed.
To develop the waste stream characterizations, we classified the solid and liquid wastes as one of three types: 1) HE residues, 2) hazardous wastes, or 3) nonhazardous wastes. HE residues are process streams containing HE materials that must be collected, handled, and treated separately from other types of wastes because of the HE content. Hazardous wastes are used to identify waste streams currently subject to the requirements of the Resource Conservation and Recovery Act (RCRA). Nonhazardous wastes are used to identify wastes not subject to RCRA requirements and which are therefore treatable as conventional industrial or nonregulated wastes.
The content of HE residues varies widely among the specific waste streams. The use of two grades, to denote either relatively high concentrations or low levels of process streams contaminated with HE materials, could be developed in the next iteration of estimating the HE residue wastes. The process streams with the relatively higher concentrations would be the HE residues most likely to merit

further processing or treatment to recover HE materials in recycle.

RESULTS OF ESTIMATES

Solid Wastes and HE Residues

Solid wastes can be either compatible or noncompactible. Compatible wastes are materials that can be treated with a mechanical process to increase packaging density for disposal or storage. Noncompactible wastes are those at or near their maximum packaging density prior to any waste treatment. Typically, compatible wastes include plastics (packing material, bags, containers, boots, gloves, tubing, bottles, hoses), paper (box liners, packing material, wipes, coveralls, absorbent paper, wrappings, boxes, cartons), cloth (filters, coveralls, lab coats, rags, mops, gloves), rubber (mold liners, hoses, gloves, pads), metal (empty cans, clamps), filters (prefilters, ventilation filters, respirator canisters) and miscellaneous items. Noncompactible wastes include wood (mold tampers, packing), filters (cartridge type), wood or metal filter frames, discarded tools (hand, power-driven), dirt (dust, floor sweepings), glass (bottles, lab glassware, mirrors, instrument tubing), conduit (tubing, cable, wire, electrical fittings), pipe/valves (pipe, tubing, valves, fittings), failed equipment, and other miscellaneous items.

Using the methodology, assumptions, waste categories, and reference HE main-charge production facility described above, we developed estimates of the solid wastes and HE residues for each of the four assumed HE production rates. These estimates are summarized in Table I and Fig. 3.

For the total estimated solids, the hazardous solid wastes (e.g., electrical vehicle batteries and light tubes) were about 2% of the total solid volumes. The nonhazardous solid wastes were not very sensitive to HE throughputs. The solid HE residues varied uniformly with HE production rate and ranged from about 20% of the total solids volume for the 100 HE units/yr case to about 60% for the 2000 HE units/yr case. The HE machining operations generated 60 to 80% of the total solid HE residues, depending on the assumed production rate, and these were also the most concentrated form of HE residues.

Liquid Wastes and HE Residues

Liquid wastes are generated during the HE production operations. The HE machining operations are a major generator of liquid wastes containing HE residues. Any oils contain trace amounts that are too low to be explicitly considered. The liquids discharged from these machines are filtered within the machining bays to recover most of the HE residues before being discharged and routed to a second water cleanup system prior to final off-site discharge.

The wet rotoclone scrubber system used to collect HE production process dust generates liquid wastes containing HE residues during its routine maintenance cycles. The maintenance and periodic replacement of the isostatic press mobilnet fluid generates nonhazardous liquid wastes containing no HE residues. The development of films as part of the x-ray inspections generates both hazardous wastes containing silver ions and nonhazardous liquid wastes.

Liquid wastes containing varying concentrations of insoluble solids (dust and dirt), detergents, chemicals and dye penetrants are collected from various laboratory drains used for inspections, mechanical testing, chemical analyses, and film processing. Some liquids generated must be collected during special HE material decontamination of equipment. Additional liquids are generated infrequently when HE lots are changed and equipment is cleaned of HE residues from a previous lot. Wet mopping of the HE production bays generates liquids containing HE residues.

To determine the types and volumes of liquid wastes, we employed the same methodology, assumptions, and reference process used for estimating solid wastes. The estimates of liquid waste volumes are shown in Table II and Fig. 4 for each of the four assumed HE production rates.

For the total estimated liquids, the nonhazardous liquids dominated the total liquid volume, with 99% of this volume being generated from the use of a once-through (e.g., no-recycle) HE machining cooling-water system. Future plants will most certainly use recycle systems and, therefore, not generate such large nonhazardous waste quantities. Thus, more meaningful comparisons of these total liquid volume estimates can be derived for design purposes by removing the once-through HE machining cooling-water quantities, and then making a new set of comparisons.

Hazardous liquid wastes are generated by x-ray film processing and analytical chemistry laboratory operations, with the former accounting for about 80% of the total hazardous liquid wastes. The liquid HE residues are generated primarily from

the wet mopping of floors and contain low concentrations of HE. The maintenance of the wet scrubber system in the HE pressing area generates the same quantity of HE residues for all HE production rates and is only 10 to 30% of the total liquid HE residues.

SUMMARY

This study identifies the source of HE residues and hazardous and nonhazardous wastes generated during the production of the main HE charges for nuclear weapons, and estimates their quantities and characteristics. The results can be used as a basis for design of future handling and treatment systems for solid and liquid HE residues and wastes at any new HE production facilities.

Because of the uncertainty of future design bases, we assumed four different annual HE main-charge production rates (100, 500, 1000, 2000 HE units) to develop volume estimates and to establish the sensitivity of the HE residue and waste quantities to production rates. The total solids (HE residues and hazardous and nonhazardous wastes) estimated range from 800 to 2800 ft3/yr and vary uniformly with the assumed HE production rate. The total liquids estimated range from 73,000 to 1,448,000 gal/yr and also vary uniformly with the assumed production rate.

We conclude that the quantities of solid and liquid wastes and HE residues generated in producing main HE charges are not very large compared with those generated by many other types of production processes. However, the solids and liquids must be addressed in the design and operation of future plants. This conclusion is particularly valid if future production plants eliminate the use of once-through machining cooling water systems, which generate large quantities of nonhazardous aqueous liquid. Further analyses of these results and the information developed in this study can provide insights into the specific HE production process operations for which wastes can be further minimized or eliminated, and for which HE residues can be most efficiently collected for reuse or eliminated.

We recognize that a new HE production facility will use facility layouts and handling operations different from those of the current Pantex general facility. A new facility will thus not necessarily generate the wastes and HE residues summarized in Table I and II. For example, the use of simple water collection and treatment systems will allow full recycling of the HE machining water rather than a once-through treat and discharge system as is currently used. This single process modification would significantly reduce liquid wastes. Nevertheless, the study provides a tabulation of the functional requirements in the HE main-charge production process and the potential waste-generating activities. Facility designers can use the information developed in this study and the results of this study, which are based on actual HE production facility staff experience, to design a new production facility that minimizes the cost among the various engineering trade-offs between waste generation, waste treatment, and waste disposal.

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RESOLVING MIXED-WASTE ISSUES DURING UMTRA ACTIVITIES

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ABSTRACT

In 1981, the U.S. Department of Energy Grand Junction Projects Office (DOE-GJPO) was assigned responsibility for the Uranium Mill Tailings Remedial Action (UMTRA) Grand Junction Vicinity Properties Project. This project required planning, characterization, design, and remediation of more than 4,000 properties in the vicinity of Grand Junction, Colorado.

The 4,000 properties were contaminated with radioactive uranium mill tailings as a result of milling operations conducted for the U.S. Government during the 1950s and 1960s. These mill tailings were used by the community as backfill in construction projects, as a sand mixture in concrete and mortar, and as a sandy loam admixture in

the native soils to enhance the growth of gardens and lawns. Congress passed the Uranium Mill Tailings Radiation Control Act (UMTRCA) in 1978 to authorize DOE to remediate and to permanently dispose of the radioactive mill tailings that were spread throughout the southwestern United States.

DOE-GJPO and its contractor, Rust Geotech, were given the task of planning and performing the remedial action of more than 4,000 residences and businesses, the largest remedial action project of its kind. These properties included heavy and light commercial businesses, some of which contained hazardous waste that had become mixed with the mill tailings (referred to as commingled waste). Rapid resolution of the commingled waste posed a unique problem to project managers because the material was regulated by both UMTRCA and the Resource Conservation and Recovery Act (RCRA). DOE-GJPO found that resolving commingled waste problems required a cooperative effort with the Colorado Department of Public Health and Environment, which is the authorized agency to manage RCRA within Colorado.

URANIUM MILL TAILINGS HISTORICAL PERSPECTIVE

From the early 1940s through the 1960s, approximately one-half of the uranium ore mined in the United States was processed by private companies under contract to the Federal government. The uranium ore was used in national defense research, weapons development, and in the commercial nuclear energy industry.

The uranium mills were shut down as government contracts terminated, leaving large uranium tailings piles on many of the millsites. Uranium mill tailings are a sand-like waste product from the milling process that contain approximately 85 percent of the radioactivity that was present in the unprocessed uranium ore. The long-term health hazards from exposure to radioactive elements in the tailings were not known at the time the millsites were abandoned. As a result, most piles were left unstabilized, exposed to the environment, and subject to other uses.

In Grand Junction, Colorado, the Climax mill was one of these abandoned uranium ore processing sites, located adjacent to the downtown area. To minimize the growing tailings stockpile, the mill operators allowed free public access to the stockpile for private use of the sandy material. The operation of the Climax mill coincided with a large building boom in the Grand Junction area, and more than 4,000 properties in the vicinity of the mill (referred to as vicinity properties) became contaminated with the tailings. The tailings were used extensively in the construction of residences, schools, churches, public buildings, and commercial structures. These uses included, but were not limited to,

- Sand substitute in concrete and mortar mixes
- Bedding sand for utility lines and concrete structures
- Backfill around foundations
- Soil amendment for lawns and gardens
- Fill material to level properties with uneven ground

In the 1960s, medical research identified potential health hazards associated with uranium mill tailings, primarily from the inhalation of radon, a decay product of the radium present in the tailings. In 1978, Congress passed Public Law 95-604 (1) establishing the Uranium Mill Tailings Radiation Control Act (UMTRCA) to authorize the U.S. Department of Energy (DOE) to remediate 24 inactive millsites and the surrounding vicinity properties in various locations throughout the United States, including Grand Junction.

The U.S. Environmental Protection Agency (EPA) subsequently published the final rule in 1983 to establish the standards for the cleanup of inactive uranium processing sites (2). The final rule incorporates the results from evaluations of Grand Junction vicinity properties to determine the hazards associated with radon. EPA estimated that for each 0.01 working level (WL) increase in radon decay-product concentration inside a residence, there is an associated risk of lung cancer for inhabitants of "something like one-half to one in a hundred for an assumed lifetime of residency." Therefore, EPA established the objective of reducing radon to below 0.02 WL in structures, where practical, and not to exceed 0.03 WL. Since 1983, awareness of health hazards from naturally occurring radon has increased, and EPA has suggested similar standards for houses nationwide. EPA also established a radium cleanup standard of 5 picocuries per gram (pCi/g) above background in the top 15 centimeters of soil and 15 pCi/g above background on the subsurface because "soil extensively contaminated at a level of 5 pCi/g of radium can lead to indoor levels of radon decay products of 0.02 WL." Although the EPA standard addressed wastes related to the milling process, the standard did not address hazardous wastes mixed

with uranium mill tailings on vicinity properties.
In 1981, DOE assigned the Grand Junction Projects Office (GJPO) with the responsibility for remediating more than 4,000 properties in the vicinity of Grand Junction. The properties were placed in the following four principal categories to manage the large volume of properties under the project: simple residential (exterior portions of a residential vicinity property), major residential (interior portions of residential properties), simple commercial (small businesses such as gas stations), and complex commercial (large businesses or a group of properties typically worth more than $500,000). To date, 3,974 properties have been remediated, many of which involved commingled waste problems.

DISCOVERY OF HAZARDOUS WASTES

In the initial stages of the program, most of the remediated Grand Junction properties were simple residential properties with no history of hazardous wastes. In 1988, during the remediation of uranium mill tailings, a material subsequently identified as Resource Conservation Recovery Act (RCRA) hazardous waste (specifically, Toxicity Characteristic Leaching Procedure [TCLP] arsenic and lead) was excavated and inadvertently transported from two commercial properties to the State-owned Temporary Mill Tailings Repository located in Grand Junction. The hazardous waste had been improperly disposed on the two commercial properties after the mill tailings had been placed there. The GJPO confirmed that the volume and concentration of TCLP arsenic and lead were not naturally occurring in tailings nor were they related to the uranium milling process.
Because there was no clear guidance on how to manage or remediate commingled waste, GJPO established a commingled waste investigation project to evaluate Grand Junction vicinity properties for the presence of RCRA hazardous waste and TSCA substances. Currently, 20 Grand Junction vicinity properties have been identified as containing levels of hazardous waste which required special management. The most common constituents of concern are TCLP lead from improperly disposed automobile batteries and waste oil tanks, polychlorinated biphenyls (PCBs) from electrical transformers, RCRA-listed and TCLP organics from dry-cleaning operations and automotive paint shops, and miscellaneous organic constituents from a 55-gallon-drum recycling operation.

VICINITY PROPERTY CHARACTERIZATION ACTIVITIES

An evaluation of Grand Junction vicinity properties for the presence of hazardous waste was conducted in a manner similar to Phase 1 real estate assessments for environmental hazards and also closely followed EPA's guidance on preliminary assessments. Typically, this process involved historical research to evaluate current and previous operations that had been conducted on vicinity properties. Site investigations were sometimes conducted to visually evaluate the property and to use direct-reading, hand-held instruments. Site-specific sampling and analysis plans were used to perform soil and groundwater sampling when the investigation identified a need for these data. Hazardous waste problems were not expected on residential properties; therefore, primarily historical research was conducted to confirm that these properties had always been residential in nature.
Most commercial properties, especially those located in or near an industrial area, were fully assessed for commingled waste. These assessments were typically site specific because each property provided unique challenges. After a few of the initial site assessments were completed, GJPO recognized the need to establish criteria for conducting site assessments and for determining the acceptability of data.
GJPO also developed a field call-out process for use during site remediation. When anomalies such as soil discoloration, odors, or debris are discovered, all work is stopped and trained personnel are called to the field to investigate. Typically, the GJPO petrology laboratory is used to conduct x-ray diffraction for evaluation of minerals found during excavations.
Approximately 100 of the 4,000-plus vicinity properties required formal site characterization and documentation to assess RCRA hazardous waste and TSCA substances. Some of the sites identified during these assessments include a drum recycling operation, underground storage tanks (USTs), a machine shop contaminated with cyanide, and landfills. These sites are discussed in more detail below.

DRUM RECYCLING OPERATION

One of the complex commercial vicinity properties assessed under the UMTRA Project was a commercial operation that recycled and refurbished 55-gallon metal drums. At

the beginning of the investigation, the property contained more than 27,000 drums, most of which were empty. During the radiological assessment, workers discovered drums labeled as hazardous waste, uranium concentrate, and hydrofluoride acid. A commingled waste investigation was subsequently conducted, and 63 locations were sampled on the property. The results of this investigation confirmed the presence of organic and inorganic RCRA hazardous waste on the property.

EPA's Emergency Response Branch took control of the vicinity property to remove the identified hazardous waste. During fiscal year 1993, EPA removed roughly 27,000 drums from the property, 125 of which contained improperly stored hazardous waste. After further assessment, EPA demonstrated to the Colorado Department of Public Health and Environment (CDPHE) that, due to insufficient process knowledge, waste materials from this vicinity property should be managed as a RCRA-characteristic waste rather than as a listed waste. In 1994, EPA used solidification and stabilization technology to treat 12,000 tons of soil and debris on the property that was contaminated with elevated concentrations of TCLP lead. However, the treated material still contained uranium mill tailings which required proper disposal under the UMTRA Project.

GJPO obtained approval from CDPHE and the U.S. Nuclear Regulatory Commission (NRC) to dispose treated RRM at the UMTRA Cheney Disposal Site. An amendment to Mesa County's conditional-use permit for the Cheney site was also required to allow for disposal of treated hazardous waste. The approval process took approximately one year from the initial meeting where the feasibility of cleaning the property was discussed by GJPO, CDPHE, and EPA. As a result of the cooperation between agencies and stakeholders, the DOE avoided the $3.1 million cost to treat the hazardous waste.

This property also had several unusual characteristics. During the removal of the 55-gallon drums, EPA discovered that ethyl mercaptan had leaked onto the ground. This material, used to create the odor in natural gas, remained even after the soil was treated. Although the odor could not be detected by instruments and no Occupational Safety and Health Administration permissible exposure limits were exceeded, the odor was strong enough that complaints were occasionally received from neighboring property owners.

Spots of yellow pigment were also discovered during excavation of uranium mill tailings. These paint chips were tested and identified as containing elevated concentrations of TCLP lead. EPA returned to the site, evaluated the material, and has made the preliminary determination that the overall waste stream does not exceed the regulatory level for TCLP lead.

UNDERGROUND STORAGE TANKS

To date, the GJPO has removed 88 USTs that were found embedded in uranium mill tailings on vicinity properties. Most of these USTs had been leaking, resulting in uranium mill tailings mixed with fuel products and waste oils that required proper management.

Working with the CDPHE, the GJPO developed a technical position that any petroleum products identified leaking from an UST (except for waste oil tanks) could be windrowed until the uranium mill tailings passed the Penske-Marten test for ignitability. GJPO also performs a test for flammability to ensure materials are properly managed.

Underground waste oil tanks proved to be a more difficult management problem. At two sites, waste oil tanks contained concentrations of TCLP lead in excess of the regulatory threshold. Although disposing of the contents of these tanks was the responsibility of the owner, not all of the waste oil sludge was removed, leaving small quantities that exceeded a de minimis volume. At both sites, uranium mill tailings were inadvertently mixed with small quantities of the sludge while the waste oil tanks were being removed. At least one of the sites will require treatment and stabilization of the TCLP lead before the commingled material can be disposed.

CYANIDE MIXED WITH RRM AT A MACHINE SHOP

One of the vicinity properties contains a machine shop that operated for more than 40 years using a high-temperature (899 to 954C) cyanide-type carburizing bath. Principal shop operations consisted of manufacturing, repairing and distributing drilling and other types of heavy equipment, and case-hardening of steel parts. A significant portion of the machine shop was contaminated with uranium mill tailings, including the area where the case-hardening process was performed.

The case-hardening process used a liquid-cyaniding (carbonitriding) bath that became

commingled with indigenous soils and uranium mill tailings in the area surrounding the bath. GJPO performed a site assessment, collected soil samples from around the carbonitriding bath, and evaluated the case-hardening process. The information collected from the site assessment was compiled into a Technical Discussion Document and presented to CDPHE. The Technical Discussion Document detailed the materials and activities associated with the case-hardening process, and assessed regulatory issues associated with the potential cyanide-based waste streams by examining the classification of the wastes with regard to RCRA. CDPHE reviewed the Technical Discussion Document and recommended performing additional site characterization activities. GJPO subsequently completed additional soil sampling and then conducted a joint facility inspection with CDPHE to evaluate the case-hardening process on a step-by-step basis.

After reviewing the case-hardening process, CDPHE concurred with the GJPO that listed hazardous waste was not an issue at the machine shop, greatly reducing the potential treatment and disposal costs for the property. CDPHE also accepted GJPO's position that characteristic reactive cyanide was below the EPA's regulatory threshold of 250 parts per million and, therefore, did not require management as a hazardous waste. Because of these successful negotiations, GJPO remediated this property at a savings to the Government of at least $50,000.

LANDFILLS

Two large vicinity properties scheduled for remediation were at one time operated as unregulated municipal landfills. Both properties are located within the 100-year floodplain of the Colorado River and both have some form of soil cover.

The GJPO is currently planning to remediate only the top 2 feet of one of the landfills to reduce the risk of the public receiving an elevated level of gamma radiation. In accordance with the requirements of EPA standards, an application of supplemental standards for this partial remediation has been submitted to the NRC and CDPHE. If both agencies concur with this application, the remaining radiologically-contaminated landfill materials will remain in place. To justify the use of supplemental standards, the GJPO has demonstrated there is a high cost of remediation, low health risk to the public, and low likelihood of buildings being erected on the site. The likelihood of building on this site is low because of its proximity to a sewage treatment plant and because it is covered partially with wetlands.

The second landfill, located adjacent to downtown Grand Junction, is scheduled for full remediation rather than the application of supplemental standards because of the anticipated change in land use by the owner. The property was also operated as an auto salvage yard, requiring the GJPO to negotiate the removal of almost 5,000 salvage vehicles to facilitate radiological assessment. A commingled waste investigation was also performed to evaluate surface soils and groundwater directly beneath the landfill. Although these studies revealed trace concentrations of various constituents, no hazardous wastes have been identified.

GJPO is planning to remediate the radiologically-contaminated landfill materials only after establishing an agreement with CDPHE to allow treatment of RCRA-characteristic inorganic wastes, if discovered. This agreement will be used in lieu of a RCRA Part B permit. CDPHE has agreed with the DOE approach to excavating the landfill as long as the necessary precautions are taken to identify hazardous waste if it is encountered. The State has also agreed that management is not necessary for undisturbed landfill materials that are not contaminated. Only a cover, comparable to the existing one, needs to be placed over these remaining landfill materials.

GJPO found that many properties adjacent to the Colorado River, while not landfills, contained large areas of subsurface debris, junk, and household trash. In locations where broken battery cases were found, elevated concentrations of TCLP lead have been identified. The lead problem will be treated with solidification and stabilization technologies if the agreement for treating hazardous wastes with CDPHE is reached.

LESSONS LEARNED AND CONCLUSIONS

In the course of remediating the Grand Junction vicinity properties, GJPO identified a number of steps that should be established in the planning of future projects to address commingled waste issues. Specifically,

Initial planning should allow for the possibility of hazardous waste being commingled with RRM. Because the uranium mill tailings were uncontrolled, it should

be assumed that other substances could have been introduced over the years.

Project participants and regulators should establish and agree upon a hazardous waste management plan that identifies how various types of hazardous wastes would be managed (e.g., RCRA-listed waste as opposed to TSCA substances).

Applicable regulatory guidance for site characterization and hazardous waste determination should be agreed upon in the up-front planning process to determine what constitutes an adequate site assessment and representative samples, and the manner in which a constituent of concern should be evaluated (e.g., how to determine constituent concentrations and appropriate confidence intervals).

Potential hazardous waste treatments should be addressed. Certain types of hazardous waste can be remediated with established and simple treatment technologies such as solidification and stabilization.

Planning for disposal of successfully treated commingled waste at an UMTRA disposal cell should be conducted early in the project to expedite the timely completion of site remediation.

Field criteria should be established during the initial planning stages to facilitate the evaluation of objects and wastes that are uncovered in the field during remediation.

Technical contacts, written procedures, and lines of authority should be clearly delineated.

In conclusion, hazardous waste issues have been successfully resolved by the GJPO through working with EPA and CDPHE. Technical Discussion Documents, outlining process knowledge and analytical results, have been instrumental in communicating with regulators to obtain their concurrence.

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DECOMMISSIONING OF THE CHORNOBYL NPP: REGULATORY AND MANAGEMENT PROBLEMS

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ABSTRACT

According to the former USSR regulations a NPP decommissioning plan should be submitted to the regulatory body in five years before an expiration of the design-based service life of NPP unit. But the Chornobyl accident of April 26 1986 caused changes in many routine procedures and approaches, particularly concerning the Chornobyl NPP decommissioning. There is a number of particular features which needs a special consideration, as for the Chornobyl NPP decommissioning, including the 30-km exclusion zone around the Chornobyl NPP, the destroyed Unit 4 and the SHELTER object (well-known as "Sarcophagus"), a relatively high level of dose exposure rate at the Chornobyl NPP site, etc. Along with both domestic and external socioeconomic and political factors it complicates a decision-making process very much and creates many problems both for regulators and operators. The ultimate decommissioning strategy for the Chornobyl NPP is not clear yet, nevertheless certain preliminary conclusions and suggestions can be made.

LIFE HISTORY OF THE CHORNOBYL NPP

The construction of Chornobyl NPP was started in March 1970. The NPP was supposed to consist of six RBMK boiling water pressure tube, graphite moderated reactor units of 1000 MWt rated power each. Units 1-4 were commissioned in 1977, 1978, 1981 and 1983, respectively. Commissioning of Units 5 and 6 was planned for 1987-1988, but after the Chornobyl accident the plan was abandoned. After beyond the design basis accident of April 26 1986 the Unit 4 has been destroyed and received a shelter building which is not consider enough by some specialists and options of overpacking by a new shelter, fully or partly dismantling etc. are under consideration. After a major fire in the turbine hall of October 11 1991 Unit 2 is out of service and will probably never be restarted. The NPP history is summarized in Table I below.

After the accident of April 26 1986 operation of all the Units was halted. During the first post-accident months accident recovery work was done along with implementation of measures intended for the NPP safety upgrading including revision of operational manuals, personnel reexamination, technological modifications. As a result Unit 1 and 2 were put into operation by November 1986, Unit 3 - by December 1987.

In spite of all the measures implemented a proper safety upgrading of the Chornobyl NPP was considered unattainable and by the Ordinance of February 17 1990 "About environmental conditions and radical improvement measures in the Ukrainian Republic" the Verkhovna Rada (Parliament) of Ukrainian Republic made the decision about withdrawal from service three units of Chornobyl NPP by 1995. The major fire at the fourth turbogenerator of Unit 2 of October 11 1991 resulted in revision of the shutdown deadline. According to the ordinance of Ukrainian Parliament of October 29 1991 the decommissioning of Unit 2 was to be started immediately and operation of Unit 1 and 3 was to be halted by the end of 1993. According to the above-mentioned Ordinances the Concept of Chornobyl NPP Units Decommissioning was developed. The Concept considered five options for unit decommissioning. The Commission on decommissioning of Chornobyl NPP Units chose the option which consisted in safe enclosure of reactor and highly contaminated equipment for 25-30 years while dismantling the other equipment with conservation of constructions and operation of life support systems. It provides for a further dismantling of all the equipment and structures till so-called "green lawn".

But the decision was not approved by the national regulatory body on account of a poor substantiation. It was noted that before selecting a decommissioning option we should have more information about a physical and radiological status of each unit, a radionuclide inventory etc. and a decommissioning option can be selected only on the basis of a complex engineering review of status of each unit taking into account all possible effects of the destroyed Unit 4 and related structures.

Nevertheless regardless of the selected decommissioning option certain preparatory actions were to be taken including evaluation of radionuclide inventory, development of supplementary heating and electricity systems etc. But nothing was done.

On October 21 1993 in the light of the crucial situation in the national economy and power generation the Ukrainian Parliament canceled its own decision on shutting down the Chornobyl NPP. Thus the shutdown date for the Chornobyl NPP became no more a political but technological issue. But it does not mean that a deadline for all Units was postponed very much.

The Unit 1 and 2 of Chornobyl NPP had been designed before nuclear safety regulations were put in force in the former USSR. These Units belong to so-called first generation units and do not have full-scope safety systems, for example, a localization system (like other first generation RBMK units situated in Russia, the Leningrad NPP Unit 1 and 2, the Kursk NPP Unit 1 and 2). Upgrading these units to acceptable safety levels is a very complicated and expensive task and appears to be unfounded and unrealistic. Thus a decommissioning seems to be a near future task, at least for these units.

REGULATORY FRAMEWORK FOR DECOMMISSIONING IN UKRAINE

During the lengthy period of nuclear energy development in the former USSR a NPP decommissioning was left out of proper consideration. It affected both a NPP designing and regulations. Decommissioning issues began to be considered in the eighties alone.

Basic requirements for decommissioning were set forth in item 5.6 of the General Provisions on Ensuring Safety at Nuclear Plants (PB-88) as follows:

"5.6. Decommissioning of Nuclear Plants

5.6.1. Decommissioning of Nuclear Plant (Unit) is to be considered during designing, operation, maintenance and repair of Nuclear Plant (NPP).

5.6.2. At least 5 years before the design-based unit lifetime expiration the Operator has to provide for development of a decommissioning plan and its approval by State Surveillance and Control Authorities.

5.6.3. Decommissioning should be preceded by a complex review of the unit by a Commission appointed by the Operator. Decision on the unit decommissioning is to be made on the basis of complex review results.

5.6.4. By order of State Surveillance and Control Authorities a NPP Unit can be decommissioned before the design-based lifetime expiration, if the status of safety systems or the unit status as a whole does not ensure NPP safety."

More detailed requirements were set forth in the Part 15 of the Sanitary Rules for Designing and Operation of Nuclear Plants (CP AC-88) and in the Part 11 of the Radioprotection Rules for Operation of Nuclear Plants (PP AC-89). The requirements consider issues to be taken into account in selecting a decommissioning plan, organizational issues of radioprotection during decommissioning, decommissioning stages, safety considerations for dismantling and waste managing etc.

The main shortages of the system of regulations in force consist in:

- an uncertainty of a licensing procedure for decommissioning;
- a full absence of provision of funds for decommissioning;
- an absence of established quantitative criteria for recycling within the nuclear field or a free release of decommissioning materials.

It should be mentioned that the Chornobyl NPP case could not be considered in a routine way in any case due to specific on-site conditions resulted from beyond the design basis accident of April 1986. In addition insufficiency of regulatory framework for decommissioning is typical for many countries and can be overcome not until a broad decommissioning experience gained (1).

During the last two years two new regulations on decommissioning are being worked out as a part of a national programme of radioactive waste regulatory framework development, namely the General Safety Provisions for Nuclear Power Plant Decommissioning and the Basic Provisions on SHELTER Safety Assurance. The continued development of the international regulatory framework including the RADWASS Safety Standard on Decommissioning of Nuclear Facilities and the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety Radiation Sources enables development specific regulations for decommissioning which may render regulatory decisions easier, more expeditions and foreseeable on the basis of the other countries' experience, but it seems to take a lot of time.

In the Chornobyl NPP case all kinds of contingencies might arise, changing the previous regulatory strategy and requiring a new approach. So the regulatory process will develop simultaneously with the decommissioning process but certain regulatory decisions should be made in the nearest future.

The first one concerns establishment of decommissioning stages needing separate licenses. For the Chornobyl NPP unrestricted release of the site is hardly attainable in a near future regardless of a decommissioning strategy. So the final goal of the first phase of decommissioning to be licensed should be determined.

The second one concerns financial provisions. In the former USSR financial issues were out of the scope of permitting process and provisions of funds for decommissioning were not established. As a result there is no accumulated reserves for the Chornobyl NPP decommissioning yet. It can affect not only timing but the safety of decommissioning operations and has to be considered in a licensing process.

The third one concerns establishment of "clearance" level for both the site and decommissioning materials which can affect decommissioning costs as well. It appears to be the most solvable problem taking into account an availability of a set of international regulations and recommendations (2-4).

The one more problem which needs a special regulatory decision concerns a long-term storage and final disposal of radioactive waste originated from the decommissioning and the lifetime operational waste stored on-site. The enormous volumes of such waste suppose on-site or nearby siting of waste treatment and disposal facilities to be a preferable option, but under an emerging regulatory regime in waste management field in Ukraine the related decision-making process needs a special concern.

TECHNOLOGICAL BASIS FOR THE CHORNOBYL NPP DECOMMISSIONING

By now we have no experience in decommissioning of NPP Units with RBMK type reactors, so do of NPP Units after a severe accident like the Chornobyl one. Thus we can consider the Chornobyl NPP case as a pilot project and many technological problems should be solved on the case-by-case basis.

Main technological problems for NPP decommissioning are usually as follows:

- reactor defuelling, fuel storage on site, spent fuel disposal or reprocessing;
- dismantling and decontamination of equipment;
- liquid and solid waste treatment, waste storage and disposal;
- dismantling or stabilization of constructions, on-site clean-up activities.

As for a spent fuel problem, it should be mentioned that Ukraine has not established a national strategy for spent fuel management yet. In the former USSR the RBMK spent fuel used to be stored in the spent fuel storage ponds on-site because of an absence

of a reprocessing facility although the fuel reprocessing was presupposed. Taking into account the socioeconomic and political conditions in Ukraine the reprocessing option is hardly feasible in a near future. Under the emerging waste management system a deep geological repository will be available in Ukraine in fifty years at best. For such a long term the existing fuel ponds seem not to be a proper option taking into consideration problems of degradation, corrosion and weather-dependence. Moreover the available capacity is not enough for the expected amount of fuel elements. Thus an additional spent fuel storage capacity should be provided beforehand, a dry-type storage being preferable.

As already mentioned above a further decommissioning was not adequately considered in the RBMK designing. So measures to facilitate equipment and structures dismantlement like optimum geometrical layout were not taken. But the design of reactors allows relatively easy dismantling, for example the pressurized tubes and the graphite moderator blocks could be removed quite well with remote-control equipment.

The other feature which can complicate the choice and implementation of dismantling technologies arises from the accident of April 26 1986 and accident recovery work. In some cases decontamination of structures or equipment to acceptable levels was not attainable due to non-favorable geometrical layout, low effectiveness of decontamination techniques or other reasons. Then in order to minimize a occupational exposure and avoid a spread of contamination the contaminated surfaces were covered by oil, plasticate etc. Thus in addition to relatively high contamination levels, large contaminated areas and relatively high percentage of actinides among contaminating radionuclides there is a covered in-depth contamination. All these features should be considered and examined during a preliminary radiological review before selecting dismantlement and waste management strategy.

The same feature is characteristic of the Chornobyl NPP site. For example, in the nearby area of the Unit 3 and 4 where surface contamination level was extremely high all debris including fuel particles and graphite were not removed but covered by a ground and concrete layers without taking account of possible in-depth contamination of ground and groundwater. And in selecting a decommissioning option all these distinctive features should be carefully examined and considered, taking into account the destroyed Unit 4 and further recovery and stabilization work at it as well.

The concept laid down at the design stage for the management of wastes at RBMK type reactors was to store the wastes on site and to postpone decisions on conditioning and disposal until the decommissioning stage. By this means wastes from operation and dismantling suppose to be handled together. Waste collection and storage systems were developed to accommodate ten years arisings of treated operational wastes with possible extension of storage capacities. The only exceptions were for the very low level solid wastes, where on-site disposal was proposed.

In the Chornobyl NPP case no modifications of design-based waste management flow-sheet were done and the lifetime unprocessed liquid and solid wastes are stored on-site. As of the end of 1993 25940 cub.m liquid wastes were stored with the total salt content of 2986 tones and the total activity of 2474 Ci. The data for solid wastes are shown in Table II.

It should be noted that after the Chornobyl accident all dry low- and intermediate-level wastes, including routine operational ones, were transported to the BURYAKOVKA disposal site operated by the Research and Industrial Association PRYPIAT in 10 km from the Chornobyl NPP. But after issuing by the Ukrainian regulatory body the license to the RIA PRYPIAT to operate the BURYAKOVKA shallow ground disposal facility the practice was halted in accordance with license conditions outlawing disposal of wastes other than considered in the facility design. So the solid waste volume will be increased rapidly.

In addition we have to take into account waste flows from the destroyed Unit 4. Contaminated water accumulated in the destroyed Unit 4 premises creates a problem already now. For example, volume of water is estimated as 2500 cub.m with activity of 3.10-6 - 10-5 Ci for Cs-134, 10-7 Ci for Sr-90, 10-8 Ci for Pu-239 and U-235 concentration estimated about 5-20 mkg/l. In so doing a water pumping-out is complicated by both high specific activity and a presence of fissionable nuclides, which needs a special pre-treatment considering nuclear safety issues. An anticipated recovery and stabilization work at the Unit 4 including a possible

shelter replacement is expected to generate a wide range of wastes and effluents which should be considered in development waste treatment and storage capacities on-site as well.
Thus in decommissioning the Chornobyl NPP liquid and solid waste treatment plants and waste storage and disposal capacities should be provided to manage both a lifetime operational waste and decommissioning waste considering waste generated at the destroyed Unit 4 as well, only for the operational waste volumes being predictable for certain. In addition due to specific contamination features conventional waste treatment and conditioning technologies should be revised and adapted.

CONCLUSION

Due to consequences of beyond the design-basis accident and influence of the destroyed Unit 4 the Chornobyl NPP decommissioning can not be accomplished by routine procedures. Key decisions on the timing and extent of decommissioning activities should be based on a thorough knowledge of a physical and radiological status of each unit, radionuclide inventory, site characteristics etc, which needs a preliminary complex radiological and engineering examination and a continuous review of the NPP Units and site.
Case-by-case decisions and a parallel development of both regulatory and management environment will enable the best practical environmental options at a reasonable timescale.

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REMOVAL OF LOW LEVEL ACTIVITY FROM THE WASTE STORAGE TANKS OF TR-2 RESEARCH REACTOR

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ABSTRACT

TR-2 is a 5 MWt swimming pool type research reactor located at ekmece Nuclear Research and Training Center (NAEM). It has been operating since 1982. The aqueous wastes collected from this reactor used to be treated by decay storage and then discharged according to the safety regulations established by Turkish Atomic Energy Authority.
The periodical checks of storage tank liquids have usually given a radioactivity level of 50-100 Bq/L for 60Co which is the most important isotope to be considered. An unusual high contamination level has been detected in the storage tanks at March 1993. The specific activities observed were higher than discharge limits. Since 60Co has a very long life time, another method of processing other than decay was needed to reduce the activity levels.
Chemical precipitation was applied in the original storage tanks. Satisfactory decontamination factors were achieved and 60Co activity decreased to about 100 Bq/L. Cleared liquid was discharged and residual sludge was cemented.

INTRODUCTION

The aqueous wastes generated by the operation of a low power research reactor are discharged after being usually treated by decay storage or ion exchange.
TR-2 has two liquid waste storage tanks located 2 m below the ground level. They are cylindrical with a radius of 1.4 m and height of 5.7 m, and each of them has a 35 m3 storage capacity. Tanks are used sequentially and liquid wastes from different parts of reactor systems are collected and retained in these two tanks as long as possible in order to discharge after the activity being decreased below certain limits. Analyses of liquid samples taken from storage tanks have been carried out periodically. Almost same levels of activities for the contained isotopes were

observed. For example, maximum values of 60Co activities were in between 50 and 100 Bq/L (1.4x10-6 - 2.7x10-6 Ci/m3).
Other frequently faced radionuclides and their typical activity concentrations were; 124Sb=1000 Bq/L, 58Co=200 Bq/L, 137Cs=25 Bq/L. The target radionuclide was 60Co in this procedure. Therefore the other 4-5 nuclides were not considered due to their small contributions or short half life compared with the 60Co. Two of them, however, are listed for only south tank in Table I.
Higher activities are not expected normally and these very low level specific activities of 60Co are managed without any difficulty.
But, an analyses in March 1993 showed that there was an unusual radioactivity increase in the tank liquids. 60Co activity was about 18.000 Bq/L and total beta activity was up to 20.000 Bq/L. There were not significant activity increase for the other nuclides.
Possible reasons of this unexpected contamination are still being searched. Control rods of the reactor might be responsible since they include a nickel alloy. It is known that nickel always contain some cobalt impurity.
Tanks had to be emptied for the normal operations of reactor, but activity level of the liquid required a treatment before discharge. Transferring of liquid to the waste treatment plant was not possible because of the lack of equipment and pipeline connection to waste treatment plant.
A convenient treatment had to be done in the tanks, themselves. But since these tanks had been considered as only collection and retention tanks, they also did not have necessary equipments for any treatment purpose.
Chemical precipitation seemed to be the easiest process in those conditions but there were two major difficulties; one was the mixing of liquid, second was the discharge of sludge from the tank bottom.
PROCEDURE
Radioactivity Analyses
Gamma spectrometric analyses were done with 100 ml of liquid samples by Canberra Series 85 spectrometer with a HP germanium detector. 1 ml of liquid was dried on a metallic disk for total beta activity analyses which were done by Tracerlab, Low Background Counting System.
Chemical Procedure
A number of jar-tests were done in the laboratory to find out the best removal conditions of 60Co from the tank liquids. The purpose of the chemical procedure was to transform the 60Co to a sparingly soluble structure from its dissolved form in the liquid.
In alkaline medium, cobalt precipitates as its hydroxide.

$$Co^{++} + OH^- \longrightarrow Co(OH)_2 \quad (1)$$

But trace concentration of cobalt corresponding to the observed activities (i.e. 20.000 Bq/L = 4.7x10-10 g/L) does not allow to form cobalt hydroxide precipitation, because of the threshold character of solubility product value. Therefore nonradioactive cobalt was added to help the formation of Co(OH)2. Iron hydroxide precipitation was also experimented together with cobalt in order to improve the precipitation.

$$Fe^{++} + 2OH^- \longrightarrow Fe(OH)_2 \quad (2)$$

$$Fe(OH)_2 + OH^- \longrightarrow Fe(OH)_3 \quad (3)$$

Although the processes given by equations (1), (2) and (3) are simple from the chemical point of view, the removal of 60Co is not easy to explain with stoichiometric rules. As we know that, some of the free Co++ ions which may still exist in the medium are adsorbed by iron hydroxide flocks. They precipitate together. Mechanism of this co-precipitation is not clear (1), but the activity removal mainly depends on the iron hydroxide flocks which carry down the cobalt hydroxide (2).
The following steps of chemical procedure were chosen according to the results of laboratory experiments:

keep PH between 10-10.5 with NaOH in the tanks,
add 0.5 ppm CoSO4 as nonradioactive carrier,
add 0.25 g/L FeSO4.5H2O,
a continuous mixing should be carried out,
a waiting period of 10 minutes between each step.

Operational Procedure

A pipe with a jet mixer on its tip was located inside the tank and compressed air was sent through it. A homogeneous, satisfactory mixing was obtained. Degree of mixing could be arranged by air pressure. A figurative illustration of operational phases are shown in Fig.1.

1.Mixing of tank liquid with compressed air
2.Suction of clear liquid
3.Suction of sludge to waste drums

NaOH, CoSO4 and FeSO4 were used from their prepared dilute solutions. After a homogeneous mixture was obtained tanks were left to be settled for two days. After that, since the analysis of clear liquid showed a satisfactory decontamination, this part of tank contents was sucked out.

Amount of sludge and its level of height in the tanks were estimated according to the laboratory experiments (3). Suction of clear liquid was stopped at 10 cm above that level. The sludge below that level was transferred to 200 L drums. 2/3 of each drum was filled and transported to the waste treatment plant.

RESULTS AND DISCUSSION

Maximum DF values obtained in the laboratory experiments were around 120 for 60Co which were less than the DFs obtained in real tank precipitations. This may be caused by the different mixing types. But most probably the mud collected during the previous operations at the bottom of the tanks, should help the activity removal by adsorbing the metal ions and taking down them during the settling period.

All specific activities and decontamination factors are shown in Table I. Residual 60Co activities are 118 Bq/L in the south tank and 78 Bq/L in the north tank. Attained DF values for 60Co in both tanks are nearly equal between 150 and 160. That DF values are quiet satisfactory compared with the values given in Ref. 4-6, which give all DFs greater than 100.

Removal of total beta activity was less than the 60Co removal. This was expected because the other beta emitting nuclides were not the subject of the precipitation. However, two of them, 124Sb and 58Co were removed to below detection limit which were less then 10 Bq/L in our case.

63.2 m3 cleared liquid containing a total 60Co activity of 6.18x106 Bq (1.67x10-4 Ci) was discharged. This amount is about 62% of monthly allowable discharge limit of safety regulations in Turkey.

About 1.4 m3 of sludge was sucked out from the tanks and cemented in 200 lt drums. 100 kg of cement was used for every drum and 12 cemented waste drums were produced. Sludge samples were taken from each drum and analyzed. Total beta activities of each drum were found between 9x107 and 1.3x108 Bq/drum while total 60Co varied from 6x107 to 1x108 Bq/drum. Surface dose rates of the products were between 5 - 15 mR/hour.

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WASTE MINIMIZATION DEPLETED URANIUM WASTE MINIMIZATION EFFORTS WITHIN THE ARMY
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ABSTRACT
The Army currently utilizes depleted uranium (DU) for shielding on Abrams tanks and as a kinetic energy penetrator for a family of tank cartridges due to the metals physical properties and performance characteristics. These tank cartridges were made famous during Desert Storm and are more widely known as the "silver bullet" which were extremely effective against Iraqi armored vehicles. The use of depleted uranium by the Department of Defense has received the highest level of Congressional interest due to unsubstantiated claims that DU is responsible for Desert Storm Syndrome.
Throughout the Life Cycle of these weapon systems large quantities of Low Level Radioactive Waste (LLRW) are generated during the testing, manufacturing and demilitarization phases. With the closure of burial sites for LLRW and sky rocketing costs of disposal the Army has instituted a number of pollution prevention initiatives to minimize the generation of waste throughout the "cradle to grave" (life cycle) of these weapon systems.
The Army has successfully demonstrated and instituted three separate depleted uranium waste minimization efforts by employing both decontamination and recycle technologies. The first effort involves the recycle of depleted uranium penetrators which have been fired at various test ranges and recovered. The recovered metal was then cleaned, remelted and recycled into commercial products such as shielding vessels and counterweights.
The second effort involves recycle of demilitarized DU penetrators from obsolete cartridges and armor scrap into state of the art weapon systems. Through the recycle and reuse of Army DU resources there is no longer a need to import more depleted uranium from the Department of Energy into the Army inventory. Lastly, the Army has been successful in developing decontamination and recycle capabilities of armor plate which have been contaminated with DU at the test ranges.
The Army is closing the loop on the use of depleted uranium by minimizing waste generation to the lowest levels that can reasonably be achieved through recycle and decontamination technologies.
INTRODUCTION.
The Army has been using Depleted Uranium (DU) metal in Army weapon systems for over the past 25 years as either a penetrator in tank ammunition or as shielding in tanks. Tank ammunition containing DU was made famous during Desert Storm and is more widely known as the "silver bullet" which were extremely effective against Iraqi armored vehicles. The use of depleted uranium by the Department of Defense (DOD) has received the highest level of Congressional interest due to unsubstantiated claims that DU could be responsible for Desert Storm Syndrome.
It has been recognized that there are a large number of Low Level Radioactive (LLRW) waste streams that result from manufacturing, testing and demilitarization of DU tank ammunition. In order to maintain compliance with the Pollution Prevention Act, National Environmental Policy Act and adherence with DoD and Department of Army policy and regulations it became necessary to identify avenues to minimize these waste streams. Since numerous unsuccessful attempts were made to replace DU with tungsten alloys, focus was placed on decontamination and recycle as the primary means to minimize these waste streams. Depleted Uranium is the material of choice for Kinetic Energy penetrators (Danesi, 1990).
Three major waste streams were identified as being candidates for waste minimization: 1) recycle of depleted uranium penetrators recovered from test ranges into commercial products; 2) recycle of demilitarized DU penetrators from obsolete cartridges and DU armor scrap into state of the art weapon systems and 3) decontamination of steel rolled homogeneous armor (RHA) plate. It has been demonstrated and validated that all three waste streams have been successfully minimized by employing recycle and decontamination technologies.
RECOVERY AND REUSE OF DEPLETED URANIUM PENETRATORS TO REDUCE RADIOACTIVE WASTE

During the 1960's, the Army began to develop armor piercing munitions containing depleted uranium (DU). The DU munitions have ranged in sizes from .50 caliber to 175mm. The munitions are test fired for accuracy against plywood or cloth targets at ranges of 1000 to 4000 meters at Aberdeen Proving Ground (APG), Maryland, Jefferson Proving Ground (JPG),Indiana and Yuma Proving Ground (YPG), Arizona. As the penetrators pass through the target and impact the ground they come to rest on or beneath the surface of the ground. The penetrators may remain intact with minimal airborne particles produced, or it may fragment into pieces from striking trees or hard and rocky soil (Stoetzel, Waite, and Gilcrest 1983).

The Low Level Radioactive Waste Policy Act of 1980 and the Low Level Radioactive Waste Policy Amendment Act of 1985 require future regional compacts to manage waste disposal and allow the existing burial sites to close. Each of the test centers is located in a different compact; APG is in the Appalachian Compact, JPG is in the Midwest Compact and YPG is in the Southwest Compact. None of these com-pacts have operational burial site and as a result are required to store their waste until their compacts are operational.

U.S. Army Test and Evaluation Command (TECOM) and the test centers considered the recovered DU penetrators a marketable commodity which could be recycled into new penetrators at less cost than obtaining DU from the Department of Energy (DOE). Arrangements were made with Aerojet Ordnance of Tennessee (AOT) to accept penetrators from YPG and JPG in 1987 and 1988 at no cost to the Army. YPG had packaged the penetrators in contaminated soil for shipment. Aerojet found the cost of disposal of the soil and the cost of recycling was more expensive than the value of the DU obtained and wanted to charge the Army for processing any future shipments. In addition, the current technical data packages for large caliber only permitted in process recycle. A costly qualification and validation effort would be required prior to accepting this scrap material as feed stock for new production penetrators.

Subsequently the ammunition developer, U.S. Army Armament, Research, Development and Engineering Center at Picatinny Arsenal, New Jersey, arranged with Nuclear Metals, Inc. (NMI) in Concord, Massachusetts, to evaluate recycling the penetrators for commercial products on a trial basis at no cost to the Army. If the costs of processing the penetrators exceeded the salvage value of the DU, then the Army would be charged for any future shipments.

NMI began to accept penetrators from the test centers in 1989 and continued until 1993. The contractor later agreed to continue to accept the penetrators from the test centers, but to charge for the processing.

DEMILITARIZATION & RECYCLE OF DEPLETED URANIUM FOR CURRENT PRODUCTION TANK CARTRIDGES

Current stockpiles of antiquated kinetic energy tank ammunition have to be demilitarized, thereby generating large quantities of DU penetrators that need to be disposed. Although demilitarization procedures exist for tank ammunition, actual studies were never conducted to validate their adequacy. Improper procedures can cause serious injury, but can easily be avoided (Bishop, 1989).

The U.S. Army Production Base Modernization Activity (PBMA) entered into an agreement with NMI, AOT and APG and to determine the technical and economic feasibility of demilitarizing old ammunition. Under these small scale (45 cartridges each contractor) studies, the following was addressed; disassembly methods, contamination levels, types and quantities of waste generated, methods for decontamination and lastly, identify any potential reuse and sale of recovered components. It was determined that the penetrators were of suitable quality to be further evaluated for potential recycle into new penetrators (Laporte, 1994).

The Army currently obtains depleted uranium tetra fluoride from the DOE and through a series of processes converts it into DU metal. However, large quantities of contaminated hydrogen fluoride and magnesium fluoride are generated and much energy is expended. If the demilitarized penetrators could be recycled these waste streams would be eliminated and the Army will no longer need to bring DU into its inventory.

A follow-up demilitarization and recycle effort was conducted with both NMI and AOT on 1400 cartridges to further establish procedures and to produce recycled penetrators in accordance with current specifications which require chemical analysis, metallography and final ballistic target penetration. Three melt cast lots were produced utilizing 100% recycled penetrators. Chemical and metallographic tests were performed prior to final machining of the penetrators.

One of the contractors used input material which contained, on average 100 parts per million (ppm) of zirconium. For the three melt lots tested for mechanical properties, there was an apparent relationship between high zirconium in the melt and low compressive yield (Laporte, 1994). Due to the high levels of zirconium found in the demil stockpile, penetrators produced by NMI cannot at this time be considered for future production until additional studies can be completed. It is speculated that at the time of original alloying of the NMI M774 penetrators that zirconium coatings were used in the graphite liners. The other contractors feed material did not contain high levels of zirconium and as a result final machining resulted in delivery of 35 specification quality penetrators (Zeman and Wright, 1994) which were later ballistically fired. The program switched gears in May 1994 when it was determined that the demilitarization production line would not be in operation to support 1995 ammunition production. Depleted Uranium Armor scrap was then identified as an alternate source of material. The material chemistry of the scrap armor is well documented, therefore this was considered a low risk effort. This material had to go through similar processing and testing requirements as the M774 penetrators. Additional processing steps that were required to recycle armor that were not required for the M774 penetrators included pickling, shearing, special furnace loading procedures and alloying with titanium.

The cast ingots from three melt lots were processed in an identical fashion to standard manufacturing practices. All melt lots met required chemistry and mechanical properties. When compared to standard melt cast lots, no processing nor acceptance yield differences were noted for the armor scrap remelt lots (Laporte, 1994). Minor additional process operations were identified by AOT such as, surface corrosion on the scrap armor which may consistently require an extra pickling operation, and careful packing of the furnace with irregular shaped pieces to enhance heat yields (Zeman and Wright, 1994) From the three lots processed through heat treatment and High Energy Rate Forming, 35 blanks were machined into penetrators by each contractor and shipped to the load assemble and pack facilities for final assembly into M829A2 tank cartridges.

Forty-five cartridges were delivered to APG for final ballistic testing; 15 rounds contained AOT penetrators recycled from old penetrators, 15 rounds contained AOT penetrators recycled from armor scrap and 15 rounds contained NMI penetrators recycled from armor scrap. The high zirconium penetrators produced by NMI were not ballistically tested. All 45 rounds were fired against modern armor targets in December 1994 and January 1995. Preliminary results indicate that all have met or exceeded the current target penetration requirements.

At the time of this writing the final determination has not been officially approved as to whether or not these materials are qualified for recycle. Since all the qualification requirements have been met, approval is expected before 1995 production of M829A2 tactical tank cartridges begin.

It has been estimated that cost savings for the production of the new M829A2 penetrator will be reduced by approximately $49 utilizing recycled penetrators and $46 by utilizing armor scrap (Los, 1993). The three dollar difference is attributed to the extra processing steps (pickling, special loading and alloying).

RECYCLING STEEL ARMOR PLATE CONTAMINATED WITH DEPLETED URANIUM

The U.S. Army presently conducts research, development and testing of DU penetrators to determine penetration depths against RHA only at APG. The RHA targets are primarily composed of rolled hardened steel. As the DU penetrator impacts the plate or target, fragments and airborne particles are formed (Stoetzel, Waite and Gilcrist 1983). The target becomes contaminated with removable DU oxides, and embedded DU fragments become tightly bound or "fuzed" with the metal surface and inside the penetration site.

APG is the Army's largest generator of non-medical radioactive waste due to this type of testing. The organizations operating the test ranges at APG, TECOM, Army Research Laboratory (ARL) and the U.S. Army Combat Systems Test Activity (USACSTA), wanted to minimize the radioactive waste produced and possibly obtain beneficial reuse of the test debris. Various avenues were explored to develop methods to decontaminate the plate. A large stockpile exceeding 3000 tons of plate accumulated before a viable alternative to radioactive waste burial was adopted.

In 1982, the ARL had approximately 350 tons of armor plate in storage that had a scrap metal value of $39,200 based on the price of $.056 per pound of steel. The laboratory estimated that $100,000 could be saved in future armor procurement by

decontaminating and reusing the steel (Markland 1982). USACSTA has made only one shipment to bury the contaminated steel as radioactive waste. In 1983, 394 cubic feet of contaminated plate, consisting of 125 pieces and weighing 8 tons, were buried at Barnwell, South Carolina at a cost of $9,550. The cost was considered to be exorbitant.

Burial as radioactive waste was considered too expensive because of the costs to package, transport and bury the waste, and the loss of the scrap metal value of the steel. Cutting the plate was costly because it was labor intensive, required disposal of the contaminated section as radioactive waste, and exposed the worker to airborne radioactive material. Three potential alternatives to burial were identified, they are 1) electropolishing, 2) liquid abrasion for surface removal, and 3) reuse as high energy radiation shielding material.

Pilot electropolishing operations conducted by Quadrex Corporation and Chem Nuclear Systems, Inc. in 1980 and 1981 appeared to be very promising (Voit and Clayton 1981). Electropolishing was considered the ideal alternative because the amount of radioactive waste produced would be minimal, exposures would be reduced, and the decontaminated metal could be sold as scrap. Electropolishing is an electrochemical technique wherein the plate is immersed in an electrolyte and electrical current is used to remove the DU from the surface of the plate (Whang 1990). Quadrex Corporation decontaminated 45 tons during an eight month trial operation. Electropolishing techniques generated too much phosphoric acid, a mixed waste, and thus did not achieve reduction of waste goals and was not perused any further.

A study by Battelle Pacific Northwest Laboratory identified liquid abrasive blasting technology as the most productive method of decontaminating the plate and targets (Glissmeyer, et al. 1987). Liquid abrasive blasting uses abrasive particles in a liquid medium at high velocity to impact a surface and abrade the contamination from the surface. The spray must be contained and the abrasive separated from the water. HEPA filters and air samplers are required to protect and monitor personnel exposure from airborne radioactive materials and hazards produced.

The system was found to be effective in decontaminating the plates and targets as quickly as they are generated. Embedded fragments and bonded metal are cut from the RHA prior to abrading, but not to the extent as was required for electropolishing. The capital investment for the equipment, construction, and the labor required to operate the facility have been offset by the sales of the scrap metal and the savings normally associated with the preparation, transportation and burial of the steel as radioactive waste.

The Army Research Laboratory did not have as much success with the liquid abrasive blasting as the U.S. Army Combat Systems Test Activity and discontinued its use in 1992. The Army Research Laboratory's system was designed differently, having the plates decontaminated in an upright, vertical position, rather than horizontally. Detailed studies have not been conducted to determine the reason for the difference in effectiveness between the two systems.

Table I provides a breakdown of the cost of the liquid abrasive blasting facilities. Table II is a listing of the plate decontaminated and salvaged by the U.S. Army Combat Systems Test Activity.

In 1976, the TECOM arranged with Oak Ridge National Laboratory (ORNL) to transfer plate stockpiled at APG for melting. The Y12 plant had an estimated 600,000 tons of metal
contaminated with DU and had planned to operate a furnace by 1980 to smelt the metal, but operations never began.

ORNL and the DoE Oak Ridge Operations Office recommended the Fermi National Accelerator Laboratory (Fermilab) as a possible means of disposal. Fermilab was seeking 100,000 tons of high density material to use as shielding for accelerators. Fermilab had been previously utilizing Navy ship scrap and off-specification steel. The Army proposed to transfer the steel at no cost to the DoE. Fermilab and the DoE would only be responsible for the transportation costs from APG, Maryland to Batavia, Illinois. The federal government would save costs associated with burial of the steel as a waste, and the purchase of new steel by DOE. Fermilab determined that the plate would not be economically feasible. This was due to the additional environmental monitoring costs which exceeded the cost of steel ($140.00/ton) obtained from other sources. Monitoring was required because the laboratory planned to bury the shielding and was concerned about possible migration of the DU contamination (45 nci from surface swipes).

Los Alamos National Laboratory (LANL) approached the TECOM in March 1986 to pursue using the plate as shielding for the Los Alamos Meson Physics Facility. U.S. Army Combat Systems Test Activity had an estimated 2200 tons of plate and targets in storage and estimated that the cost to the Army to transfer the steel to LANL rather than disposal at one of the two existing burial sites at Barnwell, South Carolina and Hanford, Washington, would save at least $648,000. Table III provides the cost estimated for burial and the transfer of the steel to LANL.

Savings for the federal government actually increased by $572,000 because the DoE did not have to purchase 2200 tons of steel at $260/ton. Packaging and labor costs were reduced because the plate did not have to be packaged in Type A containers and could be shipped by rail. Prior to shipment, the fragments were removed from the plates, holes were filled with foam insulation, and the plate surfaces painted to provide a sealant. The plates were placed in rail cars and shipped to Colorado where they were then transported to LANL by truck. The cost of the armor plate shipped in 1986 exceeded the estimate of $415,000 and actually cost the Army over $500,000, but still proved to be much less expensive than burial. LANL was only able to achieve an estimated 75 percent shielding efficiency because of the holes and uneven surfaces in the plates and recommended smelting into shielding blocks by Scientific Ecology Group's (SEG). It would cost the Army $550/ton of armor to package, ship, and process the metal into shielding blocks. The DoE and LANL would be responsible for transporting the shielding blocks from Tennessee to its final destination. The total cost to the Army was the same as radioactive waste.

However, the DoE would save by not having to purchase new steel which had escalated to as much as $900/ton in 1990.

Fourteen hundred tons were transferred to SEG in 1991 as a pilot program at a cost to the Army of $732,000. SEG's furnace became operational in 1992. This pilot operation demonstrated that melting was a viable alternative to burial. LANL has obtained optimum shielding from the smooth surfaced, easily stacked blocks. The blocks are slightly radioactive, but have no removable contamination since the DU is blended in the steel. The Army has since expanded the Army's radioactive waste program to include armor plate recycling. Thirty-two hundred tons have been transferred to SEG thru 1993, eliminating the stockpile of plate and ancillary metal equipment at APG.

CONCLUSION

Recovery of the DU penetrators reduced DU concentrations in the test ranges, minimizing the opportunity for migration and reducing the ecosystems exposure to DU. The recovery of DU at the test ranges has been complicated by the terrain, weather, the presence of UXO, and personnel being prevented from entering the area due to other explosive testing near the firing ranges. A substantial number of penetrators have been removed from the test ranges near or on the surface of the ground where the potential for migration is the greatest. Recovery of DU is a viable pollution prevention program consistent with the Clean Water Act and Resource Conservation and Recovery Act.

The recovered penetrators were recycled by the manufacturers, removing them from the test centers' radioactive waste stream. Recycling DU continues to be an integral part of TECOM's radioactive waste reduction program, conserving vital space in the low-level radioactive waste depositories.

As a result of demonstrating that depleted uranium armor scrap and obsolete penetrators can be recycled into new production tank cartridges the following benefits will be realized; no additional DU will be imported into the Army inventory, contaminated magnesium fluoride and hydrogen fluoride will be eliminated, obsolete penetrators will not be buried, DU scrap armor will be reconfigured and recycled in the same step, and most importantly, the production round will cost significantly less. Follow-up studies will be conducted to determine if the structural and compressive yield failures were due to high zirconium levels.

The benefits of armor plate recycling are: 1) the practice complies with the Resource Conservation and Recovery Act and the Low Level Radioactive Waste Policy Amendments Act by recycling material and reducing the radioactive and hazardous waste produced; 2) the State of Maryland radioactive waste production and disposal is decreased; 3) the federal government's expenses for waste disposal and purchasing shielding material are reduced; and 4) less radioactive waste is stored at APG, reducing the exposure of the ecosystems to DU.

The Army Research Laboratory has decommissioned and disposed of its liquid abrasive

blast equipment and relies on recycling its armor into shielding blocks, and decontaminating less contaminated plate by cutting the contaminated metal from the surface. U.S. Army Combat Systems Test Activity has also discontinued to operate its liquid abrasive blast facility because transferring the metal for shielding is less expensive.

The Army is reducing its radioactive waste production and has eliminated the stockpile of penetrators and armor. The Army continues to explore effective methods of waste reduction with the least adverse impact to the environment.

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CLEAN UP AND RELEASE OF RADIOACTIVE LYSIMETERS AT THE HANFORD SITE

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INTRODUCTION

This report is a record of the US Army Corps of Engineers (USACE) experience in the investigation and remediation of radioactive soil sites at the Department of Energy (DOE) Hanford Site in Washington State. This report by no means chronicles all of the efforts leading up to the actual site remediation. Environmental work at Hanford is unique in that there are various DOE contractors assigned different areas of responsibilities. The Hanford site is also governed by an agreement between DOE, the

Environmental Protection Agency (EPA), and the Washington State Department of Ecology (WDOE). This agreement is known as the Tri-Party agreement and makes EPA and WDOE direct participants in environmental actions at Hanford.

Purpose

The areas in question, known as lysimeter plots, had been sampled previously by Pacific Northwest Labs-Battelle (PNL) after the conclusion of the experimentation and the subsequent removal and disposal of the contaminated material. The results of this initial sampling indicated that no residual contamination remained in these areas; however the USACE was tasked to perform the follow up verification sampling for two reasons. First there was a concern raised by the Washington State Department of Health that the analytical methods used by PNL, (i.e., gross alpha and beta counting), might not adequately detect all of the types or quantity of radiation expected due to matrix effects and screening in the soil being analyzed. Secondly, it is US Department of Energy policy to conduct independent verification of clean closure especially in the case of radiological analyses that support releasing property for unrestricted public use such as this.

History

The Arid Lands Ecology (ALE) is a 311 square km (120 square miles) area, south of the Hanford site, that prior to 1942 contained a few homesteads and natural gas wells. In 1942 the US Government acquired the area as a buffer area for the Hanford site. In the early 1950's a NIKE missile site was constructed by the Army for protection of the Hanford site from air attacks. The Army remained until the early 1960s. In 1967, this area was established as the ALE, and functioned as a National Ecological Preserve. The area is currently managed by PNL for DOE. In 1993, USACE was tasked by DOE to investigate and remediate CERCLA sites on the ALE to allow DOE to release the ALE to non-DOE use.

In mid-July 1994, the US Army Corps of Engineers conducted sampling of two areas within the Arid Lands Ecology that had previously been used by PNL in radiological experiments. The samples were analyzed for a number of specific radioisotopes to verify that these areas did not contain significant radiological contamination. To adequately address these concerns, a sampling and analysis plan was devised by USACE in close consultation with WDH that sampled over 20 percent of the more than 500 individual locations that had contained lysimeters in the two plots and analyzed these samples for the specific radioisotopes associated with the individual lysimeters. In addition split samples were analyzed by the WDH state laboratory to provide additional Quality Assurance and coordination was conducted with Oak Ridge National Laboratories to provide further independent analyses for the DOE. The results from this sampling effort substantiate that clean closure was accomplished by PNL. The results from all of the project samples exhibit insignificant activities, with all but one of the samples having activities of < 1.2 pCi/g. The one caveat was a sample that had 53 pCi/g. This sample was taken from an area that had been previously identified by PNL as a place where ants had been discovered in one of the lysimeters and small amounts of contamination could have escaped the lysimeter tube with the ants. Although activities for this isotope (Pu-238) at this level and volume are associated with extremely small exposure risks, USACE in consultation with the Environmental Protection Agency excavated and properly disposed of soil from this area to further mitigate any exposure risks.

Lysimeter experiments were performed by PNL in the 1970s and 1980s on the Arid Lands Ecology. The purpose of the experiments were to calculate transmission and migration rates through soil and plant uptake of the radio nuclides. The lysimeters used contained I-131, I-129, Pu-238, Pu-239, Am-241, Cm-244, Np-237, U-232, and U-238. There were three locations where lysimeters were placed known as "lysimeter plots." Names are assigned to the plots sampled for the purpose of this report reflecting the commonly used name for the plot.

The first set of lysimeters are identified as the "Main Plot," 310 lysimeters were used at this site. The lysimeters consisted of a radioisotope mixed with soil and placed in PVC tubes 5" in diameter and 1 meter in length. The tubes were set inside a 6" diameter tube and sealed at the bottom and placed vertically in the ground, with the open top of the tubes flush with the ground surface. Isotopes used in the main plot consisted of 1-129, Pu- 238, Pu-239, Am-241, Cm-244, Np-237, U-232, and U-238 spiked at concentrations per tube up to 1.0 mCi. PNL decommissioned the main plot in 1992 by removing the lysimeter tubes and transporting them to the main Hanford site for disposal as radioactive waste PNL then performed a soil analysis

for gross alpha and beta at the surface and at 1 meter depth of every third lysimeter tube. The results from this first sampling event indicated that a clean closure of the lysimeters had been accomplished. To verify these results the US Army Corps of Engineers was tasked to perform another sampling event analyzing for the specific radioisotopes of concern. This sampling was conducted and coordinated in cooperation with the Washington State Department of Health (WDH).
The second location is identified as the "Old Plot". These lysimeters contained Pu-238, I-129, and Am-241. In 1979, after a brush fire swept the area, the lysimeter tubes were removed and placed in the main plot. There were a total of 128 lysimeters similar in the main plot. No prior sampling data was available from the decommissioning of the abandoned plot.
The third plot was identified as the "Kathy Moss" plot. This name was assigned in reference to the individual that performed the lysimeter experiments at that location in 1989 (ref PNL document #7298). This lysimeter consisted of direct injection of I-131 into Bunch Grass (Agropyron Spicatum). A glass tube, with a camera, was placed adjacent to the plant roots in order to observe the plant uptake of the I-131. A total of 18 plants were injected with the I-131. No confirmation sampling was performed at this location since I-131 has a half-life of 8 days and as a result any I-131 left in the soil would have decayed to background levels.
Sampling Rational and Description
The USACE designed and coordinated verification sampling of the Main and Abandoned plots. Verification was performed anticipating the Arid Lands Ecology land being eventually released for non-Department of Energy use. Prior to sampling, a site survey was performed by Westinghouse Hanford Co. (WHC) using the Ultra-Sonic Ranging and Data System (USRAD) and field survey instruments to verity that no contamination was present at levels hazardous to workers. The USRAD system consists of a backpack mounted radiation detection system with data links to a remote receiving station. The receiving station collects and correlates the detector data, and generates an accurate survey map of large areas.
Samples were collected by the WHC sampling group for shipment through a USACE contracted lab to TMA-NORCAL in Richmond, CA. The Washington State Department of Health radiochemistry lab volunteered to perform the QA for this project with no cost to the government. Along with saving monies, this allowed the WDH to directly participate in the site remediation process. An Independent Verification Contractor (IVC), Oak Ridge National Labs, also performed sampling. The IVC is a Department of Energy (DOE) unique requirement used for independent verification of the contractor sampling process. Although no sampling was slated for the Kathy Moss plot, radiation field surveys were performed to ensure no unanticipated contamination was present.
No formal clean-up level was set for this site, although an ongoing dialog was established between USACE, EPA, DOE, and WDH throughout the design and performance of this sampling. Current DOE clean-up levels for unrestrictive use of a site are 100 mrem total effective dose equivalent (TEDE) to any member of the public in one year. With the application of ALARA (as low as reasonably achievable) principles, this 100 mrem value is reduced to 30 mrem. With these general guidelines, and in consultation with WDH, analytical reporting limits were set that could be used to establish exposure scenarios at least two orders of magnitude lower than the 30 mrem level when evaluating the clean closure of the lysimeter plots.
Sampling and Analysis Design
The sampling and analysis plan (SAP) was designed to meet the data quality objectives for providing data usable in establishing verification of clean removal of the lysimeters. The sampling strategy developed in consultation with the WDH provided for sampling over 20 percent of the lysimeter holes at the surface and at 38 inches below surface. These sample points were reasoned to be the most likely areas of possible contamination after lysimeter removal because any surface contamination could have been brushed to the bottom of the holes or there could be a slight possibility of leakage of the PVC containers. A morgue sample was held from each of the sampled lysimeters and composite samples were made by combining equal amounts of the homogenized soil from five samples for the same analyte into a given composite sample. This provided for analyzing a large subset of the total set of lysimeters without exorbitant analytical expense. This plan allowed for analyzing each of the sub samples for any composite that exhibited appreciable activity to pin point any possible contamination area. The samples were taken from lysimeter holes that had been previously identified and marked using plot maps provided by the PNL

scientists who had been involved with the lysimeter projects.
The sampling was designed to meet the requirements outlined in the HASQAP and in DOE order 5400.1, as well as to conform with USACE guidance. The actual field work was performed by the Westinghouse Hanford Company (WHC) field crew designated to perform radiological soil sampling in accordance with established WHC standard operating procedures and protocols. WHC also provided Health Physics Technicians for site safety. All radioactive contaminated soil was packaged, transported, and disposed by Bechtel Hanford Inc. Coordination and oversight of the sampling was provided by USACE. The project lab chosen for the specific radioisotope analyses was Thermal Analytical(TMA) in Richmond California. TMA has been used extensively for radiochemical analyses for Hanford and other DOE installations. It is a participant in a proficiency evaluation program conducted by Environmental Measurements Laboratory for DOE, and was able to provide a deliverables data package in a format similar to the Environmental Protection Agency's Contract Laboratory Program (CLP) package to facilitate data validation. A summary of the SAP for the project follows.

Sampling and Analysis Synopsis
Next is a synopsis of the sampling and analysis plan provided to the sampling crew, and to the project and analytical laboratories involved with the evaluation of the lysimeter plots. It is provided in this report to aid in reviewing the validity and appropriateness of the data generated.
PROJECT DATA QUALITY OBJECTIVES
The purpose of this sampling event is to confirm that complete and clean removal of the radioisotopes contained in these lysimeters has been accomplished. The analytical reporting level requirements have been established to provide adequate sensitivity to determine if contaminant concentrations present an exposure risk. Specific radioisotope analyses have been chosen for this testing to avoid potential analytical problems associated with gross alpha and beta screening as was previously performed to confirm clean closure of these lysimeters. The sampling and analysis proposed was of limited scope to be used as a check of the previous analyses.
As an independent verification, the Washington State Department of Health analyzed split samples from this event on at least 10% of the samples sent to the project lab. Both the project and State lab are participants in the semi-annual PE program for radiological analyses conducted by Environmental Measurements Laboratory as outlined in DOE order 5400.1. In addition the project lab submitted Quality Assurance Program manuals, Statements of Qualifications, copies of current licenses, certifications, and proficiency results for the analyses to be employed to the USACE Mandatory Center for Expertise (MCX) in Omaha, Ne. for USACE validation prior to beginning the project.
It is typical for analytical data used in risk assessment to use CLP protocol analyses and RAS deliverables with level IV data validation. In an attempt to present data of similar quality USACE asked for a CLP like deliverables package such as that developed by TMA in a software package called DVD, and following EG&G's GRRASP protocol modified to meet the project detection limits.
PARAMETERS CONSIDERED FOR ANALYTICAL METHOD QC
Control parameters are anticipated to be within ~ 2 sigma data scatter in LCS running averages expressed in % recovery assuming activities of at least 10 X of RDL. LCS levels should not be > 20 X RDL. All recoveries that are outside of specified range shall be flagged. Any samples in a batch where associated LCS recoveries are outside 60-140% shall be reanalyzed for those isotopes which are out of control. LCS and LCS duplicates will be analyzed with each batch. A batch shall be no more than 20 samples. One sample duplicate shall be analyzed with each batch. Duplicate Analyses shall be within 30% RPD for all activities greater than 10 X project specified RDL. Samples from batches with duplicates outside this limit shall be flagged and impact on data shall be addressed in a case narrative. If duplicates are > 50% RPD at activities > l0X RDL, samples from the batch shall be reanalyzed at laboratory expense.
QA/QC SAMPLING
One composite sample out of each group of composite samples for a given analyte was split into 3 replicates. Each replicate was assigned an independent sample identification number that was to be logged with the sample location in the field notebook. One replicate was sent to the State lab, and 2 replicates to the project lab. This provided a blind duplicate at the project lab for each analyte. A

background sample was taken compositing samples taken at 4 points ~ 20 yards from the sampling site boundaries and roughly encompassing the sampling site. The grids below depicted sampling strategy.
TABLE Ia
TABLE Ib
TABLE Ic
TABLE II
Each cell represents an individual lysimeter. Those lysimeters sampled, ~ 20-25% of the total originally spiked) were assigned a number. A composite was made from samples taken from the lysimeters designated with the same number from the grid above. The samples were number coded on this grid for the specific radioisotope. A morgue sample of each sample point was retained in a 4 oz. jar. The specified lysimeters were sampled using a hand or power auger at the approximate center of each lysimeter. A 2 oz. portion of soil was collected from the surface to 3 inches in depth and again in another plug taken from 39 to 42 inches in depth. The soil from these two plugs were mixed then placed in a 4 oz. jar labeled for the analyte and the lysimeter. Composites made by taking one spoonful from each jar corresponding to the same lysimeter number in the above grid were submitted to the lab, i.e. if samples #1 through 5 above are from Pu-238 lysimeters, then 5 composites were made, 4 containing aliquots from 5 sample points, and one containing aliquots from 4 points. One of these composites was split into 3 replicates for QA/QC as referenced above.
A similar sampling strategy was followed for the old lysimeter plot. This plot consisted of 128 lysimeters. Individual lysimeters were formerly spiked with either I-129, Am-241, or Pu-238. Because the Information was incomplete on the number of lysimeters spiked or which of the three isotopes was used, we sampled ~ 20 % of the holes (27 total) at random throughout the plot. These samples were composited into 6 samples and 1 background that was analyzed for all three of the radioisotopes mentioned above. The sampling was consistent with the method used in the large plot. The background sample was composited from 4 sample sites taken - 20 ft. away on each of the four sides of the old lysimeter plot. The composite aliquots were taken using the same technique as employed in the main lysimeter sampling, i,e. by taking two plugs at surface and at 39-42 inches. All 4 aliquots from the sides away from the plot were composited in an 8 oz. jar and labeled as background with appropriate location references. The background sample was analyzed for each of the referenced analytes.
Sample Results
The samples sent to TMA were analyzed on an expedited turn around time to provide the results to be interpreted in time to allow for further action to be taken before the deadline of the Tri Party Agreement Milestone.
The results strongly support PNL's conclusion that clean removal of the lysimeters has been accomplished. All of the samples with one exception exhibited activities of less than one picocurie per gram (1 pCi/g) or less than the required detection limit (RDL) for the analysis. The one sample that had an activity of 53 pCi/g was submitted for reanalysis along with the 5 sub samples that represented the individual lysimeters sampled from this composite sample. Two of these sub samples had appreciable activities of 27 and 372 pCi/g respectively. The location of these lysimeters would correspond to cells K1 and G1 in the graph. One cubic meter of soil was excavated from each location, and follow-up sampling was performed. The follow-up sampling results showed a Pu-238 concentration for location K1 still at 35 pCi/g.
Follow Up Actions
In response to verifying and pinpointing the source and extent of this minor contamination (35 pCi/g), risk base analyses were conducted by USACE using a software package designed for radiological risk assessment called RESRAD (DOE order 5400.5) RESRAD is a computer code developed at Argonne National Laboratories for DOE that is used to calculate site specific RESidual RADioactive material guidelines (cleanup criteria) and radiation dose to an on-site resident, RESRAD calculates residual concentrations of radionuclides in soil; concentrations of airborne radon decay products; external gamma radiation levels; surface contamination levels; and radionuclides in air and water. The dose is calculated using external radiation exposure; internal radiation dose from inhalation (including radon); and internal radiation dose from ingestion e.g., drinking water, food, and soil.

The purpose of using the RESRAD modeling program by USACE at this point was to establish the TEDE for this site, and whether the clean-up action was complete. By using RESRAD, it is possible to find the point at which further sampling and remediation is not plausible. The RESRAD data follows.
Two possible future use scenarios for the ALE were used to create a RESRAD model. The primary pathway for Pu-238 in this scenario is inhalation and the standard inhalation rate used by RESRAD is 8400 m3/year.
1. Subsistence Farmer. This scenario assumes that the site would be utilized for a farmer living at the site with 50% of the time indoors and 25% outdoors. The farmer would consume 70% of the crops grown at the site and would also consume water from an on-site well. This is the "worst case" scenario. By applying the standard defaults of RESRAD, which are the most conservative, the calculated dose from the 35 pCi/g concentration of Pu-238 would be a TEDE to the farmer of approximately 0.52 mrem/year.
2. Recreational Use. This scenario assumes that the site would be used for recreational use. The occupancy rate at the site is based on a maximum of 192 hours/year (8 days camping) with no indoor occupancy. By adjusting the defaults of RESRAD to reflect this scenario, the TEDE is approximately 0.039 mrem/year. Based upon the current future of the ALE, the recreational use scenario is the most likely.
In general no scenarios could be established that presented any significant exposure risk to the public.
CONCLUSIONS
No further action or sampling is anticipated to be required to release these areas as clean. At this point in time, EPA has not made a final determination on the release of this site.
GLOSSARY OF TERMS AND ACRONYMS

AS	Alpha Spectrographic method of radiological detection for analysis
CLP	Contract Laboratory Program
DOE	Department of Energy
DVD	Data Validation and Deliverables
EG&G	A contractor for the DOE
GRRASP	General Radiochemistry and Routine Analytical Services Protocol
GS	Gamma Spectrographic method of radiological detection for analysis
HPT	Health Physics technician
HTRW	Hazardous Toxic and Radiological Waste
LCS	Laboratory Control Sample
MCX	Mandatory Center of Expertise
USACE	United States Army Corps of Engineers
WHC	Westinghouse Hanford Company

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CHARACTERIZATION OF THE LAKE CITY ARMY AMMUNITION PLANT RANGE FOR DEPLETED URANIUM
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ABSTRACT
This paper describes a radioactive material characterization of the Lake City Army Ammunition Plant firing range. This range was used for test and disposal firing of depleted uranium rounds for many years. The projectiles fired were experimental in nature and did not have penetrator properties. The characterization was completed manually using portable instruments to identify locations of impact on the range. Data collected was evaluated with Microsoft Excel Version 5.0 and Golden Software Surfer Version 5 to provide graphical display of the relative magnitude of depleted uranium present in each area of the range. Estimates of waste volume which may be generated in remediation of each area are made based on the area contaminated and the depth of projectile penetration.
INTRODUCTION
This characterization regards evaluation of an army ammunition plant firing range for the presence of depleted uranium, DU. The primary mission of the plant is development of munitions for the U.S. Army.
The plant operated under an NRC source material license which allowed demilitarization of an estimated 40,000 rounds made from Depleted Uranium (DU) in

the form of 20 mm penetrators by firing them into sand in a bullet catcher designed to minimize scatter of the rounds as they impacted the sand. This characterization identified the approximate location of source materials on the firing range, in a bullet catcher waste collection area, the sandpile area, and in a bullet catcher. The full range is about 2300 yards, 1.3 miles.
The radionuclides of concern in DU are 238Uranium, 234Thorium, 234mProtactinium, and 234Protactinium. The Thorium and two isotopes of Protactinium are short lived daughter products from the decay of 238Uranium. In this characterization, the photon radiation given off from the decay products of 238Uranium was used as the primary radiation for detection of source material in the soil and sand. Wipes of areas were counted with an alpha/beta counter, and direct measurements with beta and alpha detectors were made to determine spreadable activity.
Estimates of waste material which will be generated by various remediation methods are presented. These estimates are based on approximations of the number of projectiles found in each area. This is because prior to this characterization, the amount of material fired into the range could not be reasonably approximated.
HISTORY OF DEPLETED URANIUM USE ON THE RANGE
The present Safety Manager, conducted an interview with the previous Radiation Protection Officer at the site in September 1988. This interview revealed that in 1960-1961, test firings of rounds manufactured at the site was done daily. He said approximately 1200 rounds were fired into the air and not necessarily into a bullet catcher. However, he also said that there is no documentation regarding the quantity. He also indicated that around 1972, 40,000 to 60,000 rounds were shot into bullet catchers and the waste (DU round fragments) was "bucketed" up and sent off-site for disposal. In 1976 or 1977 approximately 40 rounds were demilitarized by shooting into the range.
An attempt at characterization for radioactive materials on the range in 1990 was inconclusive regarding the number and depth of projectiles fired into the range. That survey was done using a large area NaI(Tl) detector mounted to a wheeled cart and driven over the ground surface, it showed the presence of only spotty indication of photon emitters on the 2188 yard impact area.
SCOPE OF CHARACTERIZATION
Characterization of the range was to be conducted with specific attention to "affected areas" and "unaffected areas." The distribution of these areas was established and completed as follows:

Affected Areas:
1. 600 yard bullet catcher.
2. Mound adjacent to 600 yard bullet catcher.
3. Sandpile.
4. 1750 yard impact area.
5. 2188 yard impact area.
6. Building 45 weapons storage areas.*
7. Building 45 firing areas.*

Unaffected Areas:
1. 2400 yard bullet catcher.
2. All open areas under flight paths to:
    a. 600 yard bullet catcher.*
    b. 2195 yard impact area.*
    c. 2200 yard bullet catcher.*
3. Impact areas and bullet catchers adjacent to affected areas.
4. The holding pond near the sandpile.

Surveys of affected areas caused collection of more than 300,000 individual direct readings of radiation by portable scintillation detector, and collection of approximately 1000 samples for analysis by gamma spectroscopy to an MDA somewhat less than 1.3 Bq/gram (35 pCi/gram). Surveys of those areas classified as unaffected were made with the same methods as affected areas but in much less detail unless radioactive materials were detected, at which time, a more detailed evaluation of the area was made to more accurately locate the material, as was done in the affected areas.
SCHEDULE AND RESOURCES

The task was completed with five survey teams of five workers per team, a site coordinator, a labor supervisor, an HP supervisor, four lead HP technicians, a project engineer, and a project manager. The project manager was responsible for overall safe progress of the job and oversight of quality assurance in the collection and handling of data.
The majority of time spent on the survey was devoted to the external radiation survey as expected. Four survey teams were assigned arrays of 10 micro-R meters each and surveys were conducted in one foot increments over each of the affected areas except the building 45 areas. This survey
provided an in depth evaluation of DU present within the affected areas of the range. The fifth team was used for gridding and sampling of the affected areas. Surveys of the unaffected areas were conducted with different arrangements of personnel in order to ensure sufficient manpower to characterize these areas.
The survey was conducted in December 1993, and January 1994. This proved to be a very good time to evaluate the range due to elimination of hazards caused by insects and reptiles. The cold affected performance of instruments only minimally, some problems were experienced with battery contacts which caused erratic response in these instruments. Because nine other instruments were present in each array, identification of abnormal response was easy. The range was shut down from December 23 to January 3, this provided the opportunity to work more intensively on the characterization than is allowed when the range is in operation. During times of range operation, the crews had to work around the firing schedule, and characterization activities were primarily conducted during weekends.

ASSESSING THE SITE STATUS

Several surveys of areas of the range had been conducted in the past and were used to provide a skeletal framework to identify the most significant areas of site contamination except for the 1750 yard impact areas and bunkers in the flight path. The survey of the 1750 yard impact area proved that area to be as significant in DU contamination as the 2188 yard impact area.
Depleted uranium was found to be the only radioactive material present on the range by gamma analysis of samples. The radionuclides of interest are therefore 238U, 234Th, 234Pa, and 234mPa. Although 234U is also in the decay chain for 238U, it is a very long lived radionuclide and as such, it's progeny will not have achieved a significant activity in the time period since refinement of the 238U.

SITE PREPARATION PRIOR TO SURVEY

Prior to conducting surveys at the site, reference points were selected for the surveys and for subsequent remediation at the 2188 yard impact area, the 1750 yard impact area, the sandpile area, and the 600 yard bullet catcher area. The reference points were based on locations of monitoring wells or other immovable objects. Survey grids were established with reference to these points.
A grid system of reference was established to coincide with reference points of permanent objects. The final grid size was 3 meters (~10 feet) on a side over most affected areas.
Due to the extreme overgrowth of the sandpile area, it was necessary to remove brush from the planned survey areas. Grassy areas of the impact zones were trimmed in July and did not grow substantially in the areas cut. Very cold temperatures were expected and as a safety measure, insulated coveralls were provided to all crew members. The coveralls were worn by all personnel working on the range for extended periods of time. Even with work in areas restricted for contamination control, no personnel contamination was detected on these coveralls.
A large trailer was moved into the site at the entrance to the range, a small dressing area was built onto the outside of the trailer. This trailer housed the counting room for smear analysis, instruments, equipment, and a break room for the crew and the project manager's office.

WORKER PROTECTION

A number of hazards not typically present during radiation work are present at the LCAAP site. In general, ATG work procedures were in effect for safety of our workers and others. However, the following points were considered:
1. The first and most important is unexploded ordinance. Over the years, many different types of experimental and production ammunition rounds have been fired into the range. These included explosive rounds typically of the 20 millimeter size although other experimental rounds were known to have been fired. As a result of this, no ATG personnel or subcontractors were authorized to do large scale

excavation at the site with hand tools. No personnel were allowed to handle ordinance in the bunkers which may have been unexploded. An ordinance specialist from ATG at Fort Myers was present for the first three weeks and last week of the project. During that time, he searched all available areas of the range for evidence of explosives and could find none in the areas to be worked.
2. It was very cold in the LCAAP area during the characterization. All personnel wore coveralls, layers of cloth clothing, long underwear, a hard hat with visor, steel toe shoes, and safety glasses.
3. Samples of air were taken with breathing zone samplers during work operations which could have caused airborne materials such as sampling and removal of sand from the catch box. No airborne 238U or short lived daughter products associated with DU activity were identified. Samples did show evidence of radon daughters as expected, however, the level observed was not important from a radiological health standpoint.

CHARACTERIZATION

Decontamination of the specified sites must reduce the residual radioactive contamination to less than 1.3 Bq/g of DU (238U). The radiation surveys with portable scintillation counters (Ludlum Model 19) and the soil sample collection and analysis proved to be adequate to provide reasonable assurance that this level would be detected and properly characterize the range.

THE 600 YARD BULLET CATCHER

This area was determined from surveys to be the most affected area in terms of projectile accumulation, spallation, and the likelihood of spreadable activity. Surveys confirmed that activity deposited in the catch box was spread from the sand in that structure to sand outside the box and adjacent to the large mound outside the catcher. The activity in this area was found to be more due to spallation and degradation of the projectiles than in other areas. This is likely to be the result of weathering action in the sand which was exposed to rain and water accumulation near the bullet catcher. Many projectile fragments and oxidized pieces were found in this area.

Sand was removed as possible from the bullet catcher to facilitate survey of the structure. The surveys done in this area indicate that most of the structure can be salvaged as clean waste and only a fraction of the structural materials will need removal for disposal as radioactive material. Surveys of the structure were done with pancake probe GM detectors, alpha scintillation detectors, and micro-R meters. The sand in this catchbox is mostly contaminated with projectiles and oxidation from projectiles. The most cost effective means to remediate this area would be disposal of the sand as radioactive waste.

Samples of sand and soil in this area showed activity concentrations of DU as high as 115 Bq/g with an average concentration of 34.5 Bq/g. A total of 46 samples were taken from the sand in the bullet catcher when it was removed for survey of the structure itself.

THE 600 YARD MOUND ADJACENT TO THE BULLET CATCHER

A large mound 15 meters by 27 meters by 4.5 meters (approximately 50 feet wide by 90 feet long by 15 feet high) is present near the 600 yard bullet catcher. It appears that portions of the mound were used for dumping of sand from the bullet catcher over the years. This material has been weathered and has caused accumulations of DU and projectiles around the base of the mound and in the low spots of drainage paths from the area. Sampling has shown that the material has stayed close to the mound and has not migrated to the stream which runs adjacent to the area at less than 60 meters to the north west of the bullet catcher. Eighty-one samples were taken from the mound and the area around the bullet catcher.

Remediation of this area should consider an approximate depth of soil containing projectiles and DU oxidation products of 60 cm. The area contaminated around the mound is estimated from surveys to be approximately 75 square meters. This relates to an approximate waste volume of 45 cubic meters (1600 cubic feet).

THE SANDPILE AREA

The sandpile area was originally considered to be only minimally contaminated with dumpings from the 600 yard bullet catcher. After surveys of this area, it is obvious that material from this bullet catcher was dumped in more than one location of the sandpile area but the dumpings appear to be localized into small groupings over most of the sandpile with some very small piles dumped immediately outside the area.

A depth study was done on the piles found to contain projectiles in different areas of the sandpile. Two methods were used in this study, a scintillation detector was

used to identify activity as a function of depth by monitoring the sampling holes. This method showed that readings stayed constant to about 60 cm to 90 cm and then decreased to background below that depth. This was reasonably consistent from hole to hole. A second method, excavation of a grid at different depths, similar to that done at the 1750 and 2188 yard impact areas, was used at the sandpile to identify depth below contaminated piles. When the small contaminated piles were collected with a front loader, no activity was identified below the piles.
In addition to the external radiation surveys, 148 samples were taken in the affected area of the sandpile. The results of those samples show activity predominately near the surface. Depth samples did not show activity above the remediation level.

IMPACT AREAS

Two large impact areas were surveyed with Ludlum Model 19 micro-R meters to establish a profile of emissions from the soil. Readings were taken one foot on center and the response was plotted over the area surveyed to identify areas of higher DU concentration.
In all, 1541, 10' by 10' grid squares on the 1750 yard impact area, and 1970 grid squares on the 2188 yard impact area were established and readings were taken every square foot.
Samples of soil were taken in all areas of the impact zone and in the creek bed. Similar to the sample results at the 2188 yard impact area, samples showed localized accumulations of activity in the immediate area of the projectiles and no accumulation with depth. Studies of the depth of projectiles in the soil were done in several sample holes and by digging up one of the grids in this area. The results showed that activity did not travel into the soil and no activity could be found at greater than 30 centimeters into the soil. Over one hundred samples of soil were taken in this area.
The 2188 yard impact area was initially considered as the most important of the impact areas for DU projectiles. A large initial grid was set in the geographical center of the impact zone. This grid proved to encompass the most significant projectile impact area but did not contain the entire area. Several grid additions were made to attempt definition of the affected area. In addition, 204 samples were taken to determine activity transfer to the soil and all wells in the area were sampled for the presence of DU.

AREAS OF SCATTERED ROUNDS

When firing into the long range areas, scattering of rounds off of the soil surface or off of subsurface rocks was possible. Foot traverses of these probable ricochet areas with a scintillation detector and count rate meter was the method recommended by the NRC. This method was enhanced using of micro-R meters to locate projectile impact zones within the firing line.
Wooded areas adjacent to the 2188 yard impact area were searched with the same instruments, accumulations of projectile fragments were found within 6 meters of the woods line on the north side of the area. The searches were conducted with 5 people each with a detector, and all surfaces not extremely hindered by trees and shrubs were surveyed. Areas where projectiles or fragments were found were identified with surveyors flags.

INTRUSIVE SAMPLING

Samples were taken in all areas of the range, affected and unaffected. Each sample was a combined depth and surface activity evaluation when this was practical. The samples were collected when practical with a California hollow point auger. In some cases, the auger was not sufficient to remove soil/sand from the hole and a planters shovel was used to collect the sample. Due to the hard, frozen clay soil found in most areas of the range, a power auger was used to reach sampling depth. All sampling tools were cleaned between each sample collection, and efforts to minimize cross contamination were used as practical. The extreme cold and windy conditions made this difficult.

SAMPLE ANALYSIS

The method for measurements of soil samples taken from the various areas of the site were evaluated by comparison counting to a 238U soil standard. The focus of the analysis was the 92.3 keV and 92.8 keV photons emitted from 234Th, the immediate daughter of 238U. These photons are emitted with a photon yield of approximately 2.73% and 2.69% respectively. A region of interest around the 92.5 keV energy range allowed quantification of the 238U in equilibrium with it's Thorium daughter. The

analysis was conducted by Scientech Laboratory.

BACKGROUND SAMPLES

Background soil samples were taken primarily in four unaffected areas of the range and also in several locations off of the range. These samples were taken, similar to the range characterization samples, at depths of 15 cm, 60 cm, and 90 cm when possible. It was intended that background sample activity would be subtracted from characterization sample activity, but background activity proved to be insignificant compared to the MDA of 0.37 Bq/gram (10 pCi/gram) for most samples.

SENSITIVITY OF ANALYSIS

The comparison of sample activity results with the limiting level of 1.3 Bq/gram is to be based on a gamma spectral analysis of each sample. The detection level, due to laboratory requirements, was established at approximately 0.37 Bq/gram, approximately 30% of the limiting concentration for DU. The limiting guidelines, established by NRC, are directly related to the risk posed by the nuclide 238U in equilibrium with it's short lived daughters through defined exposure pathways.

ESTIMATES OF WASTE VOLUME

Waste volume estimates were made based on the degree of surface area found to be above background response and the accepted depth of activity based on sampling. In anticipation that soils surrounding the primary areas of projectile accumulation, the following volume could be considered above the NRC remediation guidelines:

600 yard bullet catcher area 1020 cubic meters (36,000 cubic feet)
Sandpile area 187 cubic meters (6,600 cubic feet)
1750 yard impact area: 5240 cubic meters (185,000 cubic feet)*
2188 yard impact area: 5000 cubic meters (177,000 cubic feet)*

These numbers are based on the area of significant emissions above background. In the impact areas, there was not evidence of activity spread from the immediate impact zones. Extraction of small amounts of soil around each impact would reduce total waste volume to approximately

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ABSTRACT

The Montclair/West Orange and Glen Ridge Superfund Sites are located 12 miles west of New York City in Essex County, New Jersey. The sites are contaminated with waste materials from radium-processing facilities which operated in the area during the early 1900's. The waste materials, containing radium and other radioactive isotopes were placed in three separate landfill sites. As the towns were developed, homes and streets were constructed on or near the disposal areas. Some of the contaminated materials were also used as fill around basements foundations and yards.

There are 769 residential and commercial properties, having a combined area of approximately 210 acres, included in the two Superfund Sites. The principal contaminant is radium 226, with lesser concentrations of thorium, uranium, and lead. Major public health risks are indoor radon gas build-up and indoor/outdoor gamma radiation.

In the early 1980's the United States Environmental Protection Agency, Region II (EPA) and the New Jersey Department of Environmental Protection (NJDEP) began a series of surveys, investigations, and studies to determine the boundaries of the contaminated areas.

In 1985, NJDEP began a pilot study to remediate the properties by excavation and off-site disposal of contaminated material. The study was interrupted when the disposal permit to the off-site disposal facility was revoked. Residents of four partially remediated homes were left in temporary accommodations for nearly five additional years, while a new disposal facility could be identified and utilized.

In 1989, the EPA issued a Record of Decision (ROD) which chose excavation and off-site disposal of contaminated material as the preferred alternative.

Additionally, through an Interagency Agreement, the EPA engaged the Kansas City District of the U.S. Army Corps of Engineers (USACE) to perform the remedial design. Bechtel National, Inc., (BNI) was selected by the Kansas City District Corps of Engineers to perform the extensive field investigations and provide the detailed engineering designs for the remedial work.

The purpose of this presentation is to highlight key elements of the design process for the remedial action at Montclair. Those key elements are as follows:

- Meeting community relations challenges
- Measuring radioactive contamination
- Developing plans and specifications
- Packaging of remedial action contracts
- Continually improving both the process and the designs

COMMUNITY RELATIONS CHALLENGES

Long Site History

The Montclair/West Orange and Glen Ridge Superfund Sites include three separate residential areas in Essex County, New Jersey, consisting of 769 residential and commercial properties contaminated with radioactive materials. The contamination originated from radium processing facilities which operated in the area during the early 1900's. Radium research and processing facilities were prevalent in Northern New Jersey from the early 1900's to the late 1920's. The processing facilities produced luminous paint for watch dials, surveying equipment and airplane instruments; and later extracted the radium for research and medical applications. By the early 1930's the effects of excessive exposure to radium were discovered, which subsequently caused an end to the processing facilities.

The radioactive waste materials from these facilities were disposed as landfill in what were once rural areas. As development flourished, homes and streets were constructed over the landfilled waste. In 1981 the EPA was requested by the NJDEP to conduct an aerial gamma radiation survey of the area to determine the presence of radioactive materials.

The aerial survey in Essex County identified several residential areas with elevated gamma radiation levels. Ground investigations conducted in 1983 confirmed the presence of radium contamination and identified a number of homes in the towns of Montclair and Glen Ridge with gamma radiation and/or radon decay products exceeding acceptable levels. Further contamination was discovered in the town of West Orange as the investigation continued into 1984.

The initial ground investigations at the sites focused on the identification of homes with elevated radon levels or indoor gamma radiation exposure levels. During

these initial investigations a number of homes were found to have radon levels up to 100 times the normal background for the area. At that time, the potential risks of naturally occurring radon had not been identified. Even so, these readings were considered to be extremely high.

Due to the elevated levels of gamma radiation and radon decay products, the EPA implemented mitigation actions by installing radon abatement systems to ventilate basements. These systems were designed to dilute the level of indoor radon by drawing "clean" air from the outside of the home. To reduce the gamma radiation exposure level in a home, one-eighth-inch thick lead sheets were placed in layers over areas emitting high gamma radiation until there was enough shielding to reduce the radiation exposure to acceptable levels.

Unlike most Superfund sites where residential properties are adjacent to the contaminated area, the Montclair/West Orange and Glen Ridge sites are almost entirely residential properties. The majority of the homes are one and two family residences of brick or timber-framed construction with full basements, front and back porches and detached garages.

The properties were classified into one of five categories according to their public health hazard and engineering requirements. Category I properties were core area properties which contained extensive areas of radium-contaminated soil throughout the property, including under and around the house foundation. Category II properties used a criteria of basement wall or outdoor gamma radiation levels equal to or greater than 50 micro-Roentgens per hour and extensive radium contamination. Category III properties had radon and radon decay product or gamma radiation levels above health guidelines, and with limited "hot spot" radium contamination. Category IV properties contained soil contamination above the cleanup standards of 5 picocuries per gram, but with no radon decay product or gamma radiation levels above health guidelines. Category V properties had no detectable radium-contaminated soil present in excess of soil clean-up standards.

Initial Remedial Experiences

There was a great deal of pressure on both EPA and NJDEP to initiate permanent remedial measures at the contaminated properties. In May 1984, EPA and NJDEP planned a pilot study to test the feasibility of excavation and off-site disposal of the radium contaminated soil. Twelve properties were selected for the pilot study and preliminary engineering assessments were prepared. In the fall of 1984, EPA decided to forgo the pilot study and proceed with a full Remedial Investigation/Feasibility Study (RI/FS). The sites were included on EPA's National Priorities List of Superfund Sites in a special listing in February 1985. NJDEP elected to proceed with the pilot program.

NJDEP began excavating soil in June 1985. After four properties in Glen Ridge had been completely remediated, the disposal permit was revoked. Work had begun on four different properties of the pilot study in the Town of Montclair. Excavation was halted immediately. The occupants of the homes remained in temporary housing while NJDEP attempted to secure a disposal site. A significant amount of negative publicity was generated by NJDEP's search for a disposal site and the images of drums of contaminated soil stored in the middle of a residential neighborhood. In the fall of 1987, NJDEP was able to remove the soil from Montclair. The pilot program demonstrated that excavation is technically feasible and effective. It also demonstrated that technical issues such as support of structures and building material decontamination were readily solvable. The major obstacle was finding and maintaining a permanent disposal site.

In 1985, concurrent with the pilot excavation program, the EPA completed the draft RI/FS. Because of the problems associated with the disposal of the radium-contaminated soil and the public's demands for a permanent solution to the problem, the EPA initiated a supplemental feasibility study in March 1987.

EPA Records of Decision

EPA issued a ROD for each site on June 30, 1989. The RODs called for total excavation of soil from the most severely contaminated properties. They also called for continued installation and maintenance of interim controls for radon gas and gamma radiation where public health standards were exceeded. EPA reserved a decision on the remaining contaminated properties pending further study. This remedy was expected to cost $53 million and require several years to complete.

Public comment was received until January 31, 1990, on the portions of the plan covering the properties not included in the first RODs. As a result of strong public

opinion and the recent identification of a viable disposal site, EPA issued the second set of RODs on June 1, 1990 calling for excavation and off-site disposal of contaminated material on each property where this material exceeded the cleanup criteria. The combined remedy for the sites was expected to cost over $250 million and require more than ten years to complete. EPA stated in the 1990 RODs that several factors could influence the implementation of the proposed remedy and the cleanup schedule. These factors included funding for the project and the long-term availability of a disposal site.

Overcoming Inertia

After nine years of study and unsuccessful attempts at remediation, the challenges to EPA, USACE and Bechtel were as follows:

- Regain public confidence
- Conduct detailed design investigations of 769 properties
- Address worst cases first
- Prepare individualized designs for each contaminated property
- Move quickly from study to actual remediation
- Minimize the disruption in the neighborhoods

In the spring of 1990, EPA, USACE and Bechtel began a two-phased program to jump-start the project. EPA and Bechtel began an intensive program of community relations in the three communities to inform homeowners of the overall plan for the remedial work and to obtain individual access to properties for design testing. As soon as access was received, Bechtel mobilized test crews to conduct the necessary investigations required to prepare remedial designs. By the end of the first year, investigations on almost 200 of the properties were completed. Within three years, design investigations were completed on over 740 properties.

By the fall of 1990, USACE had awarded a contract to Sevenson Environmental Services, Inc., (SES) for the removal of contaminated material from the four properties in Montclair that had remained vacant ever since the State was forced to abandon the pilot project. Due to the deteriorated condition of these properties that had occurred over five years, EPA determined that restoration would not be feasible. Therefore, the government purchased the homes and demolished them. Removal of the contaminated material on these properties was completed early in 1991. In order to maintain the momentum created by finally completing the long delayed remediation, Bechtel was able to quickly deliver final designs for cleanup of 5 more homes in the immediate area. USACE added the cleanup of these homes to the SES's scope of work.

From the fall of 1990 through the present, there has been a continuous process of design investigations, remedial actions, and final verifications underway at the Superfund sites. Remedial designs for over 200 individual properties have been completed. Four separate construction contracts have been awarded. Two of those contracts have been completed and over 100 homes have been remediated.

MEASUREMENT OF RADIOACTIVE CONTAMINATION

Early in the remedial process at these sites, EPA made a determination that it was important to provide homeowners with the most complete information possible regarding the extent of contamination on their property. For properties with extensive contamination, this information would be needed to determine whether relocation would be necessary. In other cases, the homeowner would need a sense of the inconvenience that would result from the cleanup of the property and the specifics of the restoration that would be required. Therefore, a comprehensive program for determining the extent of contamination was developed by EPA, USACE and Bechtel.

Gamma Walkovers

The first step in a property investigation is a complete indoor and outdoor gamma survey. The property is divided into 6-foot square grids. Technicians slowly move the gamma scintillator (TMA SPA-3 with 2x2 sodium iodide crystals) just over the surface in each grid. The highest and average number of counts for each grid is recorded. Inside the basement, a similar survey is conducted on a 3-foot grid. Gamma measurements are also recorded for the basement walls. Additionally, gamma exposure rates are taken using a pressurized ion chamber. The measurements are compared to normal background readings for the area. Since many building materials including some types of brick and belgian block have naturally elevated gamma radiation levels, the gamma readings may be above background in some areas. Although normal background for this area is between 8K and 10K counts per minute (cpm), a reading

above 20K cpm usually indicates that the property needs to be further investigated by taking a sample of the soil or building materials.

Exterior Radon Canisters

Recognizing that gamma radiation can be shielded by clean surface soil, Bechtel established the use of charcoal canisters to detect radon in the soil gases as a secondary diagnostic technique. Typically, twenty 3-inch charcoal canisters are distributed over the yard. They are covered with inverted 1-quart flower pots and left for 24 hours. The canisters are collected and each is analyzed for radon. Any locations with radon flux measurements exceeding 2 picocurie/square meter/second are earmarked for further soil sampling. This criterion was developed based on an analysis of flux measurements versus soil sample analyses. The use of this criterion has been 90% effective in detecting buried deposits of contaminated material.

Soil Sampling

Over the course of the project, a number of soil sampling techniques have been used. Initially, a drill rig with hollow-stem auger and split-spoons was used. The drill rig was difficult to maneuver and, at times, caused damage to lawns. Next, a portable tripod with a gasoline-powered cat-head and 140-lb. hammer driving split-spoons was employed. Setting up and moving the tripod and hammer proved to be time-consuming. It also was difficult to use this system in close proximity to houses.

A 4-inch, hand-turned, stainless steel soil auger became the method of choice. The auger is used to extract soil samples in 6-inch lift. The majority of the boreholes can be obtained using this technique. It also is advantageous while working around underground utilities, the locations of which may not be clearly known or marked. Boreholes are drilled to 5 feet or greater to insure that the vertical extent of contamination has been identified. After completion of drilling, radiological technicians perform a downhole gamma survey at 6-inch intervals. If the gamma readings are elevated at the bottom of the hole, or if they are increasing, the depth of the hole is increased. The lateral limits of the contaminated material are determined by evaluating the surface gamma readings and installing boundary boreholes having downhole gamma readings which are not elevated. For large areas of contamination, boundary holes are set at least ten feet from a contaminated borehole. In the case of a "hot spot", or small area of contamination, the boundary holes are pulled in closer to better define the extent of the contamination. Boreholes cannot fully delineate the limits of contamination on the site because the contaminated material is not homogeneous and was deposited randomly. Therefore, a borehole drilled six inches from a contaminated area may be clean. Boreholes are the best means of getting a good sense of the extent of contamination while limiting the inconvenience to the homeowner and damage to the property.

To compensate for the limits inherent in using boreholes, particular attention is paid to structures, driveways, large trees and other items of concern to homeowners. This decreases the likelihood that an item which is not scheduled to be affected by the remediation will have to be removed during the cleanup.

Field Screening of Samples

In order to better select which soil samples should be sent for full radiological analysis, Bechtel's subcontractor, TMA/Eberline uses a quick count system for preliminary screening. Radionuclides are not uniformly distributed throughout the soil, therefore, it is difficult to be sure that a selected sample is representative of the area from which it is taken. Quick-count gamma measurements are taken on undried samples using a 3-inch by 3-inch sodium iodide crystal after 24 hours. This analysis is compared to the downhole readings taken at the sample depth to ensure that the sample is representative of the area and not contaminated by material falling into the hole during the drilling. This technique is used only as an indicator in determining which samples are sent for complete analysis.

Radiological Analyses

TMA/Eberline uses standard methods for most of the measurements for the radionuclides of concern and has developed or refined techniques for the remainder. Measurements for the radiological activity of radium-226, radium-228, thorium-232, potassium-40, uranium-234 and uranium-238 use a mixed-acid digestion followed by gamma spectroscopy for detection.

Thorium-230 is not a gamma emitter, therefore, a different technique is required. TMA/Eberline has developed the PERALS (Photoelectron Rejective Alpha Liquid Scintillation) method for determining the activity of thorium-230. This procedure

calls for the selective separation of the thorium ions after acid digestion. Detection is then accomplished using alpha spectroscopy.
Measurements for both thorium ions are important for two reasons. First, the majority of the radiological measurements used on the project rely on detection of gamma radiation. Thorium-230 is not a gamma emitter. Second, measurement of thorium-230 is needed to determine the degree of radiological equilibrium of the contaminants in the soil.

Alpha Track Detectors

The final analytical step for properties that have been remediated, and for properties which do not exceed cleanup guidelines, is a 1-year radon study. Bechtel uses track etch devices for long-term radon detection. During the radiological decay of radon, alpha particles are released. The particles etch a track on a small mylar strip inside the device. These tracks are counted by microscope and can be correlated to the concentration of indoor radon. Three or four devices are installed in the basements and analyzed as follows:

- After 2 months exposure
- After 4 months exposure
- After 1 year exposure
- 10% duplicate analysis for quality control

DEVELOPMENT OF PLANS AND SPECIFICATIONS

Remedial design for a property included preparation of an excavation plan, restoration plan, sections, landscape schedule and subsurface borehole profiles. The excavation plan based on the surface walkovers and the boreholes showed areas of known radioactive materials exceeding cleanup standards. Excavation was designed to be conducted in two stages as follows:

- stage 1 - defined as primary excavation
- stage 2 - secondary excavation

The primary cut lines directed the remedial action contractor (RAC) to excavate where contamination boundaries are clearly defined. After primary excavation, the RAC was directed to conduct further gamma scans and to continue excavating until all contamination had been removed. The secondary excavation cut lines represented the anticipated limits of material removal based on "clean" boundary holes identified during the field investigations.
Cross sections through areas of excavation showed the depth of excavation in relation to the existing ground line. By design, all material removed by primary excavation was contaminated. The secondary excavation line was established to show a contractor the anticipated excavation breadth and depth. The contractor was directed to continue excavating beyond secondary cut lines if there was still evidence of contamination.
A restoration plan was developed for each property disturbed by excavation. Restoration of a property required the replacement of features such as demolished structures (garages), driveways, sidewalks, curbs, lawns, trees, shrubs, and all other disturbed features. The restoration plans required a RAC to restore the property to preconstruction conditions. In virtually all cases, the quality of the restored areas exceeded the preconstruction conditions.
With all of the above concurrent activity occurring, frequent coordination calls were made between Bechtel's Field Engineers, home office engineers, and USACE and EPA personnel. Weekly conference calls were conducted to review progress, identify potential problems, and reach decisions. Minutes of each conference call were prepared and distributed to each participant to assure that action items and relevant decisions are documented.

PACKAGING OF REMEDIAL ACTION CONTRACTS

The preparation of remedial designs was the area of greatest USACE, EPA, and Bechtel interaction. Of the 769 properties, approximately 250 have had or will require remedial action. It is not practical nor was it possible to remediate all properties simultaneously. Not all of the homeowners responded to requests for access in a timely manner. Some properties presented the far greater health risks. Finally, it was necessary politically to initiate some action in each of the communities within the first year. This resulted in a phasing of both the design and the remedial actions for specific areas or neighborhoods within the sites.
A major objective of the design plan was to minimize the impacts of remedial actions on individual residents and communities by disrupting a neighborhood only one time. In order to accomplish this, early emphasis on the investigations and designs

centered first on the worst case properties. Ideally, work would then move outward towards areas of lesser contamination. To a large degree, the phasing of work has accomplished this objective. Sporadic response from some homeowners and proverbial hardcore holdouts prevented complete designs of entire blocks.

Within individual phases, clusters of adjoining homes were selected for simultaneous remediation. The careful clustering and sequencing of work activities provided the following:

- The cluster could be isolated for safety and health considerations
- Heavy equipment could push contamination across property boundaries within the cluster
- A centralized bagging and loading point could be established
- A sense of community could be maintained (neighbors would be out of their homes and return in the same time frame)

In order to also maintain the momentum that had been established, construction packages of 25 to 50 homes were prepared as soon as test results were available.

CONTINUOUS IMPROVEMENTS IN THE PROCESS AND THE DESIGNS

Design Review

In developing the design, Bechtel initially prepared design packages at the 35, 65, 95, and 100% levels for review and approval by USACE. Design reviewers provided comments for each stage of review. Initially, these review comments were provided by letter.

Continuous improvements by both USACE and Bechtel along with the "partnering" that developed lead to substantial time savings that helped to accelerate the entire project. These improvements are as follows:

- Use of the Automated Review Management System (ARMS)
- Elimination of the 35 and 65% review cycles
- Flexibility in design submissions

Use of ARMS provided USACE a centralized and uniform mechanism of gathering comments on each design package and then transferring those comments electronically to Bechtel. ARMS also provided the mechanism to ensure that those comments were addressed.

After the first two design packages were completed, USACE felt comfortable enough with Bechtel's design performance that it was decided to progress to the 95% level as the initial review. Most of the designs were similar from property to property and the specifications required few changes.

A continuous sequencing of design reviews also allowed great flexibility in expanding or contracting individual phases of remediation. Many times, recalcitrant homeowners would be turned around when work began in their immediate area. As soon as they granted access for design, Bechtel would quickly complete testwork and prepare designs as necessary. Again based on USACE's confidence in Bechtel's designs, those properties could be added to a construction package ready for bid or in some cases added as a contract modification to an existing construction contract.

The design process was also dynamic. Since the design crews were working in one area while the RAC was remediating in another, lessons learned from the construction could be incorporated as design improvements to successive phases and clusters within phases. Many of these lessons involved discovering areas of contamination not apparent from initial test procedures. Later designs also provided greater flexibility to handle unforeseen circumstances such a radium contamination imbedded in the concrete footings and mortar mixes.

Data Management and Transmittal to Homeowners

A significant amount of data is generated during each property survey. This data is used by EPA and Bechtel to determine whether a property must be remediated and, if so, the extent of the contamination on the property. This data must also be available to anyone interfacing with a resident who may have questions regarding the status of their property.

EPA must also transmit data to the homeowner in an easily understandable format and in a timely manner. The most frequent complaints EPA received from the community during the early stages of the design process was in reference to the length of time between the data that a property was surveyed and the date that the homeowner received information regarding the status of the property. Part of the delay was a result of the time necessary to perform the radiological analysis on soil samples. In addition, Quality Assurance and Quality Control procedures required review by Bechtel's Oak Ridge office before data was released. Further complicating the matter

was keeping track of the following:
- Which of the 769 homeowners had received reports
- The status of test work on the property
- The results of the analyses

Bechtel developed a computerized database that is shared among the field site, the Oak Ridge Office and EPA's Region II office. The database contains the following types of data:
- Owner information
- Status of field testing
- Status of lab analysis
- Results of field testing
- Results of lab analysis
- Record of communications and reports to homeowners
- Schedule for current and future work on a property

Standard property reports are routinely generated from the database as soon as work is complete. In addition, the database provides a complete look at the status of work on a property.

Alternate Trench Method

The most significant concern expressed by homeowners whose homes required basement excavation was the potential damage to the structure of their home as a result of the cleanup. The classic approach to remediation under foundations and basements calls for underpinning or cribbing of the house to allow for complete removal of contamination. Once cribbing is in place, the entire foundation is demolished and any contaminated material removed. This is effective, but there are some risks. The lifting and cribbing stress the structure of the house and the process is time-consuming and costly. In addition to these risks, a number of the home in Glen Ridge are part of a National Historical District. The structure of these homes could not be damaged or altered.

Bechtel engineers suggested an alternative approach which called for the footing to be removed in altering 3-foot sections. After contaminated materials are removed, the void space is filled with concrete and the next section is removed. The house never moves and the risk of damage is greatly reduced.

ACCOMPLISHMENTS AND LESSONS LEARNED

The Montclair, West Orange, and Glen Ridge Superfund Sites have been very challenging sites to remediate. Dedicated personnel working for the government and its contractors have all contributed to the success of this project. The initial effort was to overcome the fears of the citizens living in the superfund sites. The government had to assure the citizens that we were capable of remediating their properties with minimal disruption to their lives. Previous removal action was halted by not having a disposal facility available to accept the large volumes of radium-contaminated soils.

In February 1988, this problem was alleviated when Envirocare of Utah obtained a license to dispose of Naturally Occurring Radioactive Material (NORM) from the Utah Department of Health. This facility was contracted to receive and dispose of the material from the site. All of the remaining material from the Pilot Home Project was transported to the Envirocare facility, thus ending the five year interruption in the completion of the initial project.

The coordination of the EPA and Army Corps has been considerable. EPA, being the lead agency, has had the responsibility for outreach and community relations. The EPA representatives on site are available to answer questions and concerns of the citizens. Property owners feel more secure when they see a familiar face. It was quickly realized that individual meetings accomplish more than large group meetings, especially when dealing with individual property owners.

The Army Corps' design contract with Bechtel National includes field design investigations and remedial designs. With the extensive field investigation being performed to characterize the sites, Bechtel has had personnel at the site for the past four years. These people are very helpful in explaining the technical issues to the property owners. During the individual meetings, EPA requests assistance from the field personnel to explain technical issues of the remediation. The availability of the site personnel has improved the government's responsiveness to the citizens.

Design packages were prepared using the following guidelines:
- Remediate the most contaminated properties first
- Minimize the disruption to an individual location

With the effort to have all the Category I properties investigated first and proceed on through the Category II, III, IV, and V's, property clusters were developed. These clusters were placed into biddable packages to be awarded over the course of the design. Concerned that the number of available rental properties in the area might be insufficient, the government developed remedial packages that were sensitive to the relocation issue. Award of the remedial action contracts were staggered to not have too many people out of their homes at one time. This paradigm was repeated to show continued remediation progress in the field.
The project has incorporated the team approach to remediation. All government agencies and contractors are kept well informed as to the status of activities at the sites. Meetings are formally held at least once a week to discuss site conditions.
Property owners are presented with the results of the tests as soon as possible after the testing is completed. This keeps the owners updated regarding any changes in the cleanup plan for their properties.
Each property owner's concerns are unique. The government representatives are well prepared to listen to their concerns before determining the best course of action.
The community, including local officials, citizen groups, and residents not directly affected by the cleanup are periodically updated on the status of the project. The updates are initiated at city council meetings, press releases, and via mailings.
The local officials are involved in the logistics of the cleanup on all levels including traffic control, utility work, tree selection, and building permits.
The remediation of these properties has been turned from a disaster into a success by the coordinated efforts of the EPA, USACE, and their contractors. Expectations are that the high performance efforts of all concerned will conclude this project ahead of schedule and as much as $100 million below the original $250 million estimate.

Session 48 -- USDOE Contract Reform Effort (Panel)
Co-chair: Gary Benda, US Energy;
Critz George, NUS/Brown & Root
Panel discussion not submitted.

Session 49 -- Public & Institutional Interactions
Co-chairs: Jo-Ann Holst, S.M. Stoller Corporation;
Ron Bhada, NMSU
49-1
COMMUNICATIONS IN THE FIELD OF RADIOACTIVE
WASTE MANAGEMENT IN SPAIN
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ABSTRACT
The generation of electrical energy by nuclear means began in Spain in 1968. There are currently nine nuclear groups producing approximately 38% of the country's total electricity. In addition, there are a further 1,300 installations at which radioisotopes are used for medical, industrial or research purposes and which, like the nuclear power plants, produce radioactive wastes.
In 1984, the Spanish Government set up the Empresa Nacional de Residuos Radiactivos (ENRESA) with a view to reorganizing radioactive waste management and centralizing this activity within the framework of a public company.
One of ENRESA's main tasks is to inform society of the activities it performs, since as is the case in most countries public opinion is highly sensitive to any issue relating to radioactivity, and especially to radioactive wastes. The main reason for such attitudes is a lack of knowledge of the activities and practices performed, this in turn being due mainly to a lack of information on the subject and, in certain cases, to inadequate use of the information available by certain politicians, environmentalist groups and media.
From the very beginning, ENRESA considered that there was a need for its information actions to be ordered within a Communications Plan defining continuous, long-term

interactions between the company and the public at large. This Plan, which is being applied in a series of different stages at local and national level, works well in the areas surrounding a facility for the definitive disposal of low and intermediate level wastes (El Cabril) and a disused uranium mill (AUM) which has been decommissioned and closed.
The Plan is oriented towards achieving understanding and rationalization of the tasks performed by ENRESA, establishing courses of action and projects serving to enhance conditions in the areas hosting the company's installations and to dispel whatever fantasies might exist among the population in relation to radioactive wastes and their handling.
The main activities performed in Spain in order to provide information on radioactive waste management are described in the paper, as are the most important informative efforts to be made in the future with a view to favoring the implementation of a high level waste repository. Special attention is paid to the need for widespread agreement between the main political forces, performance of an information campaign at national level and the establishment of research and demonstration centers.
RADIOACTIVE WASTE MANAGEMENT IN SPAIN
In Spain, 93% of radioactive wastes are generated at nuclear power plants. The country has nine nuclear groups at seven sites (Fig. 1) with a joint installed power level of 7,405 MWe; in 1994 these groups generated 55,314 GWh, 38.5% of the country's total electricity production (Table I). There is also a group which is now shut down, Vandells I, where dismantling operations will soon begin.
Fig. 1.
TABLE I
The nuclear program started in 1968 with start-up of the Zorita Nuclear Power Plant, and subsequently developed in three stages. Between 1968 and 1974, 1,130 MW were installed at three plants using PWR, BWR and graphite-gas technologies respectively. Between 1982 and 1985 a further five groups amounting to 4,695 MWe came on line, and in 1988 two third-generation groups with a capacity of 2,000 MWe were connected to the grid. In 1984, the Government decided to freeze the construction of 5 nuclear groups whose design and development had been initiated. The 1991 revision of the National Energy Plan ratified this decision, and the construction of new nuclear plants was put off until such time as the country's energy demand were to require them, and in no case earlier than the year 2000.
Radioactive wastes are also produced at 1,300 installations using radioisotopes for medical, industrial and research purposes.
Management of these wastes is the responsibility of the Empresa Nacional de Residuos Radiactivos (ENRESA), which carries out its work in accordance with the strategies defined in the General Radioactive Waste Plan, a document issued by the Government and periodically updated. The latest estimates on waste production consider that the current installations will produce 191,700 m3 of low and intermediate level wastes and 9,020 m3 of high level wastes in the form of spent fuel (Table II).
TABLE II
The 4th Plan, approved in 1994, defines the following activities for management of these wastes:
i. Low and intermediate level wastes
These are disposed of at El Cabril, a shallow underground facility with engineered barriers using cement as a waste immobilization matrix. The authorized capacity of this facility is for 35,000 m3 of conditioned wastes, which will allow for disposal of all the wastes produced in the country from now until the beginning of the next century.
ii. High level wastes
The Spanish strategy for the definitive disposal of high level wastes consists of emplacing them in geological formations of granite, clay or salt at great depth. In 1987, work began on determining the best suited locations for construction of a repository. It is expected that by the year 2000 sufficient information will be available for a proposal to be made to the Government regarding a site for a disposal facility.
Meanwhile, the spent fuel will be stored at the plants. In the future, as the plant pools fill up with wastes, the fuel will be stored on site in dry containers for subsequent removal to a centralized store.
iii.Dismantling

In July 1993, ENRESA concluded dismantling and decommissioning works at a disused uranium mill at Andjar (AUM) in the province of Jan. The design and performance of these works, based on projects used by the US Department of Energy (DOE) for the Uranium Milling Tailing Remedial Action (UMTRA) project, were accomplished with technical assistance from the JACOBS ENGINEERING GROUP. As regards nuclear power plants, the Spanish Plan contemplates dismantling to level on the scale defined by the International Atomic Energy Agency (IAEA) at the end of their service lifetime. ENRESA is currently working on the tasks to be completed prior to dismantling of the Vandells I plant, which was closed by Government decision in 1990.

iv.Research and development

Research and development is carried out mainly in order to increase knowledge of the technical and operational needs of the definitive high level waste disposal facility. The costs foreseen in this area, up to implementation of the facility half way through the next century, amount to some 500 million dollars.

PUBLIC PERCEPTION OF RADIOACTIVE WASTES

The various sociological surveys performed show that the attitudes and opinions of the Spanish population towards nuclear energy are mainly negative. In simplified terms, between 30 and 35% of Spaniards are radically opposed to this energy form, while those decidedly in favor amount to between 4 and 6%. Gaining insight into the underlying reasons for such attitudes is a complex problem and is addressed in Spain and in other countries from various perspectives. In general these are not clearly defined, although it is thought that in all cases three factors are implicitly involved:

Growing sensitivity in relation to protection of the environment and an ecological awareness that is becoming a majority social and cultural trend.

A lack of information regarding the nuclear issue. In spite of the interest that most people claim to have in this question, only 6% consider themselves to be sufficiently informed.

The problems posed by the nuclear installations of the Eastern European countries and the extension of these problems in the media to the rest of the world's nuclear installations.

The most important of these factors is the ecological awareness that constitutes the underlying substrate of values and beliefs supporting mainly negative attitudes towards, and opinions of nuclear issues. By way of an example, Table III shows the evolution of this trend in the opinion of Spanish people, when asked about their perceptions of the radiation risk involved in different options.

TABLE III

In Spain, radioactive wastes are considered to constitute an important drawback as regards the use of nuclear energy. In recent years, anti-nuclear groups have adopted a strategy aimed at hindering waste management actions, with the ultimate objective of forcing the closure of nuclear plants. The effect on public opinion of the Chernobyl accident has diluted with time and has not been refired by any other type of nuclear accident. Consequently, wastes have become the main source of conflict in the nuclear issue, since they are perceived as being a by-product having a major, long-lasting potential for pollution.

Radioactive wastes are considered to constitute a greater threat than nuclear energy itself, and meet with non-rational opposition regarding both their existence and the possibility of their being controlled at a disposal facility. Table IV shows the results of a survey carried out by ENRESA among several urban and rural populations, some close to waste management centers. In all cases it may be appreciated that attitudes contrary to wastes and to the location of disposal facilities exceed the negative attitudes shown towards nuclear energy. Demands for information are evident in the positive change in such attitudes that occurs when the possibility of somebody undertaking management of this particular issue is posed.

TABLE IV

CHARACTERISTICS OF THE AREAS SURROUNDING THE EL CABRIL AND AUM SITES

The El Cabril disposal facility is located in the Sierra de Albarrana in the northern part of the municipal area of Hornachuelos, in the province of Crdoba (Fig. 2). The repository is to be found in the foothills of the Sierra Morena, a mountainous area with mild winters, hot summers and scarce and irregular rainfall.

Fig. 2.

The populations of the area are small and widely disseminated, and their basic resources are agriculture, forestry and hunting (Table V).

TABLE V

The Andjar uranium mill (AUM) is located in the municipal area of the same name in the western part of the province of Jan, close to its border with Crdoba. This is the largest municipal area in the province, with an area of 957 km2 and a population of 37,000, of which approximately half live in the city of Andjar, 8 km from the site.

The city's most important economic sector is services, which represents almost half of the economy of Andjar. The industrial sector is of little importance and centers on the foodstuff and textile industries.

Until 1986, the El Cabril installations belonged to the Centre for Energy, Environmental and Technological Research (CIEMAT), which in 1961 began to store wastes in a disused uranium mine on the site and, at the beginning of the 1980's extended this practice to three buildings on the surface. When ENRESA took over this centre, the attitude of the local population was one of open rejection, since little information was available on the facility and what there was was charged with military implications or with exaggeration regarding the risks of the wastes.

In Andjar, little was known about the facility and its industrial process was not associated with the risks attributed to other nuclear cycle installations.

THE ENRESA COMMUNICATIONS AND PUBLIC INFORMATION PLAN

The Communications Plan was drawn up with the objective of informing the different sectors of the Spanish population of ENRESA's activities, such that there be knowledge of these activities and, to the extent possible, acceptance of them.

The Communications Plan was initiated shortly after approval of the First General Radioactive Waste Plan in 1986, with the activities organized in such a way that the main efforts were aimed at informing the different social sectors surrounding the El Cabril and Andjar sites, where ENRESA was to carry out specific tasks in the medium term. Complementary to the above, and with a view to facilitating longer-term tasks, the Communications Plan includes actions aimed at improving the attitudes of Spanish society towards the implementation of radioactive waste management centers.

From this perspective, the Communications Plan is articulated at local level around three objectives:

Informing all local community groups of the work to be performed, taking into account their social, economic and cultural characteristics.

Providing the authorities and opinion leaders of the areas in question with detailed knowledge of the activities involved in radioactive waste management.

Integration and interaction with the social agents in the areas in question and with the population in general.

The main actions carried out within the framework of the first of these objectives have been as follows:

Open day policy and explanation of the installations. Information Centers. This is considered to be important for modification of the opinions previously held by the population as a result of the negative connotations attributed to the nuclear issue, and specifically the secrecy surrounding it. Since 1986, when ENRESA took over the El Cabril facility, an information department manned by local people has operated at the site, the objectives being to promote knowledge of the installation and answer "on the spot" whatever questions might be posed. Similarly, a specific program of visits to the AUM was initiated in 1990. At both sites, the main target publics for visits are associations and other groups from the surrounding area capable of creating opinion regarding wastes. In 1990 an Information Center was built at El Cabril to improve the efficiency of the visits and receives 5,000 visitors every year. In the case of the AUM, the works have been explained to some 3,000 people.

A third Information Center was opened at the beginning of 1992 at the ENRESA head offices in Madrid, where information is provided on radioactive waste management.

Instruction on the basic principles of radioactivity and waste management. This has been developed as a specific part of the Communications Plan with a view to bringing about a reasonable level of knowledge of the elementary principles of waste management and the handling of radioactive materials among opinion leaders, fundamentally primary and secondary school teachers, post-graduates, students and members of local Corporations. Since October 1989, ENRESA has been organizing seminars on radioactivity, radioactive wastes and their management, in collaboration with specialists in different radioactivity-related applications.

These seminars include explanations of both the technical and scientific aspects of

waste management and of other social and economic matters referring to waste disposal and affecting the local area (Table VI).
TABLE VI
Meetings with political representatives and other opinion leaders (journalists, economic leaders, etc...).

This activity is carried out by ENRESA through three different types of meetings:
a. Periodically, and in addition to the controls established by legislation, representatives of ENRESA appear before the respective Commissions of Industry of the Congress and Senate of the National Parliament. Meetings are also held periodically with the members of Parliament and opinion leaders most closely linked to radioactive waste management, especially those representing Crdoba or Jan in the national or regional Parliaments.
b.The supply of information and contacts with local leaders are continuous efforts. In addition, there is an official meeting once a year with the Corporations of the area of influence.
c.Visits with political leaders and representatives of the media to radioactive waste management centers as a way of learning of the characteristics, design and operation of other sites and of increasing the availability of information regarding common management aspects.

Socio-economic cooperation with neighboring communities. In keeping with the needs expressed by local political leaders, attempts have been made to take advantage of ENRESA's operating requirements to promote improvement of the social and economic conditions of the areas in question. The main actions implemented with a view to achieving this objective are as follows:

a. Drawing up of a study of the socio-economic situation of the areas, with detailed information on local companies and of their capacity to act as contractors during the construction and operation of the installations.
b.Cooperation with the local political authorities in order to learn of their opinions regarding the situation of the areas and to be able to optimize the resources generated by ENRESA.
c. Training programs for potential site workers.
d. Preferential use of local resources.

By way of an example of the above, Table VII indicates the extent of contracting of local personnel, services and goods during the extension works carried out at El Cabril.
TABLE VII
Parallel to the above, in December 1989 the Ministry of Industry and Energy issued an Order which provided a certain amount of money for communities living in areas housing waste disposal facilities, this amount depending on the category and volume of the wastes in question.
Communication support resources
The different types of information described above are supported by publications, brochures and videos dealing with specific aspects of ENRESA's activities, these being adapted in each case to the target audience and to the environment in which the information is provided (information centers, round tables, debates, etc...).
Since 1986, ENRESA has been publishing the quarterly magazine ESTRATOS, a 64-page publication with a circulation of 4,500 copies which are mainly distributed among national opinion leaders. ESTRATOS is designed as a scientific and environmental journal dealing with questions related to radioactive waste management through articles on the principles and practices involved, and containing a fixed section on the situation of ENRESA's activities and projects.
The two-monthly SIERRA ALBARRANA is published in the El Cabril area; this is a local bulletin containing news on the facility and its surrounding area and has a circulation of 3,000 copies.
RESULTS
From the very date of announcement of the extension to be made to the El Cabril

facility, the local populations openly opposed the construction works, while at Andjar there was general agreement in relation to the project presented, albeit with some concern regarding the radiological risks possibly involved in moving the tailings. In Andalusia in general, and in the area surrounding El Cabril in particular, neither the population nor the opinion leaders would allow any action scheduled by ENRESA. In 1988 there was a demonstration in Hornachuelos attended by some 2,000 people. At national level, the politicians, journalists, etc... did not express overmuch confidence in the activities performed by ENRESA and the information published in the newspapers was for the most part negative. The Communications Plan was applied from the start to ENRESA activities and after a while public perception began to improve.
The current situation may be summarized as follows:

i. At regional level, the public living in the areas surrounding El Cabril and Andjar understand ENRESA's activities and believe the tasks performed to be safe.
   Most of the opinion leaders, such as politicians, school teachers, etc., have received information directly from ENRESA and have the impression that the work performed has been accomplished in a professional manner and to a high level of quality. In the case of the journalists, this conclusion is borne out by the fact that 60% of the information they publish is contrasted with an ENRESA spokesman.
ii. At national level, a fluid communications flow has been achieved with journalists and political representatives. This does not mean to say that they support whatever work is to be performed, indeed, the opposition to news regarding the location of a high level waste repository is always very severe. On the other hand, it is now very rare to find articles or news items not based on objective facts.

Attention should be brought to the support provided by the authorities of the Central Administration, and to the fact that this has led to considerable improvement as regards the efficiency of communications, as may be appreciated by comparing the results obtained with the difficulties experienced by other projects presenting similar social problems (major works of infrastructure, prisons, etc.).
THE NEED FOR INFORMATION IN RELATION TO THE MANAGEMENT OF HIGH LEVEL WASTES
ENRESA plans to initiate the characterization and construction of a high level waste repository during the second or third decade of the coming century. Meanwhile, the main tasks are oriented towards continuing research and development work and site selection studies for this facility.
In Spain, neither public nor political opinion is willing to accept the location of the repository, as is demonstrated by the fact that any geological research carried out in the field is automatically associated with ENRESA site selection plans, and soon becomes a source of strong local opposition.
The selection of a site for a high level waste repository requires a communications policy taking into account the following:

i. At political level, it is necessary to continue to stress the need for waste management and to obtain nationwide political consensus with a view to legitimizing and providing coverage for the solution adopted.
ii. A nationwide information campaign must be carried out emphasizing the benefits of an adequate radioactive waste management policy. This campaign would serve to fill in the information gaps currently detected among the members of the public and to discuss the solutions proposed in greater depth. In addition, the waste problem would be put into context, the scope of communications actions would be extended to include the most efficient media and the level of communication with society in general would be increased.
iii. The research and demonstration installations are centers at which public opinion might gain direct insight into the characteristics of the definitive management of high level wastes and the levels of quality applied in this process.
All the issues mentioned above in relation to better information for the public through well informed opinion leaders, through visits to information centers, through the availability of teaching materials for school teachers and students and through the existence of specific centers at which waste management tasks may be observed and learned of will help to facilitate decisions regarding the location of a high level waste disposal facility.

The strategy described above is backed by the experience acquired and by the results obtained locally by ENRESA at El Cabril and Andjar, where a high level of public information and appropriate understanding and consensus with the local opinion leaders and authorities have been achieved.

49-2
OPINION RESEARCH METHODOLOGY APPLIED BY NIREX TO KEY AUDIENCES IN THE UK'S DEEP DISPOSAL PROGRAM
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ABSTRACT
Opinion research is an invaluable tool for tracking a company's position with its key audiences. However, different audiences may warrant different approaches for undertaking this research. Nirex, the UK national radioactive waste disposal company has adopted specific methods for monitoring the views of its key audiences.
Telephone interviewing is used for researching the opinions of the public of Cumbria, north-west England, where Nirex is investigating a potential deep repository site. Here the public's views are sought, rather than those of the decision makers as, in this case, it would be seen as inappropriate to attempt to interview directly members of the Cumbrian local government; in December 1994 they refused the Company's application for permission to construct an underground rock characterization facility.
For assessing the views of Members of Parliament (MPs), Nirex has chosen to participate in an existing scheme, in which a panel of 100 MPs are regularly sampled through postal questionnaires. The method also allows comparisons to be made with the rest of the nuclear industry and the "opposition". MPs and other Parliamentarians are a key audience for Nirex, particularly as the Government is currently reviewing its radioactive waste management policy. The results of the latest research indicate that Nirex has to do more to get its message across.
Face-to-face interviewing is the preferred method of research for journalists, who are seen as key influencers of other major audiences. The most recent survey was of the energy and environment journalists of the media. Twenty-five per cent. thought Nirex was already "causing damage to the environment" and, somewhat disturbingly, almost all find environmental organizations the most useful source of information on Nirex. More targeting of this group is therefore seen as a priority.
Nirex has adopted specific methods of gauging the perceptions of its key audiences through adopting appropriate methods of opinion research most applicable to each group. This is paying off and giving the Company a steer as to where it should be targeting its communication effort.
INTRODUCTION
Background
Nirex is the company, set up in 1982, financed by the UK's nuclear industry, responsible for developing new disposal facilities in line with Government policy. Nirex initially proceeded to look for a near-surface site for the disposal of short-lived wastes and a deep site for long-lived ILW disposal. However, in 1987, it was decided on grounds of public perception that all ILW should go deep, coupled with LLW for economic reasons.
In 1987 Nirex started a site selection process which eventually yielded 12 sites, of which two, Sellafield in Cumbria, north-west England, and Dounreay in Scotland were selected to be studied first. In 1991 Nirex decided to concentrate its efforts on Sellafield because of its transport advantages: 60% of the waste requiring deep disposal arises there. Detailed geological investigations have been carried out at Sellafield and the next step is to construct a 650m deep Rock Characterization Facility (RCF) to assist in deciding the suitability of the site. However, in December 1994, Cumbria County Council (the local government body) refused to give permission for the RCF, thereby forcing Nirex to appeal to Government which will, in all likelihood, lead to a public inquiry.
Interaction With Whom?
To help identify and better understand the needs of its key audiences, Nirex carried out a stakeholder analysis in 1993. This confirmed initial thoughts that the following three groups were the most important:
- the people of Cumbria and their local government;

national government; and,
Nirex's shareholders.

Having identified the principal audiences it is important to determine the level of support and seek ways to maintain that support through an effective communications campaign. Monitoring of opinion is done through a mixture of Nirex's own research and that commissioned from independents, particularly when anonymity is required. The results set baselines from which progress and success of specific communications programs can be judged and subsequently adjusted. This paper compares and contrasts the approaches taken with Cumbrian audiences, with national government and with the environmental/energy journalists of the UK press, a group having significant influence in the minds of the former audiences.

LOCAL GOVERNMENT

Cumbria County Council is the major potential force within the County, although the role of the smaller Copeland Borough Council has also to be respected - certain "zoning" decisions fall within its responsibility and its views carry weight with the County and National Government. Cumbria County Council is made up of 83 representatives, 47% are Labour, 34% are Conservative, 17% are Liberal Democrat and 2% are Independent. There is therefore no overall political majority, but it must be noted that the Council's attitude towards the Nirex project is not necessarily defined by party politics.

Since 1991 Nirex has established formal links with the Cumbria County Council which allow frank and open exchanges to take place. Nirex has also taken the step of establishing its main press office locally rather than at its Harwell headquarters some 300 miles away. The local office also supports other public relations activities such as exhibitions, local newsletters and facility visits.

The most important communications issue currently facing Nirex is the increasing public awareness of the organization, without a corresponding rise in understanding of what it does.

Opinion Research Methodology

Within Cumbria, Nirex conducts its own opinion polls, but also has access to those undertaken by both Cumbria County Council and Copeland Borough Council. Here, a gauge of public opinion is important as both local councils regard it as an important driver of their attitude; moreover, direct interviewing of Councillors would be seen as inappropriate.

In 1994, Nirex assessed attitudes and awareness in Cumbria to the disposal of radioactive waste, with particular reference to knowledge of and opinions about the proposed RCF, with results broken down by district, so that attitudes of the population nearest to Sellafield could be compared with those in other parts of the county.

Nirex asked Gallup to carry out a telephone interview poll. This methodology was adopted due to the dispersed spread of much of the population of Cumbria (which is England's second largest county by area). The questionnaire was deliberately kept short so that it would be easy to administer by telephone. When the results were analyzed, the data was weighted to account for any shortfall in the proportion of people in the social grades D and E in the sample, as availability of telephones among this group can be low.

Cumbria was divided into political constituencies for sampling purposes to ensure a representative spread of interviews proportional to population size across all areas of the county. However, the views of people living in the Copeland district are of particular interest to Nirex, as the RCF investigation site, most of the deep boreholes and BNFL's Sellafield site are located within its boundaries. The sample for this constituency was therefore boosted to achieve 150 interviews.

At the analysis stage, the results were weighted by population and size of constituency. Telephone numbers of households were listed from the telephone directories for Cumbria on an "nth" basis. Postcodes (analogous to Zipcodes, but more precise) were used to identify constituency boundaries. The survey was conducted using Gallup's telephone unit based in Oxfordshire. A total of 755 telephone interviews were conducted among adults (aged 15 plus) resident in Cumbria between 11 and 20 October 1994.

Results and Discussion

By way of background, at the end of 1993, 88% of Cumbria residents as a whole were aware of Nirex. Yet, despite this very high public awareness, 40% of residents did not know Nirex's purpose (even when prompted). In fact, 50% of respondents believed

a decision that the repository would go ahead had already been taken.
The 1994 Gallup results for the public's attitude to siting a deep repository at Sellafield are shown in Table I against a similar, but not identical, questionnaire set in 1992 and carried out by Copeland Borough Council.
TABLE I
The support is encouraging, as those who "tend to oppose" might accept if reassured on factors such as safety, but the results leave no room for complacency. Nirex has historically tended to concentrate its communication efforts in the Copeland area and this can be seen in the detailed results.
When asked about the RCF, there was much misunderstanding. Although to be built purely as a research facility, many people believe that it is an actual repository, as Table II shows:
TABLE II
Note that in the December 1994 decision by Cumbria County Council to refuse Nirex permission to build the RCF, of those who attended the meeting (74 out of 83), 54% voted in favor of refusal, 45% against refusal and 1 abstention; a motion to allow Nirex full permission was defeated by 71 votes to 3. An important theme in the debate was that whilst there was little "environmental impact" reason for refusal, there is a belief in Cumbria, despite the efforts of Nirex, that Sellafield is a foregone conclusion and the RCF is a "Trojan Horse" for a repository. The public and the local politicians want to be reassured through the public inquiry process before they will be ready to accept the RCF and deep repository.
NATIONAL GOVERNMENT
One of government's roles in the UK is to set the safety targets that Nirex must meet. Additionally, government accepted in 1989 the broad shape of the disposal solution to deliver that target - a deep facility accessed from land. In May 1994, the Government began a review (the first for over 10 years) of its radwaste management policy and related matters (including site selection principles). The outcome of this review, which involved extensive public consultation, is expected in the first half of 1995.
Of influence to the policy setters are the public's elected representatives, the Members of Parliament (MPs), members of Parliamentary Select Committees and the policy advisory bodies. Of particular relevance to Nirex under the last heading are the Government's independent Radioactive Waste Management Advisory Committee and Advisory Committee on the Safety of Nuclear Installations. Also of importance for the longer term are the European Commission and European Parliament.
Opinion Research Methodology
Nirex is particularly interested in monitoring the attitudes of the Members of Parliament and a company which maintains a Political Opinion Panel (POP) of 100 Members of Parliament were chosen to provide this service. They sample the POP regularly (every four to six weeks during the Parliamentary session) to gauge the views of politicians on a number of issues. From Nirex's point of view, it was advantageous to use an existing mechanism of opinion research rather than to instigate something new and different.
The POP is an established means of anonymously testing Parliamentary opinion towards organizations. It enables perceptions of a company to be compared with other companies or organizations, and examines Parliamentary attitudes towards issues affecting the company's business. It is also useful to test the effectiveness of different arguments in support of a company's views before putting them to Members of Parliament, and helps find the most effective means of communicating those arguments to them, establishing benchmarks and thereby identifying trends.
The panel is selected to represent the balance of the political parties in the House of Commons: Conservative, 52%; Labour 42%; and 6% Others. The panel is also selected to reflect the regional balance within the Commons, length of service, size of majority and age. Six Nirex questions were put into the postal survey in early 1994, and care was taken to ensure optimal positioning of the questions in order to safeguard unbiased responses and eliminate any "order effect".
Results and Discussion
Some of the top-line results of the MPs' research are shown in the following tables. One question concerned the Members' familiarity with the activities of Nirex and other organizations with an interest in radwaste disposal. The results are shown in Table III.
TABLE III

As can be seen, the results are not very encouraging, though they will in part reflect the high media profiles of the campaigning organizations and of BNFL in the 1992-1993 debate over authorization of BNFL's new "THORP" reprocessing plant.
Another question concerned the MPs' general impression of Nirex compared to the same organizations as in the previous question. The results are given in Table IV.
TABLE IV
Here, the Company does relatively better, but given that many of the respondents had earlier admitted they were not overly familiar with our activities, the results are still worrying.
An important question covered the effectiveness or ineffectiveness of Nirex and the other organizations in their communication with Members of Parliament. The results are presented in Table V:
TABLE V
The simple and obvious conclusion is that a strong communications program is vital if politicians are to be kept informed. If they are not informed they cannot make measured judgments on Nirex's intended program for waste disposal. The other obvious conclusion is that if information is not supplied effectively, then those who oppose the industry, in particular the "green" groups, will be able to advance their positions relatively easily. Also, by not communicating, the Company is paradoxically giving strong negative messages which could be interpreted as:

politicians are not thought to be worth communicating with;
that Nirex is secretive and refuses to communicate; and,
that those who condemn Nirex are, in fact, correct in their assertions.

JOURNALISTS
A further key group is journalists. MORI, another leading opinion research consultancy in the UK, carries out regular surveys of journalists, editors and broadcasters; from time to time, they also present the views of specialist journalists. Journalists represent an important link between organizations and the general public, politicians and other audiences, providing an important source of information.
The survey is also particularly useful because companies taking part are able to nominate other companies or industries and particular topics for inclusion in parts of the questionnaire. This means that useful data on how Nirex and its activities are perceived by a target group, in comparison with, for example, companies involved in toxic or domestic waste disposal or other companies involved in the nuclear industry, can be obtained as part of a much wider survey, at reasonable cost.
Nirex had previously participated in the survey in 1990. However, because of increased media coverage about the Company's activities (much of it negative), it was considered worthwhile taking part in the 1994 survey.
The survey is designed to give those companies taking part information on:

current and future environment issues;
ways in which these issues could be tackled;
the perceived environmental record of industry sectors, companies and organizations;
how their environmental records compare with those of other sectors and companies/organizations;
perceived environmental impact and responsibility of various companies and organizations; and,
perceived effectiveness of their media relations compared with other companies and organizations.

Opinion Research Methodology
A letter requesting an interview was sent by MORI to 54 of Britain's leading environment journalists, editors and broadcasters eligible for inclusion in the survey. Appointments were then made by telephone, and face-to-face interviews with 28 journalists were conducted in July and August 1994. This represents a response rate of 52%.
Those interviewed were from a wide range of media - daily and Sunday newspapers, regional press, relevant periodicals, radio and television broadcasting. Nine of the twenty-eight journalists interviewed were willing for their answers to be attributed to them by name in the report, the remainder preferring anonymity. Twenty-seven of the twenty-eight journalists agreed to be named as having participated in the survey.
Results and Discussion

Twenty-five per cent. of the journalists questioned in 1994 considered Nirex was causing a major amount of damage to the environment, compared with 4% in 1990. This would seem to reflect their views on the nuclear business rather than Nirex's particular involvement. Some of the comments are illuminating: "Nirex has failed, except when pressed very hard, to demonstrate an acceptable and water-tight way of storing nuclear waste underground". It should be stressed that one thing Nirex has never claimed for the repository is that it will be water-tight!
Thirty-two per cent. rate Nirex's press relations as very/fairly good, but 29% rated them as very poor/fairly poor. In fact, the number of journalists nationally who specialize in radioactive waste disposal is fairly small. They, and the local media in Cumbria, know who Nirex is, where it is located and that the Company will always help them with a comment, a photograph, an interview or a story. However, perhaps the most worrying statistic to come out of the survey is the fact that 93% find "voluntary environmental organizations" the most useful source of information on companies such as Nirex.
The clear message from this survey is that much more work needs to be done with this group for the Company to improve understanding of its mission and research programs.
CONCLUSION
For Nirex, opinion research guides its communication programs for key audiences and allows important issues to be isolated and targeted. It also permits a measurement of performance and gives an indication of which initiatives are successful and which are not.
In the local context public involvement in decision making has helped the Company maintain a support for its activities in Cumbria amounting to around 70%, but this needs to be translated into the minds of the local politicians. Monitoring and maintaining the support of national government and of important influencers such as journalists is of equal relevance.
Nirex has adopted specific methods of gauging the perceptions of its key audiences through adopting appropriate systems of opinion research most applicable to each group. This appears to be paying off and giving the Company a steer as to where it should be targeting its communication effort.

49-3
THE FRENCH NATIONAL INVENTORY OF RADIOACTIVE WASTE, ALL INFORMATION OPEN AND CLEAR
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ABSTRACT
Article 13 of the December 30, 1991 Waste Act calls for Andra "to register the condition and location of all radioactive waste on national territory".
The establishment of a national inventory of radioactive waste and the broad distribution of inventory reports to ensure that it becomes a matter of public record constitute a new approach to public information and an effective means of fulfilling our responsibility to posterity. The register goes beyond the low-level radioactive waste disposal facilities to encompass "all" waste, wherever it may be, including waste in storage at generator sites. As a result, the register is multi-faceted, containing information on everything from highly radioactive waste to hospital waste collected by Andra and to repositories with very low-level radioactive material. Information must be provided about all of these widely divergent components, even though they cannot be treated alike.
ANDRA already published two inventories, demonstrating the durability of its new mission. The document now contains the inventory of radioactive waste generated by some activities connected with the defense program.
Article 6 of the December 30, 1991 Waste Act calls for Andra, before starting any preliminary works for a project of underground laboratory, to have concertation with elected people involved and with the public of the sites concerned. This concertation needs for ANDRA to be accepted as an open, reliable, fully honest and transparent partner in the discussion.
Data collection for the register therefore involves contacting the generators and working with these entities, whether nuclear industry companies, defense

organizations, non nuclear industries, or the 25 Regional Directorates of Industry, Research and Environment, the control institutions or the Environmental protection organizations.
The paper presents the management of data collection and the feed back questions and comments received after the 1994 issue of the document.
The yearly existence of this exchange of information between all the partners involved in radioactive waste management is one of the basic tools allowing ANDRA to be recognized as open and responsible, and to be more credible, fulfilling this way one of the essential criteria for acceptability.

THE NATIONAL RADIOACTIVE WASTE INVENTORY

Radioactive waste management is widely perceived to involve multiple stakes in terms of the environment and communication. Although suitable treatment methods for each category of waste must be applied, based on waste type and characteristics, waste disposal relies on a single set of principles: Straightforward information, clear-cut responsibilities, and concern for long-term environmental protection.
In this vein, ANDRA, the national radioactive waste management agency of France, was charged by the Waste Act of December 30, 1991 with developing a waste inventory which identifies the condition and location of all waste on French territory, in addition to fulfilling its other responsibilities (1).

COLLECTING, CLARIFYING AND SPREADING INFORMATION

The National Waste Register provides information on the location and condition of all known waste, including very low-level waste, whose disposal does not generally fall under the Agency's purview. However, Andra makes no claims for the completeness of the Register, for there is always the possibility that abandoned waste will be discovered in the coming decades.
Today's waste, whether defense or civilian, is generated by nuclear power plants and fuel cycle facilities, research laboratories, industry and hospitals, waste generated in the past is stored or disposed of at nuclear facilities, mines and old landfills.
The work involved in registering this data focuses on two activities: establishing a detailed picture that gives the status of all waste, including waste from ongoing site restoration, and making this information public. The underlying goal is to build collective memory about our defense and civilian radioactive waste.
The approach is to begin with existing registers and to expand from there through direct contact with those in possession of waste: Electricit de France (EDF), Cogema, the French Atomic Energy Commission (CEA) and other operators.
The presence of radioactive waste at a site is not necessarily synonymous with danger. What is most important is the manner in which the waste is managed and how the environment is monitored to ensure our long-term safety. With this in view, activities have already begun to clean up sites such as the old Bayard clock factories, which are not in satisfactory condition.

A MISSION CLEARLY DEFINED

Two editions of the Register were prepared by Andra's National Radioactive Waste Observatory. Radioactive waste did not begin with the development of nuclear power. In the first half of this century, industry used radium to make luminescent paints. Waste was not a subject of concern in that era. More recently, as nuclear power grew more widespread, new radioactive waste was generated. Radioactive products are frequently used in medicine, research and industry, although in small quantities. The use of their radioactive properties results in the creation of radioactive waste. This is a reality that must be faced by users, who must be encouraged to integrate good waste management practices early in their production process.
To produce the least amount of waste possible, to manage, process and dispose of final waste in a manner that protects our environment now and in the future: these are increasingly important requirements.
One of Andra's missions is to identify and collect existing data and to update them regularly so that the existence of this waste is not forgotten. Conveying this data to the public through publication of the Register ensures that the memory of the waste is shared by many.
By law, the scope of the register includes all radioactive waste on national territory. It therefore contains a variety of elements, from highly radioactive waste such as that stored at the La Hague nuclear fuel reprocessing plants, to waste collected by Andra from hospitals or landfills where materials that are only slightly radioactive are buried. We arbitrarily chose a total radioactivity

threshold of 0.5 Gbq, above which a descriptive data sheet must be established. This threshold makes it possible to provide representative examples of small generators in the fields of research, health, defense and non-nuclear industries without burying the report under hundreds of data sheets. Nevertheless, this is a low total radioactivity threshold equal to approximately one one-hundredth of a gram of radium.

The resolutely factual report on the Register transcribes data collected and made public by Andra without any added commentary or evaluation. In fact, it is not for Andra to take the place of government regulatory authorities with regard to safety, public health, or verification of regulatory compliance, which are the responsibility of the Division of Nuclear Facility Safety, the Industrial Environment Department, the Office of Radiation Protection, the Regional Departments of Industry, of Research, and of the Environment, and others.

We make no claims about completeness, and recent events such as the discovery of the Radiacontrle site at Pierrelatte or the story about Provaradior animal feed bags doped with radium show how important it is to be ever-watchful.

INFORMATION SOURCES

The search for data for the Register involves first and foremost direct contact with the operators and administrations that updated or approved the data sheets for the 1993 Register, or those that sent new data sheets, such as the Ministry of Defense. Andra's initial database was also expanded with the addition of documentation sent by associations such as the Commission for Independent Research and Information on Radioactivity (CRII-RAD), the Coalition of Scientists for Nuclear Energy Information (GSIEN), the Uranium Network of Action Environnement and the Western Association for Radioactive Monitoring at our request for information as part of the annual updating of our Register.

Andra therefore has multiple sources of information:

Companies in the nuclear industry (CEA research centers, Cogema, EDF, etc.);

The defense establishment;

Non-nuclear industries, i.e., small generators divided into fifteen business sectors: nuclear and non-nuclear industrial and commercial companies, research establishments, units of the National Scientific Research Center (CNRS), educational establishments, universities, university hospitals, units of INSERM, medical and paramedical-medical establishments, the Armed Forces Health Service, facilities of the defense establishment, suppliers of sealed sources, ministerial, prefectorial and municipal departments, and national public service entities

The twenty-five Regional Directorates of Industry, Research and the Environment.

In addition, the following associations were contacted: Action Environnement, Western Association for Radioactive Monitoring, Friends of the Earth, CRII-RAD, FLEPNA, FRAPNA, Greenpeace France, GSIEN, Robin Hood, and others.

1993 ACTIVITIES

The publication of the Register, as valuable a tool as it is for preserving memory and sensitizing the public, is not a replacement for managing waste properly, particularly in the clean-up of sites that stand out for mismanagement.

In this spirit, a major site clean-up operation was undertaken at the old Bayard plants in St. Nicolas d'Aliermont in the Seine Maritime Department, where luminescent radium paint was used to make alarm clocks. Contaminated soil that was potentially harmful to residents was removed and temporarily stored in an old facility of the former Bayard alarm clock plant pending selection of disposal methods. This operation mobilized numerous government departments -- the Ministry of Health, SCPRI, the Mobil Radiological Intervention Unit of the Fire Department, and Municipal Services of the commune --Andra, and the Office of Radiation Protection Assistance of the Institute for Nuclear Protection and Safety (IPSN). The Western Association for Radioactive Monitoring (ACRO) performed its own sampling and analysis. Special funding had to be secured to implement these operations, which were to cost several million francs, with Andra advancing funds for the most urgent tasks. A budget request was submitted to the Ministry of Industry, Postal and Telecommunications Services and Trade for the funds needed to complete the work as planned.

A similar but much less massive operation is also in progress under Court supervision relative to the Radiacontrle company site.

The Grande Paroisse S.A. Company commissioned a study by the Institut Garenne, IPSN and the Mining Radiation Protection Center (CRPM) on the activity of a

phosophogypsum dump site. The study showed that total combined annual exposure was 0.5 to 3% of the regulatory limit. As indicated in 1993, this data sheet is given as an example only; there are many other phosophogypsum dumps for which we have no information, and this is not a unique case.

There were two interesting stories about discoveries of radioactive materials and equipment indicative of widespread use of radium several decades ago:

a sort of urn in which a radium pellet was dipped to impart curative properties to the water, like some thermal waters; and

packages of animal feed under the Provaradior label were doped with radium as a growth stimulant (there is no record of whether they were charged with false advertising).

In both cases, the individuals who made the discoveries informed the local fire department, which retrieved the objects in question and contacted Andra.

CHANGES FROM 1993 TO 1994

The publication of Andra's first register in 1993 and its annual updating have evolved into an ethic of openness in both the defense and civil sectors. The Ministry of Defense and the French Atomic Energy Commission each established their own radioactive waste register for their defense-related activities.

The Defense Establishment

The register of the Ministry of Defense primarily includes equipment containing luminescent products and waste from the fabrication and testing of depleted uranium armor-piercing shells. Uranium was chosen for the shells due to its metallurgical properties only and not for any radioactive or nuclear reaction effect.

This register also includes waste generated by the repair of nuclear submarine reactors. Fifty-three sites are listed in the register, of which only eight exceed the arbitrary threshold of 0.5 GBQ requiring preparation of a descriptive data sheet.

The register of CEA sites and Cogema plants covers waste generated by the design, production and upkeep of nuclear weapons and nuclear submarine reactor cores. Nine sites are included, each of which is described in a data sheet.

Sealed Sources

The register of spent sealed sources, for which the French fabricator ORIS was the only company included in the 1993 edition, was expanded by the inclusion of spent sources at supplier sites and sources awaiting return to French or foreign fabricators. This register now includes seventy distributors, with nineteen of them described in data sheets.

Lightning Rods

Certain kinds of old lightning rods were made with tips containing americium or radium. Approximately 30,000 of this kind of lightning rod were produced through 1986. Since then, the use of these radioactive substances in lightning rods is prohibited. When these lightning rods are replaced, they become waste that must be returned to the suppliers, where the iron and metal are removed and the radioactive source is retrieved.

Andra collected all of the lightning rods stored at supplier locations. In addition, three large companies from the public sector -- EDF, TDF and France Telecom -- undertook the establishment of a register of their lightning rods.

DEVELOPMENT OF THE INVENTORY

While Article 13 of the December 30, 1991 Waste Act calls for Andra "to register the condition and location of all radioactive waste on national territory," the responsibility for nuclear facility safety clearly resides with the industrial companies, hospitals, research centers and others. The government is in charge of verifying regulatory compliance, particularly the Departmental Prefects and the Regional Directorates of the Ministries of Industry, Research and the Environment (DRIRE).

Data collection for the register therefore involves contacting the generators and working with these government entities.

Andra's program covers "all radioactive waste"; no limits are given for the register, nor is a minimum level of radioactivity specified.

ANDRA considers that :

The program scope includes not only waste disposal facilities, but any facility with waste in storage, even temporarily, whether it be a large nuclear facility or a medical or university research laboratory.

Non-reusable equipment and material, or equipment and material which the owner has

decided not to re-use, is included in the definition of radioactive waste. Nuclear facilities stopped, under decommissioning or under dismantling are considered as containing potential waste. The actual volume of waste will be depending of the level of decommissioning and of the date of this operation. Inversely nuclear material considered by the owner as reusable is not considered. This is the situation of fuel elements waiting before reprocessing and of depleted uranium owned by COGEMA.

Any product contaminated with man-made radioelements and any substance which has been mechanically or chemically altered in a manner which releases natural radioelements is considered to be radioactive.

Given this definition, the scope of the register encompasses a large number of sites which are widely divergent in terms of radioactivity, with differences of up to a factor of 10 to the 16th power, or a million billions. One must be careful, therefore, not to draw comparisons between what is not comparable.

USE OF THE INVENTORY

The data sheets are organized geographically in the register by region and by department to facilitate the public's understanding of them, particularly those close to a storage or disposal site.

Maps for each region are cross-referenced to the data sheets of registered sites. The geographically cross-referenced lists were also prepared alphabetically by site name, by waste category and by the name of the company holding the waste. These lists make up the register's first appendix.

Each data sheet contains administrative and technical information on the site, a summary description of the radioactive waste, an estimate of the waste's total and specific activity, and a list of the known radionuclides contained in the waste. Information sources are identified. Despite the very wide variety of situations encountered, we preferred a certain consistency of format to make the data sheets more readable.

Twelve categories of radioactive waste generators/sites were identified:

- nuclear power plants,
- uranium mines,
- Cogema reprocessing plants,
- other nuclear companies,
- CEA research centers,
- Andra storage and disposal facilities,
- decommissioned facilities,
- small waste generators,
- distributors of sealed sources
- non-nuclear industries
- waste dumps.
- the defense establishment

OPENNESS, RESPONSIBILITY, ACCEPTABILITY

By the end of 1989, fierce public opposition to the drilling program for the other three sites had developed. The Prime Minister of France declared a one year moratorium on all drilling activities and ordered a comprehensive review of the various aspects of the backend of the fuel cycle and waste management.

The Government asked advice from the College de la Prvention des Risques Technologiques (Technological Risk Prevention Board), group of 12 independent experts created in 1988. After various hearings the College published in april 1990 a first opinion. In this document the College underlines the necessity of underground disposal for some waste and considers that no waste can be buried in a non-reversible way before the potentiality of partitioning and transmutation has been studied. The College also gives some technical recommendations among which the creation of several underground laboratories to come to the best choice of one or more disposal sites.

In the same time, in accordance with the wish of the Prime Minister, the Parliament (National Assembly and Senate) addressed the Office Parlementaire d'Evaluation des Choix Scientifiques et Techniques (Parliamentary Office of Technology and Scientific Assessment). The Office, after contacts with a large number of people representing institutions, associations and electives, and after visiting several foreign facilities, and after four days of public hearings, reported to the Government on December 17, 1990.

The outcome of this review was the major new legislation on radioactive waste

management, enacted by Parliament on December 30, 1991.
The conclusion of the report insists on the fact that " Responsibility, transparency and democracy are the words that must now drive any process concerning waste disposal and more generally concerning the civilian nuclear policy.

RESPONSIBILITY, because we have to dispose our waste at home in France and banish any post-colonialist idea of foreign disposal. Furthermore we have to solve this question now and avoid transmitting to the next generation, to our children, the search of a solution.

TRANSPARENCY: The nineties must see the end of the secrecy culture in nuclear issues ; the military origins of this industry can explain this irritating persistency of the secrecy culture which is no more justified when anybody in France profits of the nuclear energy. We now need a transparent approach.

DEMOCRACY: Till today an autoritary and administrative method prevailed. It can have corresponded to a development without obstacle of nuclear energy in France. Without doubting of the potential of researchers and technicians it comported the risk of a technologic and scientist dizziness. Things are different today, the public requires information and wish to participate in decision making. As far as long-lived radioactive waste management is concerned, one can see that the conclusions of the report is similar to the initial objectives. BUT IT IS PROPOSED TO TAKE AN OTHER WAY using the resources of democracy and information. To this sophisticated form of energy production that nuclear is, must correspond an open decision making process. The future of nuclear energy in our country relies on our capacity to develop democracy."

The publication of the yearly updated and enlarged inventory is one major element in the volunteer trend of ANDRA towards responsibility, transparency and democracy as requested by the Parliamentary Office of Technology and Scientific Assessment in its report.
This publication is one of the supports for our credibility and later for the acceptability of new projects, in particular for the siting of the underground research laboratories to study the conditions of a potential geological waste disposal facility , reversible or not.
To be acceptable one must be credible, to be credible one must be transparent and responsible, to be transparent and responsible one must be independent.
But being independent, transparent, responsible and credible is not enough to be acceptable.
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DETERMINING ORGANIZATIONAL LEARNING PERFORMANCE: EVALUATING THE U.S. DEPARTMENT OF ENERGY'S (DOE'S) NATIONAL TECHNOLOGY INFORMATION EXCHANGE (TIE) WORKSHOPS
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ABSTRACT
In this paper we describe an organizational learning paradigm we used to evaluate how well the U.S. Department of Energy's (DOE's) National Technology Information Exchange (TIE) Workshops are achieving their mission. The National TIE Workshops are described as an organizational learning process because their mission is to increase efficiency and reduce overall costs in DOE's environmental restoration programs by sharing lessons learned. We evaluated TIE's effectiveness by measuring whether lessons learned were actually being shared and applied to other problem solving experiences. This paper outlines the TIE process, defines organizational learning and performance, and characterizes the methods we used. We also present evaluation findings and provide an evaluation approach that could be applied to other DOE meetings or group events.
INTRODUCTION
The U.S. Department of Energy (DOE) hosts a wide variety of workshops, meetings, and other events in an effort to improve performance among its program offices and sites. These gatherings bring people together to increase learning on an individual and organizational level. Defining, measuring, and demonstrating this learning is vital in today's tightening fiscal climate.
DOE's 1994 strategic plan outlines a total quality management (TQM) approach to managing its national programs in a post-Cold War era: " The world has changed....Our challenge is to move away from the Cold War economy, invest in people and technology to strengthen the economy and protect the environment, and reinvent a government that is efficient, serves the American people, and provides more services with fewer resources." Part of the transformation to a TQM environment is becoming a learning organization. In fact, one of DOE's TQM success factors is assessing how information is communicated and trust is built.
In learning organizations, successes and failures are captured and made available, responsible people have access to information when they need it, and the sharing of information becomes a performance measure (14). Organizational learning is the process through which these activities occur. We suggest organizational learning includes sharing lessons learned, applying the new knowledge, and sharing lessons learned again. DOE is moving in the direction of becoming a learning organization.
TECHNOLOGY INFORMATION EXCHANGE (TIE) WORKSHOPS
Within DOE, the National Technology Information Exchange (TIE) Workshops exemplify an organizational learning tool. The National TIE Workshops were founded to provide environmental management (EM) workers an interactive forum to share information about existing environmental restoration and waste management problems, technology needs, and potential near-term technology solutions. The intended result is new and existing environmental restoration and waste management technologies will be applied more efficiently, less expensively, with better results, and in compliance with regulatory requirements. TIE's theme is "Using Today's Technologies Better."
TIE's primary participants are those who actually do the environmental management workthe engineers, geologists, chemists, etc. who implement restoration and waste management technologies. TIE's secondary audience is made up of technology developers, both within DOE and industry, and other stakeholders, such as regulators. TIE attracts DOE and contractor personnel across the complex and encourages participation from other federal agencies, regulators, and industry.
Each TIE Workshop features a variety of interactive formats designed to help peers share information about their success and failures. These formats include: panel, poster, and interactive poster sessions; spontaneous and scheduled breakout sessions; concurrent sessions; and site tours. Topics and presentations for the TIE Workshops are selected by approximately 25 volunteers from DOE sites who serve as members of the Field Area Technology Representatives Steering Committee.
The TIE Quarterly newsletter continues the information exchange begun at TIE

workshops. Distributed to approximately 3,000 people, the TIE Quarterly covers lessons learned presented at each workshop and environmental management activities and technologies at DOE sites.

The TIE Workshop as a Learning System

We view TIE in the context of a cyclic learning system. Figure 1 illustrates this system. Participants from across the DOE complex bring Lessons Learned (inputs) about implementing environmental restoration and waste management technology to a National TIE Workshop. Sharing these lessons learned with peers changes and increases the knowledge of other participants (outputs). Participants use this Shared and Changed Knowledge in the field (DOE Field Actions) on environmental restoration and waste management projects (outcomes), benefiting and improving DOE Performance in environmental restoration and waste management. In turn, participants have the opportunity to share lessons learned from applying their changed knowledge at future National TIE Workshops. However, the TIE process is not a closed system. Lessons learned are continually being fed into the process by Other Sources from within DOE, other government agencies, regulators, and industry. In summary, TIE converts inputs to outputs for an outcome to its customers: DOE environmental management workers and ultimately the DOE environmental management program as a whole. Seen as a cyclic, systemic process, the National TIE Workshop is a model of DOE organizational learning.

Fig. 1.

Organizational Learning

Garvin (6) defines a learning organization as "an organization skilled at creating, acquiring, and transferring knowledge, and at modifying its behavior to reflect new knowledge and insights" (p. 80). The process of sharing knowledge is called organizational learning. The intent of organizational learning is to improve organizational performance through the use of knowledge (4). Knowledge gained from experiences is transferred through lessons learned.

A lesson learned is a "catchall phrase describing what has been learned from experience" (8). Lessons learned are developed, stored, retrieved (distributed), and interpreted by individuals or groups of individuals in an organization as a result of problem solving (7,3,12,4,). Organizational learning includes gathering or describing data, relating the data to the current situation, and applying the data to build information and knowledge (9,14).

Figure 2 illustrates the process of organizational learning. The solid rectangles in Figure 2 represent the four functions of organizational learning. The first function, Problem-Solving Experience, occurs when an individual produces a lesson learned or organizational knowledge through applying some method or data to a situation and understanding the results. DOE Program Activities exemplify such problem-solving experiences.

The second function, Storage and Refinement of Organizational Knowledge, is the process "by which knowledge is stored for future use" (7) and organizational memory is refined or updated. Organizational Memory is refined by supporting applications or actions that met expectations and reducing or disconfirming applications or actions that didn't meet expectations. The third function is the Distribution and Retrieval of Organization Knowledge, or accessing the knowledge gathered from problem-solving experiences. The second and third functions, with organizational learning, are represented by the National TIE Workshops.

TIE is an example of a formal mechanism to refine, store, distribute, and retrieve lessons learned in DOE's environmental management program. Lessons learned gained from a problem-solving experience are stored and refined in organizational memory as knowledge. This knowledge is retrieved, distributed, and interpreted to help solve an analogous problem. For example, a team at the Lawrence Livermore National Laboratory (LLNL) was faced with the problem of chromium (Cr^{6+}) contamination in the ground water. After testing several alternatives, they developed an ion exchange resin to filter the chromium. The team brought this lesson learned to a National TIE Workshop.

The fourth function, Interpretation and Use of Lessons Learned, is applying new information to another problem-solving experience. This function represents the decisions and actions TIE participants implement after National TIE Workshops to begin a new problem-solving experience. Without a mechanism to store and retrieve lessons learned (organizational knowledge), an organization will be unable to improve decisions and actions based on experience.

Returning to the problem of chromium contamination, an individual presented the lesson learned at the National TIE Workshop and the information was also stored and distributed to a larger audience in the TIE Quarterly. An individual from Hanford retrieved this experience and applied it to the chromium contamination problem at the Hanford 100 Area. TIE provides the learning mechanism, but individual decisions and actions determine the outcome of the process.
Fig. 2.

ORGANIZATIONAL PERFORMANCE

Sink and Tuttle (13) define organizational performance in terms of seven criteria or types of measures:

- effectiveness
- efficiency
- quality
- productivity
- quality of work life
- innovation, and
- profitability/budgetability.

To evaluate TIE's performance expectations, we identified the desired outputs and outcomes described in its mission statement. TIE's mission statement discusses all seven performance criteria. In some organizations, these criteria are not found in mission statements, but rather in a statement of objectives or other document. TIE's mission is:

> to promote the sharing of information concerning the environmental restoration and waste management problems, plans, actions, successes, failures, and related technology needs and development among persons directly involved in environmental restoration, waste management, decontamination and decommissioning, and related technology development activities. The focus is on the sharing of lessons learned among "hands-on" field personnel who can benefit from a more effective utilization of their current technologies, in particular, those which can achieve a cost savings. The TIE Workshops have an ongoing "shoptalk" environment which technologically "ties" the sites together. The goal of TIE is the identification and implementation of the best available, cost-effective, and appropriate technology to address DOE environmental restoration and waste management problems today.

Because TIE is a model of organizational learning, a measure of effectiveness is constantly sharing and using lessons learned and other information. Both Sink and Tuttle (13) and TIE's mission define additional measures. TIE (and organizational learning) is concerned with quality. Better use of technologies improves efficiency and productivity. Cost savings improves budgetability. The TIE process of involving DOE workers from all DOE sites improves quality of work life. Using the most appropriate technology addresses innovation. TIE achieves its performance expectations if DOE environmental management workers increase their knowledge and reduce DOE costs as a result of attending National TIE Workshops.

To TIE's organizational learning performance, we decided to examine both the outputs and outcomes of TIEthat is, (1) whether participants learned about a technology, technique, or approach at TIE (shared and changed knowledge) [outputs) and (2) whether participants then made decisions and took actions to apply that shared and changed knowledge at their sites [outcomes], and (3) whether TIE participants anticipated cost savings or improved efficiency as a result of their decisions and actions after attending a National TIE Workshop [outcomes].

METHODOLOGY

We measured TIE's effectiveness in three phases. The first phase was gathering participant feedback on benefits they received from the National TIE Workshop. We used oral and written surveys during the first five TIE workshops. During the second phase, we compiled the survey feedback and conducted a content analysis to understand the feedback more completely. A content analysis is a process of culling meaning and categories of meanings from qualitative data. We looked for patterns in the expected and reported outputs and outcomes participants said they gained from National TIE Workshops. We then created a revised survey based on the content analysis and used it at the Sixth National TIE Workshop. The third phase was an interim survey between the sixth and seventh National TIE Workshops. (The seventh workshop will be held in April 1995.)

At the first five TIE Workshops, both the oral and written surveys predominantly focused on lessons learned, benefits, and expected performance outcomes of applying

knowledge gained at the workshop. Most participants reported they exchanged lessons learned and made valuable contacts with their peers. Some participants anticipated saving significant development and labor costs. At later workshops, some participants reported outcomes realized from applying lessons learned from a previous National TIE Workshop. Based on participants' feedback from these surveys, we concluded TIE was meeting its mission on a workshop-by-workshop basis.

But after five workshops, we wanted to understand how well the National TIE Workshops were performing as a whole. We wanted to identify and evaluate the outputs and outcomes of TIE. We recognized an inherent time lag between the outputs and outcomes of a TIE Workshop that couldn't be accounted for through surveys at a particular workshop. We wanted to document actual performance outcomes that may have been realized over the past three years.

We developed and executed a process to evaluate the outputs and outcomes of TIE:

1. Identify performance expectations by examining TIE's mission statement.
2. Conduct general surveys at the first 5 National TIE Workshops.
3. Analyze TIE survey feedback by performing a content analysis of past TIE survey results.
4. Develop a conceptual model of TIE based on the content analysis (see Fig. 3).
5. Develop and administer a structured survey based on the conceptual model to measure TIE outputs and expected outcomes.
6. Analyze the survey results.
7. Develop an interim (between workshop) survey to document additional outcomes.
8. Administer the interim survey to all past TIE Workshop participants.
9. Analyze survey results.

Conceptual Model of TIE Workshops

Figure 3 shows a conceptual model of the relationships of TIE outputs and outcomes. We developed this model based on our content analysis. According to our analysis, the National TIE Workshops produce two kinds of outputs: Knowledge and Relationships (the first part of the Fig. 3). TIE participants develop new relationships with peers across the complex. These relationships support the exchange of information and knowledge during and after the workshops. The TIE process results in shared or changed knowledge. We categorized the types of knowledge participants acquired at TIE:

1. general: suggestions, feedback, studies completed, current literature to use as a reference, benchmarking;
2. status of DOE complex: the goals and scope of the problems facing the complex;
3. specific discussions of methods/technologies: what worked, what didn't work, resources needed, alternative methods, how methods support regulatory compliance;
4. cases: examples of general approaches for program successes and failures; and
5. need: needs of other sites, mutual problems, opportunities for sharing technologies and efforts.

The next section of the figure is Individual Actions. The knowledge and relationship outputs support actions taken by individuals in the field. Participants reported they made decisions to pursue or not pursue an approach to solving their site problem as a result of the information and knowledge gained at the National TIE Workshops. They also reported following-up with other participants by calling them or establishing information exchanges.

Fig. 3.

Individual Actions lead to Program Level actions. Participants identified program level actions focusing on technology and coordination. Participants reported information shared at the workshops resulted in technology being bought, sold, borrowed, or demonstrated between sites and industry. Coordination of activities included sharing and re-using technology between sites, eliminating duplication, increasing consistency, and saving resources.

Program Actions lead to Program Performance. Participants identified the following program performance outcome measures as a result of attending TIE: compliance, time, public acceptance, dollars, efficiency, quality, and safety. Participants believed they could bring a project into compliance, meet the regulations more quickly, avoid non-compliance, and comply with regulations they didn't know about. Participants reported they would save time as measured in person days, years, and workload in development and implementation of methods learned about at TIE. The public's acceptance of DOE, confidence in DOE, and perception of DOE's credibility would increase as a result of the individual and program level actions derived from TIE.

Participants believed the Environmental Management program would operate more efficiently with greater quality and more safely. Participants believed all these benefits would result in the dollars saved (though few were able to quantify). They also identified cost reductions in marketing, equipment development, research and development, analysis, and travel cost.

Based on this conceptual model, we developed a survey to evaluate and quantify the outputs and outcomes TIE participants said they received. We used subjective measures because 1) the participants were surveyed at the workshop and 2) there is an inherent time lag between the change in knowledge and the use of the knowledge. The survey included lists of outputs and outcomes based on previous participant feedback identified in the content analysis. For example, we asked participants to tell us how many of the following outcomes they received from attending a National TIE Workshop:

- communicate more with other sites
- coordinate environmental restoration efforts with another site
- save time
- save person hours
- save other resources
- adapt existing technology methods to your site (instead of developing new ones)
- apply successful methods from another site
- adapt existing data from studies to your site
- work better with regulators
- transfer technology to another site
- share equipment with another site
- increase credibility inside DOE
- increase credibility outside DOE
- immediately impact projects
- increase consistency across the DOE complex
- reduce workload in out years
- abandon bad ideas

We gave this survey to the participants at the Sixth National TIE Workshop, hosted by Richland Operations Office.

RESULTS AND DISCUSSION

At the Sixth National TIE Workshop, the return rate was almost double the response rate of past workshops. (At most workshops we get about a 10 percent return rate; at the sixth workshop, our return rate was about 20 percent.) Ninety-four percent of the survey respondents believed TIE is achieving its mission. Ninety-six percent thought there was increased cooperation among DOE sites doing environmental restoration work because of TIE. Respondents said they gained knowledge and made contacts that would impact their work, benefiting both the Environmental Restoration program and DOE as a whole:

Workshop Outputs

88% will follow up with other participants after the workshop.

41% learned something that will prevent them from improperly implementing an environmental restoration technology.

45% will explore the option of transferring technology or equipment from another site as a result of TIE.

48% learned of an opportunity to collaborate with another site on a project.

Performance Outcomes

56% will be able to apply information gained at TIE to save costs and improve efficiency.

76% said duplication of effort among DOE sites conducting ER work will be reduced as a result of TIE.

69% said their organization will share technology and projects with other sites as a result of TIE.

64% said their organization is likely to use methods or technologies discussed at TIE.

After the Sixth National TIE Workshop, we developed an interim survey to gather information on what TIE participants did with the knowledge they gained at TIE after the workshops. This survey was mailed to all past TIE participants (about 1200 people). For the interim survey, we had a 17% return rate, an expected rate given the method we used (we mailed the surveys with no follow-up). Twenty-four percent of respondents reported they have applied technologies or methods at their site based

on information gained at TIE. The time delay for implementing a technology or method ranged from zero to three years. (However, the workshop has only been held for three years.) For example, one participant reported that Oak Ridge implemented the SAFER approach to characterization immediately after TIE. Another participant reported that his/her organization entered into a Cooperative Research and Development Agreement (CRADA) with Lawrence Livermore National Laboratory as a result of TIE, and he/she expected the resulting technology to be implemented in three years. Sixty-eight percent of respondents said they expanded their communications or peer groups as a result of TIE. Most participants built networks of contacts for specific technology applications. Many visited other sites to observe or present technologies. Some developed and improved relations with their regulators by inviting them to TIE. And, some participants even formed their own mini-TIE's. For example, as a result of TIE, the Savannah River Site and Oak Ridge National Laboratory hold quarterly meetings to exchange environmental restoration lessons learned. These examples represent TIE's primary audience, DOE environmental restoration workers in the field.

Many respondents gave specific examples of ways they have saved money, increased efficiency, and/or brought a project into compliance because of TIE.

Hanford saved $109,000 on ground water cleanup technology recommended by Lawrence Livermore National Laboratory (LLNL). Took one year to implement.

One group at Rocky Flats saved three months and $30,000 when they found another group at Rocky Flats that had completed a data study similar to one they were planning to do.

By using the six-phase heating technology developed by Pacific Northwest Laboratories, Rocky Flats will reduce the time of subsurface remediation of NAPL from 10 years to less than one by increasing the mass removal rate of subsurface contaminants.

Our demographics section of the survey revealed that many survey respondents were unable to identify specific examples of benefits because they were part of TIE's secondary audience. TIE's secondary audience, technology developers, both within DOE and industry, and other stakeholders, such as regulators, though essential to the success of TIE, are contributors of knowledge, not implementers of technology solutions. The outputs and outcomes for TIE's secondary audience differ from that of the primary audience but contribute to TIE outcomes. For example, the DOE-Oakland Operations Office invited some of its regulators from the State of California to the Sixth National TIE Workshop. The regulators learned that 1) the dense nonaqueous phase liquid contamination problem is not limited to Lawrence Livermore National Laboratory (LLNL) and the problem is much greater at other sites and 2) other sites are having success with more innovative technologies than currently outlined by LLNL's/California's regulatory agreement. The outcome of this shared and changed knowledge led to LLNL/DOE coming to agreement "with its regulatory agencies to broaden application and testing of innovative remediation technologies at LLNL's Site 300 Building 834 Operable Unit." The ultimate outcome will be bringing this project into compliance, potentially saving time, money, and resources, and forging a stronger relationship with regulators.

CONCLUSION

In evaluating the National TIE Workshops, we examined the desired performance outcomes described in its mission. We defined effectiveness in organizational learning as constantly sharing and using lessons learned and other information. We found TIE is successful in meeting its mission, and we found it to be an effective organizational learning process. In fact, survey results suggest that performance is being improved across the spectrum of performance criteria. TIE participants are sharing lessons learned and translating this shared knowledge into outputs and outcomes that improve the Environmental Management program's performance in implementing environmental restoration and waste management technologies. TIE serves as the Environmental Management program's organizational memory for this knowledge. The TIE process can be improved by 1) improving its inputs and the workshop itself through feedback from the field and 2) improving the link between its outputs and outcomes through follow-up and evaluation of decisions which lead to actions. Within DOE, the success of activities like the National TIE Workshops is related to the ability of participants to apply the knowledge they learned at the workshops. DOE promotes organizational learning and respective performance improvement by hosting TIE Workshops, sending people to the workshop, and creating avenues for individuals

to affect change after the workshop.

LESSONS LEARNED FROM OUR PROCESS

Participants at the National TIE Workshops work with knowledge and information. Their actions during and after the workshop determine the level of improvement DOE experiences as a result. Total quality management, organizational learning, and other improvement activities are based in the actions and perceptions of people. The National TIE Workshops are focused on people. Our evaluation process was based on understanding and organizing people's reported perceptions of the National TIE Workshops to evaluate the outputs and outcomes. Following are our lessons learned:

As in all TQM activities, the evaluation process is cyclic. We used data from phase 1, which at first glance appeared unimportant to the second phase evaluation, to help us construct the conceptual model and detailed survey.

The evaluation process must be linked to the desired outcomes described in a well-formed mission statement.

A mission statement may not include all possible or desirable outcomes (or the right outcomes) and may need to be revised or expanded. In some organizations, a statement of objectives will contain desired outcomes. (Although not directly related to this evaluation, the TIE mission statement is currently being revised).

Constructing a conceptual model that described TIE as a process identified previously unidentified outputs and outcomes.

Success is often evaluated based on dollars saved. Though very important, this perspective may overlook equally important but less tangible outcomes, such as saving person hours or preventing duplicate tasks. Determining organizational learning performance requires a broader definition of success than the bottom line. One way of broadening it is to use the seven performance criteria described above.

Using an organizational learning and process perspective helped identify TIE's outputs and outcomes and provided a conceptual framework for developing a survey instrument.

This process can be used to evaluate the effectiveness of any ongoing meeting/process. These are the steps to follow:

- Examine the mission and/or objectives statement to identify desired outputs and outcomes.
- Administer general surveys longitudinally (i.e., over a period of time).
- Analyze the content from these survey responses for common themes, outputs, and outcomes by performing a content analysis.
- Develop a conceptual model of the process based on the content analysis.
- Develop and administer a structured survey based on the model and content analysis to measure outputs and expected outcomes.
- Analyze the survey results.

Next Steps

This evaluation process focused primarily on TIE's primary audiences-DOE environmental management workers. A model is needed for measuring the performance of TIE with secondary audiences of technology developers, both within DOE and industry, and other stakeholders, such as regulators. In support of organizational learning and performance improvement, TIE is effective at facilitating the sharing of information at the workshops. Individuals apply this information in the field. We don't know how often they return to TIE with lessons learned from the application of changed knowledge. This could help further define the effectiveness of National TIE Workshops. Currently, the National TIE Workshop's mission is being further defined. The mission change is significant. TIE is moving from merely facilitating information sharing to more proactively promoting technology implementation. Future evaluations will address this more comprehensive mission:

> To advance the accomplishment of the DOE Environmental Management mission by helping to promote the prompt and efficient application of the best available and tested commercial technologies and, when those are lacking, by helping to promote the development and application of appropriate new or modified technologies. (Draft TIE Mission, 1/24/95)

Continued attention should be placed on documenting performance outcome improvements objectively to complement participant's perceptions of performance improvement.

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APPLYING RESEARCH TO PUBLIC INVOLVEMENT PLANNING AT FERNALD

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ABSTRACT

This paper addresses the importance of applying public relations research techniques to develop credible and effective public involvement programs, particularly at U.S. Department of Energy (DOE) facilities. Specifically, this paper discusses the experience at the Fernald Environmental Management Project, where two major research efforts -- a community assessment and an internal communications audit -- have been completed. The assessment and audit results are being used as a benchmark for measuring the effectiveness of future program activities. In addition, the Fernald site's public involvement program has been fine-tuned, partly in response to the findings of these two studies. Public relations research is a valuable tool for planning and implementing two-way communication between an organization and its publics, because it measures changes in awareness, attitudes, opinions, and behaviors both inside and outside the organization. Measurement and evaluation are particularly helpful for gauging program effectiveness when resource constraints require management to select the most meaningful public involvement activities.

INTRODUCTION

Fernald Cleanup Budget Dwindling
Clinton Proposes Deep Cuts
Radioactive Waste: Pay Price of Cold War Victory, But Don't Get Soaked by Abuses
Superfund Spending in Line for a Trim? Hope Fernald Is Not Relegated to Back Burner
Fernald Watchers Concerned: GOP Ascendancy Could Affect Cleanup Funding

Newspaper headlines herald the new reality for environmental remediation efforts at U.S. Department of Energy (DOE) weapons complex sites: shrinking resources are requiring managers to be more selective with their programs. When budgets barely allow for the completion of mandated remediation activities, managers are finding themselves looking carefully at public involvement efforts. Gone is the luxury of being able to apply any and all types of public involvement techniques. Additionally, the DOE is requiring measurement in all areas of performance as a way to show effectiveness and prove value.

The DOE, acknowledging past weaknesses in its own and contractor performances in its report "Making Contracting Work Better and Cost Less," noted that DOE officials "are not in a position to ensure prudent expenditure of taxpayer dollars in pursuit of our principal missions. Now is the time to reform our management practices, empower our managers to effect these reforms, and measure their success." Indeed, DOE's contract reform initiative is guided by several key principles advanced by Secretary Hazel O'Leary, including developing meaningful ways to measure DOE and contractor performance with system performance criteria and measurement mechanisms. Crafting clearly stated, results-oriented performance criteria and measures is critical, according to the DOE's Contract Reform Team report.

Although measurement typically is associated with activities such as financial accounting, safety statistics, or production quotas, it is equally valid for public involvement activities. Research is recognized as being vital for planning communication programs and for determining the success of programs.

All these factors have contributed to creating an environment in which public relations research is not just desirable, but essential.

BACKGROUND

But what is public relations research, exactly? Broadly speaking, public relations research is that activity "conducted to detect problems and assess the status quo, on the one hand, and...to evaluate the planning, implementation, and impact of public relations programs on the other." (Dozier and Repper, 1992) The first type of public relations research often is used for defining problems, while the second kind of research -- evaluation research -- is designed to determine how well public relations programs work. Both types of research have value in DOE programs, although evaluation research is currently most relevant to the Fernald program.

Prior to initiating an ongoing measurement program as a routine component of the public involvement program at Fernald, there were some sporadic evaluation efforts. Employees were surveyed in 1985 and 1990, while neighbors living near the Fernald site were interviewed in a 1989 community assessment conducted prior to development of the Community Relations Plan. The Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) requires a community assessment to characterize residents' concerns about a contaminated site. At the time, the DOE did not seek to go beyond requirements and expand the scope of the assessment to support broader management objectives. These measurement efforts were not designed to support management evaluation of overall program effectiveness. Rather, the primary means of determining program effectiveness was the "gut" sense of the public relations staff working at the site.

The only other type of measurement consistently practiced at the Fernald site prior to implementation of the current measurement program was the practice of distributing questionnaires at meetings. These meeting evaluation forms, however, were not crafted to elicit a broad range of information; most simply asked if the attendee thought his or her questions were adequately answered by the presenters at the meeting.

In 1994, DOE and the Fernald Environmental Restoration Management Corporation (FERMCO), DOE's prime contractor at Fernald, recognized that a consistently-applied evaluation program was essential. Not only did the site's Community Relations Plan require revision to reflect public involvement activities during the project's remedial design and remedial action phases, but managers realized that they needed some way to determine the effectiveness of specific program activities. The community assessment and internal communications audit were designed to meet both regulatory requirements and management's needs.

METHODOLOGY

1994 Community Assessment

The DOE conducted a comprehensive community assessment in May 1994 to improve its understanding of community concerns, needs and interests. The community assessment

consisted of a series of interviews with members of the public who are impacted, or potentially impacted, by activities at the Fernald site.
The assessment involved 50 face-to-face interviews with community leaders, including business owners, government officials, educators, local news media, representatives of the Fernald Citizens Task Force, members of the Fernald Residents for Environmental Safety and Health, and others. To gain a sense of general public perceptions about Fernald, the assessment also included 365 telephone interviews with residents within a 20-mile radius of the Fernald site. The interviews were conducted by DOE and FERMCO Public Affairs personnel.
By conducting the community assessment, DOE can establish a baseline for use in monitoring changes in public interests, needs and concerns as the Fernald facility begins its transition from the remedial investigation/feasibility study phase to the remedial design and remedial action phase.
Internal Communications Audit
In addition to the community assessment, DOE and FERMCO commissioned a separate internal communications audit of Fernald employees. The audit was conducted by the University of Cincinnati (UC) Center for Environmental Communication Studies in July 1994. The audit was designed to gather information on current communication practices and provide assessment of the organizational culture at Fernald.
The UC research team used three main techniques during the audit -- senior management interviews, a survey questionnaire, and employee focus groups. Additionally, all informal contacts with Fernald employees during the study period were documented and integrated into the qualitative data where relevant.
The survey questionnaire was distributed to all FERMCO employees, 80 DOE-Fernald employees, and about 300 subcontractor employees. About 950 questionnaires were returned, representing a 36 percent response. The research team analyzed the responses and reported its findings to DOE and FERMCO management.
Feedback from both the community assessment and communications audit has been used to benchmark Fernald's effectiveness in delivering messages to employees and the community and to develop new or better approaches for informing and involving all stakeholders, including employees.
FINDINGS
1994 Community Assessment
Community concerns and issues raised during the 1994 community assessment can be linked to four core issues:
- Providing accurate information about the Fernald site and site activities
- Involving stakeholders in the decision-making process during all phases of the project
- Desire for site remediation without wasting taxpayers' money
- Public health and safety and economic impacts.

Providing accurate information: The 1994 community assessment indicates most community leaders are satisfied with the frequency, quantity and quality of information they receive from Fernald sources, though some commented on the need to present information in a more understandable form. More than half of the community leaders say their understanding of Fernald issues has changed due to information they have received. Most community leaders have established direct, personal contacts with Fernald personnel and are satisfied with them. Community leaders indicate information from Fernald is useful and say they are encouraged by recent efforts to reach and involve the public.
Findings suggest respondents in the 20-mile areas are not as informed and are less positive about Fernald, in general, than community leaders. These respondents almost exclusively rely on mass media for information about Fernald, unlike community leaders, who primarily receive their information directly from Fernald sources. Community leaders' ratings of Fernald management's efforts to inform the public are higher than those of the general public. One-third of general public respondents do not know who within the Fernald organization to contact about Fernald-related concerns, and many say they would contact elected officials.
Involving stakeholders: Half of the community leaders indicate they are receiving enough information to be involved in the decision-making process. Of the 50 percent who said they are not receiving sufficient information, some feel they can make no impact in the decision-making process and some simply do not envision themselves as part of the decision-making process.
Assessment findings indicate individuals living or working closer to Fernald are

more likely to participate in public involvement activities than people living or working farther from the site. Most community leaders say they are satisfied with their current level of involvement and do not want to be more involved. Some indicated that they would be strongly opposed to any diminishing of public involvement opportunities as the project progressed to the remedial design and remedial action phases. Findings indicate individuals farther from the Fernald facility prefer to be informed rather than actually involved.

Respondents were asked to state the most positive thing they could say about Fernald. Community leaders responded that the most positive aspect of the Fernald facility is improvement in public involvement. In comparison, general public respondents closest to the facility mention jobs and other economic benefits, and those farther from the facility either do not know or cannot say what is positive about Fernald.

Fernald remediation schedule and cost: Through the 1994 community assessment, public interest in timely, cost effective cleanup emerged as a major concern. This contrasts with past assessments which showed health and safety issues as the chief concerns.

Community leaders ranked concerns over budget and costs equally with concerns about public and worker safety and health. Some of the community leaders indicated concerns regarding the ultimate cost of remediation, and some were concerned about availability of funds to complete the project. Remediation progress was the sixth-ranked top concern among community leaders, but was tied for first among general public respondents within a 5-mile radius and second-ranked within a 20-mile radius.

Among the community leaders, opinions regarding cleanup progress were mixed. Forty-two percent said they believe progress is slow, but 30 percent perceived cleanup is progressing. Some said they believe the cleanup has been stalled or delayed or that the paper work required to prepare for remediation is burdensome. (General public respondents were not asked to respond to this question.)

Public health and safety and economic impacts: Most community leaders report no personal or family health problems which they attribute to Fernald. However, most community leaders say they know other people, outside their own families, who attribute health problems to Fernald site activities. Within a 5-mile radius, 16 percent of general public respondents attribute health problems to Fernald, compared to seven percent within a 20-mile radius.

Almost half of the community leaders believe their property or a family members' property or quality of life has been negatively impacted by the Fernald facility. In comparison, 19 percent of general public respondents within five miles of the site and eight percent within 20 miles of the site believe their property or a family members' property has been impacted by Fernald activities. Some community leaders mention groundwater contamination as an example of how Fernald has negatively impacted local property values and residents' quality and cost of living.

Internal Communications Audit

Major audit findings can be summarized in two categories:

- Information patterns
- Organizational culture

Information patterns: Several general themes are apparent in the data on information patterns. First, employees expressed a need for more information about job restructuring issues and how management makes decisions that affect their jobs and their futures. Second, employees indicated eagerness for information about Fernald policies and activities or for government information relevant to Fernald (e.g., the Federal budget process). They are also eager to provide input into Fernald policies and procedures, but they feel that they have little opportunity to do so. Third, it is apparent that employees are receiving safety information. The importance of safety has been effectively communicated to employees. Employees expressed a need for more job and situation specific safety information, indicating they have accepted the importance of safety as a goal. Fourth, employees indicated they receive a great deal of information through the informal grapevine, though they would prefer to get it from management (see Fig. 1). Finally, most employees indicated they are overloaded with written communications and electronic mail. Employees were split along job classifications on their perceptions of internal publications.

Two other key findings were revealed by the internal communications audit. Employees

said that they want face-to-face information from their bosses. The analysis of the audit also indicated a high need for employees to communicate to their supervisors on certain issues.
Fig. 1.
Organizational culture: Two sets of questions within the survey were aimed at characterizing the existing culture. The first set was designed to reflect work relationships at Fernald and the second set was focused on identifying the level of organizational commitment at Fernald.
The data indicated that employees are generally very positive about their co-workers and immediate supervisors. Employee responses indicate a less positive relationship with top management. All job classifications indicated a high degree of trust in fellow employees and supervisors, but there was a trend toward a distrust of top management. That trend however, was not as pronounced within the manager classification. The data also showed that Fernald workers are very committed to their jobs and their individual work groups. The only negative comment by many employees focused on the fact that they wanted to accomplish their work group's goals faster, but felt that they were being held back by regulatory agencies or a lack of cooperation from other work groups.
Crossover Questions
Several questions about the future of the Fernald site were included in both the community assessment and the internal communications audit to provide information on public preferences to the Fernald Citizens Task Force. Formed in 1993, the task force will develop recommendations about cleanup solutions and future courses of action at Fernald.
Cleanup Levels: Most community leaders and Fernald employees do not believe the Fernald site should be cleaned to a "pristine" condition. Conversely, more than half of all general public respondents believe the site should be cleaned to pristine levels, even if achieving that cleanup level would require spending more taxpayer money than needed to meet cleanup levels mandated by government regulations.
Waste Disposal: The most common preference for disposal of Fernald waste indicated by community leaders, general public respondents, and employees is to dispose of waste in arid western states at existing government facilities, if possible. Eighteen percent of community leaders, 23 percent of employees, and four percent of general public respondents acknowledged that some waste should be, or would have to be, stored on site.
Future Use: Regarding the future use of the Fernald facility, most community leaders and employees said they would like the Fernald facility to return to a natural setting, such as a wildlife/nature preserve. Of the general public respondents, one-third offered no specific suggestions on what should be done with the Fernald facility once remediation is completed.
PLANNING
Equipped with the results of the community assessment and internal communications audit, Fernald management applied these findings to public involvement and communications planning during Fall 1994.
Even as plans were being developed, results of the assessment and audit were communicated to employees and the public. Summaries of the research were written and distributed, and availability sessions were held for employees to meet with the UC researchers. Management recognized that simply by conducting the research, expectations of responsiveness would rise, and reporting the findings promptly provided a means of assuring respondents that their concerns had been heard. It also established a basis for management to seek public and employee involvement in solving communications problems it could not solve alone.
Data from the community assessment were used to revise Fernald's Community Relations Plan, a regulatory required document that identifies ways in which the DOE will involve the public in decisions at the Fernald site. Moreover, assessment findings were applied to evaluate individual public involvement program elements. Established program elements were examined to determine their value given the assessment results, and new or refined techniques were adopted. The following examples of planned activities are illustrative of how the assessment has shaped Fernald's public involvement program:
Continued emphasis on person-to-person communication is planned to sustain the level of satisfaction expressed by community leaders with the information received. This communication is occurring either directly between Fernald decision-makers and

community leaders or through the Fernald Envoys, a group of DOE and FERMCO employees who have volunteered to develop relationships with community opinion leaders. Reaching a broader audience with more easily understandable information is being pursued by expanding distribution of a new, summary level community newsletter, the Fernald Progress. A community access telephone line was established as an alternative to public meetings for informing citizens. To address the need to inform the general public more effectively, an information campaign emphasizing mass communication tactics is planned.

A strategy for coordinating public meetings among Fernald Operable Unit project teams and other projects is planned to avoid "stakeholder burnout." This acknowledges community leaders' level of satisfaction with the current involvement and the increasing demands for public involvement as the project's formal decision-making process nears its conclusion.

A commitment to public involvement during remedial design and remedial action was made in the revised Fernald Community Relations Plan. The plan outlines continuing public participation throughout all phases of environmental restoration of the Fernald site.

Elements of FERMCO's employee communications program were similarly examined and adjusted. Example initiatives and program enhancements taken in response include:

A series of "cascading" meetings to establish supervisors as chief communicators with employees was planned and conducted. Top management kicked off the series at an All Supervisory Meeting where information was provided on topics employees expressed a high need to receive. Supervisors were asked to hold meetings with their work groups within ten days to convey the information throughout the organization, and questionnaires were provided for employees to indicate whether the meeting met their information needs. Small group follow up sessions with supervisors were planned to share and discuss the information needs, barriers, and recommended communication processes.

Small group employee roundtables were continued weekly. In these meetings, top management meets face-to-face with employees and discuss issues affecting Fernald.

Targeting information media to the needs of various audience segments was planned, including publishing a single monthly employee publication and publishing abstracts of available information weekly on the electronic mail bulletin board.

Providing more information on human resource policies and budget and restructuring decisions was planned by continuing or establishing regular feature articles or special editions of employee publications on these topics.

NEXT STEPS

Fernald's planned public involvement and communication program activities, including the examples described above, are in various stages of implementation. Each has been designed to accomplish specific communications objectives. Project-specific communication plans now specify measurement at predetermined stages of implementation to assess whether objectives were reached. Qualitative and quantitative data will result, enabling both subjective and objective evaluation of DOE and contractor performance in meeting public involvement and communications objectives. Initially, these results will be compared to the benchmark established by the 1994 community assessment and communications audit to determine effectiveness.

To provide added value for future communications planning, current emphasis is on developing formal measurement and evaluation of the overall impact of program activities on the relationship between the Fernald organization and affected publics. Coorientation research techniques are being developed to ascertain the level of agreement and accuracy between views expressed by people both inside and outside the Fernald organization. Using this approach, gaps in one side's perceptions of the other's views and the other's actual views can be identified and factored into planning. This is particularly important in a public involvement communications environment where the ultimate objective is to bring organizational and public interests into alignment.

CLOSING THOUGHT

During times of budget constraint, public relations research gives managers the information they need to confirm the value of public involvement programs and make informed choices about where to apply resources for maximum impact. The Fernald experience is showing this to be true in the DOE communications environment.

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COMMUNITY COLLEGE NETWORK (C2NET)
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ABSTRACT
In 1990, a proposal was made to the U.S. Department of Energy (DOE) that community colleges associated with DOE facilities be given the means to interact with DOE and each other to more rapidly and consistently develop programs that deal with DOE's workforce restructuring. A specific objective in the initial proposal was the development of associate degree curriculum in environmental restoration and waste management. Representatives of eight colleges attended initial meetings in Albuquerque, New Mexico, in July and December of 1991. Since then, meetings have been held each fiscal year, with two meetings each in fiscal years 1993 and 1994. The Community College Network, or C2NET, helps member colleges progress in educational programs, while remaining flexible to deal with DOE's evolving

assessment of its workforce needs. Twenty colleges currently are members. C2NET uses networking and clearinghouse methodologies to allow each college to move forward with its own program and to develop consistency among the colleges' curricula. Semiannual meetings provide DOE Headquarters workforce perspective, lessons learned by the colleges and their facilities, and a forum for joint curriculum work. Site-specific meetings foster the relationship of each college with the local DOE office and its contractors. The Grand Junction Projects Office (GJPO), as the administrator for C2NET, maintains a database of information on curricula, courses, and instructional materials of the various colleges and makes reports available to all members. Bulletins issued quarterly provide current information on DOE workforce needs, college curricula, and C2NET activities.

During the last 4 years, DOE has continually modified its appraisal of its environmental workforce. The DOE focus has evolved from supplying several hundred thousand scientists, engineers, and technicians by "filling the pipeline" with students to "retraining and retaining" displaced weapons-plant workers. Two main reasons caused the shift in focus: 1) the end of the Cold War and the reduction of the nuclear stockpile and 2) a more experienced understanding of the personnel needed for environmental cleanup.

In fiscal year 1994, C2NET sought and was granted funding from the DOE Task Force on Worker and Community Transition. Guidance for the organization is now under Section 3161 of the 1993 Defense Authorization Act with its requirements for alleviation of the impact of DOE defense mission cutbacks on workers. The current focus is on development of a condensed curriculum to retrain displaced workers to work as environmental technicians. To do this, the colleges are identifying the core competencies from the collective associate degree programs. The goal in fiscal year 1995 is to produce a retraining curriculum that can be taught in the time frames allowed under Section 3161.

BACKGROUND

In 1989 and 1990, the U.S. Department of Energy Grand Junction Projects Office (DOE-GJPO) in Grand Junction, Colorado, collaborated with Mesa State College (also in Grand Junction) to develop the curriculum for an associate of applied science degree in Environmental Restoration Technology. That effort evolved out of the GJPO need for qualified environmental technicians and technologists and the desire to retrain some of the GJPO technicians who had only high school educations. Personnel from the GJPO operating contractor, Rust Geotech, performed most of the preliminary curriculum development in support of the Civil Engineering Technology staff at Mesa State.

During the development process, we sought help from DOE and from other colleges that might have established similar programs. In doing so, several elements became apparent:

- DOE had limited knowledge about community colleges and, therefore, had little interaction with them.
- All DOE sites we identified had a nearby community college, sometimes more than one.
- A few of those community colleges already were developing similar environmental programs, many more were considering such programs.
- Almost all of the colleges were approaching curriculum development on their own with little help from other colleges or from DOE.

In fact, some colleges with environmental programs in place had success with placement of graduates with DOE contractors but still could not get acknowledgment, let alone definition, of workforce needs. It was clear to us, if a number of community colleges were going to develop programs to support DOE's environmental workforce needs, then DOE would benefit by providing those colleges with information and by helping them to work together to develop those programs more rapidly and consistently.

In 1990, a Technical Task Plan submitted to the DOE Office of Technology Integration and Environmental Education Development (EM-522) proposed providing the community colleges with an annual meeting to share curriculum information with each other and to gather current DOE information on workforce needs. We also proposed to act as a clearinghouse of information on curricula and colleges. Funding was approved late in fiscal year 1991. In July and December of 1991, meetings were held in Albuquerque, New Mexico, with DOE representatives and faculty from eight of the community colleges associated with DOE facilities. The twofold purpose of the meetings was to

provide DOE with the capabilities, roles, and needs of the community colleges and to help the colleges gather information about DOE's perception of its workforce needs. College representatives also stressed their institutions' capabilities to perform services for DOE in nondegree areas such as training and research.
Even at that early stage, the college participants in those meetings articulated several distinct needs:
- Accurate and current information on DOE workforce needs from authoritative sources.
- Better and more official communication with DOE at the facility level.
- DOE expertise, especially in the form of adjunct faculty.
- Start-up funding when DOE wants a capability developed in a hurry.
- Equipment for highly technical courses.
- Internships for faculty and students.
- Scholarships.

A spinoff of the 1991 meetings was a DOE EM-522 attempt to initiate a "community college program." A notice of program interest was published in 1992, and funds were designated. The program's concept was that colleges would compete for grants or cooperative agreements. On what basis the colleges would compete for work at a national level when their roles were local was not established. The program did not proceed.
Our national community college program really became what it is in 1992. We began to compile a database of information on the colleges and their curricula, course content, and instructional materials. We contacted additional colleges and hosted a meeting in Denver, Colorado, in August. Thirteen colleges received invitations, and most sent representatives to the meeting.
The Denver meeting set the tone for what has truly become the DOE Community College Network (C2NET). Even though rough at times, the meeting was productive in presenting a variety of viewpoints on DOE's workforce needs and its need for the services of community colleges.
Fig. 1.
It became apparent to the colleges that DOE did not speak from one mind on these matters. One issue to be overcome was terminology with respect to "workforce." One speaker stated DOE's needs for people as being for DOE Federal employees only and predicted no real hiring of technicians. Other speakers saw the need as including contractor and even subcontractor personnel with the need for community college graduates increasing accordingly. In addition, two authors of separate assessments of DOE workforce needs were featured speakers at the meeting.
The meeting was the first time we presented the DOE-related speakers in a panel format and encouraged questions from college and industry participants. Some lively discussion resulted, even among the panelists. In addition, representatives from some of the colleges with active programs were asked to speak on educational or retraining accomplishments and lessons learned, again in a panel format. This time, with the DOE and contractor attendees asking questions, the discussion again led to what the community colleges could provide for DOE and what DOE needed to do to enable the colleges to meet DOE's needs. These topics continue to be a common theme of discussion in all meetings to date.
The Denver meeting was also the first time we formally facilitated with the assistance of a facilitator and recorded the meeting discussion on a flip chart. All meeting participants received reductions of the flip chart notes of the discussion. We found this format preferable to transcribed notes as it more accurately captures the "flavor" of the discussion and evokes discussion that was not recorded. Because of the large amount of information provided and the ensuing discussion, only a limited amount of time was dedicated to work on curriculum at that meeting. In fact, it became apparent that there was too much to be accomplished in a 2-day meeting. One of the needs expressed by the college representatives to DOE was for two meetings each yearone for information sharing and one for intercollege work on curriculum development.
Subsequent meetings were held in Albuquerque, New Mexico (spring 1993); Grand Junction, Colorado (summer 1993); Oak Ridge, Tennessee (spring 1994); and Idaho Falls, Idaho (summer 1994). We have continued the format of one meeting each year for information sharing and one for curriculum development.

CURRICULUM DEVELOPMENT

At the spring meeting, a curriculum development meeting, in Albuquerque in 1993, the

college representatives agreed to seek consistency among their various and diverse curricula. The first idea on how to build that consistency was through "competencies," the smallest building block they could identify. Competencies, as used by the C2NET colleges, refers to discrete elements of learned knowledge or skill that are single teaching objectives. The consensus of the C2NET participants, at that time, was the disparity among colleges in the curriculum structure and content of courses was too great to use as starting points.

Because of the isolated way the environmental degree programs had been developed, courses with the same name had different content at different schools, courses with different names had the same content, and subject matter contained in one course at one college was distributed in several courses at another. The typical example of the latter is Environmental Law and Regulations, taught as a single course at some colleges but distributed as supporting material in several courses at other colleges.

The first attempt to extract competencies was abortive. Only two schools had really appraised their curriculum in those terms because their governing systems required development of a competency-based curriculum. In addition, the individual who was driving the competency effort resigned during the information gathering effort.

The 1993 summer meeting in Grand Junction was an information-sharing meeting. During the working portion of that meeting, participants discussed that competencies without a framework or agreed parameters were too difficult to build. The discussion also led to the idea of a "core" of technical courses. The idea of courses again was problematic because of the mismatches among colleges. However, the concept arose that a theoretical core of technical courses, containing those subject areas and competencies that were common and essential, might be determined by consensus. The time was too short to expand on those ideas, but a diagram of a common core of courses was drawn and kept as the basis for the next year's discussions.

The agendas of the two 1994 meetings were reversed to accommodate the introduction of a new funding sponsor, the DOE Task Force on Worker and Community Transition. The spring meeting in Oak Ridge had only a half-day working component. Within that time frame, we revisited the idea of technical core courses. Through much discussion, the participants approved a preliminary list of 10 courses. To satisfy some of the concern about terminology, we agreed to refer to these as "course equivalents." The colleges are interested in attaining common academic content, not common courses. They agreed to submit their lists of competencies within that framework of course equivalents.

Our new funding sponsor was interested in the curriculum development and participated actively in the working portion of the meeting. The sponsor's interest was expressed in terms of wanting the capability to retrain displaced weapons plant workers to perform environmental work. That training might have to take place within a time frame as short as 120 days.

The 1994 summer meeting in Idaho Falls, Idaho, was truly a working meeting. Minimum time was devoted to speakers and formalities. Several of the colleges submitted competencies that fit within the 10-course equivalent framework. We compiled those competencies into a combined set, unedited, with much redundancy. We used the meeting as a facilitated session to condense the list, to eliminate redundancies, and eventually to achieve consensus. The meeting was difficult, cordiality was tested, but we accomplished our goal.

Our intent is to proceed down two paths. In terms of building commonality into the associate degree programs, the speed of C2NET seems to be workable and suitable to the colleges. Their change processes could not move much faster. The commonality ostensibly will be expressed in terms of a crosswalk of competencies. The benefits are that course work will be transferable from one college to another and that one degree will be comparable to another and considered equivalent in terms of competencies. Beginning with the 1995 meetings, the idea of such a crosswalk will be introduced and developed

The other path will be a shorter term project and is in response to DOE's need for retraining of displaced weapons plant workers in as little as 120 days. Section 3161 of the 1993 Defense Authorization Act aims at mitigating the effects on workers and communities as the DOE workforce is restructured at defense nuclear facilities. Section 3161 was written because of the anticipated worker displacement that is due to the "end of the Cold War," the reduction of the nuclear stockpile, and the resulting cutbacks in DOE's weapons production program.

One of the requirements of Section 3161 is the retraining of displaced weapons plant workers to perform environmental cleanup work. DOE funded the Accelerated Retraining Curriculum (ARC) project, which will use the core competencies from C2NET to develop a 16-week retraining curriculum. Curriculum development started in January 1995, and five C2NET colleges will receive subcontracts for the ARC project this year. Developers will build the curriculum around the 10 core technical courses from C2NET with a modular structure that is suitable for a number of time formats, the shortest and worst case being 16 weeks.

WORKFORCE INFORMATION

One of the basic needs of the community colleges associated with DOE is to understand DOE's workforce needsthe need of any community college associated with any industry. Unfortunately, DOE has, in the past, not used or communicated with the community colleges to any great extent. Recently, DOE has been in a transition of its mission that has caused it to be unclear about its workforce needs. When C2NET started, DOE was on parallel missions to continue with its nuclear-weapons-related mission as well as to clean up the environment at its facilities. The best information available on workforce needs at that time indicated there would be a great shortage of technical environmental personnel for years to come. On that basis, many programs were initiated to move people into the appropriate educational "pipelines" to meet the perceived need.

Developments associated with the end of the Cold War caused DOE to plan for reconfiguration of its weapons production system and reduction of the nuclear stockpile. This change in scope meant reduction in the weapons production mission of DOE, consolidation or closure of some facilities, and associated downsizing in its workforce. Communities and congressional districts where facilities would be lost or downsized expressed great consternation about the changes.

Action by some legislators directed DOE to alleviate the impact of such moves on the existing workforce. Section 3161 of the 1993 Defense Authorization Act requires DOE to retrain displaced "Cold Warriors" for environmental cleanup work as well as other mitigation measures. The workforce picture changed from needing new people to retraining existing people. As you may imagine, this picture change has not been altogether simultaneous or unanimous within DOE.

After 2 years of struggling with the implications of Section 3161 and its retraining requirement, it is only clear that retraining is required for compliance. It is not clear what jobs will be available for the retrainees, especially at their previous salaries. At this point, the emphasis is on developing the retraining programs so they are available for those who want to be retrained under a facility restructuring plan.

As the workforce-needs picture has changed in the last 4 years, change will continue until the needs picture coincides with reality. We will retrain some people. People earning upward of $50,000 per year in weapons production are probably not going to be retrained for $25,000-per-year environmental technician jobs. Many, if not most, of the higher technical environmental positions are already filled. Most workforce projection is still done on a budget basis rather than on a technical-needs basis. As DOE gains more experience in actual cleanup, its workforce picture will become clearer. C2NET has tried to present a broad picture of DOE's perceptions of its workforce needs to the colleges and will continue to do so as long as possible, or until it is no longer necessary.

CLEARINGHOUSE

The main component of the clearinghouse is an information database on the colleges and their curricula, course content, and instructional materials. The database, or set of relational databases, is in Paradox and is maintained by Rust. Our data gathering method was, and still is, telephoning the colleges quarterly to update and gather new information. Given the schedules of the college personnel, we are on the telephone almost dailywe are very persistent. We have had many requests for information from colleges, mostly from those not associated with C2NET, that are trying to start similar programs.

One of the initial efforts was to act as a broker for communications among colleges. If one of the colleges needed some information, a representative would call us and we, in turn, would contact other colleges in search of the information. In fact, the number of times this happened was to be one of our success measurement criteria. After an early increase in such activity, it decreased, then died almost totally. In trying to determine why, we were told that once the college personnel became

acquainted, they saw no reason to include us in their conversations. We had to consider our broker endeavor a success, even though our measurement criteria indicated it was a failure.

As a clearinghouse, we also issue a quarterly bulletin to update our members and stakeholders on C2NET activities. The bulletin also includes updates on college accomplishments and programs and on other items of interest to the colleges, such as funding sources.

DOE SUPPORT

One of the services we provide for DOE is representation of the DOE-associated community college perspective and the C2NET program at DOE meetings and workshops and through information dissemination. We also try to represent the community college resource in DOE discussions on such issues as training and workforce planning. Community colleges have been so underutilized by DOE that it sometimes does not regard them as a ready and economical resource unless it is brought to its attention.

LESSONS LEARNED

Along the way some hints on how to improve the coordination of these activities have become obvious to us, sometimes painfully:

Meeting room configuration and seating arrangements are important factors. In Denver, two columns in the room that were shown as very thin on the hotel floor plan turned out to be 3-feet thick and blocked participants' views. U-shaped seating permits better discussion than classroom (parallel rows of chairs) seating.

To manage a successful database, you must aggressively update the database as well as aggressively seek to disseminate the information. If it's not used, it's just data.

Community colleges have very limited funds for travel. We have paid travel expenses for college representatives to attend our meetings.

Arriving at a product by consensus is difficult work. Longer work sessions at one meeting are just not productive. Two 4-hour work sessions, with one 4-hour information session and one field trip, is about the length of time that people can stay enthusiastic.

DOE is very interested in working with community colleges but just not experienced. A culture populated with scientists and engineers has had a natural affinity with universities and sometimes cannot envision what a community college can do that a university cannot.

Not all community colleges are equal. We have had to arrive at a working definition. We consider a community college as one that offers associate degrees and interacts directly with local industry and the community for training and other human-resource development needs. Some of our C2NET colleges have different names, such as technical college or state college. Some are 4-year colleges that also serve as community colleges because the size of the community will not support two colleges.

THE FUTURE FOR C2NET

Community colleges exist near the doorstep of each DOE facility. They are an economical and "tappable" resource for education, training, and other human-resource development needs. C2NET provides DOE with a single, convenient way to interface with that group of colleges and provides the colleges with a forum for working together to develop training and curricula to meet DOE's needs. As DOE facilities and their contractors find themselves smaller and unable to support large internal training and other human-resource development capabilities, community colleges will be there to fill the need. We see C2NET continuing to evolve to help DOE articulate its workforce needs to the colleges. We also see C2NET continuing to facilitate interaction among the colleges. The hope is that each college will eventually be a true partner, supporting its DOE facility. Until then, C2NET will have a role encouraging their interaction. Perhaps the future for C2NET will be a true consortium of colleges doing business with DOE.

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DETERMINING TRAINING NEEDS FROM SUPERVISORS' ASSESSMENT OF STAFF PROFICIENCY IN TASKS AND SKILLS*

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ABSTRACT
To provide the basis for establishing training opportunities, this project investigated supervisors' views of three components of staff activities. The project established the tasks that staff perform, identified staff's level of effectiveness in performing these tasks, and investigated staff's level of proficiency in performing the skills underlying these tasks. Training opportunities were then determined in those areas where knowledge and skills could be improved for staff to perform their tasks more effectively.
Staff currently perform their tasks sufficiently well. Furthermore, supervisors indicated that for the most part staff do perform the tasks they should perform. In carrying out these tasks, staff use primarily critical thinking, problem solving, and communication skills rather than discipline-specific skills. Although staff generally have working knowledge of most of these skills, additional training in critical thinking and problem solving, program and project management techniques, and communications is appropriate to further improve the organization's effectiveness.
INTRODUCTION
To successfully implement environmental restoration, federal agencies must have skilled staff to carry out the range of activities and tasks needed to achieve the agencies' missions. This project investigated the training needs of an environmental organization by examining the tasks that the staff must perform, staff effectiveness in performing these tasks, and staff proficiency in the skills and knowledge necessary to perform these tasks.
This project sought answers to the following questions:
  What tasks do staff currently perform and are these all or nearly all of the tasks they should perform?
  How well do staff perform these tasks?
  How proficient are staff in applying the knowledge and skills needed to perform these tasks?
Answers to these questions provide guidance on what training would most benefit and improve staff's knowledge and skills.
The organization examined in this project is one headquarters office within the U.S. Department of Energy (DOE), the Office of Environmental Management (EM). This office is responsible for directing and overseeing programs rather than managing and implementing specific projects. Under this mission, DOE EM is charged with carrying out tasks under the broad functional areas of award fees, budget, communications, documents, guidance, integration and coordination, operations and contractor management, quality assurance, policy and planning, and safety (1).
METHODS
This section briefly describes the procedures used to examine tasks and the associated knowledge and skills needed to accomplish them. These procedures are further discussed in a complementary study by Young and Hensley (2).
The 4 supervisors* of a headquarters organization of DOE EM completed a workbook that asked questions about the tasks their 24 staff members perform and the knowledge and skills needed for these tasks. The workbook listed 35 tasks and 57 knowledge and skill areas that staff might use to perform their daily activities. Supervisors specified the number of staff who perform each task. In addition, even though staff might not currently perform a task because of limitations such as work priority or budget constraints, supervisors specified how many staff members should perform each task. Supervisors further indicated how well staff perform each task and staff levels of expertise in relevant knowledge and skills. Responses from the supervisors were keyed into a spreadsheet and tabulated. The data were then examined to determine the following:
  Proportion of tasks that staff currently perform and should perform,
  Mean scores in staff's effectiveness in performing their tasks, and
  Staff level of knowledge and proficiency for skills associated with these tasks.
A previous study (2) provided baseline information on the knowledge and skill areas necessary for each task. Training opportunities were determined and prioritized by looking at the knowledge and skills needed for tasks performed least well.

RESULTS

This section presents the supervisors' views on the tasks that staff currently perform and those tasks that they should perform. It also describes staff levels of effectiveness in performing tasks. Finally, this section discusses the knowledge and skills needed to perform

The various tasks that staff currently perform and staff proficiency levels in performing these skills.

What Tasks do Staff Perform and are These all or Nearly all of the Tasks They Should Perform?

Environmental restoration tasks needed to carry out the mission of a headquarters EM organization are quite diverse (Table I). These tasks can be grouped into broad categories that range from communication activities, such as preparing and reviewing documents; to policy tasks, such as "advise, assist, and provide authoritative guidance"; to administrative activities, such as monitoring schedule constraints and managing files and records; to technical oversight tasks related to the National Environmental Policy Act, land use, and waste management.

TABLE I

The organization assessed in this project is functioning essentially as it should. Supervisors indicated that, for the most part, staff are performing the tasks that they should (Table I). Furthermore, these tasks are generally consistent with staff position descriptions and with the roles and responsibilities delineated for EM (1). Although a majority of staff currently perform all these tasks, a larger proportion of staff should be interfacing with other program elements both within and outside of the organization, conducting briefings and presentations, and planning for budget and personnel required to implement projects (Table I). In addition, more staff should be overseeing several aspects of headquarters operation: land use activities; decontamination and decommissioning; independently verifying that cleanup and decontamination activities are adequate; and checking surveillance, monitoring, and inspection activities.

How well do staff perform their current tasks?

For any organization to carry out its mission effectively, staff need to be proficient at the tasks they perform. Therefore, the project looked at how well staff accomplish these tasks as well as whether the tasks performed by most staff are also the tasks in which they are most effective. In general, those tasks that the largest proportion of staff perform are also those tasks that they perform most effectively (Table II). These include "coordinate document reviews"; "interface within and outside of DOE"; "prepare and review documents"; and "advise, assist, and provide authoritative guidance." However, four tasks that a relatively large percentage (75%) of staff should perform are performed between somewhat and sufficiently well (mean scores 2.95 to 2.74) (Table II): "establish cost, schedule, and technical baselines"; "provide analyses to determine effects of remediation projects"; "develop, coordinate, and disseminate program management policies and procedures"; and "formulate, define, and modify budget." Training would be appropriate for the underlying skills associated with these four tasks to improve proficiency because they are performed by a majority of staff.

TABLE II

What Knowledge and Skills do Staff Need to Perform Their Tasks?

For an organization to operate smoothly, staff should be proficient in all knowledge and skill areas needed to carry out the tasks they perform. A previous study determined the knowledge and skills needed for each task (2). Because the tasks that staff perform for environmental restoration are complex, the knowledge and skills needed to perform these tasks are also quite diverse (Tables III, IV, and V). Thirteen knowledge and skill areas applied to 12 or more of the 35 tasks that staff currently perform (Table III). These include knowledge of the DOE mission, keeping abreast of current developments, defining criteria applicable to solving problems, and being able to work within schedule constraints. As Table III illustrates, skills needed to carry out the majority of tasks are not technical or regulatory in nature, such as knowledge of the Resource Conservation and Recovery Act; rather, these skills are generally in the categories of problem solving and critical thinking. Table IV further supports the finding that communication skills, program management techniques, and process evaluation skills are needed to perform several tasks successfully. In contrast, several unique knowledge and skill areas are needed to carry out only a few tasks (Table V). These specialized skills include knowledge of

radiation effects, health physics issues, land use issues, and emergency response procedures. Tasks associated with these specialized knowledge and skill areas are performed by fewer staff members. Consequently, although training in these areas may be appropriate for selected staff members, it would not necessarily provide the greatest benefit to the whole organization.
TABLE III
TABLE IV
TABLE V
How Proficient are Staff in the Relevant Knowledge and Skills needed to Perform Environmental Restoration Tasks?
Once the relevant knowledge and skill areas associated with 35 tasks that staff perform were identified (2), supervisors rated staff proficiency levels for each of the 57 knowledge and skill areas needed to carry out these tasks.
In a smoothly operating organization, staff first should be proficient in the knowledge and skill areas that are needed for most tasks they carry out. Once staff are proficient in these widely applicable skills, they can be trained in skills that apply to only a few tasks. Staff proficiency levels for all skills are adequate, although not outstanding (mean scores 3.6 to 2.6 on a 4-point scale) (Table VI). Staff are proficient in several of the skills common to many tasks (Table VI). Skills common to many tasks in which staff are quite proficient include working under schedule constraints, keeping abreast of current developments, providing clear guidance and direction to accomplish tasks, and prioritizing based on criteria.
TABLE VIa
TABLE VIb
On the other hand, staff proficiency levels are only moderate for some knowledge and skill areas that are common to many tasks (Table VI). These include knowledge of the DOE mission, knowledge of DOE management strategies, the ability to write succinctly and clearly, and the ability to assess strengths and weaknesses of alternatives (Table VI). Furthermore, staff are quite knowledgeable about the National Environmental Policy Act process and waste treatment/disposal technologies even though these skills are not common to many tasks (Tables V and VI).
CONCLUSIONS
Supervisors in this headquarters office of DOE EM indicated that most of their staff perform the majority of their tasks quite effectively. Supervisors further indicated that, collectively, staff have working knowledge, although not expert understanding, of the subjects and skills associated with each task they perform.
Thus, training is suggested for the knowledge and skill areas associated with those tasks performed by most staff that are performed relatively least well. These knowledge and skill areas fall under the general categories of program and project management, critical thinking and problem solving, and communications. Program management skills include knowledge of EM's mission, DOE management strategies, and defining applicable criteria. Many of these skill areas are dynamic, so training is necessary to keep staff current.
Critical thinking and problem solving skills include examining trade-offs among alternatives, comparing existing conditions to preset standards, and assessing strengths and weaknesses of alternatives. Communication skills include the ability to write clearly and succinctly, interpersonal facility, and providing clear guidance and direction to accomplish tasks. Because these are the primary skills common to many tasks (Tables III, IV, and V), training in these areas is appropriate to help the organization continue to function effectively.
The knowledge and skill areas related to communications, problem solving, and project management have been further prioritized on the basis of the margin of improvement needed to achieve expert knowledge or highly proficient skill levels and how many times the skill is needed (Table VII). Training in the knowledge and skill areas identified in Table VII will strengthen this organization so it can better carry out its roles and responsibilities. This training will further benefit the organization by allowing staff to reach their potential and make their jobs more satisfying (3,4).
TABLE VII
Different types of training opportunities should be explored. Existing training activities should be continued, and new training programs should be developed to address the diverse skills that underlie the tasks that staff perform. Supervisors should look beyond traditional training courses to improve performance. For example,

supervisors may empower their staff by sharing opportunities usually performed by management, such as conducting briefings on projects with which staff have been greatly involved and delegating authority to staff members who are capable of performing new tasks. Job proficiency may also be improved by pairing experienced staff with less experienced staff, using in-house mentors, and encouraging short-term assignments in other organizations.

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Session 50 -- Characterization of Radioactive & Mixed Wastes - Session I
Co-chairs: Gary B. Semones, EG&G-RF;
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50-1

DEVELOPMENT OF RADIOLOGICAL PROFILES FOR U.S. DEPARTMENT OF ENERGY LOW-LEVEL MIXED WASTES*

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ABSTRACT

Radiological profiles have been developed by Argonne National Laboratory for low-level mixed wastes (LLMWs) that are under the management of the U.S. Department of Energy (DOE). These profiles have been used in the Office of Environmental Management Programmatic Environmental Impact Statement (EM PEIS) to support the analysis of environmental and health risks associated with the various waste management strategies. The radiological characterization of DOE LLMWs is generally inadequate and has made it difficult to develop a site- and waste-stream-dependent radiological profile for LLMWs. On the basis of the operational history of the DOE sites, a simple model was developed to generate site-dependent and waste-stream-independent radiological profiles for LLMWs. This paper briefly discusses the assumptions used in this model and the uncertainties in the results.

INTRODUCTION

The U.S. Department of Energy (DOE) Office of Environmental Management (EM) has undertaken a Programmatic Environmental Impact Statement (PEIS). This EM PEIS addresses the environmental and health risks associated with the management of DOE radioactive and hazardous wastes. The treatment, storage, and disposal (TSD) of radioactive low-level mixed wastes (LLMWs) represent a major effort in the EM PEIS. LLMWs are wastes that are both hazardous under the Resource Conservation and Recovery Act (RCRA) (Title 40, Part 261 of the Code of Federal Regulations (CFR)) and meet the definition of low-level waste (LLW) under DOE Order 5820.2A (1).

The major sources of LLMW under DOE management considered in the EM PEIS are 1) currently stored LLMWs awaiting treatment and projected generated wastes from future operations, that is, Waste Management (WM) LLMWs, and 2) Environmental Restoration (ER) wastes, which are obtained from site restoration and decontamination and decommissioning (D&D) of DOE facilities, and sent to WM for TSD. Approximately 430,000 m3 of WM LLMW are expected (inventory and future generation) through 2014. More than 2,000 waste streams of LLMW at 44 separate DOE sites are identified in the 1994 Mixed Waste Inventory Report (MWIR-2) (2). In the ER "semirestricted" alternative, it is projected that the volume of ER-derived LLMW needing treatment will be approximately 7.3 million m3 through 2030. The specific ER secondary waste streams that will be sent to WM have been described in the Automated Remedial Assessment Methodology (ARAM) database provided by Pacific Northwest Laboratory (PNL) (3,4).

To support the EM PEIS effort to determine environmental and health risks associated

with the TSD of DOE LLMW, Argonne National Laboratory has established a methodology for estimating radiological and chemical profiles for the more than 2,000 waste streams at 44 DOE sites. Radiological characterization of these waste streams given in the various DOE databases on LLMW is clearly inadequate for the stated task. The databases can, however, be useful in verifying and providing data to establish some simplifying assumptions.

LLMWs are classified into four separate radiological categories. The largest volume (>99%) is contact-handled (CH) waste, which has a surface activity of less than 200 mrem/h. Remote-handled (RH) waste has a surface dose greater than 200 mrem/h. CH wastes are treated separately from RH LLMW. The type of radiation emitted also determines LLMW classification and can affect environmental and health risks. The majority of LLMW (approximately 75%) is non-alpha waste (less than 10 nCi/g transuranic (TRU) radionuclide alpha activity). The remaining 25% of LLMW is alpha LLMW (between 10-100 nCi/g TRU alpha activity). Because of regulatory concerns, the handling and routing for treatment of all LLMW are performed separately for alpha and non-alpha wastes.

The following sections of this paper cover the procedures and assumptions used to estimate the radiological profiles for LLMW for each site and include a discussion of the uncertainties associated with these results. The estimated chemical profiles associated with LLMW are presented in another paper for this conference (5).

RADIOLOGICAL PROFILES FOR LLMW

Determination of radionuclide concentrations for LLMW is difficult because there is insufficient information in MWIR-2 or in the current Waste Management Information System (WMIS) databases to quantify the concentration of the various radionuclides. Given the current data, it is not possible to construct precise radiological profiles for each of the LLMW streams at all sites.

The operational mission(s) is primarily responsible for the radionuclides generated at each site. The presence or absence of RCRA contaminants in a waste stream distinguishes LLMW from LLW and is not very likely to affect the relative proportion of radionuclides in the waste. Given that the same mix of radionuclides tends to contaminate both LLW and LLMW at a site, it is assumed that radiological profiles for LLMW are site-dependent but waste-stream-independent. The LLMW radiological profiles developed in the EM PEIS are derived from information on the radiological content of LLW contained within the 1991 Integrated Data Base (IDB) (6).

The IDB has information, accumulated over many years, on the operational history of the larger DOE sites. This information on each site is subdivided into waste volumes, activity concentration (Ci/m3), and source of activity. Because the data have been accumulated over a number of years, they represent the cumulative result of the various DOE waste-generating activities over those years. Five major DOE programs generate distinct sources of radioactivity. These sources are 1) fission products, 2) induced activity, 3) uranium/thorium, 4) TRU alpha, and 5) tritium. The IDB data provide radiological profiles near the time of generation for each of these five operations that produced LLW (see Table I).

TABLE I

In estimating the radiological profiles for the LLMW in the EM PEIS, it has been assumed that at the time of waste generation, LLMW radiological profiles for each site are the same as those for LLW because the same operational mission(s) is responsible for generating both LLW and LLMW. Furthermore, it has been assumed that radiological profiles are the same for all waste streams at a given site. Treatment facilities for LLMW are not generally expected to be available for treatment until 2003, whereas many LLW treatment facilities are already in operation. Thus, the time interval between generation and treatment will be much greater for LLMW than for LLW. Therefore, it is much more important to account for radioactive decay for LLMW than for LLW. LLW is normally disposed of soon after generation; thus, its radiological profile at the time of disposal will not be much different from that at the time of generation. On the other hand, because the proposed treatment of LLMW is to begin in 2003, the age of the LLMW at the time of treatment will be much greater than that of LLW. Therefore, the radioactive profile of LLMW at the time of treatment will differ considerably from its profile at the time of generation.

To estimate the time interval between LLMW generation and treatment, the treatment date for all LLMW has been assumed to be 2008 (the midpoint of the proposed treatment period of 2003 through 2013). The average generation date estimates are different for different sites and for the three different waste groups, that is,

current inventory, waste projected to be generated over an ensuing 20-year period, and ER waste. The generation dates were arrived at by estimating the average date (given site history) of radionuclide generation, except for the uranium/thorium source term. For the uranium/thorium source term, the generation time is defined as the time of the most recent chemical or isotopic concentration/isolation of the material.

Thus, the principal assumptions on which LLMW radiological profiles were estimated are as follows:

At the time of waste generation, LLMW radiological profiles for each site are the same as those for LLW at the same site.

At the time of waste generation at a given site, activity concentrations (Ci/m3) are the same as those for LLW.

At the time of treatment, the radiological characteristics of LLMW will be different from those of LLW only because of the greater age of the waste.

The treatment date for all LLMW is 2008.

Radiological profiles are assumed to be independent of the waste stream category at a particular site.

The average generation date of LLMW depends on the site and on whether it comes from inventory, projected generation, or ER.

The information on LLW does not distinguish between alpha-contaminated (TRU content = 10 nCi/g-100 nCi/g) waste and non-alpha- contaminated (TRU < 10 nCi/g) waste. This analysis assumes a geometric mean of 33 nCi/g (TRU) for the alpha-contaminated LLMW and 33/20, approximately 1.7 nCi/g (TRU), for the non-alpha-contaminated LLMW. These values are equivalent to .082 (Ci)/m3 and .004 Ci/m3, respectively, in TRU activity (estimated density = 2,500 kg/m3 for the two types of waste).

For sites with gaseous diffusion plants (GDPs) or that store GDP waste, adjustments were made to the uranium/thorium source term by the addition of technetium-99 (Tc-99), which through carryover of gaseous technetium (Tc-99) fluoride in the GDP process, is present (in small amounts) in this specific waste type.

The proportional LLMW source term distribution by site and the site-dependent activity percentages for the CH LLMW (non-alpha) and CH LLMW (TRU alpha) are given in Tables II and III. These tables were derived from IDB data on the accumulated radioactivity (in curies) for LLW at each site. The tables list the activity percentages near the time of generation of the radionuclides. For major sites, the assumed average times of radionuclide generation/isolation are given in Table IV. The coupling of the appropriate waste ages, activity percentages, and source term radiological profiles will generate site-dependent radiological profiles and activity concentrations for LLMW. The individual site radiological profiles are reported in a technical support document (9). Table V lists the estimated radiological profiles for WM LLMW at the average time of treatment (2008) for a number of larger DOE sites. The methodology presented here predicts substantially different radiological profiles expected at the time of treatment for the different sites. Such a result is consistent with the widely varying programmatic missions of the different DOE sites.

TABLE II

TABLE III

TABLE IV

TABLE V

This procedure can be carried beyond the time of treatment of the waste to times of interest for disposed waste. Figure I illustrates the behavior of radioactive hazard with time for a typical DOE waste stream containing plutonium. This activity time plot was obtained by ordering radionuclides according to increasing half-life and plotting cumulative sums of their activity concentrations against time. The activity concentrations were multiplied by a weighing factor that represents the radiological risk to human health (10). This fig. illustrates as a function of time the ingrowth of americium-241 (Am-241) and later of neptunium-237 (Np-237). An example might be Rocky Flats Environmental Technology Site (RFETS). Here the projected radiological profile is dominated by plutonium-241 (Pu-241) near the time of treatment. Over a time scale of hundreds of years, the Pu-241 decays away, but the resultant Am-241 grows in to become the dominant radionuclide. In a time scale of thousands of years, the Am-241 decays and Np-237 becomes prominent. Such information is useful in

determining the environmental and health risks associated with disposed waste from RFETS.
Fig. 1.
The DOE sites included within Tables II and III contain more than 98% of the total LLMW. Each of the additional small DOE sites were approximated as being similar in radiological profile to one of the large DOE sites included within Tables II and III. Table VI shows the assumed similarities between the radiological profiles of the small and large DOE sites used in this analysis.
TABLE VI
The determination of radiological profiles for ER LLMW was hampered by incomplete radiological characterization of the waste in the ARAM data. Examination of ER-derived LLMW radionuclide concentrations indicated a similarity to the site-based radiological profiles used for the WM LLMW. For this study, the site-dependent ER LLMW radionuclide source terms were assumed to be identical to those used for WM LLMW with the following three differences. First, the mean time for generation of radionuclides in all ER wastes was assumed to be 1965. This produces a different adjustment to aging for the ER wastes compared with non-current-inventory WM LLMW at each site (see Table IV). Second, the longer time for radioactive decay and dilution of ER wastes by nonradioactive external materials, such as soil, old packaging, and groundwater, tends to lower the average concentration per unit volume of ER LLMW compared with similar WM LLMW. The ARAM database indicates that the average expected activity concentration (Ci/m3) for ER wastes is about 10% of the WM LLMW activity concentration estimate derived from IDB LLW information. Thus, the radionuclide concentration of ER LLMW is taken as one-tenth the activity concentration for equivalent WM LLMW at a given site. Finally, the ARAM database lists the ER waste at Sandia National Laboratories (New Mexico) (SNL-NM) as being contaminated with uranium. Thus, in Table V, the radiological composition of SNL-NM ER LLMW is similar to that of the Fernald Environmental Management Project (FEMP) and not that of the Savannah River Site (SRS) as shown for WM waste.
UNCERTAINTIES IN RADIOLOGICAL PROFILES
The major source of uncertainty in the radiological profiles is the lack of adequate quantitative radiological characterization of LLMW streams compiled in the MWIR-2 database. For the majority of the waste streams, radiological content is described only qualitatively. For example, a waste stream will be described as containing TRU radionuclides but identities and concentrations are not given. In other cases, the radionuclide quantities are given but are not appropriately aged and the waste generation date is unknown.
Because reliable data providing contaminant characterization of waste streams are not available, models were constructed for radiological profiles. The uncertainties in radiological profiles arise from the assumptions made in developing the profiles. Two assumptions in particular contribute to the uncertainties for radiological profiles: 1) radiological profiles depend only on the site and not on the particular waste stream and 2) total activity concentrations in LLMW streams at the time of generation are the same as those in the LLW streams.
The first assumption, namely that radiological profiles do not depend on the waste treatment category, is likely to introduce the largest errors. Because different radionuclides have different solubilities, their proportions in aqueous and organic liquids will be different than in solids and sludges. For example, the proportion of tritium (H-3) may be underestimated in liquid wastes and overestimated in solid wastes, thereby affecting H-3 estimates in air and water emissions. Variations in radiological profiles among the various solid waste streams are less important because such variations will have a much less significant impact on emissions source terms.
The second assumption is that at the time of waste generation, total activity concentrations in LLMW streams are the same as in LLW streams. It is reasonable to assume that LLMW streams are similar in radionuclide composition to LLW streams because the radionuclides present in both streams are generated by the same processes. However, the total initial activity concentration in LLMW is not necessarily the same as that in LLW. The uncertainty in the total initial activity concentrations will introduce uncertainties in total radioactivity content in emission source terms. However, the impact of this uncertainty on comparisons among the different WM alternatives will be less significant because the errors in total activity concentration estimates will tend to be in the same direction for all

alternatives.
Other assumptions pertain to the age of the LLMW, values assumed for the TRU content, and Tc-99 content of wastes from GDPs. Uncertainties introduced by these assumptions are expected to be of lesser importance than those arising from the two assumptions discussed above.
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50-2
CHARACTERIZATION OF LOW-LEVEL MIXED WASTES AT THE LOS ALAMOS NATIONAL LABORATORY
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ABSTRACT
Los Alamos National Laboratory (LANL) and its subcontractors recently completed a program to conduct characterization of all low-level mixed waste (LLMW) materials in storage that were accepted into the inventory prior to current-day waste acceptance criteria implementation. Characterization review was directed not only toward RCRA compliance issues, but also addressed requirements for eventual treatment or disposal of the older, or legacy, waste inventory. Waste management records for legacy wastes were found to be inadequate for a thorough characterization of all LLMW items. Further requirements to identify packaging and treatability were imposed to assess waste disposition and prioritize waste management action. Over 6,000 individual LLMW items required RCRA-certifiable characterization. This was done most economically and expeditiously through Knowledge of Process (KOP) characterization of over 95 percent of the waste items.
An environment of staff reductions, inter-divisional transfers, and personal liability concerns proved to be a challenge to those charged with obtaining the generator's knowledge of the waste. KOP interviewers were required to use existing records together with their knowledge of LANL past and present programs and

generator or waste coordinator input to construct the proper understanding of the waste-generating process. After all this information had been compiled, a robust quality assurance process for completeness, correctness and consistency allowed for characterization of the waste. Characterization was sometimes made difficult where there was not always clear guidance from a regulatory authority. The paper presents methodology for knowledge of process characterization and shows how the results provide an understanding of LANL LLMW which meets both regulatory and waste management needs.

INTRODUCTION

This paper discusses the purpose and results of a program to characterize low-level mixed wastes (LLMWs) in storage at Los Alamos National Laboratory (the Laboratory). The program was conducted to maintain regulatory compliance and to support ongoing waste treatment and disposal activities. The characterization team conducted a characterization review of wastes stored at the Laboratory that contain both a low-level radioactive and a hazardous component. The team addressed only those wastes that were generated prior to January of 1993. The wastes which were reviewed, referred to as legacy wastes, had been generated before the implementation of comprehensive waste acceptance documentation procedures which assure regulatory compliance for more recently generated wastes. The review was performed to verify existing RCRA code assignments and was required as a component of the Laboratory's Federal Facility Compliance Agreement (FFCA). The review entailed identifying all legacy LLMW items in storage, collecting existing documentation, contacting and interviewing generators, and reviewing code assignments based upon information from knowledge of process (KOP) as allowed by RCRA. Project findings were entered into tracking and characterization databases.

The characterization team identified 7,546 legacy waste items in the current inventory, and determined that 4,200 required further RCRA characterization and documentation to satisfy the FFCA requirement. The remaining waste items had already been characterized by sampling or appropriately identified for treatment by existing or planned treatment methods. KOP characterization was successful for accurately assigning RCRA codes for all but 117 of the 4,200 items within the scope of work. Sampling and analysis requirements for complete characterization of the 117 remaining items were outlined in the project report. As a result of KOP interviews, 714 waste items were determined to be non-hazardous, while 276 were determined to be non-radioactive. Other wastes were found to be stored as suspect radioactive because of administrative requirements due to their generation in an area of incident isotope usage which was not likely to have contaminated them. Many of the suspect radioactive wastes were certified by the generators as non-radioactive and will eventually be removed from the mixed-waste inventory following radioactivity screening. Table I provides a breakdown of the Laboratory's legacy wastes.

THE PURPOSE AND BASIS OF CHARACTERIZATION

Characterization of the Laboratory's legacy LLMW was necessary not only to satisfy regulatory requirements, but also to provide important waste profile information for a rigorous schedule set forth in the FFCA to treat all LLMW in the inventory. The Laboratory's FFCA currently requires that it initiate the design of two new skid-mounted treatment processes each fiscal year through fiscal year 1997 and that it have full characterization and planned treatment for every waste item in its inventory by March, 1995. The required treatment designs can not be effectively performed without prior detailed knowledge of the nature of the waste streams involved, including much more information than the RCRA code for each waste would provide. Properties such as physical state, waste matrix, packaging, and radionuclide contamination are of critical importance when designing a safe and effective waste treatment. The capture of these types of data was an equally important task to the satisfaction of RCRA in successfully completing the characterization effort.

The diversity of waste streams encountered by the team reflects the Laboratory's mission of research and development. Much of the waste in storage was generated by such uncommon research activities as nuclear rocket propulsion, biological effects of radiation, and radioisotope power sources for space stations and medical applications. The matrix and nature of these waste streams are often such that available treatment technologies do not directly apply. Much of the characterization effort was focused on identifying similarities in the diverse inventory so that organized treatment could proceed, while maintaining distinction between truly

differing wastes. Prior to the characterization task, the Laboratory's more than 7,000 legacy LLMW had been segregated into only 23 broad treatment categories, based primarily on RCRA codes. Much more definition was required to accurately plan for waste treatment, for, as the characterization data now indicates, there are more than two hundred distinct waste streams which will require unique consideration for appropriate treatment. Fig. 1 depicts an example of characterization by RCRA code versus characterization for treatment.
Fig. 1.
RCRA Characterization
The RCRA regulations allow waste generators to apply knowledge of the hazardous characteristics of the materials in the waste and the waste generating process to determine whether the wastes are hazardous. In the case of the Laboratory, this waste characterization process consisted of reviewing existing Laboratory waste data and new process knowledge information gathered from the generator interviews. The comprehensive data were then evaluated to confirm that the Laboratory-assigned RCRA hazardous waste codes were appropriate. The process knowledge approach was chosen by the Laboratory as the most efficient means for characterization for several reasons, including:
the need to maintain personnel exposure that is required for sampling and analysis as low as reasonably achievable;
the need to keep characterization costs as low as possible; and
the notion that generator-supplied information about process wastes would be more effective than typical statistical sampling, especially when evaluating RCRA-listed properties. For example, sampling and analysis cannot distinguish between D021 characteristic and F002 chlorobenzene, which have differing treatment standards.
Characterization for Treatment
In addition to assigning RCRA hazardous waste codes, the characterization process included a compilation of the data necessary to satisfy treatment technology acceptance criteria. The questionnaires which were used to conduct the generator interviews were developed with the assistance of waste treatment design staff members at the Laboratory to provide details about packaging, complete chemical constituency, and radionuclide content. The recently developed DOE treatability coding scheme was used to segregate characterized wastes into standardized waste stream types. Process knowledge interviews also verified proper assignment of waste storage codes, and Department of Transportation (DOT) container codes to ensure safe waste storage and transportation. See Table II for a list of the types of DOE Treatability codes used by the characterization team.
RETRIEVAL AND MANAGEMENT OF INFORMATION
Hardcopy Records and Knowledge of Process Forms
Documentation for generated wastes at the Laboratory had been developed over a period of time as new regulatory and waste storage requirements became more relevant issues. Legacy wastes which were generated prior to the implementation of the comprehensive waste management system had little information available in the waste management records to support RCRA or treatability characterization. For many of wastes, generators were not required to document process knowledge nor did the waste acceptance criteria always identify all the hazardous properties of the waste. The characterization team was tasked to supplement any existing waste profile information with a standardized questionnaire, which included information such as:
complete description of the waste and its generating process;
a process flow diagram, where applicable;
chemical constituents and concentrations;
detailed packaging information, both at the primary container level and at the drum level;
radionuclide identification including activity level;
information about the area of waste generation;
physical properties of the waste including:
- physical state;
- flash point;
- pH;
- reactivity; and
- PCBs;

heavy metal identification and content; and
organic content with respect to listed halogenated and non-halogenated compounds.

Because legacy drums often contained many waste items of differing chemical, physical and radiological properties, interviews were conducted on an item basis rather than on a drum basis. Often many separate sets of characterization forms were required for one drum. Some of the legacy labpacks contained as many as 50 distinct waste items. Drums of identical wastes from the same generator were grouped together where possible into one data folder. All characterization forms and related Laboratory data were assembled into packages referred to by the team as folders. Each folder was a stand-alone document for a drum or drum group. LLMW Characterization folders were organized with new process knowledge and characterization segregated from existing Laboratory hardcopy data on the two sides of a double-sided binder. Material Safety Data Sheets (MSDSs) were also included in the files and often proved to be difficult to obtain, since many of the original suppliers had long since ceased to exist. Figure 2 outlines the organization of waste information into data folders.

Fig. 2.

Quality Assurance

The assurance of accurate and precise information was the most important criteria in the success of the characterization task. Quality assurance reviews were performed during each step of the program. The first QA/QC review was conducted after folder assembly was completed, before the folders were issued to the field team for the KOP interview. Each folder was evaluated to determine if the assembled data package was consistent and to its verify status as a stand-alone document for a drum or drum group of like wastes. Particular attention was directed toward ensuring that all available existing Laboratory documents associated with the identified waste items were present.

The second critical QA/QC review was conducted on a folder-by-folder basis following successful completion of the KOP interview. The interview forms were reviewed to verify that the level of detail was adequate to verify or legitimately modify existing Laboratory data, confirm RCRA characterization, and provide additional data in sufficient detail to meet waste treatment objectives. After it was established that the folder was complete with respect to Laboratory and Team-generated documentation, a review of the existing Laboratory data was made to compare information obtained during the KOP interview with the information contained on Laboratory documentation. In general, if inconsistencies were present, the QA/QC team verified that an explanation of the inconsistency was included in the folder. A related review was conducted of the a waste summary form which would later be used to modify electronic data pertaining to each waste item. Each data field of this form, which included all of the Laboratory's databased information for a particular item, was evaluated for consistency with the KOP interview forms and existing Laboratory data. A rationale sheet was then completed to document the reason for any modification to the electronic data.

A third critical QA/QC review was conducted following waste characterization. Waste items were characterized based on a review of all physical, chemical, and KOP information available; a waste characterization form was completed for each waste item or group of similar waste items. The form identified the RCRA hazardous waste code and the rationale for assigning the code, and also listed all Laboratory-assigned waste codes that were not properly assigned and provides an explanation. The waste characterization form also contained a DOE Mixed Waste Treatability Code assignment for each waste item and provided a rationale for that assignment. All information on the form was subject to a complete QA/QC to verify all responses. Due to the specialized expertise necessary to determine proper characterization of waste items, this review was performed by an independent member of the Characterization staff using the same review procedures employed by the QA/QC staff. Fig. 3 depicts the interview, quality assurance, and characterization loop.

Fig. 3.

Conducting the Interviews

The coordination and completion of accurate generator interviews was the most challenging aspect of the program. Over 1,500 separate interviews were necessary to complete characterization at a level consistent with the FFCA requirements. Interviewing activities were organized with the use of a tracking database and were conducted by a team of eight process- and RCRA-knowledgeable engineers. Most of the field team had prior experience with certain groups at the Laboratory from earlier tasks. The project tracking database included preliminary descriptions of waste

items, as well as information about the generating groups and locations. This information allowed for efficient coordination of interviews for many different wastes and many different generators in a short period of time. Interview assignments to team members were prioritized based upon familiarity with generators, understanding of waste streams, and specific interviewer backgrounds.
Throughout the interviewing phase, problems arose when generators perceived the necessary exchange of information as a potential for personal liability regardless of the fact that they had correctly adhered to waste management practices at the time of waste acceptance into storage. Other difficulties arose when generators had retired or moved to other facilities. The team relied heavily on the support and cooperation of Laboratory management in resolving these issues. As a result, the team was able to meet with individuals who were both familiar with wastes and willing to discuss generation details in all but the most difficult cases. A handful of drums still remains that will require sampling and analysis due to lack of generator process knowledge documentation. Most of the drums requiring sampling and analysis were insufficiently characterized because of complexities in the generating process that could not be resolved through KOP.

CHARACTERIZATION APPROACH

RCRA characterization was performed in accordance with the provisions of 40 CFR 261. Figure 4 details the process used to assign RCRA hazardous waste codes. After a complete review of all physical, chemical, process, and generator information, the Laboratory-specific waste characterization form was completed for each waste item, or group of similar wastes in each drum. The form identifies the RCRA hazardous waste code and the rationale for assigning the code. In addition, the form lists all Laboratory assigned hazardous waste codes that did not reflect waste characteristics and were properly assigned. Rationale were provided to explain why the codes did not apply to that particular waste item.

Fig. 4.

To complete characterization, each waste was evaluated to determine if it contained any regulated spent components or if it was an unused commercial chemical or spill residue. In addition, each waste was evaluated for ignitability, corrosivity, reactivity, and toxicity characteristics. MSDS sheets were helpful in many cases in providing property information for labpacks and unused materials. The team utilized RCRA-experienced chemists in determining the properties of mixtures, when necessary. The pH, flashpoint, and chemicals of concern were evaluated to identify any hazardous characteristics. Listed and characteristic codes were assigned to each waste as appropriate.

A conservative approach was applied while characterizing LLMW to ensure that all the potential waste codes were assigned. In situations where the Laboratory-assigned RCRA code could not be confirmed as correct or not, the code was left as assigned and a notation made. Due to the fact that RCRA characterization regulations and waste codes are somewhat ambiguous, EPA has developed and issued guidance materials over the years that clarify questions on various waste codes and waste management issues. The team subscribed to several guidance periodicals and solicited Laboratory policy guidance in many of the more difficult cases.

Approximately fifty percent of the RCRA hazardous waste codes originally assigned by the Laboratory were confirmed in the characterization process. Several waste streams were consistently assigned an incorrect waste code in the absence of detailed process knowledge. These waste streams are discussed in the paragraphs below. Some examples are listed below.

Many wastes had originally been assigned a D001 ignitable solid code. This code is easily assigned to liquids but it is more difficult call to assign this code to solid wastes because the wastes have to be spontaneously combustible, or create a fire upon exposure to moisture and burn so vigorously and persistently as to create a hazard. Often times this code is applied to wastes that can cause a dust explosion that flashes and burns out. According to EPA, this is not the intention of this hazardous waste code. Solid wastes that were assigned the D001 code were evaluated to determine if the code was applicable. In addition, the D001 code was applied to unused chemicals that had not been identified as DOT oxidizers.

Many legacy cylinders of gas were assigned a D002 corrosive waste code. These codes were changed based on the fact that the RCRA corrosivity characteristic is applicable only to liquid or aqueous wastes. Also, a D002 waste code was applied to solid wastes which is not the intention of the RCRA regulations, therefore the codes

were changed.
Several toxicity characteristic codes were deleted due to concentrations of the component in the waste or simply because the component was not in the waste. For example, several wastes were assigned a D009-mercury or D008-lead waste code and the Laboratory data and generator interview both noted that these contaminants were not in the waste. Frequently, wastes from analysis for heavy metals were coded with the analyte metal code, which was reversed based upon KOP. For example, a waste originally labeled as UV SPEC WASTES FROM LEAD ANALYSIS would be assumed to contain lead because of the description when it was actually just a dilute buffer solution. The waste codes were changed in these situations.
Spent solvents that were used for degreasing parts or cleaning oily or greasy equipment were identified as F001 if the solvent appeared on the F001 list. Although EPA's intention with this F001 category was to regulate solvents used in large scale degreasing operations (such as vapor cleaning), the Laboratory policy was to remain conservative and identify these wastes as F001 listed for degreasing. Wastes that were identified as F002 and were used for degreasing/cleaning parts were changed to F001 to remain consistent with Laboratory policy.
The F003 spent solvent code was incorrectly applied in several situations. This is a common area of confusion. The F003 code applies to those listed solvents only if one of the following is true: (1) the waste consists of solely an F003 listed component, (2) the waste consists of a mixture of F003 listed components, or (3) an F003 listed component is mixed with an F001, F002, F004 or F005 listed waste, or a combination of those listed wastes.
Wastes which were originally assigned solvent F-listings but were actually generated by a cooling process were changed to non-regulated or assigned an appropriate characteristic code. Cooling is not a solvent property according to EPA regulations.
In all cases, any changes in RCRA coding were documented and databased with an appropriate rationale. All of the routine errors in original RCRA coding were reported to the Laboratory waste acceptance managers to avoid the potential for similar instances in the future.

PROJECT RESULTS

Of the 7,546 legacy LLMW items in the inventory, 4,083 were RCRA and radiologically characterized by KOP at the conclusion of the project. These items were also assigned DOE treatability codes. 117 items could not be KOP-characterized due to insufficient information. The remaining items fall into other categories and will not require characterization. A significant fraction of the wastes which were reviewed by the team may be eliminated from the LLMW inventory after certification either as non-radioactive or non-hazardous. Many more unused, labpacked waste items may be re-evaluated after surface decontamination.
A preliminary sampling and analysis plan has been prepared to address the remaining legacy items. The majority of the items were RCRA-characterized by KOP but were not adequately radiologically profiled. The team has recommended varying degrees of radiological screening for these waste items, and interviews support the conclusion that most of them are surface contaminated only. Most of the RCRA-uncharacterized items are labpacked unused chemicals that will require a visual inspection to read container labels and screening to verify radioactivity data.

USING THE CHARACTERIZATION DATA

As mentioned above, the characterization task was only one element of a broad compliance effort set forth in the Laboratory's FFCA. Since completion of the characterization program, the Laboratory has continued to implement several programs which pertain to the FFCA and make use of characterization information. Many of the data from the characterization program have been captured in an electronic database, which is being made available to process engineers and waste treatment planning staff. The waste folders and all of the documentation they contain have been digitally imaged on CD-ROM and converted microfilm to provide a format which is shareable by many individuals.
The characterization information is currently being used for the conceptual and detailed design of several skid treatment processes. The feed handling systems in these designs rely on accurate packaging and matrix information from the folders. Characterization information has been provided to operators of existing treatment programs at the Laboratory. A few waste streams are currently being treated on-site including surface contaminated lead and scintillation vials. The volume and nature

of these streams was estimated prior to the development of treatment; whereas complete characterization data has now allowed detailed feed information and an inventory of all applicable wastes for these treatments.
The Laboratory is also using characterization data to store drums in the most compatible and efficient groupings. Several wastes for which RCRA coding changes were made were relocated to provide more consistency in the waste inventory. Similarly, future storage requirements are being evaluated based upon waste generation history and projections from the characterization program.
EVOLUTION OF LABORATORY WASTE MANAGEMENT
While most of the inventory of stored wastes has been addressed by KOP or other means, the continuation of environmental restoration and R&D activities suggests that the generation of LLMW will not decrease substantially in the near future. As a result of the characterization team's experience with waste generators and the historical waste management and documentation practices, the Laboratory has been able to make several improvements in maximizing the effectiveness of the waste acceptance process. Several measures have been implemented and continue to be developed to ensure that future wastes are fully documented as received and to minimize the need for characterization activities in the future.
Historically, the Laboratory's Waste Management groups have had to maintain responsibility as generator for large amounts of waste which the groups themselves did not produce. These groups have had important custodial responsibilities, but did not necessarily have much authority or control over the waste streams they receive. Much of the Los Alamos waste management structure was implemented prior to the promulgation of current waste regulations and was not prepared to follow the cradle-to-grave approach to waste stewardship. In addition, management of the waste was distributed amongst several different groups, making coordination of information difficult.
Efforts are now under way to provide a more centralized and complete tracking mechanism for generation-to-treatment waste documentation. More complete documentation is being required of waste generators, using concepts which were introduced in the characterization effort. The two databases which in the past have separately maintained storage and characterization data have been linked together to provide a more focused source for waste information. Most importantly, the Laboratory is pursuing a system to increase generator awareness and involvement in the management of their own wastes. Waste generator training has been revised to increase awareness of RCRA regulations and Laboratory policies regarding waste generation. Each waste-generating group now has a full-time waste coordinator who provides support to both the generator and the waste management groups. An interactive waste management system is being studied to allow waste generators or waste coordinators to log on to an intelligent database to characterize their waste for appropriate storage and treatment with a series of guidance questions.

5O-3
CHARACTERIZATION OF RADIOACTIVE WASTE GENERATED AT ABERDEEN PROVING GROUND
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ABSTRACT
A study to characterize radioactive waste generated by tenant activities at the Aberdeen Proving Ground (APG) was conducted during the spring of 1994 for the installation's support Health Physics Office (HPO). The APG HPO was assigned responsibility for managing waste materials, coordinating disposition and providing interim storage while awaiting disposition. In order to ensure adequate interim storage, the HPO needed to characterize generation and evaluate the impacts of proposed minimization practices. This information is needed to gauge operations at the HPO's 670 square meter low level radioactive waste storage facility which will be used to temporarily store radioactive waste generated by APG tenants until the Appalachian Compact sites and builds a low level radioactive waste disposal facility.

The results show that the total annual volume of solid radioactive waste generated is 955 cubic meters (m3), of which sixty-six percent (2/3) is recyclable metal and thirty-three percent (1/3) incinerable and compactable. Only a small fraction of the waste could be classified as a mixed waste (3E-05). Depleted uranium usage is the largest single use of any radionuclide. This study also evaluated the impact of proposed waste minimization practices.

The results of the study were also used as a planning tool to develop procedures for packaging, storage and processing radioactive waste. Additionally, the findings were used to develop a inventory management system to manage and track radioactive waste stored or in processing at APG's low level radioactive waste storage facility.

INTRODUCTION

This study was conducted in order to characterize the types, form, components, quantities and generation rates of the low level radioactive waste from tenants in the Edgewood and Aberdeen Areas of Aberdeen Proving Ground (APG). Until 1989, tenants were responsible for arranging disposal of their own radioactive waste. In assuming overall responsibility for interim storage of radioactive waste generation at APG, the installation Health Physics Office (HPO) needed to establish a baseline generation rate, characterize the components of the radioactive waste stream and determine waste generation practices. The study included interviewing the tenants that use radioactive materials in order to determine the nature, type, subtypes, quantities and volumes of radioactive waste generated. The results of this study will be used by the APG HPO in order to plan future disposition of radioactive waste.

Aberdeen Proving Ground is a Test and Evaluation Command installation within the United States Army Materiel Command. The mission of the installation is to develop and test military materials and to train officers and enlisted personnel in the use and maintenance of Army equipment. The installation was established in 1917 as two separate military reservations, APG and Edgewood Arsenal. Both were consolidated into APG in 1971. APG is located in northeast Maryland, has 73,000 acres of land and water and has 59 tenant organizations. Figure 1 shows the Aberdeen Proving Ground.

METHOD

In order to effectively characterize the APG's radioactive waste, a standard form was developed for interviewing potential generators. The form requested basic information such as: types, quantities and usage rates for each radionuclide; type and form of radioactive waste generated and generation rates for each; mixed waste generation rates; segregation practices; waste minimization plans; anticipated mission changes affecting generation rates; typical container types; and storage locations. The radiation safety officer was contacted in each tenant organization to determine points of contact for generators. Each point of contact was then contacted, interviewed and generation practices discussed.

RESULTS AND DISCUSSION

Of the 59 tenants, only 11 use radioactive materials. Of the eleven tenants, a total of 29 different organizations within those tenants generate radioactive waste. The tenant name, generator name, and the activity conducted by the generator resulting in radioactive waste generation are listed in Table I, cont'd.

The type of generator activity is categorized as follows: depleted uranium munitions testing (8), laboratory research (8), radioactive commodities (6), radiological oversight (5), radiation detection instrument calibration (1) and nuclear weapon effects simulation (1). Of the twenty-nine generators only eighteen generate on a routine basis. These routine generators represent: 1) Army Research Laboratories (ARL), 2) Combat Systems Test Activity (CSTA), 3) Medical Research Institute for Chemical Defense (MRICD), 4) Army Environmental Hygiene Agency (AEHA), and 5) Edgewood Research, Development, and Engineering Center (ERDEC). A description of the type of activities generating radioactive waste follows.

Depleted Uranium Testing

ARL and CSTA conduct depleted uranium (DU) munitions testing. Depleted uranium is depleted of the natural isotopic distribution of U-235 typically found at 0.72%. The distribution of U-235 is 0.24% in DU. The other radioisotopes in DU are U-234 at 0.0054% and U-238 at 99.75%. DU munitions consist of DU incorporated into the penetrator of a projectile. Tests involve firing projectiles from small caliber as well as large caliber weapons. Tests are conducted in enclosed ranges that have air pollution controls to reduce the amount of airborne DU released to the environment from the impact of the projectiles on targets.

Material within DU test ranges becomes contaminated with DU when projectiles contact targets causing explosive debris to impact on items within the test range. Targets are the most contaminated as molten DU melts the target and becomes embedded. Airborne DU dust and debris eventually settles-out contaminating exposed surfaces. Airborne DU is filtered through a filter train consisting of roughing filters, prefilters and high efficiency particulate filters (HEPA). Larger ranges utilize baghouse upstream of the filter train to reduce filter loading. Contamination is spread throughout the interior of the range through subsequent movement of equipment and personnel. The vast majority of materials exiting the ranges are discarded as radioactive waste.

Typical material found in the range waste stream includes cotton protective clothing, wood pallets, wood used to frame x-ray film, brake screens, yawl indicators and target material. Plastic and rubber shoe covers and gloves also become contaminated. Periodically the roughing, pre and HEPA filters are changed out. The largest component is target material, which consists of metal and non-metal components. The metal stream includes hardened steel and small amounts of aluminum, titanium, copper and tungsten. The non-metal target components are glass, fiberglass, plastic, glass reinforced plastic and ceramics. In large caliber ranges the steel backstop, shot table and the interior range liner, become warped or too riddled and eventually require replacement. Testing also produces debris consisting of small pieces of wood, target material, metal, plastic, paper and glass which falls to the floor. These floor sweepings are picked up and are also disposed as radioactive waste. A liquid waste stream is produced from laundry of cotton coveralls and range decontamination. CSTA recycles their liquid radioactive waste, and until recently, ARL was releasing their liquid radioactive waste to the sanitary sewer per the Nuclear Regulatory Commission's requirements (1). Table II shows the annual volume of waste produced from ARL's and CSTA's DU testing.

Laboratory Research

Several of the MRICD, AEHA, and ERDEC generators are engaged in evaluating or analyzing the effects of toxic compounds and chemical agent on biological systems and tissues. These generators utilize in-vitro and in-vivo techniques to determine uptake, retention and excretion of radiolabeled materials. Effects of compounds on animal metabolism are also tested. Typical radionuclides used in research include H-3, C-14, P-32, S-35 and I-125.

An other AEHA generator analyzes environmental and biological samples for radioactivity. They use radionuclides such as H-3, C-14, Sr-90/Y-90, Ra-226, Uranium, Th-230, Am-241 and Cs-137 for standard preparation, analysis, sample spiking and instrument calibration.

The radioactive waste stream from laboratory generators consists of solid (mixed compactables and incinerables), liquid and other wastes. The liquid waste stream consists of liquid scintillation counting media (LSCM) and aqueous solutions. The LSCM consists of liquid scintillation cocktail, radiolabeled compounds and other organic material. Aqueous solutions consist of waste radiolabeled compounds and organic material. Tenants for the most part have switched to a non hazardous liquid scintillation cocktail solution, thus as long as the sample does not contain material considered hazardous under the Resource Conservation and Recovery Act (RCRA) (2), the LSCM is not regulated under RCRA.

Solid waste material consists of paper, cloth, plastic, glass, rubber, and wood. Of the total solid material, most is paper and plastic. Only a small portion of the solid waste stream is wood. The other waste stream consists of radioactive commodities and by products from tests conducted on small rodents.

The annual volume of radioactive waste from the laboratory generators are summarized in Table III.

Radioactive Commodities and Instrument Calibration

Radioactive commodities include those instruments and devices described in Ref. 5 found in equipment. Commodities include dials, gauges, sights, lenses, electron tubes, indicators, clocks, compasses, calibration sources, check sources and counterweights. They are also found in chemical agent detection equipment. There are eight generators that routinely utilize commodities containing radioactive material. Several of these generators use foreign radioactive commodities. Typical radionuclides include Ra-226, H-3, C-14, Am-241, Sr-90, DU and Pm-147. Generators are directed to turn in equipment if it is no longer serviceable, obsolete or otherwise usable. Test Measurement Diagnostic Equipment has a radioactive commodity

license to calibrate radiation detection equipment. As part of their calibration service they leak test detection instrument check sources and calibration sources. Leaking check and calibration sources would be turned in for disposal.
Although the rate at which commodities are turned in is sporadic, the impact of commodity volume to the installation volume is small. However, the specific activity of these commodities can be quite high, and their contribution to the installation's quantity of radioactive material for disposal can be significant.

Radiological Oversight

There are five independent tenant health physics organizations that provide oversight of generator operations conducted under their tenant license. These organizations collect and analyze smear, water, air, vegetation, soil samples and other media as necessary to ensure license conditions are being met. They also provide health physics services in addition to serving as a focal point for radioactive waste collection. Radioactive waste generation within the tenant Health Physics Offices occurs largely from analysis of samples. The waste stream consists of contaminated surgeon's gloves, planchets, sample holders and sample media, LSCM aqueous media and occasionally commodities. Excluding commodities, the combined health physics waste stream was 1.1 cubic meters per year (m3/y) for ARL, CSTA, ERDEC and MRICD.

Nuclear Effects Testing

CSTA operates a facility that simulates effects of neutron and gamma radiation from nuclear weapons on military systems and equipment. Test protocols and materials are reviewed to assure that low atomic number materials are used to the maximum extent possible thereby reducing the amount of neutron activation during tests. Facility construction was designed to maximize the amount of low atomic number materials used in construction. Use of these low atomic number materials minimizes neutron activation of components, support structures, enclosures, test pads, etc. Most of the induced radioactivity in exposed materials is short-lived. Therefore exposed materials can be stored on site for decay, reducing the volume of radioactive waste generated. Test materials resulting in long-lived radioactivity are generally returned to the experimenter if the experimenter has a radioactive material license, if not, CSTA retains possession. Quantities of radioactive waste generated from unlicensed experimenters remain low.
Tools and equipment that become activated/contaminated are reused in radiation areas. All paper, plastic, glass and small amounts of metal are placed in the same container, the annual generation rate (including outages and routine maintenance) is 0.32 m3/y.

CURRENT MINIMIZATION STRATEGIES

Tenants have been asked to segregate incinerable and compactable waste streams and maximize the use of incinerable items wherever possible. These actions will ultimately result in lowering the compactable volume and reduce interim storage requirements at APG.
Both ARL and CSTA have programs to reduce their radioactive waste generation. ARL has plans to replace wooden pallets with metal pallets. This should reduce the amount of wood to be incinerated and increase the volume of recyclable metal. ARL estimates a 90% reduction in the wood stream from 112 m3/y to 1.2 m3/y. To reduce the volume of radioactive metal, ARL plans to install a coring device that will cut most of the radioactive portion out of shot target plate. ARL is also investigating the use of abrasive materials to decontaminate targets. ARL anticipates the combination of coring and abrasives will reduce their metal waste volume by 1/3 to 2/3.
CSTA, at one time, operated a liquid abrasive unit to decontaminate target metal and cut contaminated metal from shot target. Funding for the program has been reduced and no longer operates. However CSTA's "Superbox" range is able to achieve a significant volume reduction over that of a similarly sized range. Superbox, which is located at Ford's Farm, surveys all materials in the controlled area to segregate contaminated material from uncontaminated material. Reusable material that is contaminated is marked and reused in the controlled area.

SUMMARY OF ALL WASTE STREAM TYPES

The total volume of radioactive waste generated by the tenants is illustrated in Fig. 2. Based on 1993 data and historical generation rates, APG generated 1409 m3 of radioactive waste last year. Almost half of the waste volume is metal. As discussed previously, the majority of the metal waste stream is generated by ARL and

CSTA. Most of the metal waste volume is recycled as shielding material for DOE projects.
Almost one-quarter of the annual volume is incinerable and compactable waste. This waste stream is also largely attributable to ARL's and CSTA's DU testing. Until recently, the incinerable and compactable waste stream was not segregated. With the waste reduction initiatives underway, the wood component is expected to be reduced as wood pallets are replaced with metal pallets, and compactable items are replaced with incinerable items.
The figure also shows that approximately one-third of the volume, 454 m3 is liquid waste. Most (99.9%) of the low level aqueous radioactive liquid waste volume is attributed to decontamination operations at ARL's Range 9 and Range 14. The ARL liquid volume consists of DU in solution with detergent soaps and water. Of the remaining liquid radioactive waste volume (0.04%), most is attributed to generation located from the Edgewood Area tenants. This non-range volume contains mostly H-3, C-14 in solution with LSCM or in solution with aqueous material such as waste animal fluids and other dilute laboratory chemicals.
Excluding the liquid radioactive waste stream volume from the total volume, leaves an annual solid waste volume of 955 m3/y. Of that total, 591 m3 is metal waste (62%), 334 m3 is incinerable and compactable (34%), and 30 m3 of other radioactive waste types. The vast majority of the metal is recyclable and at least half of the commingled compactable and incinerable stream is incinerable.
Figure 3 presents the contribution to the solid waste stream by tenant. As the figure shows, ARL's and CSTA's DU testing dominates the solid waste stream, where as the other generators contribute only a small fraction.
Place Figure 3 here.

MIXED WASTE SUMMARY

APG does not generate a significant quantity of mixed waste. However, the potential for generating a mixed waste does exist, given the types and nature of the that use hazardous and radioactive materials. At this time no mixed waste is generated on a routine or reoccurring basis.

CONCLUSIONS

This study summarizes the results from interviews of 29 generators at APG that generate radioactive waste. Of the 29, only 18 generate waste on a routine basis. The tenants that routinely generate radioactive waste are ARL (7), CSTA (4), MRICD (4), AEHA (2) and ERDEC (1). These generators produce over 1409 m3 of radioactive waste per year.
Of the annual volume routinely produced, 99.7% of the waste volume is produced by CSTA and ARL. Greater than 95% of the annual volume is produced by ARL's Range 9 and 14 and CSTA's Superbox and Bomb Throwing Device. Both of these tenants are engaged in DU penetrator testing, hence the principal radionuclides present in the waste are U-238, U-235, U-234 and progeny. The remaining 0.3% of the radioactive waste (approximately 3.7 m3/y) is produced by Edgewood tenants (MRICD, AEHA and ERDEC) which produce laboratory wastes. The principal radionuclides in these lab wastes are H-3 and C-14.
Excluding radioactive commodities, the radioactivity in the APG waste stream averages 3-4 Curies per year (approximately 100 to 150 billion Becquerels (Bq)). A relatively small fraction of the waste volume (less than the Edgewood generation rate) results from turn-in of commodities from domestic and foreign ordnance. In comparison to the average APG radioactive content, the quantity of radioactive material in a single commodity (which ranges from several million Bq to several hundred billion Bq) could exceed the average radioactivity in the combined volume of all waste streams generated at APG.

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50-4
SUCCESSFUL CHARACTERIZATION OF RADIOACTIVE WASTE AT THE SAVANNAH RIVER SITE

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ABSTRACT

Characterization of the low-level radioactive waste generated by forty five independent operating facilities at The Savannah River Site (SRS) experienced a slow start. However, the site effectively accelerated waste characterization based on findings of an independent assessment that recommended several changes to the existing process. The new approach included the development of a generic waste characterization protocol and methodology and the formation of a technical board to approve waste characterization. As a result, consistent, detailed characterization of waste streams from SRS facilities was achieved in six months.

SITE DESCRIPTION AND PROCESS OPERATIONS

Construction of SRS was begun by the United States Government in 1950. The site is owned and operated by the U. S. Department of Energy. Certain daily activities are carried out by managing and operating contractors, including the Westinghouse Savannah River Company (WSRC). Historically, the primary purpose of the SRS was to produce special nuclear materials, primarily plutonium and tritium. The process operations at SRS can be divided into three main categories. Raw materials such as uranium and lithium were processed into fuel and target assemblies in a fabrication facility. Next, fuel and targets were loaded into nuclear reactors where the fuel was used to bombard the targets with neutrons. This produced the desired products, including tritium and plutonium. Lastly, the products were separated and purified in chemical separations areas and then distributed to other areas for further processing or to end users as applicable. This process produced high-level radioactive waste products which are stored in underground tanks for future processing and subsequent shipment to a federal repository for permanent disposal. The recent change in the mission of SRS has shifted most operations to a fourth category, that of waste management and environmental restoration. Accordingly, increased emphasis on the management of this part of the mission has resulted in changes in the way SRS dispositions radioactive waste.

SRS Conversion from Shallow Land Burial to Concrete Vaults

In September 1988, DOE issued DOE Order 5820.2A, "Radioactive Waste Management"(1). This Order established new performance objectives for disposal of low-level radioactive waste. The Savannah River Site realized current practices of shallow land burial would no longer be acceptable because this practice could not meet the performance objectives. Prior to this time, guidance had been received from DOE Headquarters which suggested disposal practices move toward separation (isolation) from the land. At the time the guidance was received, SRS was planning for the expansion of its current disposal practices. SRS took the guidance from DOE Headquarters and a decision was reached that the expansion would be to build concrete strong vaults that met the performance objectives of the Order. These vaults are currently operational and are referred to as the E-Area Vaults.

Chapter III of the Order details policy and requirements for management of low-level waste and mixed waste (containing both radioactivity and RCRA hazardous constituents). In Chapter III, a hierarchy of requirements are laid out to ensure protection of public health and the environment. First, four performance objectives, that must be met by all waste management operations are listed. Then, for waste disposal facilities, a radiological performance assessment (PA) is required. The purpose of the PA is to demonstrate reasonable assurance that the performance objectives will be met over a span of 10,000 years or more. A Peer Review Panel (PRP) was established to ensure technical quality and consistency in the development of PAs (2, 3). Waste Acceptance Criteria (WAC) are required to be developed for each storage treatment and disposal facility. For disposal facilities, results of the

performance assessment are key inputs into the development of the WAC. Finally, waste characterization and certification are required to ensure that the waste accepted for disposal comply with the disposal facility WAC.

The Performance Assessment and Waste Acceptance Criteria at SRS

The SRS E-Area Vaults for the disposal of low-level waste was the first new facility in the DOE complex to be required to prepare and maintain a PA. The performance assessment primarily utilizes a geo-hydrological modeling process to estimate the long term environmental effect of radioisotopes disposed of in the vaults.

The performance assessment radioisotopes and their estimated concentrations that maintain long term compliance with the Order performance objectives are entered into the E-Area Vaults Waste Acceptance Criteria. In addition to the Order requirements there are also site safety analyses that establish radioisotope limits that cannot be exceeded during operational circumstances and are also entered into the WAC. In combination, then, these comprise the radioisotopes that must be characterized and the detail required for characterization by generators who send low level waste to the vaults for disposal.

These characterization requirements are documented in the SRS Waste Acceptance Criteria Manual (4) and proceduralize how SRS will comply with DOE Order 5820.2A. In 1993 the process to approve the certification programs for the 45 facilities at SRS that generate low-level waste commenced. Radioisotopic waste characterization proved to be an obstacle early on and resulted in only 3 of the 45 waste generators being certified in the first 15 months of the program.

INITIAL UNSUCCESSFUL CHARACTERIZATION EFFORTS AND LESSONS LEARNED

In the beginning, the E-Area Vaults Waste Acceptance Criteria were developed with an emphasis on all radioisotopes that were possibly important in the Performance Assessment for the Vaults, even though some of these isotopes could not be measured in the waste streams in any statistically defensible way. The then-existing certification process took the criteria and interpreted it to mean that all of these radioisotopes were important, so they all had to be measured. Concurrently, the term "process knowledge" was perceived by the waste generators as something the certification process deemed unreliable because of the importance of isotopes to the PA and would therefore not be accepted. The result of all of this was that a technical debate on characterization became established between the waste generators, the disposal facility and the certifying organization and became an impediment to the certification process. An assessment by senior waste management program and engineering managers determined that a fresh start was needed that incorporated the lessons learned in this protracted first phase. Specifically, focus on Order requirements, establish "technical" guidelines for characterization methodology at SRS to set the standard for assessments, develop a site model for radioisotopic characterization that waste generators could use as an example and compare the SRS program to commercial and other DOE sites.

CHANGE IN WASTE CHARACTERIZATION STRATEGY

These lessons learned were addressed at a site level by commissioning an independent task force to identify those necessary efforts needed to bring the Savannah River Site into compliance with the waste certification requirements of the DOE Order. The task force was led by senior executives with experience across several environmental, industrial and technical business areas. Members of the task force were drawn from SRS low-level waste generators, commercial technical consultants and internal assessment organizations.

The task force concluded early on that an impediment to waste certification had been the lack of an established and accepted method to characterize the radioisotopic constituents in low-level waste. Accordingly, it was decided that the task force should develop this methodology, apply it to sufficiently complex test cases and utilize the process as the preferred process for characterizing waste as a part of the certification program. The characterization plans developed by the task force would then serve as models for SRS waste generators and would establish the standard against which waste certification assessments could be performed.

The characterization plans were developed using a logic base that included consideration of the risk of exceeding the E-Area Vaults limits for specific radioisotopes. The underpinning of this characterization methodology lies in the ability to develop technical arguments that provide reasonable assurance that low-level waste is characterized with sufficient accuracy to permit proper segregation, treatment, and disposal as required by the specific Treatment, Storage

and Disposal (TSD) facility. Combining the above considerations into a rational approach, the task force developed a waste characterization strategy, which is shown in Fig. 1.
Fig. 1.
This strategy was applied to two separate waste generator situations and example characterization plans were produced over a period of two months. The process served as a high level forum to resolve complex issues associated with waste characterization and also provided comprehensively documented characterization packages that were made available to the SRS waste generators as standards. The two example facilities were chosen to represent the spectrum of SRS waste generators. One generator had extensive historical process knowledge that included criticality analyses, solubility studies, chemical partitioning of process solutions and health and physics radiometric surveys. This facility also had multiple radioisotopes and complex process chemistry. The other facility had rather straightforward process chemistry and simple radioisotopics but, due to the purity of their product, had very low amounts of radioactivity in their waste such that special characterization strategies were required. In the course of developing the two example plans, it became clear that the generators and TSD organizations would benefit from a better understanding of chemical and radiometric analysis principles that would assist in what could be expected from analytical laboratories. The task force also provided guidance in this area (5).

THE SRS WASTE CHARACTERIZATION BOARD

The institutionalization of the task force strategy is embodied in the Waste Characterization Board (WCB), which is functionally within the Waste Certification organization. The Board is composed of 10 voting members appointed by the Solid Waste Management Department (SWMD) manager and selected from various WSRC organizations such as laboratories, waste management, environmental compliance and the operating divisions. Members of the Board are senior level scientists or engineers and possess a knowledge of chemistry, radiochemistry, RCRA regulations as well as a knowledge of SRS conduct of operations and work practices. There are also two independent consultants who are voting members. The Board reviews and approves the SRS waste generator's characterization plans and associated documentation. In this way a consistency of approach and methodology is achieved for the characterization of the hundreds of waste streams at SRS.

The Board review is first step in the waste certification process. The waste characterization plan review typically consists of technical documents prepared and defended by the facilities' engineering or technical staff in conjunction with that division's regulatory programs branch. This group develops the characterization outline, the characterization plan, any sampling and analysis plans needed and the necessary documentation of the radioisotopic distribution baseline. These plans are developed in conjunction with the Solid Waste Management group who provides technical support and consultation to the waste generators on what the essential plan elements are that the Board will require. The key elements of the characterization plan are the following:

1. A brief description of the facility and the processes that generate waste.
2. The rational used to identify, separate and segregate the various waste streams.
3. A description of the physical forms, the radionuclide content and isotopic distribution in the waste.
4. Information on the content of, or methods of exclusion of, hazardous materials.
5. Details of the strategy used for routine curie calculations.
6. A description of the methods used to accurately determine, document and periodically validate the physical, chemical and radiological properties of the waste streams.

In the event a generator uses a characterization strategy that involves sampling for initial characterization or verification of process knowledge, the generator is expected to provide a sampling and analysis plan. The plan details the following:

7. The objectives of the sampling.
8. The rationale for the locations and numbers of samples.
9. The justification for the samples representative nature.
10. The methods of collection.
11. The analyses to be conducted.

The members of the Board receive the waste characterization plan from the generator facility technical staff for formal review of the plan in an open session. During

this session the facility engineering support or technical staff appear before the Board to defend the waste characterization plan elements and address any technical concerns raised by the Board. This is in effect a technical "oral board" process during which a majority of the board members must be convinced of the viability of the waste characterization. The generator's waste characterization plans can be disapproved, approved, or approved with action items which are then tracked to closure before actual certification of their program can occur.

RESULTING ENHANCEMENTS AND FUTURE PROGRAM DIRECTION

This strategy for accomplishing waste characterization has proved to be very successful at SRS. During the first six months, the Board approved 45 waste characterization plans representing over 100 individual waste streams. Most of the plans were approved during their first formal appearance before the board; some required multiple appearances before the board before the plans achieved approved status. This has had the effect of front end loading the overall waste certification process and eliminated some of the false starts that were present in previous efforts and has resulted in estimated overall cost savings to the site of $250,000. It has had the additional effect of dropping the cost of an individual waste certification assessment by 10-15 percent.

In addition to its specified role, the Board expanded as an advisory group for SRS waste management policies, due to its acquired expertise both in the function as an approval authority and as the repository of the site's collective waste characterization strategies. The Board has been asked to evaluate soil characterization strategies, facility operations designed to separate transuranic and hazardous waste and project strategies for SRS waste treatment and disposal facilities not yet operational.

Future waste characterization programs at SRS involve the disposition to the Waste Isolation Pilot Plant (WIPP) of transuranic waste currently in storage and the certification of SRS waste generations to use the Consolidated Incinerator Facility (CIF), which is scheduled to begin operations in late 1995.

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5O-5

PROGRESS IN THE ANALYSIS AND SPECIATION OF ORGANIC CHELATOR COMPOUNDS IN HANFORD WASTE TANKS

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ABSTRACT

Twenty-three of the 177 high-level waste storage tanks at the Hanford Site in southeastern Washington are being watched closely because of the possibility that flammable gas mixtures may be produced from the mixed wastes contained in the storage tanks. One tank in particular, Tank 241-SY-101 (Tank 101-SY), has exhibited episodic releases of flammable gas mixtures since its final filling in the late 1970's. It has been postulated that the organic compounds present in the waste may be precursors to the production of hydrogen. Six samples of core segments from Tank 101-SY, obtained during the window E core sampling, have been analyzed for organic

constituents using derivatization gas chromatography/mass spectrometry (GC/MS), liquid chromatography (LC), and LC/MS. Four of the samples were from the upper region, or convective layer, of the tank and two were from the lower, nonconvective layer. The combined techniques of derivatization GC/MS and LC have accounted for approximately 73 to 93% of the total organic carbon (TOC) depending on the sample.

INTRODUCTION

Large quantities of EDTA and HEDTA (at least 240 and 1500 tons, respectively) have been used in defense related activities at the Hanford site in southeastern Washington State (1). These chelators are now part of the complex mixed wastes found in large storage tanks at Hanford. Mixed waste is defined as waste containing both hazardous chemicals and radionuclides. The chelators form water-soluble complexes with most heavy metals, thereby enhancing the migration of heavy metals in soils. For example, studies at Oak Ridge National Laboratory demonstrated that EDTA caused the low-level migration of 60Co from intermediate-level liquid waste disposal pits and trenches (2). Another study at the Maxey Flats commercial low level waste disposal site revealed that Pu-EDTA and 60Co-EDTA migrated (3). In addition, after approximately 40 years of storage under radiolysis and favorable conditions for chemical degradation, other chelator fragments may have formed with equivalent or more complexing capability than the starting materials. Recent studies with simulated wastes indicate that chelator degradation is, in fact, quite vigorous (4,5). As a result, the presence of chelators is an important consideration when handling and storing mixed wastes.

Of the 177 high-level waste storage tanks at Hanford, approximately 23 tanks are being watched closely because of the possibility that flammable gas mixtures will be produced from the mixed waste contained in the storage tanks. One tank in particular, Tank 241-SY-101, has exhibited episodic releases of flammable gas mixtures since its initial filling in the late 1970's. Studies of simulated waste mixtures (SWM) have indicated that the gas generation and retention are influenced by chelator concentration. It was postulated that the chelators form hydrophobic surfaces on solids in the SWM. These hydrophobic surfaces are more conducive to bubble attachment, which leads to flotation of the solids and eventually crust formation (6). The presence of chelators becomes very important for the understanding of crust formation and gas release. In addition, among the degradation products of the chelators are a number of small organic acids, some of which may be linked to the production of flammable gases such as hydrogen.

There are no accepted methodologies for the analysis of chelators, chelator fragments, and low molecular weight acids (LMWAA) in radioactive mixed hazardous wastes. As a result, methods have been developed to analyze these organics in tank wastes. The chelators represent a class of compounds whose polarity and non-volatility preclude direct analysis by GC/MS; therefore, chemical derivatization is required. Toste et al. (7,8) used boron trifluoride/methanol for the analysis of organics in radioactive wastes. An LC method was developed to analyze LMWA. In addition, thermospray LC/MS was used to confirm the identification of LMWA (9). Waste from Tank 101-SY was analyzed using derivatization (boron trifluoride/methanol) gas chromatography/mass spectrometry (GC/MS), liquid chromatography (LC), and liquid chromatography/mass spectrometry (LC/MS) (10-12).

EXPERIMENTAL

Waste Sample

The waste sample represented a composite of several segments of actual waste obtained by core drilling the waste in Tank 101-SY. This material was highly radioactive, contained a high concentration of nitrate and nitrite, and the OH- concentration was 2M. A 5-g aliquot of the waste sample was weighed and then stirred with 10 mL of doubly-distilled water overnight. The aqueous phase was filtered through a 0.45 m filter. The filtrate was then passed through a cation exchange column (10 g, AG50W-X8 resin), rinsed using an additional 10 mL of water to quantitatively remove the organic carboxylates from the resin. The resulting sample volume was 20 mL. After surveying the samples for radioactivity levels, the aqueous solution containing the organics was removed from the hot cell facilities. Aliquots (2 mL) of the sample were transferred to Teflon-sealed vials, evaporated to dryness using nitrogen blow down techniques, and derivatized with 2 mL of BF3/methanol. The sample was heated to 100C for one hour. The sample was then hydrolyzed with a KH2PO4 solution and extracted with chloroform. The chloroform extract was then analyzed by GC/MS.

GC/MS Conditions

The instrument was equipped with a HP 5980 GC and a HP 5988 mass spectrometer operated in the splitless mode. A fused silica column (DB-5, 30m X 0.25 mm i.d., 0.25 mm film thickness, J & W Scientific) was used. The oven temperature was typically programmed in the following manner: 50C for 1 min, 8C/min to 300C, and hold at 300C for 5 min. The mass spectrometer was tuned daily with perfluorotributylamine (PFTBA). In these studies, the mass spectrometer was scanned from 50-500 amu and operated in the electron impact mode (70 eV). The source temperature was 200C, the injector port temperature was 250C, and the interfaces were also at 250C.

Chemical ionization was carried out with both methane and isobutane in both positive ion and negative ion chemical ionization modes. The temperature of the source for positive ion chemical ionization mass spectrometry was 200C and 120C for negative ion chemical ionization. The mass spectrometer was scanned from 100-600 amu in the negative ion mode and 70-500 in the positive ion mode.

LC

Anion exchange chromatography was performed with a model 510 solvent delivery pump, a model 712 WISP autosampler, and a model 484 variable wavelength UV absorption detector (Waters Associates, Milford, MA). The column was an Ion Pac A54A (250 mm x 4.0 mm) equipped with an Ion Pac AG4A guard column (50 mm x 4.0 mm) (Dionex, Sunnyvale, CA). The chromatographic data were collected and processed by Maxima 820, v3.02, software (Dynamic Solutions, Ventura, CA) on a model APC IV NEC computer (NEC, Foxbourough, MA). The mobile phase was 1.3 mM NaHCO3 and 1.4 mM NaCO3 at a flow rate of 2 mL/min. Detection was accomplished at 210 nm. The sample volume injected for analysis was 10 mL.

Ion exclusion chromatography was performed on a model 1090 high performance liquid chromatograph equipped with a diode array detector, and a data collection system (Hewlett-Packard, Palo Alto, CA). The column was an IC-PAK 50 , 300 x 7.8 mm, 7 mm, equipped with a guard column (Waters Associates). The mobile phase was 0.01 N H2SO4 at a flow rate of 0.6 mL/min. Detection was accomplished at 210 nm. The typical sample volume injection was 20 mL.

LC/MS

The thermospray analyses were performed using the HP 5988A LC/MS in the isocratic mode. The ion source was 275C and the stem initial temperature was 100C. Typical values for the tip and the vapor were 230C and 272C, respectively. Filament operation was in both the on-and-off modes. The multiplier voltage was 2692 V and the scan range was either 79-500 or 92-300 amu, depending on the mixtures being analyzed. The injection volume was either 10 mL or 20 mL, depending on the concentration of the sample.

The instrument was tuned with polyethylene glycol (PEG) tuning solution to give maximum abundance of ions. It should be noted that the thermospray ion source should be cleaned approximately once a week to remove nonvolatile chloride salts resulting from the use of HCl as the mobile phase. Also, since HCl possesses corrosive effects on metal surfaces, water was flushed through both the HPLC and thermospray system at the end of the day. As long as these cleaning and flushing procedures were followed, no detrimental effects of the thermospray or chromatographic equipment were observed.

Preparation of Deuterated-EDTA

Synthesis of this material was accomplished by the method of Vnylos (13), using d4-ethylenediamine purchased from Isotech (Miamisburg, Ohio) and sodium chloroacetate.

Recovery Study on 101-SY Matrix

An aliquot of 101-SY matrix was spiked with chelators and key chelator-related fragments in concentrations 3-5 fold above that found in initial analysis of the sample matrix. Additionally, a sample of the unspiked matrix was derivatized at the same time so that the contributions of these materials native to the matrix could be conveniently subtracted from the spiked samples. The design of this experiment was to determine the effect of the matrix on consumption of the derivatizing agent or other interference with the conversion of chelator acids to methyl esters.

RESULTS AND DISCUSSION

Six samples of core segments from Tank 101-SY, obtained during the window E core sampling, have been analyzed for organic constituents. Four of the samples were from the upper region, or convective layer, of the tank and two were from the lower,

nonconvective layer.
Values for total organic carbon (TOC) ranged from 1.0-1.4%; this range is consistent with TOC data obtained from other sub-samples of core segment composites. Among the samples analyzed, the bottom of the nonconvective layer has the highest TOC value. The combined techniques of derivatization GC/MS and LC have accounted for 73-93% of the TOC depending on the sample. Table I summarizes the data for LMWA, chelators, and chelator fragments. These component types constitute the majority of the carbon accounted for.
TABLE I
Chelators, chelator fragments, and several carboxylic acids have been quantitated in the current samples using derivatization GC/MS. The major components detected were EDTA, nitroso-iminodiacetic acid (NIDA), NTA, citric acid (CA), succinic acid (SA), and ethylenediaminetriacetic acid (ED3A). The chelator of highest concentration was EDTA in all six samples analyzed. The amount of organic carbon accounted for by derivatization GC/MS varies from 20-52% of the TOC depending on the sample. Preliminary results indicate the chelators and chelator fragments constitute more of the organic carbon in the convective than the nonconvective layer.
Liquid chromatography has been used to quantitate LMWA (oxalic, formic, glycolic, and acetic acids, which are present in the waste as acid salts). Approximately 23-61% of the TOC is accounted for by these acids. Oxalic acid constitutes approximately 40% of the TOC in the nonconvective layer samples. The concentration of oxalate in the nonconvective layer is approximately 3-4 times higher than the convective layer. The nonconvective layer sample from the lowest layer has the highest percentage of water-soluble organic carbon as low molecular weight acids. Previous work with 101-SY composite samples has identified normal paraffin hydrocarbons (NPH) as components of the water-insoluble portion of the 101-SY TOC. The NPH fraction was isolated by extracting the sample with methylene chloride. Analysis of the methylene chloride extracts by GC/MS indicated that this material accounts for an additional 4-13% of the TOC of the samples.
The elemental compositions of twenty-five components in the derivatization GC/MS total ion chromatogram have been determined using high resolution mass spectrometry (HRMS). Several nitroso compounds have been unambiguously identified. However, the nitroso compounds may be formed during the derivatization procedure; they have not yet been found by methods that do not involve acidic conditions of derivatization or separation. Thermospray LC/MS has been used to identify of many of the small organic acids including citric, acetic, formic, and glycolic. Several components in both the LC and LC/MS chromatograms are unidentified.
Nitroso compounds have been identified by GC/MS after derivatization of the waste extracts with boron trifluoride. The reaction of iminodiacetic acid (IDA) spiked in a dried, simulated waste sample containing nitrate, nitrite, and hydroxide with boron trifluoride/methanol produces NIDA as the major product. These results demonstrate that IDA, if present in the original waste, is probably a precursor for the production of NIDA during the boron trifluoride/ methanol derivatization. Under derivatization conditions, NIDA is stable. A probable precursor for two other nitroso compounds found in the derivatization product is ED3A. The question remains whether nitroso compounds are present in the original waste. A procedure for direct analysis of the waste samples for nitroso compounds by LC/MS under basic conditions that do not involve derivatization was developed. Using a mobile phase of NH_4CO_3 at pH approximately 10.5, a sample of real waste was analyzed. The mass spectrum of a major component and a standard of IDA were identical. The major ions observed included m/z 134 $(M + H)^+$ and m/z 151 $(M + NH_4^+)^+$. These results indicate the nitroso compounds were artifacts of the derivatization procedure and formed under the acidic analysis conditions.
Deuterated Chelator Probes
Deuterated chelators and/or chelator fragments have many distinct advantages over the introduction of other organic acid materials that might be used as yield probes. The methylation chemistry should be identical with that of the ordinary chelator found within the matrix, thus giving a direct measure of the efficiency of the methylation process for that particular chelator. If the labeled material is not subject to deuterium-exchange processes with the highly basic matrix, then the deuterated material can be used as a tracer; added at any point in the process, the probe acts as a monitor for loss of that material during any set of steps. The major criteria for use of these materials involves 1) deuterium stability, 2) well-defined

amount/response ratios in quantitation, and 3) MS resolution for analysis of particular fragment ions. Unfortunately, the deuterated compounds either co-elute or overlap the parent unlabeled material to some degree, necessitating ion selective monitoring by MS in order to utilize their benefits.
D4-succinic acid was purchased for initial trials. This material was found to withstand deuterium exchange if added directly to the highly basic Tank 101-SY matrix and derivatized immediately. Deuterium exchange on carbons alpha to carbonyl functions is expected to take place upon prolonged contact with base; thus d4-succinic acid is likely to suffer from some exchange during contact with the matrix. No apparent hydrogen/deuterium exchange was observed in our initial experiments, which was somewhat surprising.
A better probe would not possess exchangeable deuteriums, and an attempt was made to obtain labeled EDTA with the deuteriums on the N-N' bridge carbons. This material was not readily available commercially, and quotes from a manufacturer of labeled materials indicated that purchase would be prohibitively expensive. Methylation using BF3/methanol and subsequent GC/MS analysis indicated that the material was essentially free from chelator-type byproducts; a trace of the corresponding ED3A methyl ester was detected. High resolution MS performed with the JEOL High Resolution tandem mass spectrometer confirmed the expected deuterium content of this material. Fragment ions were observed at m/z 293, 192, 176, and 148. The structure of methyl ester of deuterated-EDTA is shown below.
Eq.
In general, the recovery was shown to be nearly quantitative; with the exception of HEDTA, which has been shown through prior work by Toste et al. (14) to undergo only partial conversion to a volatile derivative. Recovery data is shown in Table II. In addition, the use of smaller sample sizes resulted in higher recoveries.
TABLE II
Since derivatization GC/MS is time consuming and costly, other methods for direct analysis of waste are being evaluated, such as electrospray MS, atmospheric pressure chemical ionization MS, and capillary electrophoresis.
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NON-DESTRUCTIVE ANALYSIS OF THE RADIONUCLIDE INVENTORY OF LLW-DRUMS BY TRANSMISSION & EMISSION TOMOGRAPHY

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ABSTRACT

Temporary storage, transportation and final disposal of radioactive waste containers have to meet certain requirements of governmental regulations, one of which is the specification of the radionuclide inventory. In this respect a need emerges to thoroughly examine the containers' contents. This can either be achieved by destructive or non-destructive assay methods such as gamma scanning or tomography. Since destructive methods require reopening of the containers and also time consuming drilling within their contents the non-destructive analysis is generally the more preferred method. This paper describes an accurate gamma ray assay method allowing to identify all detectable radioisotopes present in Low-Level-Waste-drums (LLW) and to quantitatively determine their activity. The analysis is performed by applying a combined transmission and emission tomography technique. The paper gives a comprehensive insight of this tomographic assay procedure and describes the tomography assembly, which is extensively in operation at the KFA. Actual measurements on real LLW-drums are a special subject of the paper, and additionally, reconstructed images of the density and activity distribution of the drums' contents are included.

INTRODUCTION

The characterization and disposal of nuclear wastes is a problem of national and international importance. Governmental regulations have made it necessary to certify radioactive wastes in compliance with reliable assay techniques. The technique of segmented gamma scanning is currently the most important NDA-system (Non-Destructive Assay) to determine the radioactivity contained in large volume drums. In the case of a homogeneous waste matrix with densities between 0.1 and 3.0 g/cm3 the determination of the specific and total activity is fairly straight-forward. However, because the calculated activity is strongly dependent on the absorption characteristics of the waste matrix one or more correction factors must be included if inhomogeneous structures are present (1). A better estimation of the activity can be achieved by applying tomography techniques.

This paper exclusively focuses on 200 liter drums containing Low-Level-Waste with a maximum surface dose rate of 2 mSv/h. The radioactive waste in the drum is often compacted or cemented. If a cemented matrix of relatively high density is used to condition the waste, a considerable fraction of the inner radiation is absorbed in the drum itself. This has to be taken into account in an effort to obtain reliable results using the technique of emission tomography to quantitatively determine the specific and total activity of the waste drum. The analysis becomes even more difficult if the activity and the density of the material are distributed inhomogeneously within the drum, or if e.g. the drum contains canisters of unknown kind, number and size. The preliminary objective is the measurement of the density distribution of shielded structures and heterogeneous fillings inside nuclear waste

drums by using a transmission tomographic method, followed by the analysis of the activity distribution of the radionuclides contained in the drum by applying the technique of emission tomography.

TRANSMISSION TOMOGRAPHY

The aim of Transmission Computerized Tomography (TCT) is the determination of the density distribution of the waste drum by attenuation measurements. The general physics of TCT and its necessary components are quite elementary: the intensity of a gamma beam radiated by an external source and attenuated by the object (waste drum) is measured at various positions of the object. The drum is turned to various angle positions ranging from 0 to 180 and moved linearly through the gamma beam by a stepping motion. The attenuation data is taken at each single step. The source emits gamma rays from the decay of a radioactive isotope, such as 60Co. Typically, a high activity source is used to reduce the measuring time necessary for each drum position. Because of the fact, that an external source is required TCT is termed to be an active method. Knowing the count rate I0 of the detector of an unperturbed gamma beam, the waste drum's attenuation through a distance L can be deduced from the formula:

$I/I0 = \exp(-L)$

where I is the count rate measured at a particular drum position. Each detector reading represents a single data line for the waste drum. As the drum is moved these lines form a complex matrix of overlapping data. The work is then left to a computer reconstruction program which, in this case, iteratively constructs an image of the observed cross section of the drum by matching the plethora of measured data. The reconstruction technique, applied here, uses an ART-algorithm (Algebraic Reconstruction Technique) (2).

EMISSION TOMOGRAPHY

The aim of Emission Computerized Tomography (ECT) is the determination of the activity distribution of the radioisotopes contained in the waste drum by applying emission measurements. The radiation of various gamma sources located inside the drum is monitored by a detector at different drum positions. An external gamma source is, therefore, not required (passive method). The drum is moved in the same way as described in the transmission case, however, with angle positions ranging from 0 to 360. In general, the inner attenuation of the emitted radiation of the nuclides contained in the drum can not be neglected. For this reason the attenuation coefficients obtained from the transmission measurements are applied to correct the emission data. The sources inside the drum consist of a great variety of radioisotopes emitting gamma rays at numerous energy levels. The transmission analysis using the 60Co-source, however, is carried out at one of the two well-defined energy peaks of 60Co, preferably the upper peak at 1.332 MeV. Due to the fact, that the attenuation coefficients are strongly energy dependent the reconstruction software requires a well-fitted approximation of the energy-attenuation-relationship in terms of mathematical equations. The attenuation coefficients of various materials at different energies can be taken from literature tables (3). The reconstruction software uses a special density corrected ART-algorithm.

TOMOGRAPHY ASSEMBLY

The setup of the tomography assembly is of the translation-rotational type. The drum is positioned onto a turn table, which itself is mounted onto a linear moving bench. Both axes (rotational and translational) are driven by stepping motors, which are powered by two PC-controlled power drives. The stepping resolution of 1000 steps per revolution ensures exact positioning of the drum, which is necessary in TCT and ECT applications. Due to the high density of the cement filling a strong gamma source of 260 GBq 60Co (7 Ci) was installed as the external source within the tomography assembly. To keep radiation low for the operating personnel the gamma source is shielded by 82 kg of the very dense material 238U. The gamma beam can be exposed by remote control. The detector, which is used to collect the quanta passing through the drum, is a high purity coaxial Ge-detector of a relative efficiency of approx. 20 %. The gamma rays can be collimated by a variety of collimators different in diameter ranging from 6 to 40 mm. The material of all collimators is lead. Both the gamma source and the detector including the collimator can be manually moved in vertical direction in order to observe different horizontal cross-sections of the drum. A pre-amplifier, an amplifier and a Multi-Channel Analyzer (MCA) serve as nuclear electronic instrumentation. The nuclear data acquisition, the

synchronization with the mechanical movement of the drum and the tomography reconstruction are carried out by a standard PC. The visualization software enables the operator to quantitatively determine the density and the activity distribution of the drum by calibrated density and activity scales.

MEASUREMENTS ON A TEST DUMMY

To check the reconstruction software and the accuracy of the tomography assembly a test dummy was designed and manufactured. The dummy embodies 3 different materials, different in density (lead, steel and plastic material). These materials are enclosed by a matrix made of concrete. Since the size (diameter 56 cm) as well as the materials chosen for the dummy are those of a real waste drum the dummy allows to perform precise measurements under realistic conditions. Figure 1 shows a reconstructed horizontal cross-section of the test dummy including the insets of different materials. The calculated density distribution and the calibrated density scale enables to quantitatively determine the density of all parts of the dummy. The plastic material (right side of the figure) is only barely visible, because the type of plastic used had similar attenuation properties as the cement matrix. The image is set up by a 301x301 pixel matrix. It is based on the data of 156 angle positions, each consisting of 100 linear scan steps. The measuring time of each single step was set to 5 seconds. The quanta emitted by the source and traversing the object were collimated by a 6 mm collimator.

Fig. 1. Calculated density distribution of the test dummy.

The dummy also contains several holes and hollow spaces, in which radioactive samples can be inserted in order to simulate a gamma ray emitting waste drum. In nuclear waste management the key nuclides 60Co and 137Cs are of great interest. To simulate a self-emitting gamma spectra by the drum matrix with an energy range from 662 keV to 1332 keV several radioactive samples, namely 60Co and 137Cs, were inserted into the holes of the dummy. Inner shieldings and heterogeneous fillings are then simulated by the different insets enclosed by the concrete matrix. On this setup emission measurements were carried out and the activity distributions of both nuclides were calculated. Fig. 2 shows the result of the reconstruction of the 137Cs-activity distribution of five Cs-samples of 31 MBq each. The image is set up by a 127x127 pixel matrix. It is based on the data of 64 angle positions, each consisting of 50 linear scan steps. To keep statistical noise low the measuring time of each single step was set to 10 seconds. In order to collect enough quanta emitted by the sources a 15 mm collimator was necessary. The maximum count rate of approx. 770 c/s was monitored close to the Cs-sample near the edge on the left side of the dummy.

Fig. 2. Activity distribution of 5 137Cs-samples inserted into the dummy.

In nuclear waste management it is often of great importance to quantify the activity of a single source located at a particular position inside the waste drum (hot spot). Once a tomographic image is reconstructed this can easily be achieved by integrating, i.e. summing up, the pixel values of a certain region, in which the source is located. This technique then quantitatively gives the activity of the particular source. The five circles surrounding the Cs-samples in Fig. 2 indicate the regions of integration and the numbers near the circles display the calculated activities of these regions, i.e. the activities of the five sources. Keeping in mind, that each sample has approx. 31 MBq (10%) the technique yields results of appropriate accuracy.

As mentioned above the reconstruction algorithm uses attenuation correction of the emitted rays along the ray paths from the source to the detector. This correction has a great influence on the reconstruction of the activity distribution. Calculation of an activity image by using the identical algorithm and the same input data, but without applying density correction, gave only unsatisfactory results. In the case of the setup shown in Fig. 2 the calculated image permitted to localize only the very left source near the surface of the dummy, whereas the value of its activity was not calculated correctly.

MEASUREMENTS ON WASTE DRUMS

Figure 3 shows the density distribution of a 200 l waste drum of approx. 450 kg in weight (surface dose rate approx. 0.35 mSv/h). The density distribution was obtained by performing transmission tomography on the drum. The drum contains an inner canister surrounded by a thick cemented filling. The inner canister's contents is cemented material, which is distributed quite inhomogeneously showing a few, well defined cracks. A hole of 6 cm diameter caused by a preliminary destructive assay

can also be recognized. It is quite likely, that the inner canister contains radioactive material and that the outer cemented region is used for shielding purposes and will probably not contain any radioisotopes. This assumption is confirmed by the corresponding ECT-image (Fig. 4). The image shows the 60Co-activity distribution of the same drum at the same horizontal cross-section. All of the radioactivity is contained within the inner part of the drum, whereas the outer region is basically just a shielding. The ECT activity image also shows the above-mentioned hole, which obviously cannot contain any radioisotopes. This activity hole would not be visible if the reconstruction algorithm had not used density correction of the emission data. The activity scale of the image is not calibrated, but does give an impression of the relative quantity of the activity concentration within the observed cross-section.

Fig. 3. Density distribution of a LLW-drum (200 liter).

Fig. 4. 60Co-activity distribution of the drum shown in Fig. 3.

CONCLUSION

The experimental tomography assembly described in this paper revealed that the contents by density of nuclear waste drums can be determined non-destructively by the technique of TCT. In addition, all detectable radioisotopes can be localized within the drum volume by applying ECT. Furthermore, by applying density corrected ECT, i.e. combining TCT with ECT, the radioisotopes can be determined quantitatively. The knowledge obtained by operating and running experiments on the assembly was an aid in the construction of a new tomography assay system. The design of this advanced system has been recently completed at the KFA and the system is currently being manufactured. The new advanced tomography assay system will have extensive tomography capabilities (transmission and emission tomography), will perform gamma scanning and digital radiography, and is designed for fast assays of the waste drums. The assay system will be applied to inspect their contents in order to declare and certify their radionuclide inventory.

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LOW LEVEL WASTE DRUM GAMMA SCANNER; THE UNITED KINGDOM ATOMIC ENERGY AUTHORITY (WINFRITH)

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ABSTRACT

This paper describes the specification, procurement, commissioning and early operation of a solid waste drum gamma scanner. Batches of up to ten 220 litre drums are loaded by conveyor on to a turntable. Four fixed electro-cooled germanium detectors can use three alternative waste matrix correction methods for counting the gamma emitting isotopes present. The procurement, testing, operational procedures and initial results are discussed. It is expected to be fully operational by April 1995.

INTRODUCTION

Disposal of low level radioactive solid waste in the United Kingdom is subject to regulatory control. It is becoming a requirement to directly measure at least a proportion of waste arisings. This paper describes a design by AEA Technology, Harwell Instruments, for a Low Level Waste batch drum segmented gamma scanners (SGS) already installed in the United Kingdom. Early operational results are discussed,

prior to full operation in April 1995.
The scanner is located at the Winfrith site of the United Kingdom Atomic Energy Authority and will directly measure gamma emitting isotopes in waste contained in 220 litre drums. This site has arisings from irradiated fuel, power reactor decommissioning and waste from other commercial companies. The drums are counted before compaction and disposal at the Drigg waste repository operated by British Nuclear Fuels.

SPECIFICATION

A comprehensive specification was drawn up and submitted to Harwell Instruments, AEA Technology, Harwell, Oxfordshire, England. Harwell Instruments have a commercial agreement with EG & G Ortec Instruments for the supply of electronic equipment. The following criteria were included in the technical specification:

Nuclear industry safety standards and codes of practice.

Waste drum size to be 220 litre.

A minimum caesium-137 detection limit of 100 KBq in a matrix with density 0.4 g/cc using a drum cycle time of 10 minutes.

Ability to handle drums up to 1 tonne in weight.

Ability to take batches of 10 drums.

Ability to accommodate radiation levels from background up to at least 2 mSv/h contact.

Ability to run automatically with minimum operator intervention.

Ability to be largely unaffected by radiation from adjacent drums.

Ability to be flexible in matrix corrections and be able to act as a research instrument when required.

To have adequate warranty conditions.

To be delivered within nine months from time of order.

To have a capital cost that did not significantly exceed that of other available scanners.

GENERAL DESIGN FEATURES

The design opted for four vertical electro-cooled detectors. These were judged to require the least maintenance and provided the neatest detector arrangement.
The operator inputs the drum identification parameters and if required a 'fingerprint' to attribute any associated alpha and beta activities. Drums are fed automatically from a commercial input conveyor and are rotated on a turntable in front of the fixed 25% high purity germanium detectors. After counting they accumulate on an output conveyor (Fig. 1).

Fig. 1.

The data is processed by an industrial standard 486 computer. A drum analysis summary is written to a printer for immediate inspection. Full spectra and analyses details are written temporarily to the hard disk under reusable batch filenames. An abridged version of the printer data file is permanently stored, and can subsequently be down-loaded to the site waste data base system. Each batch spectra and analyses files are retained initially on the SGS for inspection by the system supervisor if required. As each drum generates about 1.5 MBytes of files, these are allowed to be overwritten by the next batch of drums to prevent excessive build-up on the hard disk. They can of course be stored permanently under a different filename if needed.
Peak identifications and analyses are carried out using an Ortec Maestro pulse analyser emulator and Ortec Omnigam peak analysis software. Harwell Instruments software is provided for instrument control and to allow drum matrix corrections, using the following three alternative methods:

a) Europium-152 transmission sources through the waste drum.
b) Intrinsic differential peak absorption from a multi-gamma isotope in the waste.
c) Bulk density from weight measurement and inbuilt correction factors.

The supervisor can select the method preferred, although the machine will default automatically from the differential peak option to weight correction in any segment where no suitable intrinsic isotopes can be found.
This is believed to be the only SGS in the UK with the capability of using these three matrix correction techniques.

MANUFACTURE COMMISSIONING AND OPERATION

Our Project Manager maintained continual close contact with the supplier throughout the stages leading up to delivery. This proved invaluable in deciding quickly upon detailed and sometimes unforeseen options. Close liaison was welcomed by Harwell

Instruments and proved crucial to the successful delivery within specification and time scale.
The machine was completely assembled at the Harwell site and made ready for a comprehensive series of mechanical, electrical and sensitivity tests. All aspects of the specification were individually verified by Harwell Instruments and the Project Manager in Works Acceptance Tests (Appendix 1).
After agreement had been reached, the SGS was dismantled and transported to Winfrith where much of this testing was repeated as a Site Acceptance Test. All details became part of the SGS Commissioning Quality Plan.

CALIBRATION

Energy and Absolute count-rates

Standard point sources in the centre of an empty drum have been used for calibration of the SGS. This was easier than making homogeneously distributed sources in as large a volume as a 220 litres. The effects of changing between the point and homogeneous distributions at different drum densities can be readily calculated.
The SGS is energy calibrated with caesium-137 and cobalt-60 sources. Efficiency calibration is by a single standard europium-152 source of 3.5% accuracy. The source is placed in the centre of an empty drum directly opposite each detector in turn. From the known energy and abundance of each peak, energy and efficiency calibrations are filed for future use.
After calibration the computer requires an immediate check by a verifier drum. This is automatically demanded again at future intervals, presently set at seven days. The drum contains four standard caesium-137 sources in the same vertical positions as used in the europium calibration. The measured caesium activities must agree within a pre-set percentage with the europium calibration or no further waste assays are allowed. Any differences are normally well within 3% and act as confirmation that the original calibrations have been carried out correctly.

Geometry and Matrix Corrections

Factors have been incorporated into the algorithms that take account of different drum densities. They also convert the calibration point source geometry into equivalent counts for a uniform distribution, which is assumed for unknown waste drums.
The differences due to activity positional errors have been investigated experimentally. Point sources have been counted in a series of tests in specially constructed drums fitted with vertical re-entrant plastic tubes. These enable sources to be introduced at varying heights across the drum that was filled with various matrices.
A single 2.97 MBq caesium-137 source is used in an empty 220 litre drum. Measurements are made at 5 cm vertical intervals in centre (Fig. 2) and edge tubes. It can be seen that whilst there is significant variation with position, there is no significant 'cross-talk' between the detectors. This has been repeated with the drum packed tightly with PVC (bulk density about 0.22 g/cc). Changes were small at the drum centre but the variations increased near the edge. Further tests are planned.
Fig. 2.
Each value shown has been already been corrected by the SGS software factors. The totals (MBq) from both experiments are shown (Fig. 3, centre source in-filled). If the compensations over a complete drum height and diameter are correct, the mean values for the sets of the 17 vertical measurements and every single point should be 2.97 MBq. The individual values are actually seen to vary in extreme cases by as much as 20% to 30% and 50% to 60% at the centres and edges respectively, depending upon the actual measurement positions chosen. However the overall mean values for all points are accurate to +/-2%.
Fig. 3.
The count rates decrease when the sources are positioned midway between the detector centres. This effect is least when the source is at the drum centre and greatest at the very edge. It is even just possible to miss the presence of a point source at the very edge of the drum, but only over a vertical height of less than one centimetre.

MATRIX CORRECTION METHODS

Transmission Sources

This is the least demanding and most generally applicable method. The drum segment counting is followed by counting with the four 60 MBq europium-152 transmission sources exposed for between 10% and 50% of the original drum count time. The

percentage may be varied as required. Corrections are made for each drum segment and take into account any europium already present in the waste.
The greatest errors occur when the waste has high density and a high intrinsic count rate. This makes the transmission source low energy peaks difficult to measure, with corresponding uncertainties in the correction factors at these energies. The problem is compounded if europium-152 is also present in the waste.
Differential Peak Absorption
This requires the waste to contain an isotope with gamma peaks ranging from low to high energy values. By assuming that other radioisotopes present are also similarly distributed, this method may be utilised to provide an elegant method for correcting matrix density effects.
Special requirements include a suitable isotope being present in sufficient quantity to be counted statistically accurately over a range of peak energies, that the detector efficiency calibrations are very accurate and the matrix density does not reach some limiting value. As with transmission corrections, problems also arise with a high waste intrinsic count rate and high densities.
Experiments so far on the SGS have suggested that these requirements are often difficult, but the method is capable of giving good agreement with transmission source work. In some cases, when conditions are right, it may be more accurate. However, in spite of the theoretical attractions, it seems less robust than transmission correction.
Bulk Density
The drums are accurately weighed during counting, allowing the bulk densities to be calculated. Factors measured from calibration drums of different densities, correct the measured values as a function of gamma energy.
For matrix corrections based on bulk densities, no concession of any kind has been made to inhomogeneity. With transmission and differential peak methods, corrections are made for each segment. Thus the method is ideal for waste drums with uniformly distributed materials.
Some waste drums may have heavy items at their bases making a bulk density correction inappropriate. This method is used as a default when the other two have been ruled out.
PRACTICAL INSTRUMENTATION AND INSTRUMENTAL PROBLEMS
Electronic Stability
SGS ambient conditions vary between 10C and 30C. Temperatures inside the control desk even exceeded this maximum value in very hot weather. A cooling unit has now been fitted and augmented by spectrum stabilisation to maintain proper energy calibrations under these widely varying conditions. The latter is achieved through the main amplifiers by automatically opening the europium transmission sources between drum movements for checking and feedback corrections.
The operation of gamma equipment in an industrial environment can cause other problems. As well as extreme temperature differences, electrical interference and environmental pollutants can cause impaired performance. It has proved essential to guard against these. If it is not possible to control the immediate environmental conditions, the equipment should at least be properly sealed and some temperature limiting introduced.
Initial difficulties associated with earth loops were also encountered, which only started after moving the equipment from the contractors premises to the Winfrith waste plant. The effect was a broadening and sometimes severe deformation of the peak shape. This was eventually minimised by isolating the flexible metal detector cooling pipes from each other. No other earth linking procedures seemed to be reliable. The insulated system now operates close to the designed resolution of 1.95 keV for all detectors.
Prior to using insulation, unpredictable changes took place in which peaks could fail to be properly identified. Single peaks occasionally appeared as doublets and sometimes the system would reject broad or misshapen peaks, giving inaccurate answers. Such occurrences might be overlooked if only the final summarised output data from unknown constituent drums are inspected. Only frequent inspection of spectra and full width half height value ensures good data.
High Count-Rates
Accurate pulse dead-time corrections at high count-rates must be assured. These can reach over one hundred kilohertz of accumulated counts in each detector, from drums with fission products at contact doses of 2 mSv/h or 5 mSv/h from cobalt dominated

waste. Both correspond to 80% dead-time.
Practical tests have been carried out using low level caesium-137 and cobalt-60 sources counted initially by themselves and then with gradually increasing count-rates from a separate europium-152 source. Experiment showed an accuracy of 0.3% on cobalt and 2% on caesium was still maintained at 100 kHz total count-rate. This only increased to 0.4% for cobalt, but to 11% for caesium at 300 kHz. These errors also include a component for peak area measurements where a very significant background is present, especially at lower energies. At these dead-times the accumulation times have become very long however and lower efficiency detectors would be preferable for such real waste material.

Sources

The present europium calibration source has a quoted accuracy of +/- 3.5%. Whilst better accuracy is available, this is probably adequate bearing in mind other errors and assumptions in the overall method. The cross check with the caesium sources in the verifier drum agrees to at least this accuracy.

Calibration

Great care has to be taken with system calibrations. It has been found essential to ensure sufficient source counting times, so as to get very good peak shapes and statistics. Discrepancies between measurements in earlier commissioning were traced to this cause. The present system collects spectra in only four thousand channels, although this is being changed to eight thousand.

The analysis software requires peaks to be within 1 keV of their library value, or the isotope is not recognised and is returned as an unidentified peak. In this way even a major isotope could go unreported.

DATA GATHERING

The SGS generates and uses much data. The deductions it makes should be at least as good as those of an experienced analyst. Only the system supervisor has access to manually inspect these spectral and analytical files. However except for unusual cases, it is only normally possible to look at data occasionally. Correct initial settings and conditions must be maintained to make this satisfactory.

Peak analysis will identify not only major and genuine isotopes but also a very large number of high uncertainty and low abundance peaks. In mixed fission and activation product waste, the process may start with as many as two hundred possible peaks present, many of which may be based on uncertain statistics. Although rejection of most peaks normally follows automatically from unacceptable shapes, resolutions, non-exact energies, or non substantiating additional peaks, operator examination is still needed in some cases.

REPORTING OF ISOTOPES

The unidentified peak energies from many initial drums of waste measured on the SGS were examined. Starting with only a basic isotope library, many unidentified peaks were reported. These were listed and sorted on a spreadsheet, which enabled more commonly occurring values to be seen and attributed to real minor gamma constituents. The isotope library was then extended to include these isotopes. If this procedure had not been not followed, then the great number of unknown peaks would have caused a time consuming evaluation problem for the system manager. There is significant time saving in only needing to examine the small number of remaining peaks.

A problem remains of how to deal with the reporting of those isotopes that cannot be accurately quantified by gamma spectroscopy and would be better measured by a different technique. The decision to eliminate some or all of these from the results, seems to depend upon the experience and common sense of an operator.

In our system this function can only be carried out by the system supervisor. The isotopes over which some doubt exists are isolated as a group in the print-outs and data files. The actual spectra from each drum segment are then examined and considered on their merits. The removal of unwanted data is accomplished on a separate computer using the DOS Editor, or more usually by reading the file into a spreadsheet. For those data sets where standard changes are required, macros are available to make these alterations quickly and accurately.

COMPARISON OF MEASUREMENT METHODS

Data from the different matrix correction methods is still being accumulated. A typical example for a drum of well-characterised waste stream of uniform waste with density 0.25 g/cc, shows quite small differences between total activities as measured by (1) SGS transmission, (2) direct assay from fingerprint and radiation

measurements and (3) SGS bulk density/weight corrections (Fig. 4).
Fig. 4.

SYSTEM SAFEGUARDS

The use of this SGS at UKAEA Winfrith is replacing alternative methods for determining activity contents, each of which also involves manually transcribing data on a keyboard with the attendant possibilities for errors.

The SGS system can only be accessed by the supervisor, preventing any tampering with the SGS system or the introduction of unauthorised software. The SGS specific software employs a tier password system. No access at all is available to unauthorised persons. The first password allows entries only those menu items for entering drum identities and for starting and stopping the system. All remaining menus governing the way the system is run, parameters used, analytical and all data files, require a second password known only to the system supervisor. These analyses options are held on disk and so are less subject to errors. They include counting times, type of matrix correction used, calibration parameters, tolerances and system management parameters.

The SGS will not allow drum assays unless the system verifies within set error limits.

FURTHER DEVELOPMENTS AND LIMITING FACTORS

Matrix Corrections

Only four drum measurement segments are measured with this SGS design due to a need to minimise throughput times. Other systems with less time priority can afford to measure more segments, with an attendant potential improvement in corrections. Although this machine cannot readily be altered in this respect, better information on waste matrices could bring further improvements in accuracy.

If the latter were possible, source and density configurations other than those assumed in the calibration could be calculated and the factors applied to the measured values. This may be a way forward to more exact analyses. X-radiography, counting using tomography or counter arrays giving spatial distributions may provide this information.

Perhaps the biggest error of all lies with a dependence upon an accurate knowledge of the non-gamma constituents. This has to be obtained from destructive analyses and may account in some cases for two thirds of the total activity.

LIMITS OF DETECTION

The original SGS specification called for a limit of detection of 100 kBq caesium-137 in a matrix of density 0.4 g/cc in a 10 minute cycle time. Subsequent experiment suggests that less than 40 kBq uniformly distributed may be detectable in a low background environment and using extended counting times, even with drums of density of 1 g/cc. This is equivalent to 0.2 Bq per gram and less than the UK Free Release limit.

Free Release of waste is the topic most likely to give the best financial return on effort invested. The cost differential between dumping at active repositories and public waste tips is very significant when disposing large volumes of demolition waste.

Our furthest advanced Free Release project has been on drums filled with used fire detectors with their americium-241 removed. Preliminary tests show that a point source of 37 kBq of americium-241 can easily be detected in the centre of a very low density alarm case matrix drum. Further special drums are being made to see if Free Release Limits can be achieved. Calculations suggest that a similar uniform distribution would give a 30% better limit of detection.

ACKNOWLEDGEMENT

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APPENDIX 1

Items considered in Works and Site Acceptance Tests

- Accessibility to replaceable items
- Safety checks on electromechanical parts and radiological surface dose rates
- Loading and off-loading batches of drums
- Turntable operation
- Drum weighing
- Correct and safe transmission source operation
- Power failures and emergency stop situations
- Drum sequencing and verification checks

System information outputs
Effects on assay from nearby active drums
Uninterrupted power supplies operation on mains power failure
Correct close down sequences
Software tests
Matrix correction tests
Sensitivity and limits of detection
Correct use of 'fingerprint' factors for beta and alpha constituents
Activity distributions against height
Documentation checks

50-8

A PORTABLE SAMPLE MONITOR FOR LOW-LEVEL ALPHA CONTAMINATION

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ABSTRACT

A portable swipe monitor (PoSM) has been developed at Los Alamos National Laboratory for bench top or field use. This rugged detector system uses Long-Range Alpha Detector (LRAD) ion transport technology, which monitors ionization created by alpha interactions with ambient air. The PoSM has been designed to be easily portable and insensitive to mechanical noise. Real-time background subtraction is built in, adding to its convenience in field use. Preliminary tests reveal that the PoSM is an inexpensive, fast device capable of monitoring a variety of samples in the field with very reliable statistics. Sensitivity to alpha contamination is on the order of a few dpm/100 cm2.

INTRODUCTION

In early 1994, Los Alamos National Laboratory (LANL) began developing a radiation sample monitor for assay of swipes or smears. The goal was to develop a device that could make fast, reliable measurements in a mechanically noisy environment. The primary design consideration was that it be insensitive to vibration and still yield reliable statistics in a short time. We used Long-Range Alpha Detection (LRAD) ion transport technology because we expected that it would perform well under these conditions. A prototype, which we call an LRAD-based Portable Swipe Monitor (PoSM), has been built and is being tested.

We believe that the system will have applications beyond the initial intent of the project. The PoSM is likely to be an improvement in terms of cost, sensitivity, and convenience over conventional swipe/smear monitoring technology. The prototype is 16.5 cm x 19 cm x 33 cm, weighs less than 5 kg, and can be easily carried with a shoulder strap. Its light weight and small size make it potentially very useful to radiological control technicians (RCTs). Currently, swipe monitors that are sensitive to low-level contamination tend to be large, stationary devices, so RCTs have to take swipes out of the field to be analyzed. The PoSM is a fieldable alternative.

Contamination collected on swipes is by definition removable. Transporting swipes before analysis introduces cross-contamination issues and the problem of diminished accuracy when contamination is lost from the swipe. If the contamination level has diminished, then 1) the contamination has spread and 2) the original object of interest has been mischaracterized. If the swipes had been assayed when they were taken, the likelihood of these two problems would have been greatly diminished. Other problems inherent with removing the swipes from the field involve the efficiency of work done in nuclear facilities. Many kinds of work at nuclear facilities are supervised by RCTs who must make decisions based on the analysis of these swipes. The ability to take these measurements quickly in the field could eliminate work delays that occur while waiting for the analyses. Having a PoSM in the field could also alleviate much of the paperwork that is required when keeping track of large numbers of swipes and free up time for overburdened technicians.

DETECTOR CONCEPT

The strong alpha signature of transuranics, uranium, radium, and many other radioactive contaminants and the relatively high energies involved in alpha decays make alpha detection an excellent method for locating and measuring radioactive contamination. Alpha particles, however, travel only very short distances in air before all of their energy is dissipated in interactions with air molecules. The alphas deposit their energy in air by separating the charge that resides on air molecules. The lifetime of the ions that are created is on the order of seconds. Researchers at LANL have found that ions generated in air can remain at over half their initial concentration for as long as 8.4 s (1). Ions can therefore be transported much further than the alpha's range, which is typically 3.5 cm.

Long-Range Alpha Detection (LRAD) is the technique of detecting alphas by means of their ion progeny. In the LRAD swipe monitor, these ambient ions are transported to a detection electrode by a high-voltage electric field and generate currents on the order of 10-15 A. Research at Los Alamos has shown that this current is linearly proportional to the alpha activity (3) and is therefore a measure of alpha contamination.

There are two types of LRAD detectors distinguished by the ion transport mechanism employed. Airflow LRADs use fans to transport all of the air, ionized, and non-ionized molecules indiscriminately, to the detection electrode. Electrostatic LRADs, such as the PoSM (Fig. 1), transport only the ionized molecules to the detection electrode by means of an electric field. Of the two types, electrostatic LRADs are capable of greater efficiencies and higher signal to noise ratios.

Because the PoSM is intended to be a conveniently fieldable device, the background is subtracted in real time to alleviate the need to recheck background each time the device is relocated. Figure 1 shows an elevation view of the PoSM. The sample is placed on the bottom of the drawer, which seals when closed. The bottom of the drawer and walls of both volumes are kept at the potential of the negative post of the battery. The signal planes are kept 300 V above this. All of the negative ions that are generated in the sample volume are collected on the lower signal grid. Only those negative ions that are generated in the background subtraction volume are collected on the upper signal grid. The two chambers have equal volumes.

Fig. 1.

The background signal can be attributed primarily to three sources: 1) leakage current, 2) penetrating radiation, and 3) radon.

Because of the very low current levels involved, care must be taken in the design to avoid leakage currents along the surface of the insulating standoffs. These currents can be of the same order of magnitude as the currents that constitute the detector signal and so are capable of swamping the detector. To avoid these currents, the standoffs in the PoSM are guarded. The guards consist of metal rings (as shown) that are kept at the same voltage as the signal planes. Because this eliminates the potential gradient along the surface of the insulator between the signal plane and the guard ring, no charge flows between them. The leakage that takes place between the guard rings and ground is not sufficient to affect the potential of the rings and so has no effect on the signal plane. In this way, leakage currents along the insulators are effectively eliminated.

The ionization in the decay volume caused by radon results from the decays of both the radon isotopes and the alpha-emitting radon daughter products. Because radon is a gas, the amount of background contributed by disintegrations of radon isotopes for a given concentration is directly proportional to the volume (4). Because the chambers have equal volumes, we can eliminate the portion of the background due to the radon disintegrations by subtracting the top chamber's signal from the bottom's. The elements created by the decay of radon are not gaseous at room temperature, so they plate out onto the walls of the chambers. Assuming that on average the radon concentrations are the same in both chambers, the amount of alpha emitting daughter products plated to the walls and electrodes of both chambers will be the same within a few percent. Subtracting out the top chamber's signal therefore eliminates this source of background as well.

Penetrating radiation deposits very little energy in the detector volume compared to what is deposited by the alpha contamination on the sample. Gammas caused by cosmic rays and terrestrial background radiation combined deposit only about 4 MeV per minute per liter of detector volume compared to the approximately 516 MeV per minute that just 100 dpm of 239Pu contamination would deposit. The two decay volumes used

in the PoSM are 3.1 L, so even without the background subtraction, background penetrating radiation is not significant.

RESULTS

Figure 2 shows a calibration set taken with National Institute of Standards and Technology (NIST) traceable 239Pu sources. As shown in the figure, this detector sees 166 fA per 1000 dpm. The average alpha energy from 239Pu is 5.16 MeV and 34 eV are required on average to ionize one air molecule. If we assume that ionization is the only mechanism for energy loss in air, we find that about 152,000 ion pairs are created by each alpha. The PoSM sees only one ion for every pair because the positively charged signal planes collect only negative ions. Because half of the alphas will have trajectories which take them back into the source, 1000 dpm of 239Pu can be expected to generate around 203x10-15 coulombs of free charge per second in the air. This charge would become 203 fA if the signal planes picked up all of the negative ions that were generated. The 166 fA response indicates that the PoSM is about 82% efficient.

Figure 3 shows the results of a proof-of-principle test intended to determine how well the detector performs in mechanically noisy environments. A rough running vacuum pump was used as a makeshift mechanical vibrator. The outputs of the two electrometers are shown as the detector is picked up and set against the pump. (The top plate signal has been shifted up 2 mV so that it can be distinguished more easily.) The plot shows 60-s averages.

Fig. 2.

Fig. 3.

This figure shows rather dramatically that the noise in the signal is not affected by the vibration. The standard deviations of the points before and after setting the detector against the pump are 0.14 mV and 0.13 mV respectively for the top and bottom plates. The standard deviations while the pump was vibrating the detector are 0.14 mV and 0.18 mV. Of course this simple test does not guarantee that the PoSM will perform well bolted to a truck with its engine running, but the results are encouraging. The opportunity for that sort of field test has not yet arisen.

COMPARISON OF LRAD SWIPE MONITOR AND ONE TRADITIONAL METHOD

The comparison data shown in Fig. 4 was taken with an LRAD-based sample monitor (y axis) and a Berthold LB770 10 Channel Low-Level Planchet Gas Proportional Counter (x axis). The LRAD-based device is a prototype that was developed primarily for looking at small soil samples. It preceded the PoSM and is very similar in design. The figure shows results from 13 swipes, taken in a known contamination area at a nuclear facility in Los Alamos. The R2 value reflects a high degree of linearity between the two technologies.

Fig. 4. Comparison of LRAD-based swipe monitor and a commercial monitor based on gas proportional technology. The square of the regression coefficient, R2=0.999.

One advantage that the gas proportional counter and other traditional swipe monitoring technologies have over the PoSM is that they provide both alpha and beta information. Future plans could include modifying the PoSM to give beta information as well. The modification would involve sandwiching a beta-sensitive scintillator between the signal planes and adding the necessary electronics. We anticipate that the heavy lead shielding required by the Berthold device (which would severely compromise the PoSM's portability) could be avoided by adding a pulse shape discriminator to the electronics. The lead shielding is necessary to exclude ambient gamma radiation, but the pulse shape that gammas produce in scintillators is distinguishable from the shapes produced by betas. The PoSM could thus retain its portability, although it would be made somewhat more bulky by a photomultiplier tube.

SUMMARY

The work on the prototype PoSM being developed at Los Alamos has demonstrated that an LRAD-based swipe monitor will work quite well, is extremely efficient, and is capable of operating in mechanically noisy environments.

The PoSM could fill an important role in the field of radiological control. A lightweight, convenient, portable swipe monitor for in situ assay could be a significant improvement on the current practice of collecting swipes and taking them to a lab to be counted. Having a PoSM in the field could have a significant effect on the efficiency of RCTs and the work that they supervise.

ACKNOWLEDGMENTS

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Session 51 -- Decommissioning & Decontamination
Co-chairs: Sue Mitchell, NUMATEC;
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51-1
PROGRESS WITH STAGE 1 DECOMMISSIONING OF THE STEAM GENERATING HEAVY WATER REACTOR (SGHWR)
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ABSTRACT
The Steam Generating Heavy Water Reactor (SGHWR) at the Winfrith site was a 100 MWe (330 MWth) power reactor, with a design based on the use of discrete vertical pressure tubes, with heavy water moderator and light water coolant at a sufficiently high pressure to directly drive a turbo-generator. The reactor operated as a successful prototype from 1968, until October 1990 when the decision to stop operations and to start Stage 1 decommissioning was taken. The Stage 1 program is scheduled to complete in March 1997.
Financial responsibility for the decommissioning has been accepted by the Department of Trade and Industry (DTI) and responsibility for ensuring that the program is carried out in compliance with safety, and other statutory regulations lies with the Government Division of the United Kingdom Atomic Energy Authority (UKAEA-GD). All decommissioning work is carried out within a policy agreed between UKAEA-GD and the DTI, which is to ensure that the overall plan for the facility is optimized to provide a minimum lifetime cost (discounted at 6%) and that value for money for all work undertaken is obtained, primarily by exposing as much work as is practicable to competition.
To meet the above policy a work program was devised with the primary objective of bringing the facility to a state where it can be left, for a minimum of twenty years, with the minimum of care and surveillance. This program has five main elements, namely:
The removal of all fuel and the decommissioning of the fuel handling and transport equipment.
The emptying of fuel pond water, debris and removal of pond equipment, and decontaminating or sealing the pond surfaces, as appropriate.
The removal from service of all systems not required for the resulting care and surveillance of the facility.
To clean, seal and provide ventilation as appropriate for the primary and secondary containments.
To demolish as required, all facilities external to the main containment buildings. This includes for example, the cooling towers, auxiliary boilers and emergency generators.
AEA Technology have been appointed the Managing Agency to provide the project management and to ensure that the DTI/UKAEA-GD policies are met, particularly in using competitive tendering to award implementation contracts for work undertaken. All individual tasks above 10k in value are subject to competition, unless it can be demonstrated that this is not cost-effective.
Good technical progress has been maintained over the first four years of the

project. The critical path contains the tasks associated with the fuel and pond program and early work has concentrated on these. All fuel was removed form the site ahead of schedule, and furniture and debris has been removed from the fuel ponds. Surveys to define the state of the facility have been completed and removal of plant and equipment that would result in an excessive long term maintenance burden (such as the cooling towers) is nearing completion. Most of the work to rationalize systems required in the longer term, such as power and instrumentation, is also complete. The main tasks remaining involve the decommissioning of the fuel ponds and the rationalization of the containments ventilation system.

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INTRODUCTION

The Steam Generating Heavy Water Reactor (SGHWR) grew out of studies, undertaken by the United Kingdom Atomic Energy Authority, into the possible alternatives to the gas-cooled, graphite moderated reactors on which the UK nuclear power program had been launched. The SGHWR design was based on the use of discrete, vertical pressure tubes, with light water coolant at sufficiently high pressure to provide steam directly to the turbo-generator. The moderator was unpressurized, heavy water, which also acted to control reactor power by varying its level in the calandria. The majority of the primary systems was contained in a reinforced concrete primary containment structure.

Construction of the prototype reactor of 100 MWe (330 MWth) at Winfrith started in February 1963. Synchronization to the national grid was on 24 December 1967 and the reactor was declared available for sustained generation at the design output in January 1968. The facility operated as a prototype, providing support to the evolving UK PWR program and generated power for the national grid. The decision to stop reactor operations was taken on 2 October 1990, at which point the decision to undertake Stage 1 decommissioning was taken. During its lifetime of 23 years, the reactor operated at a load factor approaching 60 per cent. A diagram of the facility is given in Fig. 1.

Responsibility for ensuring that decommissioning is carried out in compliance with safety, and other statutory regulations lies with the Government Division of the United Kingdom Atomic Energy Authority (UKAEA-GD) as owner of the Winfrith site. Financial responsibility for all work associated with decommissioning and maintaining the SGHWR has been accepted by the Department of Trade and Industry (DTI) who are funding the work through the Decommissioning and Radioactive Waste Management and Operations (DRAWMOPS) Program letter agreed with UKAEA-GD. The whole DRAWMOPS portfolio consists of well over 100 major facilities, of which SGHWR is one. All decommissioning work is undertaken within a policy agreed between UKAEA-GD and the DTI. In essence, decommissioning is to:

Ensure a safe condition which complies with statutory requirements at all times.

Ensure that value for money is obtained for all work undertaken, primarily by exposing as much as is practicable to competitive tendering.

Ensure that all work necessary for safety, environmental or planning consent reasons, or which is justified on economic grounds, is carried out.

Consistent with the above, ensure that the overall plan is optimized to provide minimum lifetime cost (discounted at 6%) within the constraints of available funding.

Fig. 1.

DECOMMISSIONING OBJECTIVES

The main technical objective of the decommissioning process is to provide overall lifetime minimum cost subject to satisfying safety and other regulatory requirements. A conventional three Stage decommissioning strategy is envisaged for the facility and early studies showed that it was economic to proceed immediately with Stage 1 in order to reduce the hazard posed by the facility in its immediate post-operational state. That is, the cost benefits obtained through the reduction in care and surveillance costs following Stage 1 decommissioning were seen to more than compensate for the costs of undertaking the Stage 1 activities. Stages 2 and 3 being more extensive in scope and not having as great an economic benefit on their completion, were not judged to be worthwhile undertaking on an early timescale. In particular there is no national repository for intermediate level wastes and waste storage facilities on-site are limited. Thus Stages 2 and 3 are likely to be delayed until the next century when disposal facilities will be available.

Following on from the general objectives for Stage 1, more specific objectives were defined as the basis for the development of a specific work program and schedule. The Stage 1 process was split into sub-Stages with objectives as follows:

Stage 1.1 (Preliminary Decommissioning)

To render the reactor inoperative under any circumstances by defuelling the core;

To reduce the hazard posed by the tritiated heavy water moderator by discharging it from the reactor circuits and removing it from the SGHWR complex.

These activities were part of the standard shutdown and refuelling operations of the facility and were completed within a few months of closure of the facility.

Stage 1.2 (Preparation for Care and Surveillance)

To remove all fuel from the Winfrith site.

To provide appropriate long term atmospheric control of the primary and secondary containments.

To remove, fix or contain all contaminated liquids, loose or readily removable contaminated materials and other hazardous substances.

To rationalize, modify or remove plant, buildings and services where justifiable on cost and dose grounds.

To provide appropriate methods for the management of generated and accumulated wastes.

To provide appropriate equipment for monitoring, surveillance and emergency response.

To secure buildings against unauthorized entry.

These activities are programmed to be undertaken over a six year period, and will be followed by a period of care and surveillance with a requirement:

To monitor and maintain the residual reactor plant, systems and buildings, and

to perform any remedial work or other activities that become necessary for either safety or economic reasons.

These activities will be ongoing over at least a twenty year period.

As well as the technical objectives given above the project has the commercial objective of reducing the overall cost of the program to the minimum, consistent with an appropriate sharing of project risk, by offering as much of the work as is practicable to competition.

DECOMMISSIONING PROGRAM

Specific Features

A number of specific features of the facility were relevant to planning and undertaking the Stage 1 decommissioning program. The most significant of these are as follows:

Being a water reactor some corrosion has occurred in the primary circuits and as a result cobalt-60, which arose from activation of circuit materials, is an extensive source of contaminant activity. Most of the more highly radioactive components are confined within the concrete primary containment and thus operations within the primary containment need to be limited as much as possible to keep personnel dose rates as low as reasonably practicable (ALARP). This, together with the relatively short half-life of cobalt-60, made it desirable to delay decommissioning operations within the primary containment.

The facility has a direct steam cycle design. This has given rise to extensive, but mostly low level, contamination within the secondary circuits.

A number of potentially hazardous materials had been used in the facility. In particular asbestos lagging had been used on the primary and secondary circuits, some of which was in poor condition and needed either removal or treatment.

Water seepage was known to be occurring from the spent fuel storage ponds, which put some urgency in removing the spent fuel from the reactor site.

The fuel pond contained an element which had been damaged during handling and could not be dealt with using the existing fuel handling equipment and flasks. This required special equipment to be developed and proven, in order to recover and straighten the element prior to its removal from the pond.

The plant and buildings had been generally well maintained throughout their life, and thus there was no overriding need to take urgent remedial action to refurbish structures.

Program

The work program to achieve the stated objectives is shown in Fig. 2. The program started in October 1990 with the removal of fuel from the reactor core and is scheduled to complete by the end of March 1997. The main critical path for the

project includes the transfer of fuel off-site and the emptying and decommissioning of the fuel ponds. The reactor closed before a detailed decommissioning plan had been developed. In accordance with early studies, reactor and project staff systematically examined a comprehensive list of all facility systems in order to determine the most likely options for dealing with each. A work breakdown structure was drawn up and an initial decommissioning plan evolved. The physical work was divided into 105 tasks in five program areas as briefly described below.
Fig. 2.
Fuel Program
This includes the transfer of all new and irradiated fuel from the Winfrith site to the British Nuclear Fuels plc (BNFL) facilities at Springfields and Windscale - At the time of shutdown the core and fuel ponds contained the equivalent of 255 irradiated fuel elements; The repacking of all oversized and non-standard elements for transport, including the recovery of the damaged element; The decommissioning of the fuel handling and transport equipment.
Pond Program
This includes the characterisation, recovery and disposal of fuel pond sludges and debris, the removal of redundant pond furniture and other equipment, and decontaminating or sealing the pond surfaces as appropriate.
Operational Systems Program
The main objective of this program is to remove from service all systems not required for the resulting care and surveillance state of the facility. Other systems required for long term surveillance (such as some power and instrumentation systems) are to be rationalized and reduced to the minimum. Together, these actions provide for a cost-optimized long term care and surveillance program.
Containment Program
The objective of this program is to leave the primary and secondary containment structures in a state suitable for long term surveillance under a minimum facility maintenance regime. This involves a strategy of sealing, cleaning, and rationalizing the ventilation systems. Prior to determining the actual strategy a detailed survey to establish the inventory of radioactive and other hazardous materials was carried out.
External Building Program
This program is to survey, and demolish as required, all facilities external to the main containment buildings. This includes, for example, the emergency diesel generator, auxiliary boilers and cooling towers.
Commercial and Financial Considerations
The DRAWMOPS Program placed the initial contract for decommissioning with AEA Technology due to their previous experience as operators of the SGHWR. A team with experience of decommissioning, project management and SGHWR maintenance and operations was assembled, which ensured that continuity of plant knowledge was brought into the decommissioning planning and operations. In November 1992 the project management was restructured into its current form, specifically to enable competition to be brought into the decommissioning tasks and thus maximizing the value for money. The value of work let competitively has steadily increased with time, and for the final three years of the project (April 1994 to March 1997) approximately 60% by value of the decommissioning task will be undertaken by contractors awarded work on a competitive basis. All individual tasks above 10k are subject to competition unless it is demonstrated that this is not cost-effective.
Costs falling to the project come from three main sources, which are approximately comparable in magnitude

Those from the actual decommissioning work, including planning, surveys, studies and record keeping to support these tasks.

Those from work to support the facility including system operations and maintenance, health physics and dosimetry, and overall project management.

Those for services provided by the Winfrith site, including waste processing and disposal, power and utilities, security and regulatory services.

The best estimate total cost to undertake Stage 1 is approximately 77M as shown in the Table I.
Actual expenditure to the end of 1994/95 is anticipated to be 59M, which compares favorably with the budgeted estimate of 62M.
PROJECT MANAGEMENT AND CONTROL
The management of the decommissioning activities of UKAEA-GD is undertaken by the

DRAWMOPS Directorate (DD), which has contracted to AEA Technology to act as the "Managing Agency" for the Stage 1 decommissioning of the SGHWR. The day-to-day responsibilities of the managing agency include:

- Project planning, management and control
- Management of safety
- Selection of contract strategy
- Selection and supervision of contractors to undertake the decommissioning tasks.

Overall direction of the project, and responsibility for ensuring that the project meets the objectives and constraints set by the Department of Trade and Industry, the ultimate customer, lies with UKAEA-GD. A simplified diagram of the project management arrangements is shown in Fig. 3.

Fig. 3.

Stage 1 decommissioning of a power reactor is a complex project made up of, several thousand individual activities. An optimum strategy for decommissioning each system, or possibly single items of equipment, needs to be developed in order to identify what work needs to be undertaken, and perhaps more importantly, what work need not be undertaken since a high level objective is to meet the Stage 1 requirements at minimum overall cost. In addition the activities need coordinating to ensure compatibility with overall program, safety and logistical constraints. The key player in the management team is the senior project manager who is responsible not only for all of the project management, but also for ensuring that decommissioning is carried out in a manner that ensures the safety of the plant and personnel at all times. The senior project manager has reporting to him project managers responsible for defined tasks within the project, as well as a facility manager (also responsible for safety), a contract services manager and a project services manager. These latter provide support services across the whole project in relation to the preparation and negotiation of contracts for work to be undertaken, and the provision of project planning, financial and progress monitoring respectively. A task coordinator is appointed to take charge of each program and to supervise the activities of the managers of the individual tasks, each of which is progressed and controlled from inception to completion within the framework of a task flow diagram as shown in Fig. 4. The most significant aspect of this process is to show that the implications of each task have been properly considered and that an appropriate range of options has been reviewed to ensure that the optimum strategy for the task is to be followed. These studies define, for example, the potential interaction with other tasks, the hazard to be removed, radiation dose expected, waste arisings expected, safety documentation requirements and include cost benefit analysis of the various possible options. The senior project manager will authorize the task to start once he is satisfied and following advice received at the relevant planning meetings. On completion of the task a completion report is prepared which, as well as being a record of all work carried out, ensures that all ancillary actions (such as amending the relevant drawings) have been completed. The plan for each task is the responsibility of the task manager; the plans for all tasks are integrated by software link to form the complete project plan from which the senior project manager derives an overall view of the project. This two tier approach to planning has proved a valuable tool in providing effective control, coordination, resource planning and reporting, and has assisted the facility manager in the duties of safety supervision and day to day coordination for which he is responsible.

Fig. 4.

PRINCIPAL ACHIEVEMENTS

After completing the initial tasks, including the removal of fuel from the core and draining of heavy water, the first major objective to be achieved was the despatch of all fuel to BNFL at Sellafield for storage prior to possible reprocessing. This included damaged elements, dismantled elements and material from post-irradiation examination much of which was prepared for despatch in the active handling facilities at Winfrith against exacting BNFL acceptance specifications. A high standard of planning, coordination and liaison with interested parties was therefore needed. The completion of the work in May 1993 removed a significant source of risk from the project.

Despatch of fuel allowed a major objective of the Operational Systems Program to be tackled, namely the ending of shift working, which had been in place since 1966, and permanent manning of the control room. This required the provision of a comprehensive fire detection and alarm system and a computer based plant monitoring

system which would relay detailed information to the Winfrith site control centre outside normal working hours. The implementation of these systems, and the changes in working which they supported, were closely monitored by the Nuclear Installations Inspectorate, liaison with which comprised an important element of the task.
While these two programs were in progress, consideration was given to the approaches to be adopted for the pond system and the reactor containments. The pond system includes a fuel pond and transfer tunnel, two containment suppression ponds, and two quench ponds all forming an integral part of the main concrete structure. Their characterisation could not be completed until all fuel had been removed. By this time, it had been decided to maximize the use of competitive tendering for selecting contractors for decommissioning tasks. A strategy was devised in which three prequalified teams of consultants undertook option studies covering both the method for decommissioning the ponds and the end condition to be achieved during the current Stage. These studies led directly to the submission of a tender from each team. This approach involved a wide range of expertise and proprietary methods, and encouraged the submission of realistic and cost effective proposals with minimal constraint. The integration of contract expertise within the project team has contributed greatly to the effectiveness of this approach as well as the rapid introduction of competition throughout the project.
The reactor has two containments, a five feet thick concrete primary and the secondary which is formed by the double skinned, steel clad reactor building enclosing the primary containment and the turbine, feed train, pond and support systems. A ventilation system has maintained the pressure within the containments slightly below atmospheric throughout the life of the facility to ensure that no contamination can leak out. This will be expensive to operate and maintain for the whole of the care and maintenance Stage. A strategy has been developed to identify the most cost effective combination of decontamination, sealing, enclosure and airborne contamination monitoring necessary to allow the ventilation system to be turned off without challenging the external environment. This strategy also provides a consistent framework for resolving containment issues in other tasks. Because of the regulatory interface, and supporting development work, this work has been managed directly, but the work to rationalize the ventilation system will be exposed to competition. The primary containment will enclose the principal hazards (radiological and asbestos) until Stage 3. The amount of work planned to be done within it is limited but was sufficient to justify the development of a "Surrogate Tour". This is based on a collection of photographs covering all accessible detail of the primary containment and stored on a video disc. The display of these photographs in the form of a tour is managed by windows-based PC software, which also allows annotation with supplementary data and updating as plant changes are made. This tour has greatly reduced the number of visits necessary to the primary when planning work and has proved both dose and cost effective.
A key requirement of the current work is to provide a comprehensive set of information for those who will maintain the facility, undertake the later Stages of decommissioning and dispose of the wastes without the benefit of staff who were involved in the design, construction or operation. As part of this, comprehensive radiological and asbestos inventories have been drawn up, requiring close integration of surveyors, engineers and database specialists. Extensive use was made of asbestos during construction, and after initial assessments it was decided to remove all asbestos from the secondary containment as it complicated many planned activities. Conversely, asbestos has been left in the primary because it is well contained, little other work is planned and removal would involve substantial dose issues.
Implementation of the decommissioning of the ponds and the modification of the ventilation system are the two most significant remaining tasks.

CONCLUSION

The project is approaching the end of the fourth year of its six year program. All fuel and moderator have left the site; a comprehensive study of the options for decommissioning the ponds has been undertaken in readiness for placing a contract; rationalization of the mechanical, fluid, electrical and instrument systems is largely complete; an economic containment strategy has been devised; a systematic approach to the buildings associated with the facility is being implemented; and planning for handover to care and maintenance has begun. Substantial economies have already been made in the manning and maintenance of the facility.

The decommissioning is now being undertaken in a fully commercial environment, with the majority of the decommissioning tasks being awarded through competitive contracts. The project has successfully kept within the original safety, cost and time parameters by the integration from the outset of reactor and project staff within a comprehensive project management culture. The systematic approach which this encourages has permitted good delegation, planning and coordination; has provided a firm base so that changes in the external environment of the project can be assimilated; and has thus permitted the SGHWR Decommissioning Project to successfully meet its objective of providing a cost effective solution for the ongoing management of the SGHWR facility.

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RADIATION SURVEY AND ACTIVITY ASSAY SYSTEMS IN SUPPORT OF DECOMMISSIONING OPERATIONS AT THE SELLAFIELD REPROCESSING COMPLEX

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ABSTRACT

This paper describes the radiation monitoring systems and strategies developed as part of BNFL's 'Decommissioning and Decontamination Development Program' to support the Company's current and future decommissioning operations. A number of novel monitoring systems and techniques are described covering data capture and planning phases, through to dismantling and, finally, waste disposal. Systems described include: planning and survey tools such as the computer code 'DECOM', radiation imaging systems and a modular Passive Neutron Coincidence Counting system known as 'DISPIM'; Piece and Drum Monitors for Plutonium Contaminated Materials (PCM) arising from dismantling operations and finally; a system for monitoring walls and floors of redundant buildings, targeted at segregation of materials at the Very-Low-Level Waste/ Low-Level Waste boundary. The ongoing development of all the systems described has been carried out in close collaboration with decommissioning personnel at all stages of the decommissioning process. It is concluded that such operational experience is an essential part of the process of instrumentation and strategy development.

INTRODUCTION

The Sellafield site in North West England is operated by British Nuclear Fuels plc (BNFL). As with any nuclear site which has existed over 45 years a number of facilities over time have become redundant as a result of mission change or other factors. In many cases, shutdown facilities entered a Surveillance and Maintenance (S&M) phase. In other cases, where space within certain buildings was needed for new construction, some facilities were dismantled and secured pending future decontamination, size reduction or packaging.

With a significant number of already closed facilities awaiting decommissioning and the prospect of a further large number of facilities becoming redundant at the end of Magnox reprocessing, a decommissioning program for the Sellafield site was developed in 1981. The purpose of the program was to develop plans to reduce the backlog of facilities requiring decommissioning and to develop necessary skills and expertise to access the program of work. The mid-80's also saw the creation of a dedicated decommissioning team who would carry out the upcoming decommissioning activities. Since 1985, the Decommissioning Unit (DU) at Sellafield has grown to a manning level of over 160. Initial decommissioning plans underwent progressive development culminating in 1988 in a 100 year decommissioning program for the Sellafield site. This plan covers currently redundant buildings, those still operational and those yet to be commissioned.

Decommissioning at Sellafield is now guided by a rolling 15 year plan. This plan is updated regularly and is formulated in agreement with the regulatory body. In order to ensure that the aims of the decommissioning program could be met, a

decontamination and decommissioning development program was established in 1988 with the aim of generating all the necessary expertise and technology required in areas such as plant data capture, remote handling and radioactivity assay as well as improving the accuracy of decommissioning cost estimation and developing decommissioning strategies.

This paper describes some of the strategies and techniques in the areas of radioactivity assay that have been developed as part of the decontamination and decommissioning development program.

CHALLENGES FOR DECOMMISSIONING ASSAY

Clearly, the requirements for radioactivity assay depends very much on the particular application, for example whether the plant has handled mainly alpha active material or beta/gamma activity. However, in all cases there are a number of general driving forces underpinning the need for assay, these include:

- Safety
- Dose minimization
- Disposal cost minimization (by decategorisation and volume minimization)
- Regulatory requirements

In the case of plants dealing with alpha active contamination, in particular plutonium bearing materials, the specific requirements for assay information is dominated by the need to maintain criticality safety at all stages throughout the decommissioning operations. In addition to the general practical difficulties of making measurements in such plants, the plutonium contaminated material is often of unknown and variable isotopic and chemical composition which provides a major challenge in the design of accurate assay technology.

In the case of beta-gamma contaminated plant the requirements are usually to support dose minimization and to provide inventory information to support decategorisation and disposal regulations.

The following sections provides a description of some of the methodologies and instrumentation developed to support decommissioning operations within BNFL and, in particular, on the Sellafield site.

METHODOLOGIES FOR DECOMMISSIONING ASSAY

Alpha Plants

Strategies for decommissioning alpha plants at Sellafield have been developed around a number of lead projects (1), these include:

TABLE I

In general, the monitoring strategy adopted for alpha plants at Sellafield has taken the following form:

In order to provide fissile inventory data for planning purposes and developing criticality safety cases, a modular Passive Neutron Coincidence Counting (PNCC) system known as DISPIM (Decommissioning In-situ Plutonium Inventory Monitor) has been developed. Depending on the strategy adopted for the plant being decommissioned, two courses of action in relation to subsequent monitoring may be followed:

If levels of fissile material levels are known, or expected, to be sufficiently high that criticality safety during the dismantling phase is a major issue, because of possible redistribution of material, then 'Piece Monitoring' is employed to monitor individual pieces, after size reduction, and to provide a running total of the fissile content as these pieces are deposited into a waste drum or other storage container as appropriate. The piece monitoring system has been developed utilizing a combination of Passive Neutron Coincidence Counting and High Resolution Gamma Spectrometry and is able to deal with items of unknown Plutonium Isotopics and unknown Chemical Compositions (e.g., oxide and fluoride). In order to provide verification of the total drum fissile content prior to storage a drum monitoring system has been developed utilizing similar techniques to the piece monitors.

In some cases throughput benefits can be achieved by monitoring prior to size reduction, for example when large items are expected to contain only small amounts of fissile material. As such, piece monitoring may not be an appropriate course of action and consequently, larger monitoring systems are required. Such an approach is currently being adopted in support of finishing line decommissioning.

During the course of the dismantling phase of decommissioning operations relating to alpha plants, the DISPIM system is often re-employed to re-establish the database of fissile inventory information to provide ongoing criticality safety assurance.

Beta Gamma Plants

In a similar way to alpha plants, decommissioning strategies for beta gamma plants have been developed around a number of lead projects (2), these currently include:
TABLE II
In these cases, the monitoring strategy has adopted a slightly different form to that described for alpha plant. The first phase of monitoring is generally targeted at providing initial radiation source distributions to enable selective decontamination of 'Hot Spots' to efficiently minimize the dose uptake of decommissioning personnel. Additional monitoring requirements are usually aimed at waste segregation and decategorisation (e.g., Intermediate-Level Waste (ILW) to Low-Level Waste (LLW)) and in support of providing the necessary inventory information for storage and final disposal.
Two approaches have been adopted to assist in the determination of radiation source distributions. The first is based around inverse shielding calculations, using the DECOM computer code, to derive approximate source distributions based around Health Physics dose survey data. The second approach is based on direct mapping of radiation sources in 2D and 3D using a variety of radiation imaging tools, some of which are described in the following section. In the case of storage ponds decommissioning a variety of other instruments are currently under development to assist in the mapping of submerged radiation sources.
Work in support of decategorisation and inventory measurements has so far been centered around the production of bespoke systems to meet the specific requirements of individual projects but is generally aimed at making measurements as close to the point of origin as possible, in order to minimize mixing of different waste streams. Depending on the application, such systems range in complexity from simple Low Resolution Gamma Spectrometry devices through to major multi-technique assay stations encompassing combinations of High Resolution Gamma Spectrometry, passive and active neutron techniques and radiographic measurements.
SYSTEMS FOR SUPPORTING DECOMMISSIONING OPERATIONS AT SELLAFIELD
The following sections provide information relating to a number of systems and techniques that have been developed so far to support current and future decommissioning operations within BNFL.
The DECOM Code
As stated above, decommissioning work often involves estimations of the activities of the sources in a plant which may not be known, perhaps because they have decayed over time or because accurate information characterizing them is not available. DECOM uses information on the size, position and composition of the shields and radiation sources in a plant, together with measured dose rates taken around the plant, to calculate the activities of the sources. These calculated source activities can in turn be used to evolve a decommissioning strategy, or to calculate dose rates at other positions.
DECOM utilizes a new core calculation method which is, on average, a factor of 4 times more efficient than the point kernel method commonly used by other codes. A source subdivision algorithm is implemented automatically by the code to further increase the efficiency of the code. The subdivision implementation requires no user input.
DECOM can also be used as a conventional gamma ray shielding code and has a novel graphical interface allowing the user to view/check their model; interactively calculate the dose rate at a point; calculate and display colour dose rate contour plots and interactively change their models to answer "what-if" questions, e.g. what is the change in the dose rate at this point if I remove this shield?
The graphical interface can also be used to estimate the improvement in the precision of the calculated source activities due to additional measured dose rates before actually taking those measurements. This can be used to calculate the most beneficial position to take additional dose rate measurements whilst minimizing the exposure to the person taking the reading.
Decommissioning work so far undertaken using DECOM has included a test case involving the B41 dry storage silo at Sellafield. Each of the 5 silos in B41 contains both radioactive and non-radioactive waste of many different sizes and materials, e.g. gloveboxes, reactor hold-down weights, Magnox Swarf. No information was available as to the positions of each of the waste's many components, however, the total amount of each waste component was known.
Calculating source activities in this model presented two problems since neither the source positions nor the self shielding effects of the waste were known. The problem

was approached in the following manner.
Since most of the waste components are small it is possible to "smear out" the discrete geometry of the waste and to model it as one homogeneous material whose density was equal to the average density of the waste. The average density of the waste is calculated using the relative amount and density of each of the waste components. This is a valid and widely used assumption for gamma photons since the mass attenuation of a material is mainly dictated by its density and not the material itself, i.e. concrete (density 2.32g/cm3) and aluminum (density 2.7g/cm3) have very similar mass attenuations.
Each silo was modelled as a number of discrete blocks of known position. Modelling the silos in this manner also reflects the fact that waste of different activities were placed in the silos at different times and this will tend to form layers of waste of varying activity. These discrete blocks were then treated in the usual manner to find the source strength of each block. In this manner the source strength is calculated as a function of position.
Although there are no direct means of validating the calculated source strengths, when they were used to calculate the dose rates at the measured points it was found that 207 of the 238 calculated dose rates matched the measured dose rates to within 80% and were, on average, approximately 30% different. The remaining 31 calculated dose rates were up to 2 orders of magnitude different from the measured values. Inspection of these dose positions found them to all lie on the same vertical plane. This suggested that some significant shielding feature at this particular height had not been modelled.
The calculated dose rates were considered to give a very good fit to the measured dose rates considering the assumption made in "smearing out" the waste material geometry and in the build-up factor approximation inherent in the calculation method.
Radiation Imaging Systems
From the outset the aim of the development programs in support of radiation imaging has been to generate colour radiation maps which can be super-imposed on to TV camera views of the same scene. In this way, both the energy and intensity of incoming radiation can be presented, together with features such as hot-spot identification and isotopic analysis. Such images could also be incorporated into computer models of plant layouts, both as a tool to assist in the planning of decommissioning activities and to allow operator access to plant to be optimized in order to minimize doses. Our requirements are two-fold:
Large, fixed systems for use in equipment decontamination or in waste sorting applications. These systems will need to be very sensitive, possibly working in real time, and will have some features in common with existing medical and space systems.
Lightweight, yet rugged, devices will be required for remote deployment which will include decommissioning and incident response applications. These systems will need to use innovative design to achieve a suitable balance between portability and speed of response, and will be very different from existing systems used in other industries.
In order to meet these requirements a number of distinct radiation imaging systems have been, or are being, developed and are summarized in the following table.
TABLE III
Of the systems described in Table III the 'Cyclops' device is the furthest developed and is currently being used on plant in a variety of applications. This particular system utilizes a 18mm x 18mm photodiode coupled CsI detector with tungsten collimation giving a choice of 'fields of view' ranging from 5 to 10. This is directly coupled to a camera system and the whole assembly mounted on a computer controlled 'Pan and Tilt' mechanism. The output of the device is in 'real-time' with count rate information directly overlaid on the video recorded image. Energy discrimination may be employed if maps of individual isotopes are required. Figure 1 shows a schematic of the system.
Fig. 1.
The 'Cyclops' system has so far been used in support of many decommissioning projects, including decommissioning redundant gloveboxes in the Sellafield R & D building and preliminary survey measurements on storage ponds, and is receiving considerable general attention as a useful tool in decommissioning operations.
The DISPIM System
The Decommissioning In-Situ Plutonium Inventory Monitor (DISPIM) is a modular

Passive Neutron Coincidence Counting (PNCC) system that has been developed to assist in planning and criticality safety assessments of alpha plant decommissioning projects. The system enables measurements of the Pu-240 equivalent content of, for example, gloveboxes, vessels and pipework, to be carried out prior to and during dismantling operations. This information combined with a knowledge of the associated isotopic breakdown, obtained from plant knowledge, in-situ gamma-spectrometry or sampling, enables the fissile content of the plant item to be determined.
Considerable operational experience has been gained with the system through its successful use in a number of decommissioning projects on the Sellafield site.
Figure 2. shows the system in use for an 'on plant' glovebox measurement. The performance of such a system is dependent on the conditions relating to a particular measurement, i.e. detector geometry, matrix characteristics, background radiation levels and plutonium isotopics. However, for a typical, civil grade, plutonium contaminated glovebox monitoring exercise, with a count time of the order of a few hours, a detection limit of a few grams Pu is easily achieved.
Fig. 2.
The DISPIM system is based on up to 30 polyethylene moderated 3He detector modules combined with charge amplifiers and shift register coincidence electronics. The use of neutron coincidence counting allows the time correlated neutrons from spontaneous fission events in 240Pu and other even mass number isotopes of plutonium to be recorded separately from the randomly emitted neutrons originating from (a,n) reactions on light elements. The emission of the latter is dependent on factors other than the plutonium content, in particular the chemical composition of the plutonium compounds and the waste matrix. In-situ calibration of the system is performed via the 'add a source' technique using a Cf-252 of known activity to determine the neutron detection efficiency of the particular arrangement for fissile material in a variety of locations within the item being monitored. For criticality safety measurements, worse case data is always used by the software such that the plutonium mass calculated is based on the lowest efficiency measured during the calibration exercise. Similarly statistical fluctuations are included in the error calculations so as to provide an upper limit to the actual plutonium mass.
Figure 3. shows a schematic of the hardware making up the DISPIM system. The outputs of several detector modules are combined and connected to a single amplifier. The outputs from all the amplifiers are summed to form a single input to the neutron coincidence counting unit. The operator interface is via a terminal and keyboard with data output to both screen and printer.
Fig. 3.
DISPIM has fully QAed software written to BS5750. The package is entirely menu driven and extremely user friendly. Indeed, the system has been successfully used on plant on a routine basis by operators with minimal training.
Piece and Drum Monitoring Systems (Alpha Plants)
Both the Piece and Drum Assay Monitoring Systems utilize a combination of passive neutron and gamma measurements to determine the plutonium content of PCM waste packages or 200L drums respectively.
240Pueq content is determined by Passive Neutron Coincidence Counting using a measurement system incorporating helium-3 (3He) gas filled thermal neutron proportional counters and neutron coincidence electronics. The percentage 240Pueq relative to total plutonium is determined by High Resolution Gamma Spectrometry (HRGS) using an intrinsic germanium detector and system electronics with multichannel analysis capability.
Both systems incorporate a number of features to alert operators of the presence of materials that could produce erroneous results. For example, the presence of uranium is detected by measurement of the 186 keV gamma radiation from 235U or one of the 238U gamma-rays at 766 and 1001 keV and the operator alerted if statistically significant amounts are present. Similarly, the presence of very high fluoride contents, which can cause problems with PNCC due to extremely high (a,n) emissions, is flagged by measurement of gamma-rays simultaneously emitted from such reactions, in particular at 583 keV.
In addition, considerable attention has been paid to the issue of measurement controls, so as to ensure that all measurements are reliable and not subject to system fault or external influence. For example, neutron measurements are corrected for the effects of other decommissioning and plant operations by measurement of the background radiation at regular intervals whilst the system is in use. Similarly,

standardization checks of the gamma system is carried out on a regular basis by the use of an actuator driven radionuclide source.
Figure 4 shows a photograph of one of the piece monitoring systems supplied so far to the Sellafield site, this particular one being supplied to support decommissioning the miscellaneous facilities identified in Table I
Fig. 4.
Very-Low-Level Waste (VLLW) Assay System
This system has been developed to assist in the decategorisation process for walls, floors and other large areas of similar geometry which are at the Very-Low-Level-Waste (VLLW) / Low-Level-Waste (LLW) boundary, taken as 0.4Bq/g above natural activity levels in the UK (4,5). The radiometric monitoring sub-system employs High Resolution Gamma Spectroscopy (HRGS) techniques and incorporates a high efficiency Germanium detector, associated electronics and system software which controls both the deployment system, data acquisition, data analysis and display functions. The detector is heavily shielded such that measurements of sufficient accuracy may be made at any one location over a relatively short period. At the present time, the system has been optimized for use on uranium contaminated buildings, although the system software may readily be adjusted for other types of contamination.
The computer controlled deployment system can position the detector anywhere within an x-y plane on a wall or floor. A system of special tracks are fitted to the wall or floor and electrically propelled carriages fitted to the tracks are used to position the detector assembly anywhere within the boundary defined by the tracks. The deployment system is of modular design thereby allowing coverage of areas ranging from about 6m2 to 40m2 or more. When the monitoring of one area is complete, the whole system may be dismantled and then reassembled at the next wall/floor area which is requires to be monitoring. In this way, the majority of the internal wall and floor area of a building may be monitored with each measurement position (i.e. approx. 0.1m2 area) being sentenced as either VLLW or LLW. The degree of contamination is expressed in Bq/cm2 and the mean penetration depth of the contamination in the building material is estimated by exploiting differential absorption of gamma ray peaks in the uranium spectrum. Appropriate decontamination techniques may then be applied to the areas identified as being LLW in order to re-categorize them to VLLW. Each measurement position is permanently identified using an automated paint spray system so that areas of wall which are classed as LLW may be readily identified for future decontamination treatment.
The system is currently about to undergo active trials at the BNFL Springfields site in support of decommissioning the Residue Processing Plant associated with fuel fabrication activities.
DISCUSSION
The monitoring systems and techniques described in this paper represent only a small subset of the output from the ongoing development program to support decontamination and decommissioning operations within BNFL. In all cases the development process has been guided from the outset by discussions with decommissioning project managers and future operators in order to properly define the measurement requirements at all levels. Nevertheless, the development process is a cyclic one making full use of experience gained during initial plant operations to enhance the design and capability of the particular system. As such, the development process is ongoing in support of the various 'lead' decommissioning projects identified with the aim of having a complete suite of tools to assist in all aspects of future decommissioning projects.
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DISMANTLING OF THE JAPAN POWER DEMONSTRATION REACTOR BIOLOGICAL SHIELD BY CONTROLLED

BLASTING

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ABSTRACT

Controlled blasting technique is, in general, able to demolish efficiently only specific portion in the massive concrete structures. In the Japan Power Demonstration Reactor (JPDR) decommissioning program conducted in the Japan Atomic Energy Research Institute (JAERI), the controlled blasting technique had been developed and tested to be applied to dismantling the most part of the biological shield which volume is 1,400 tons. The dismantling activities using the controlled blasting started January 1992 and completed January 1994. Through the dismantling activities, various data on industrial and radiological safety were collected to verify the work procedures using the controlled blasting technique.
Based on the data obtained, the controlled blasting technique was confirmed to be useful for dismantling the massive concrete structures in reactor facilities.

INTRODUCTION

The JPDR (BWR, 90Mwt) started to generate electricity for the first time in Japan in 1963. After training of operators and some irradiation tests, the JPDR ended its service life in 1976.
In consideration of future decommissioning of commercial nuclear power reactors, the JPDR decommissioning program was initiated in 1981 by JAERI under a contract with the Science and Technology Agency (STA) of Japan to develop techniques necessary for decommissioning of nuclear power reactors and to demonstrate these techniques in the actual dismantling work.
After the development of various decommissioning techniques such as dismantling machines, estimation of radioactive inventory, decontamination, waste management and radiation control, the actual dismantling of the JPDR started in 1986 and will completed in 1996.
By the end of October 1991, the highly radioactive components such as the reactor pressure vessel (RPV), the reactor internals, the pipes connected to the RPV and the highly activated part of the biological shield were removed by newly developed dismantling techniques with remote operation.
As a result of removing protrusion of the biological shield, the maximum dose rate in the cavity of the biological shield decreased to 3 x 10-2 mSv/h or less, where workers could approach to work by hands. The controlled blasting technique having an enormous dismantling capability was therefore applied to dismantling the remaining portion of the biological shield.
In the first step of dismantling activities using the controlled blasting, the inner layer (about 40 cm in depth) of the biological shield, which had relatively low radioactivity, was dismantled during January 1992 through October 1992. All wastes generated by this work were loaded into steel containers for storage in the waste storage facility in JAERI. In the next step of dismantling activities, the outer portion of the biological shield, which had extremely low radioactivity, dismantled during April 1993 through January 1994. All wastes were loaded into the flexible polyethylene fiber containers, and these will be disposed in the test shallow land burial place in JAERI for the study on migration behavior of radioactive nuclide.

JPDR BIOLOGICAL SHIELD

Figure 1 shows the cross sectional view of the JPDR biological shield showing area classification relating to the applied techniques and the maximum radioactivity of the concrete. The JPDR biological shield was made by reinforced concrete. Steel reinforcing bars, 29 mm in diameter, were located at interval of 150 to 200 mm. The wall was 1.5 m to 3.5 m in thickness and lined with 13 mm thick steel plate. It had a cylindrical shape with a central cavity, where the RPV was located. The sizes of

the cavity was 2.7 to 3.5 m in diameter. At the mid-core level portion, it had the inner protuberant part (a height of 3.6 m, a wall thickness of 0.4 m) to prevent the neutron irradiation. The concrete had a compressive strength of about 30 MPa(300 Kg/cm2). There were various openings and the ventilation ducts on the wall surface. A number of cooling pipes (34 mm in diameter) and neutron detector guide tubes (265 mm in diameter) were embedded in the concrete structures.
Fig. 1.
CONTROLLED BLASTING TECHNIQUE
Explosives for Controlled Blasting
Generally, the controlled blasting technique is used to make cracks in concrete structures to be demolished with minimum shocks and vibrations. In addition, the damage to surrounding structures is prevented by the use of relatively mild explosives. Recently, the controlled blasting technique has been applied to construction of tunnels and destruction of buildings.
Urbanite, which was an mildly improved form of Dynamite was used in this dismantling work. As compared with Dynamite, Urbanite has low detonation velocity and high gas pressure. When Urbanite was blasted, shock waves generated by blasting made cracks in the concrete, and then the gases produced expanded the kerf of the concrete. For this reason, Urbanite was adequate to dismantling only the specific portion of the biological shield. The amount of explosives for each blast was determined using the Hauser's empirical formula:
L = C x W x D x H
 L: Weight of explosives at each blasting hole (Kg)
 C: Blasting factor
 W: Shortest distance between blasting hole and free-surface (m)
 D: Distance between adjacent blasting holes (m)
 H: Depth of blasting hole (m)
The blasting factor (C) was determined by the mock up tests, which were conducted before the actual dismantling work.
Charging Directions
The explosives were charged vertically or horizontally. The charging direction was selected on the depth of the biological shield being demolished and the working space in the cavity of the biological shield. The vertical charge was mainly employed to demolish the inner layer of the biological shield. The blasted blocks by the vertical charge were about 2 m in width, 0.4 m in thickness, and 1.8 m in height as shown in Fig. 2 (1). In the vertical charge, the blasting force was concentrated toward the center of cavity. Therefore, the damage to the outside wall was limited to the planned portion. If the outside wall of the blasting location was thin, the blasting holes were drilled in two layers instead of one layer to prevent the development of cracking on the outside wall. The horizontal charge was mainly employed to demolish the outer portion of the biological shield. The areas being blasted by the horizontal charge were all the circumferential direction, about 2 m in thickness, and 35 cm in height as sown in Fig. 2 (2). In the horizontal charge, the blasting force was concentrated above the planned blasting areas. The circumferential areas were widely blasted at a times.
Fig. 2.
Procedure of Dismantling Work
As a preparatory work, the inner layer of the biological was slitted together with the steel liner and the reinforcing bars to a depth of about 0.25 m at the boundaries of blocks being blasted by an oxyacetylene torch and a diamond cutter. Drilling, charging of explosives and mechanical crushing were performed by workers only using the jack-hammer and the mechanical breaker on the movable platform in dismantling the inner layer. However, in dismantling the outer portion of the biological shield, a drilling machine and a back-hoe shovel were introduced to forward the work of drilling and crushing as shown in Fig. 3. The procedure for vertical charge blasting and horizontal charge blasting is summarized as follows.
  Holes were drilled on the blasting location by the use of jack-hammer or drilling machine on the platform, which was lifted by a polar crane.
  Explosives were charged in each hole. The holes were then sealed with tamped sand.
  The areas being blasted was covered with mats and sheets to prevent the scattering of concrete debris and to minimize the spread of airborne dusts.
  The explosives were set off with ignition time delay intervals of 0.025 or 0.25 seconds.

The cracked concrete was fractured using a mechanical breaker and a back-hoe shovel. The undetached reinforcing bars were cut with an oxyacetylene torch.
In dismantling the inner layer of the biological shield, the dismantled concrete and metals wastes were put into two-hundred-liter drums and one-cubic-meter containers for storage in the waste storage facility. While, in dismantling the outer portion of the biological shied, a small-wheel loader was used to load the concrete waste into flexible polyethylene fiber containers for disposal in the test shallow land burial place.
In both of dismantling the inner and outer portions of the biological shield, the above procedure was repeated starting at the top and progressing to the bottom of the biological shield.
Fig. 3.
DATA OBTAINED FROM THE DISMANTLING WORK
Through the dismantling activities, various data such as manpower expenditure, collective dose, waste generation, noise and vibration, were collected to verify the usefulness and safety of the controlled blasting technique. Table I shows the data on the dismantling work.
TABLE I
Manpower and Collective Dose
The total duration and manpower expenditure required for dismantling the inner and outer portions of the biological shield were 362 days and 8,600 man-days, respectively. The breakdown of the total duration and manpower expenditure were 153 days and 3,700 man-days for 207 tons of the inner biological shield (18 man-days/tons), 209 days and 4,900 man-days for 1260 tons of the outer biological shield (4 man-days/tons). Since the heavy machine such as the drilling machine and the back-hoe shovel were introduced to forward the dismantling activities in the outer portion of the biological shield, the manpower expenditure per unit weight of dismantled concrete was less than that of dismantling the inner layer of the biological shield.
The total collective dose was limited to be 2.3 man-mSv. Most of the radiation exposure occurred while workers engaged in the work of drilling and mechanical crushing at the mid-core level portion.
Waste Generation
The total weight of concrete and metal wastes generated by the controlled blasting was 1,500 tons. In dismantling the inner layer of the biological shield, 180 tons of concrete wastes were put into 708 two-hundred-liter drums, and 30 tons of metals which consisted reinforcing bars and steel plates were put into 16 one-cubic-meter containers. In dismantling the outer portion of the biological shield, 1190 tons of concrete wastes were loaded into 1012 flexible polyethylene fiber containers, and 70 tons of metals wastes were put into 107 two-hundred-litter drums and 29 one-cubic-meter containers, respectively.
The steel containers (two-hundred-liter drums and one-cubic-meter containers) were transported to the waste storage facility in JAERI. While, the flexible polyethylene fiber containers will be disposed in the test shallow land burial place in JAERI.
Blasting Work
The total volume of blasted concrete structures was about 580 m3 and the total weight of explosives used was 170 kg. The explosives of 250 to 500 g were required to blast the volume of 1 m3. The amount of explosives tend to increase with the weight of metals in the biological shield. By analyzing the data obtained in dismantling the inner layer of the biological shield as shown in Fig. 4, it was found that approximately 260 minutes of the total working time were required to demolish the typical blasting location; 25% for drilling using a jack-hammer, 24% for making and charging of explosives, 13% for covering the protective mats and sheets, 38% for mechanical crushing using a mechanical breaker.
Fig. 4.
Noise and Vibration
In dismantling the inner and outer portion of biological shield, the blasting noise and vibration generated by the blasting were measured in the reactor enclosure and outside of the reactor enclosure.
As an example of the noise data, the blasting sound pressure level was measured at two points, third floor in the reactor enclosure and at a distance of 10 m from the blasting location outside of the reactor enclosure. The maximum sound pressure level was about 130 dB(A) in the band of 25-250 (Hz) in the reactor enclosure. It

decreased to about 100 dB(A) outside of the reactor enclosure. Generally, the critical value of the sound pressure level to the human organization is considered to be about 130 dB (A) regardless of its duration. In this dismantling work, 130 dB (A) of the sound pressure level was measured some times in the reactor enclosure. However, since the workers were evacuated from the reactor enclosure when the blasting was initiated, it had no influence on the workers.
To evaluate the effect of vibration on the surrounding components, the acceleration of the vibration was also measured at three points: 12, 17, and 27 m from the blasting portion. The acceleration varied from 20 to 400 gal (cm/s2). Generally, the acceleration range between 100 and 400 gal corresponds to a large earthquake. However, since the vibration generated by blasting had the characteristics of high frequency and short duration, there were no damage on the remaining concrete structures and no influence on the integrity of components in the reactor enclosure.

LESSONS LEARNED
In dismantling the biological shield, it was proved that the controlled blasting technique was useful for demolishing exactly the planned portion. The lessons learned are listed as follows :
  The slits cut in the boundary and the back-hoe shovel used were very effective in improving the demolition performance of the massive steel-lined reinforced concrete structures.
  The amount of airborne dusts were effectively limited by the protective mats and sheets, the sprinkling water on the concrete to be demolished.
  For dismantling the highly activated concrete structures using the controlled blasting technique, it will be necessary to develop the drilling and charging methods with remote operation to reduce the radiation exposure.
CONCLUDING REMARKS
The controlled blasting technique was successfully applied to demolishing of the low level radioactive portion. The safety of the controlled blasting was proved by the various data on industrial and radiological safety.
After dismantling the biological shield, the major activities in the JPDR decommissioning program are decontamination of the floors and walls in the buildings together with the final survey of the radioactivity. In the beginning of 1995, the radiation controlled area will be released gradually to demolish the buildings in the JPDR site.
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DECOMMISSIONING OF NUCLEAR FACILITIES THAT WERE ABRUPTLY SHUTDOWN - A LOOK AT THE UNITED STATES DEPARTMENT OF ENERGY URANIUM REFINERY
AT FERNALD, OHIO
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ABSTRACT
The Fernald Environmental Management Project site, formerly the Feed Materials Production Center, produced purified uranium for Department of Energy enrichment plants and production reactors for more than 35 years until production was suspended in 1989 for reasons of decreasing demand for purified uranium and increasing environmental accountability. Operations were suspended without a formal and complete shutdown being performed. This resulted in large amounts of liquid and solid materials being left in process lines, furnaces, and intermediate storage areas. The site has undergone a change in management since the days of production and is presently engaged in environmental restoration and waste management as its sole mission.
There are many challenges faced by the engineering personnel in evaluating the existing systems, providing for a safe work environment, and designing environmental remediation solutions.

For many systems the first step in decommissioning is to empty the system of all internal fluids and solids. The uranyl nitrate hexahydrate (UNH) neutralization project is tasked with neutralizing, filtering, and drumming for disposal approximately 760,000 liters (200,000 gallons) of low pH UNH, essentially an uranium-bearing nitric acid solution in water. The UNH is presently stored in 18 tanks scattered across 5 different areas of the site. The UNH was an interim product of the production process and was routinely denitrated to form uranium oxide (UO3) solid. The wastewater collected from the refinery operation was treated with magnesium oxide to precipitate the uranium. The slurry material was then returned to the refinery for processing. This involved miles of stainless steel piping, numerous pumps, and a multitude of storage tanks.

In the several years since operations ceased, there has been a rapid deterioration of system integrity resulting in numerous leaks and a loss of confidence in the ability of the system to adequately contain the UNH. The decision to use the absolute minimum amount of existing equipment in the design of the new neutralization system was an effort to restore the confidence in the system and ensure the safe processing of UNH.

INTRODUCTION

The Fernald Environmental Management Project (FEMP) site, formerly the Feed Materials Production Center, produced purified uranium for Department of Energy (DOE) enrichment plants and production reactors for more than 35 years until production was suspended in 1989 for reasons of decreasing demand for purified uranium and increasing environmental accountability. Operations were suspended without a formal and complete shutdown being performed. This resulted in large amounts of liquid and solid materials being left in process lines, furnaces, and intermediate storage areas. The site has undergone a change in management since the days of production and is presently engaged in environmental restoration and waste management as its sole mission.

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In the several years since operations ceased, there has been a rapid deterioration of system integrity resulting in numerous leaks and a loss of confidence in the ability of the system to adequately contain the UNH. The decision to use the absolute minimum amount of existing equipment in the design of the new neutralization system was an effort to restore the confidence in the system and ensure the safe processing of UNH.

Configuring the existing systems to be used presented a large challenge. The integrity of the existing tanks and support equipment had to be determined, and a logical interface with the new system had to be established. The UNH storage tanks have been sitting without agitation, in some cases for over 8 years, and have settled out into layers with viscosities ranging from that of water to that of castor oil, and in some cases had crystallized into a solid mass. This paper will detail the system configuration, UNH processing, and site construction and operation that must be achieved in order to decommission this nuclear facility.

SYSTEM CONFIGURATION

Existing process equipment, storage tanks and process lines were used to the extent appropriate following an engineering evaluation of the equipment. The plant's systems had been modified significantly from their original construction to meet the requirements of the process systems needs. These modifications were rarely

incorporated into system drawings and minor modifications were frequently made without a design review. This resulted in equipment that differed greatly from the design drawings, or "as-builts," if they existed at all. The process systems had been cross-connected to allow operators to move process material virtually anywhere with any piece of equipment through any piping system be it a process line, sump line, or water line.

The UNH project team set out to develop a current set of "as-builts" to allow the maintenance personnel to work from a certified drawing. This allows performance of the necessary tank isolation to ensure no material could enter or leave the tank except through designated lines and valves whose integrity had been verified to the maximum extent possible. This task required frequent walkdowns of the system components and interfacing with the construction personnel to ensure that all interference equipment and piping was identified sufficiently so that it could be safely isolated and removed if necessary.

The maintenance personnel were required to remove connecting piping and insert blanks in the lines or install blind flanges on all penetrations to the tanks above the liquid level. Lockout and Tagout procedures were followed for all penetrations below the liquid level. These storage tanks contained UNH where a severe chemical and moderate radiological hazard required extensive Environmental, Safety and Health and Radiation Protection personnel review and supervision. The piping going into and out of the storage tanks had never been flushed, and due to cross connection of the various systems, the exact contents of the pipes could not be determined by visual inspection alone. So in order to ensure safety and protection to the personnel and the environment, conservative requirements were instituted to prevent contamination or injury through inhalation or skin contact.

The Environmental, Safety and Health and Radiation Protection requirements included the use of glove bags, secondary containment, and forced air respirators for the initial opening of tanks and process lines. After inspection and sample characterization if required, the handling procedures were adjusted to fit the actual hazard involved. The buildings that housed the process equipment and several of the storage tanks also presented hazards of their own. Transite panels that were manufactured with asbestos were disintegrating and falling off of the ceilings and walls, presenting a falling object and airborne asbestos hazard. The buildings were open to birds and the threat of histoplasmosis* required frequent Environmental, Safety and Health evaluation as well.

The maintenance staff required extensive training to bring their level of knowledge to the industry standards of today, such as the use of torque wrenches and material certification. These were two specific areas that were discovered to be inadequate during the walkdowns to develop the "as-builts." Bolts were sheared off and flanges were left installed with less than the required number of fasteners. Many leaks were identified in places where material was used that was not suitable for the low pH nitric acid environment.

The final decommissioning of the UNH piping and process equipment is a follow-on activity to the UNH neutralization project. The system will be in an environmentally benign status after processing as all lines and tanks used in the process are being cleaned and rinsed after their final use. The processing system has been identified to be potentially available for future processing of other site materials should their remediation include similar processing.

UNH PROCESSING

The UNH processing flow diagram is shown in Fig. 1. The UNH is pumped from the remote storage tank to the dilution/neutralization tank that has been previously filled with a specified amount of water to reach the required dilution factor for optimum processing. Magnesium oxide is added and the tank is agitated until a neutral pH is obtained. The slurry is then pumped out of the bottom of the dilution/neutralization tank to the filter feed tank and then to the vacuum filter. Fig. 1.

The vacuum filter pulls the liquid portion through a canvas cloth coated with a diatomaceous* pre-coat. The filtrate is then characterized and either discharged to the Great Miami River, reprocessed, or treated in the biodenitrification facility until discharge limits are met. The filter cake is removed from the vacuum filter with a knife edge that allows the pre-coat to remain on the cloth, and the filter cake is loaded into 208 liter (55 gallon) drums via a discharge chute. The drummed filter cake is characterized and prepared for shipment to the Nevada Test Site for

ultimate disposal.
The UNH process system was designed with a minimum number of flanged unions and valves to maximize the length of welded pipe. This reduces the number of places where a leak can occur and also reduces the amount of secondary containment required. Due to the construction of the process areas at the FEMP, most of the pipe unions were able to be placed over existing secondary containment areas. All piping runs were sloped to aid in the cleaning and flushing of the tanks and process lines using the minimum amount of process water possible. The entire UNH processing system underwent an extensive program of hydrostatic testing and operational testing using clean water to ensure that the system performed as required.
SITE CONSTRUCTION
Construction activities started in May 1994 and were still ongoing as of January 1995 to complete minor alterations. In an effort to utilize stock on hand, all procurement items were first screened through the on-site stores system to determine if the required material was available on site. This effort resulted in frequent delays and the identification of material that was similar but not exact according to the design specifications. When material was procured, the Federal Acquisition Regulations requirements inserted additional delays as frequently the low bidder was a Small Disadvantaged Business who rarely had the material in stock. When the material arrived on site, receipt inspections often resulted in discrepancies where the supplier chose to substitute what they believed was equal but design review showed noncompliance.
Construction activities were delayed as material did not arrive as sequenced by the construction activities schedule leading to individual systems being only 90% complete when the schedule required that system to interface with another system. This resulted in modifications to test plans, fabrication of "dummy" components to be installed to allow the systems to be made up to connecting systems, and the removal of these dummy components and installation of the required component when it arrived from the supplier.
The manufacturer of the progressive cavity pumps delivered the pumps as required by the design specification with one important exception. Four of the six initially delivered pumps had cracks in the casing in the same location indicating a flawed manufacturing process. The pumps were returned to the manufacturer for replacement. Due to this serious impact on the schedule, it was decided to suspend the order for the pumps and to procure double diaphragm pumps that were available on-site from the on-site stores system. These new pumps were installed with minor modifications to the process system.
The extensive Environmental, Safety and Health and Radiation Protection requirements imposed on the maintenance staff resulted in delays of the turnover of storage tanks and process equipment to the construction organization. These delays were somewhat mitigated by the construction manager rescheduling activities out of sequence so that downstream activities were completed ahead of schedule.
OPERATIONS
The operation of the UNH neutralization process was initially attempted in April 1993 using existing equipment. This effort resulted in a minor spill of UNH and provided the impetus to reevaluate the system conditions and ultimately led to the design and modification of the hybrid system intended for initial processing on January 17, 1994.
The process was to undergo an Operation Readiness Review (ORR) conducted by DOE Headquarters personnel to ensure that personnel and equipment performed satisfactorily to prevent and/or mitigate any potential releases of UNH during the process. DOE-HQ decided to conduct this ORR in parallel with a complete FEMP site ORR as one had not been completed since the site shifted from production to environmental restoration. The site was unable to be ready for the ORR when the UNH project required evaluation, so a Readiness Assessment was performed for the UNH project. This required the UNH project team to process clean water from a storage tank to demonstrate the process systems performance as well as the overall knowledge and professionalism of the operators and supervisors.
LESSONS LEARNED
The UNH neutralization project is one of the largest and most technically challenging projects faced by the FEMP site to date, and very likely in its entire remediation process. The difficulties encountered resulted in many lessons learned; several are summarized below:

The project has had five Project Managers in little over a year. A Project Manager should be selected in the beginning who has experience in the scope and scale of the project and the leadership skills necessary to bring the job from inception to completion.

An incentive-based procurement strategy should be established that not only meets the requirements of the Federal Acquisition Regulations, but also meets the requirements of the project as well. For example: quality and timeliness should be significant factors in addition to cost.

A facility should be shut down in a controlled and planned manner to ensure that process material is in discrete known locations to mitigate the "surprises" that may be encountered when piping systems are opened for decommissioning.

Process systems should be designed to the level commensurate to the actual risk as determined by a Safety Analysis, and not to arbitrary conservative requirements that increase cost and lengthen schedule without significantly lowering the risk to human health and the environment.

CONCLUSION

In the case of decommissioning a facility where both chemical and radiological hazards exist, it is even more important to thoroughly plan the suspension and completion of operations, the short term safety and integrity of the system, and the long term final remediation. This task can be made substantially less difficult by maintaining impeccable records of process systems and performing periodic verification of the system "as-builts." All design modifications should be evaluated as to how they impact the eventual decommissioning of the system, not just how they satisfy the immediate needs of the process requirements.

For those facilities that have been thrust into the decommissioning arena and have not been fortunate enough to be prepared for this eventuality, I suggest that the technical staff not allow themselves to be forced into arbitrary remediation schedules that do not allow for the technical review required to safely and completely execute the required task. This may seem like an impossibility, but every minute spent in planning will certainly save hours in the execution.

For those facilities that are relatively new and do not face decommissioning for many years, the unique opportunity exists to plan for the eventual decommissioning on a day-to-day basis. This will make decommissioning a smoother and more cost efficient task that is friendlier to the environment and safer to human health.

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AN INTEGRATED BUILDING DEMOLITION AND WASTE PLANNING MODEL FOR THE FERNALD SITE

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ABSTRACT

The Fernald DOE site will begin full-scale remediation of buildings under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) during the 1995 fiscal year pursuant to a signed Record of Decision. This effort is currently estimated to cost $350 million and span a minimum duration of 8 years, if funding is not a constraint. The identification of the most viable sequence and schedule for the effort involved the development of an integrated planning model and the commissioning of a sitewide planning team. The resulting work product represents the best combination of assumptions and calculations possible at this time and provides information necessary for compliance with the CERCLA Remedial Design documentation requirements for the over 230 component structures governed by the decision.

Sequence and integrated schedule development for the decontamination and dismantlement (D&D) of Fernald structures has involved evaluation of current and future utilization of structures, availability of waste storage and staging space, the needs and impacts of other on-going Fernald projects, Resource Conservation and Recovery Act (RCRA) waste management and remediation projects, the layout of site utilities, site hydrology, and the potential sizing, location, and construction rates for an on-property disposal cell.

BACKGROUND

Under a 1994 signed Record of Decision (1), the Fernald DOE site must initiate remediation of the site buildings before October 22, 1995 and maintain a continuous remediation program within the operable unit thereafter. The site currently plans to follow the completed remediation of Plant 7 (performed under an expedited response action) with the removal of Plant 4 (the former uranium hydrofluorination plant) followed by the removal of the majority of the Plant 1 complex (a multi-purpose sampling, milling, and staging facility). These facilities were chosen due to their availability, their relatively low amount of usefulness, and the need for additional cleared space within the former production area of the site.

During the identification of the Plant 4 and Plant 1 complexes as the initial set for remediation, it became clear that a sitewide integrated project planning effort would be required to support development of optimized sequencing for D&D projects. The approach would be required to fulfill EPA requirements, while integrating the D&D project with the other major CERCLA projects to be undertaken at the site, the waste management functions currently underway, and the RCRA driven actions mandated at the site.

INTEGRATED PLANNING TEAM

In order to ensure a comprehensive understanding of the site situation and develop an integrated approach to the sequencing, a planning team, referred to as the Sequencing Design/ Engineering/Construction (DEC) Team, was commissioned with members representing all major divisions of site management and the DOE Fernald project management staff. The major drivers and constraints were initially evaluated by the Sequencing DEC Team to determine the internal and external influences on the decision-making.

Schedules from other major projects at the site were reviewed to determine all planned uses for site facilities. Where no schedules were available, best assumptions were documented and initially considered constraints to building demolition. Date constraints for each facility were documented as milestones in a scheduling software to be used in developing a conforming schedule.

In addition to D&D constraints, several major site projects (e.g., the soil remediation and on-property disposal cell projects) assume that many of the site structures will be removed prior to their initiation or completion. Assumptions from these projects were handled as drivers for the D&D. Other drivers, major assumptions, and constraints considered by the Sequencing DEC Team for a preferred sequence are provided below:

Site hydrology (above and below grade) drives the remediation to begin in the north and move southward to prevent recontamination of remediated soils and perched water zones.

Soil use within the on-property disposal cell would require a minimum 3 to 1 ratio with respect to D&D debris to maintain adequate compaction capability.

In the absence of adequate quantities of soil, debris such as concrete and steel would be bulk stored in piles until soil could be generated.

On-property disposal cell construction is anticipated to begin receiving debris for burial in August 1997 and progress at an average rate of 95,600 cubic meters per year.

Due to inclement weather, cell construction activities are assumed to cease for three months during the winter.

D&D sequencing must progress to maximize availability of large zones for soil remediation, starting with the north and generally progressing to the south over the course of the action.

Two adjacent structures of significant size cannot simultaneously undergo dismantlement, to avoid project interferences.

No more than 5 major dismantlement projects can be undertaken simultaneously, since site construction management resources would be strained with more projects undertaken.

Safe Shutdown (removal of equipment hold-up residues, utility disconnects, and gross surface decontamination) activities must be completed in each structure prior to initiation of dismantlement.

Due to its importance in the mitigation of risks to human health and the environment, the Safe Shutdown Program is assumed to take budgeting priority over dismantlement.

Defining Remediation Complexes

Using the concept of economies of scale, the expenses for a decontamination and dismantlement project can be reduced significantly by addressing multiple components in a single project instead of remediating components as individual projects. The cost and time involved in the development, review, and submittal of contracts, work plans, health and safety plans, and other supporting project documentation are relatively independent of the number and sizes of structures within a D&D project. Other expenditures, such as subcontractor training, establishing control zones, mobilization and demobilization of construction equipment and crews, and air monitoring are also relatively independent of the number and sizes of structures within a project. Therefore, the above-grade portion of 165 individual structures were combined into twenty-three groups (called complexes) to reduce remediation costs. Similarly, the at- and below-grade portions of the operable unit was divided into three complexes (i.e., north, central, and south).

The structures were assembled into complexes based on many considerations, such as relative location of the structures to minimize impacts between dismantlement activities and the daily operations of the site. If possible, complexes were confined to a distinct area, such as a city block, that could be safely partitioned into a construction zone without adversely affecting other projects.

A second consideration for grouping structures into complexes was the current and/or future use of the facility. For example, components that support the distribution of electricity across the Fernald were combined into the Electrical Station Complex, although these components are not all located together. Two advantages to grouping structures based on their related use are that they have a high probability of becoming available altogether and are likely to be constructed of the same types of materials, making design and dismantlement activities more simple and, therefore, cheaper.

Prioritization of Complexes for Remediation

The prioritization of these complexes for remediation was initially performed by the Sequencing DEC Team without consideration of funding as a constraint. Since actual funding for a longer term project is dependent on many factors, the actual schedule of the D&D project is expected to follow the prioritized sequence, but at the rate that satisfies the funding constraints. Several average annual funding level estimates were utilized to test the viability of the preferred D&D sequence. However, since possible schedules may vary from as little as 8 years, if unconstrained by funding, to up to 30 years for minimal funding levels, the D&D sequence was required to demonstrate viability at each of the assessed funding levels. Since the preferred sequence is driven primarily by external project factors and structure reusability factors, the duration of the schedule for D&D was found to have relatively little impact on the determination of an optimal sequence.

The first step in developing a remediation schedule that is not constrained by funding is to establish the earliest possible starting date that the complex will be available for remediation. There were many structure-specific scheduling constraints that had to be factored into the schedule because many facilities are necessary to either support remediation activities or required site activities (e.g., wastewater treatment, RCRA warehouses, and ongoing maintenance) and cannot be scheduled for removal until these activities are relocated, replaced, or are no longer necessary. Although the objective is to first decontaminate and dismantle the complexes that lie within the footprint of the proposed on-property disposal cell and to clear a path for at- and below-grade remediation to proceed from north to south, some of these complexes may not become available until remedial activities are well underway. If the highest priority complex is not immediately available, remediation of that complex will be deferred until it becomes available. The availability of the second highest priority complex would then be assessed, and so on.

The second step is to factor in the impacts of several logistical constraints. Reasonable limitations on the number of workers in a given area, traffic patterns, and waste handling routing must be established to prevent overcrowding and to minimize potential health and safety hazards during remedial activities while pursuing the overall goal of a timely, efficient, and cost-effective remediation. Therefore, it was assumed that no more than five construction projects would occur at the same time. Also, careful consideration was given to avoid two construction crews working on adjacent complexes at the same time.

The final step in developing a funding-unconstrained remediation schedule is to ensure that the prioritized sequence of remediation will not heavily impact the

network of required site utilities. Fernald site utilities include electricity, plant air (used for air-supplied respirators), instrument air, natural gas, propane gas, fire protection water, sanitary water, process water, steam, sanitary sewers, storm water runoff systems, cooling water, roadways, and telephones. If the utilities are not required for the safe, efficient, and cost-effective removal of a complex, the utility lines will be capped or terminated near the boundaries of each complex (for above-grade activities) or remediation area (for at- and below-grade activities) before dismantlement begins. Utility connections to the occupied areas of Fernald will be maintained by temporary or rerouted connections, as needed. Figure 1 shows the prioritized remediation of complexes that resulted from this approach. The dashed lines represent the division between the three below-grade remediation areas (i.e., north, central, and south).

Fig. 1.

A prospective project schedule using the preferred sequence and an anticipated minimal average annual project funding level was developed to support preparation of draft remedial action compliance milestones with USEPA and the Ohio EPA. The publication of the schedule as part of the Remedial Action Work Plan will represent the first use of many intended uses for the scheduling tool developed by the Sequencing DEC Team. Annual updates are expected to be prepared for submittal to the regulatory agencies and the schedule should also represent a powerful tool for projecting and evaluating annual budget scenarios for the D&D project.

A significant portion of the schedule evaluation was for determination of waste generation rates. As a means to evaluate the impacts of waste generation rates resulting from the schedule, the known and projected waste flows were compared to handling, storage, and disposition assumptions in a material balance model.

MATERIAL BALANCE MODEL

Continued management and disposition of the production products and wastes from the site, as well as D&D wastes is a significant facet of the current site mission, requiring the reutilization of many of the former production facilities to provide covered storage. The impact of the additional waste resulting from D&D projects and the net reduction of available facilities over the course of the project was evaluated to demonstrate material flows versus capacities throughout the project. Although much of the D&D waste is expected to be eligible for disposal in the proposed on-property disposal cell, the cell is not planned to be available for D&D wastes prior to late 1998. Under the most aggressive D&D schedule, 105,000 cubic meters of debris would be generated in advance of that availability, requiring the use of 44,600 square meters of existing storage space. Additionally, several facilities currently utilized for storage would be removed under the aggressive scenario to facilitate the construction of the disposal cell, thereby reducing the overall availability of covered storage facilities.

To determine the impact of the remediation schedule on Fernald's capacity to store materials, the material balance model uses a general mass balance equation. The general equation for determining mass balance for material that enters and leaves a system is as follows:

Eq.

This general mass balance can be modified as follows to apply to material at Fernald:

Eq.

This mass balance equation considers volumetric material flow on a monthly basis. The first term in the equation, Off-Property Receipts, represents the current DOE anticipation that Fernald will not receive off-property materials for on-property storage or disposition. Therefore, in the mass balance equation, the volume of material to be received from off-property sources drops out of the mass balance equation.

The Material Generation term represents the volume of generated material that may require on-property storage prior to on- or off-property disposition. The Material Generation term specifically does not include volume estimates for uncontaminated office trash and recycled materials because these materials are dispositioned off-property in a timely manner and, therefore, do not require temporary storage at Fernald. The Material Generation term represents the total material burden, current and future, that may potentially require storage facilities prior to disposal. Table I defines the types and quantities of materials to be generated and provides the information required to calculate the Material Generation term of the mass balance

equation.

TABLE I

The Off-Property Disposition and On-Property Disposal terms represent the volumes of those materials listed in Table I that are anticipated to be either shipped off-property for disposal or recycling, or buried in the on-property disposal cell. By using the estimated volumes in the mass balance equation, the difference between annual material generation and annual material disposition equals the amount of material that requires temporary storage at any point during the project. Storage is organized into three types: hazardous and mixed waste storage; covered storage of low-level radioactive materials; and uncovered storage of low-level radioactive materials. By comparing the need for these types of temporary storage on an monthly basis with the maximum on-property storage capacities, the resulting material balance determination will indicate whether or not there is a need to provide for additional temporary storage facilities during specific periods of the project.

By performing the material balance model at monthly intervals throughout the duration of the remediation, using the information available or estimated for each of the parameters, a set of waste generation/disposition and available storage space curves were generated and compared. Figure 2 provides an example waste curve that compares the amount of needed uncovered storage space with the projected available uncovered storage capacity for a 21-year project scenario.

Fig. 2.

Rates of waste generation anticipated for each of the D&D projects were provided by evaluation of an aggressive D&D project schedule, since this scenario would arguably result in the most burdensome waste storage problem. Other waste generation activities, such as daily operation and maintenance activities, removal actions, and remedial actions of the four other operable units at Fernald, were also tracked in the model, since the resulting wastes compete for the same available storage. Waste disposition rates were developed from existing planned disposition projects or estimated for waste streams with no detailed schedules for disposition. Existing stored waste quantities were known by waste type and storage footprints (floor space utilized) were also known at the start of this analysis. Waste disposition rates for materials anticipated to be destined for disposal in the proposed on-property disposal cell were reported by the cell design effort, underway as part of the remedial design in several of the CERCLA operable units at Fernald.

MODELING RESULTS

The aggressive D&D schedule used in the material balance model, derived from the work performed in the sequencing activity, that the entire D&D effort at Fernald could be performed in an 8-year period, if funding were not a limitation.

The 8-year schedule represented the highest waste generation rate prior to development of an on-property disposal cell, and also the case under which the most storage facilities would be removed most quickly, thereby resulting in a worst case for storage needs versus capacity at the site. The analysis of this worst case demonstrated that, although the site has severely limited current storage capacity, during the course of the action, sufficient storage capacity for covered, uncovered, and hazardous/mixed wastes exists to avoid construction of new facilities. The analysis also demonstrated that the preferred sequence, in combination with other existing planned and assumed activities, was viable with respect to material disposition issues.

Since the evaluation demonstrated the necessary storage capacity was available for this worst case, and since the sequence was determined to be relatively insensitive to schedule extension, no additional modeling would be required to demonstrate feasibility of any proposed D&D schedules with respect to storage, handling, and disposition capacity.

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ABSTRACT
This paper discusses changes made in a demolition project at the Fernald Environmental Management Project (FEMP), a site on the National Priorities list (NPL), owned by the Department of Energy.
The project, to demolish fourteen uranium ore silos and their structure, was based on a Removal Action Work Plan, submitted and approved by the United States Environmental Protection Agency (USEPA), that integrated Comprehensive Environmental Response Compensation and Liability Act (CERCLA) requirements to remove the source of contamination and threat to public health and the environment. After the demolition contractor defaulted at 30% complete, completion of the project by the USEPA deadline was threatened. The recovery plan included re-evaluation of project documents in addition to the schedule. It was determined that re-interpretation of the removal action criteria, including design and Removal Action Work Plan, would eliminate road-blocks, and optimize resources, resulting in project completion by the original deadline even after lost-time in mobilizing another contractor.
This presentation will discuss the "lessons learned" by the project team and illustrate how simplification of construction methods resulted in enhancements to the environmental controls, improved material handing, and created a safer work environment.
INTRODUCTION
The Fernald Environmental Management Project (FEMP), owned by the Department of Energy (DOE), was used to process uranium for U.S. Defense programs. On-site production ceased in 1989 and the facility was placed on the National Priorities List (NPL) for remediation under the CERCLA process.
The former production area included the Plant 1 Ore Silos, which were constructed in 1953 and used to store cold metal oxide by-products from processing pitchblende and lower grade uranium ores, until they were "abandoned in place" in 1961. The Plant 1 Ore Silos included four forty-four feet high, and four ten feet high, glazed tile silos and six ten feet high reinforced concrete silos. The silos, built on open steel structures, were thirty-eight feet above grade. There were platforms and material conveying equipment above and below the silos. Residues remained in the silo cone bases with depths from one foot to about six feet.
Fig. 1. Overview of Plant 1 Silos looking north.
When residues leaked from the deteriorated silos in 1991, demolition of the Plant 1 Ore Silos became a CERCLA Removal Action to speed the removal of radiological and safety hazards until final remediation of the Fernald site could be completed. A Removal Action is used to abate, minimize, stabilize, mitigate, or eliminate the release or threat of release of hazardous substances. A structural evaluation determined that, due to deterioration, the silo structures were overstressed and unable to withstand high wind forces. Demolition of the entire structure was warranted.
ORIGINAL DEMOLITION DESIGN
The initial Removal Action Work Plan listed 58 detailed, sequential steps to demolish the silos, one at a time. The demolition design required that all work be done in containment to prevent emissions of radiological contamination into the air. This containment consisted of scaffolding built around the silos and covered with plastic sheeting.
The containment sheathing had two curtain walls, an inner wall from the floor slab to the bottom of the silo and an outer wall that started above the slab near the bottom of the silo and enclosed the entire scaffold.
The ventilation system for the containments operated on three 4,000 cubic foot per minute (cfm) air-filtration devices (AFDs) equipped with High Efficiency Particulate Air (HEPA) filters. The units ventilated through a common duct and stack, which was monitored for emissions. The containment was ventilated using ducts routed to each silo which directed makeup air through the silo from outside the containment. To maintain a balance and steady number of air exchanges, multiple sizes of ductwork

and configurations of air handlers were required.
The containment scaffold around the tile silos was designed to carry hoists that would suspend a large debris slide or "funnel." The funnel would fill the shipping container immediately below the cone. Only one silo was to be demolished at a time because a multiple hoisting system was to be used to suspend a waste container and debris slide inside the silo to catch dislodged tiles as they were thrown onto the debris slide. The intent was that only the filled container would be lowered to ground level to be disconnected and replaced with another container.
The proposed hoists, required to raise and lower a debris slide, would have been individually operated with little tolerance between the slide and the perimeter of the silo. There would have been no allowance for uneven movement of the slide, which could have jammed against the silo wall.
Additional scaffold height would have been needed to install the hoists over the silos. This height would have limited access to repair any mechanical problems. The designed weight of the hoists added dynamic loading to the scaffold and complicated the scaffold design and erection.

SUBCONTRACTOR DEFAULTS

In November 1993, the construction demolition subcontractor defaulted at 30 percent completion. It would have taken five months to rebid and mobilize a new contractor to complete the work. Those five months would have been lost-time and the project would not have been completed by the December 1994 USEPA deadline and could have resulted in major fines. Therefore, the site service contractor was chosen to complete the project.

CHANGE IN STRATEGY

Reevaluation of project documents revealed ways to improve the project schedule and worker safety through design changes and activity resequencing. Project limitations, imposed by the existing work plan, were also evaluated. This evaluation was outside the scope of normal construction review.
Because the Removal Action Work Plan was a USEPA approved legal document, the construction project team proposed the enhancements to CERCLA/RCRA Unit (CRU) project engineering, environmental engineering, and safety assessment team members. This project team wrote a document detailing the changes, "Plant 1 Ore Silos Removal Action Enhancements," which was presented to DOE and USEPA. Both agencies concurred, for both worker safety and environmental reasons, that the enhancements should be implemented.
The enhanced work plan outlined the criteria and Applicable or Relevant and Appropriate Requirements (ARARs) as concepts, rather than discrete steps. Work packages were written for the different types of tasks done, rather than for each step. Worker safety was enhanced by the elimination of work under a suspended load. Environmental controls were improved by fully enclosing the containment. The changes were made during the construction phase and incorporated construction supervision and worker input on work methods during the development of work packages for the field.

PROJECT ENHANCEMENTS

Many Plant 1 Ore Silo structural design assumptions and decisions were based on evaluations to determine the life expectancy of the silos during production. New structural surveys were performed based on demolition requirements. With some additional column repair, the Plant 1 Ore Silos had sufficient structural integrity for the work to be done.
The new demolition sequence made the structure and work area directly accessible to the workers, eliminating the need for manlifts while removing auxiliary platforms, decking, etc. For safety, worker access was still limited by the number of people allowed on scaffold planks.
The containment design was changed so that all sheeting went to the floor and divided the facility into three chambers of two tall and four short silos each.
Fig. 2. Scaffolding and plastic sheeting was used to construct the containment around the tile silos.
Thus the ventilation equipment would have adequate capacity to ventilate each chamber. More silos could then be demolished at a time. Make-up air was provided at the top of the containment to avoid worker exposure to airborne contamination. Additionally, bringing the sheeting to the floor provided better control of debris from demolition of the tile.
Since the previous subcontractor had not fabricated the ductwork for the tile silos,

alternative duct routing was planned, which could use the existing duct from the concrete silos containment.
Additional savings resulted because the remaining ductwork was changed from stainless steel to galvanized and corrugated flexible duct.
The ductwork was run to the containment wall, with a branch to each chamber, to provide a minimum of seven air exchanges per hour in accordance with the plan requirements, within each chamber being worked in. The extensive ductwork of various diameters routed to the base of each silo was deleted.
To prevent the containment from becoming "dusty" with air-borne contamination, additional portable HEPA air-filtration devices were used at the base of the debris chute during the demo of each tile silo. The use of portable air-filtration devices meant more filter changes on the local units, which protected the filters in the 4,000 cfm units, which were difficult to change. The chute itself had a dust cover with duct taps sized to fit over a waste container. After the waste container was positioned under the chute, plastic was used to provide cover to the waste container to seal all voids.
Fig. 3. Waste container placed under the debris chute to collect tile silo debris and minimize emmissions due to dust.
This reconfiguration of the ductwork resulted in the duct being installed once, and allowed flexibility to do the same task on all the silos versus completely demolishing one silo before going to the next.
The decision to delete the hoist completely eliminated the safety risk to workers who would have been under the suspended load of the debris slide while connecting and disconnecting the shipping container. Eliminating the hoists allowed the scaffold roof to be lowered and sloped for drainage. The installation of the temporary roof was simplified and improved to a tube and clamp type of scaffold system verses extensive beam and headers bridging the scaffolding over the silo. Eliminating the use of the hoist made the scaffold safer by eliminating a sizable weight and reduced the anchoring needed for the scaffold footings.
Finally, the new approved plan used a combination of debris chutes and conveyors on the ground floor for movement of the waste containers. Personnel would not be required to move beneath the debris as it was being collected and moved to the waste containers. The use of chutes eliminated the need for additional electrical service and decreased the potential for mechanical failure.
Since the actual weight of the tile was unknown and there was concern over controlling the impact of falling tile into the waste container, a chute with a 45 degree off-set was used. The advantage of the off-set was that the crew loading the container, and monitoring the filters, were never under the crew breaking the tile. The chutes were made of steel to withstand impact loads and designed to prevent plugging. The silo cone bottom remained in place and funneled the debris into the chute. The impact of falling tile on the cone bottom and chute reduced the tile size so that waste containers could be efficiently loaded in compliance with the shipping weight limit.
Two sets of conveyors were mounted on the ground floor to slide the waste containers in and out of the containment. Use of the conveyor provided excellent control of the waste containers through the scaffold legs. Thus, a larger opening in the frame was not required for heavy equipment and access (i.e., fork truck). An airlock and staging area was set-up at the containment wall for the wipe-down of full waste containers to prevent spread of contamination.
An equipment mounted shear was used to demolish the steel structure rather than removing it piece-by-piece, by hand, using torch and rigging. Beside the speed at which it can cut, using the shear reduced worker risk of falls from elevated heights and exposure from lead-based paint and air-borne radiological contamination.
SUCCESSES
The Removal Action was completed safely with minimum worker or environmental exposure to contamination. The debris was containerized for shipment to an approved off-site disposal facility, the structural steel was staged for future disposition, and the concrete pads were locked-down for removal later in the demolition or the remediation of the site.
Working together, with a regard for worker safety paramount in all planning steps, resulted in a hazardous job done without incident. Although requirements, such as 100 percent tie-off of personnel, made work difficult at times, the labor force willingly complied with safety directives.

Fig. 4. Workers shown dismantling tile silo.
Summer work hours were changed to night shift because of excessive heat inside the containment. When the final structure had been demolished, this intensive eight month, 49,473 man hour, effort resulted in only four minor injuries.
Implementing changes to the Removal Action Work Plan resulted in the successful demolition of the Plant 1 Ore Silos one month ahead of the USEPA commitment date.
The lessons learned during this Removal Action include:
1. Use conventional demolition techniques, such as debris chutes.
2. Keep it simple. Don't overdesign.
3. Use a detailed work plan for field implementation, that parallels the intent of the regulatory document.
4. Be flexible. Unknown field conditions frequently happen during remediation projects.
5. Involve construction in the design phase. Communicate ideas and practical field implementation methods. Design change is expensive and results in lost time for field work.
6. Keep lines of communication with regulators open. Don't be afraid to sell enhancements, especially ones that improve safety and environmental outcomes.

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DECONTAMINATION IN THE MIDST OF ACTIVE RESEARCH AND ADMINISTRATIVE OPERATIONS
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ABSTRACT
Many decontamination projects are performed in buildings where ongoing operations have ceased and corresponding occupants have vacated the facilities. In the case of the Battelle Columbus Laboratory Decommissioning Project (BCLDP), remediation has to proceed simultaneously with, directly adjacent to, and literally in the middle of, ongoing research and administrative operations. This has resulted in some additional planning, operational, and interfacing challenges not found in more traditional decontamination projects. Keys to successfully meeting these challenges include planning properly by taking into consideration the "extra" time and resources needed, communicating openly with facility occupants during all phases of the project, and being flexible and responsive in addressing occupants' concerns as much as practical.
INTRODUCTION
The Battelle Columbus Laboratory Decommissioning Project (BCLDP) is a Department of Energy (DOE) Environmental Management (EM) project to decontaminate and decommission (D&D) 15 privately owned buildings in central Ohio. Remediation began in 1989 and, when completed, will allow use of all facilities without any radiological restrictions. Most of the buildings and equipment items became contaminated as a result of various government sponsored nuclear research activities dating back to the 1940s. The cost of the project is shared between DOE (90%) and Battelle Memorial Institute (10%).
Battelle's Building KA-4 and Building KA-A are two of the 15 facilities and are still being used for non-nuclear research and office operations. D&D work to date in the BCLDP has been performed primarily in buildings where ongoing operations have ceased and corresponding occupants have vacated the facilities. This is a common scenario in D&D and environmental remediation projects throughout the United States. Cleanup work can proceed without major concern for other operations utilizing the facilities. However, in the case of BCLDP's Building KA-4 and Building KA-A, cleanup has had to proceed simultaneously with ongoing research and administrative operations. This has resulted in some additional planning, operating, and interfacing challenges. The remainder of this paper will explore these unique challenges and the effective methods instituted to meet them.
ADDITIONAL CHALLENGES TO THE DECONTAMINATION PROCESS
First, a criterion needs to be developed to determine what really constitutes an "additional challenge" to the decontamination process. For this paper the following criterion was used:
  Additional Challenge - an item that resulted in an activity being performed that either would not have been performed at all, or would have been performed at a significantly reduced effort if the building had been vacated and not had operations ongoing during remediation.

The additional challenges can be grouped into three categories presented below, along with descriptions of the challenges themselves:

Planning Challenges

1. Using historical records, process knowledge, and available data, specific rooms are identified as being radioactively "suspect" and are included in the project. A challenge at times is to determine the internal organization responsible for these rooms so items can be removed and schedules for relocation and remediation properly coordinated.
2. Many rooms and items identified as being suspect are in use at least partially for research or administrative operations. These rooms are properly rendered completely unusable when decontamination activities are initiated. At times, this "squeezes" organizations out of money making space. It is a challenge to plan for and accommodate the many requests for accelerated room remediation and return for unrestricted use.
3. Some rooms to be remediated are located in the middle of plush, well-furnished, and highly populated facilities. A planning challenge is to mobilize remediation equipment and personnel while minimizing the impact on the surrounding operations.

Operating Challenges:

4. It is to everyone's benefit if decontamination can be performed without relocating active functions. A challenge is to identify those "low-impact" remediation activities that can be performed while laboratories and offices remain occupied.
5. A major remediation activity is the removal of contaminated drain lines from the buildings. A challenge is to remove these lines while keeping active the intersecting uncontaminated drain lines originating from ongoing research activities remaining in the building.
6. The nature of the work in Buildings KA-4 and KA-A is such that decontamination may be performed in rooms literally surrounded by occupied offices, laboratories, and hallways. A constant challenge is to effectively isolate remediation operations from these adjacent activities.
7. In many instances, it first appears that active operations will need to be moved to allow for effective remediation. A challenge is to develop and apply "in-situ" characterization and decontamination techniques to prevent expensive relocation of operating laboratories.
8. In practice, most of the time the only practical alternative is to relocate some functions to comparable facilities. A challenge is to integrate this activity into the overall operation of the BCLDP.

Interfacing Challenges:

9. As previously mentioned, many decontamination efforts are surrounded by other research and administrative operations. It is a challenge to perform remediation and implement associated safety and operational controls in areas directly adjacent to individuals who have limited knowledge of radioactivity and/or decontamination operations.
10. Another "co-location" challenge for the BCLDP is working adjacent to research operations that are not conducive to remediation and construction type work.

METHODS TO MEET ADDITIONAL DECONTAMINATION CHALLENGES

Methods instituted that were effective in meeting the additional challenges are listed below and numerically correspond to the 10 previously presented items.

Planning Methods

1. Utilizing building diagrams and other organizational information supplied by the internal facilities department is the first step in determining ownership or occupancy of specific rooms. Personal contact is then made with individuals managing and working in targeted rooms. After a period of negotiation, plans for remediation are agreed on between both parties, preventing "surprise" remediation work crews showing up in the middle of a laboratory experiment or other important task. As much as possible, remediation is planned to coincide with operational down or slow times, thus minimizing the financial burden on the impacted organizations.
2. Priorities of the research operations are accommodated as much as possible. Many of these operations can't afford significant down time. A list of critical equipment in suspect rooms is developed and survey, decontamination (if necessary), free release and relocation of these items are scheduled as a priority. Requests for schedule acceleration to return an entire room for unrestricted use must often times be denied due to factors such as planned neighboring remediation work, Independent

Verification Contractor (IVC) schedule, and restoration activities. However, plans have been implemented that allow for acceleration of specific building floors being released for unrestricted use.
3. In mobilizing decontamination personnel and equipment, a key item is communication with surrounding occupants on remediation plans. Then both parties can plan the most efficient and least impactful methods of moving equipment and personnel into the work areas. Some of these methods include moving large equipment items after normal working hours and placing protective coverings (particle board, plywood) over plush carpeted surfaces.
Operating Methods
4. The most common activity able to be performed in occupied areas is removal of contaminated drain lines. Drain diagrams are obtained or developed so the exact rooms to be impacted can be identified. Occupants are then contacted by the remediation team and the activity is discussed, including the safety measures to be used. A schedule is agreed upon, extra precautions taken to protect items in the offices, and every effort expended to complete the job as promised. Of critical importance is leaving the office as clean, or even cleaner, than before the performance of the task.
5. In Building KA-4 and Building KA-A, contaminated drain lines can not be removed without consideration for intersecting "clean" lines. These other lines are in use constantly and provisions need to be made to keep them operating. Drain line diagrams and knowledgeable facilities personnel are consulted to explore various options. In most cases, clean drain lines in use are rerouted to other clean lines, leaving the contaminated lines free to be removed. Occasionally, the active clean drain lines need to be shut down as rerouting activities are finalized. This is coordinated with building occupants and preferably performed on weekends.
6. The most effective point of isolation in the affected buildings is at the ends of hallways. During high-impact activities such as concrete wall and floor removal, soil removal, and underground drain line removal, access must be limited to only remediation workers. Communication with affected individuals concerning inaccessibility of the hallways, planning alternative passage routes, and installing barriers at the ends of hallways proved to be effective in limiting access to active areas. During certain room surface decontamination efforts, adjacent rooms and hallways were able to remain open. Room walls actually provided a sufficient isolation barrier that negated major impacts on the neighboring operations.
7. Normal procedure for Building KA-4 is to relocate operating groups that occupied suspect or contaminated rooms prior to beginning remediation. In the case of a Class 100 Clean Room occupying laboratory space above suspect underground drain lines, an alternative method was investigated because of the potential expense involved in moving this type of facility. The drain lines were internally cleaned and flushed. A snake like probe detector was then used to obtain direct survey measurements from inside the drain line in an attempt to verify the absence of contamination in the pipe and surrounding soil. In this specific instance, results from the probe indicated contamination above release limits remained at various locations along the drain line. The laboratory was then relocated and the entire drain line excavated and removed. Direct survey measurements of the drain line during removal correlated closely with the results obtained using the snake detector, instilling further credibility in this type of detection system. Under the BCLDP, a custom-made probe able to fit through many types of drain lines is currently being developed. The goal is to use this probe to free release other drain lines or to identify only specific sections for removal as opposed to the entire line.
8. Relocating active operations is treated just as any other piece of work in the project. Activity scopes, costs, and schedules were developed by individuals in the groups moving and BCLDP staff. This information was submitted to the local DOE office for approval and then added to the formal project baseline. BCLDP staff then assumed responsibility for managing facility relocation and modification tasks in addition to their remediation roles.
Interfacing Methods
9. An important step is to assume that individuals from ongoing operations will have questions and anxieties once decontamination activities are scheduled near their facilities. This is the case in Buildings KA-4 and KA-A and questions are also routinely fielded from past occupants of areas scheduled for decontamination. Personal contact is made with impacted individuals by the BCLDP Project Manager or

Field Supervisor prior to beginning remediation activities. The nature of suspect or confirmed contamination is discussed along with what to expect operationally during remediation. Industrial safety and radiological protection hazards and controls are also discussed in as much detail as needed. Continuous communication takes place between the two parties during remediation efforts and this helps prevent "alarmist" type reactions from individuals.

10. Several types of activities not naturally conducive to decontamination and construction operations are located near the remediation areas. These include sensitive engineering research, clean room optics research, and administrative work by other Battelle organizations, government agencies, and a university function leasing some space. Special efforts taken to address these activities include coordination with affected parties on use of construction equipment, heightened attention to keeping dust and dirt levels low, and use of signs, postings, and physical barriers.

CONCLUSION

For obvious reasons, decontaminating a building while the facility still houses ongoing operations is not a preferred option. However, in the case of the BCLDP and perhaps other projects, it may be the only practical option. Two customers now need to be satisfied - the client funding the project and the owner/occupants of the facility. In situations like this, keys to success include planning properly by taking into consideration the "extra" time and resources needed, communicating openly with facility occupants during all phases of the project, and being flexible and responsive in addressing occupants' concerns as much as practical.

Session 52 -- DOE Spent Fuel: Issues, Challenges & Options
Co-chairs: John Jicha, Jr., USDOE;
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DOE-OWNED SPENT FUEL PROGRAM STRATEGIC PLAN: ISSUES, CHALLENGES, AND OPTIONS

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ABSTRACT

The Department of Energy faces a significant challenge in managing and disposing of the inventory of spent nuclear fuel (SNF) generated by past production and ongoing test/research reactor operations. The Office of Spent Fuel Management was created within DOE's Environmental Management program to address issues associated with, and coordinate activities related to, the management of DOE-owned SNF. This paper discusses DOE's efforts to plan for and implement management of its inventory of SNF, which are embodied in the DOE-owned SNF Program Strategic Plan issued in December. The SNF Strategic Plan presents an overview of issues, challenges, and options for the SNF program; another purpose is to establish a context for the other papers in the conference session that present more detailed discussions of these topics. The strategic plan indicates that the planning basis for the path forward for the ultimate disposition of DOE-owned SNF is to provide new interim storage pending placement of the SNF, along with vitrified defense high-level waste, in the first repository. This new interim storage will be designed and operated to national consensus standards as embodied by NRC requirements.

INTRODUCTION

The Department of Energy (DOE) faces a significant challenge in managing and disposing of the inventory of spent nuclear fuel (SNF) generated by past production and ongoing test/research reactor operations. SNF is composed of nuclear materials or heavy metals such as plutonium, uranium, and thorium that have been withdrawn from a nuclear reactor following irradiation. The Department currently owns and stores approximately 2700 metric tons of initial heavy metal (MTIHM) at a number of locations throughout the country. Ninety-eight percent of this SNF is stored in facilities near Richland, Washington; Idaho Falls, Idaho; Aiken, South Carolina; and West Valley, New York. Figure 1 presents the location, amount, and types of SNF in the DOE-owned SNF inventory.

SNF will need to be stored for decades because a decision regarding ultimate

disposition of DOE-owned SNF has not been made and because DOE is curtailing processing of SNF to recover strategic materials (although limited processing of SNF to meet waste acceptance criteria is under consideration). However, even after a disposal solution is identified, a number of program and policy issues affecting SNF management will remain. Pursuing the option of disposing of DOE-owned SNF in a geologic repository requires that DOE resolve questions regarding a suitable disposal site, acceptable disposal forms, and methods of storing, packaging, and transporting the spent fuel.

The Office of Spent Fuel Management was created within DOE's Environmental Management (EM) program to address these issues and coordinate/consolidate activities related to the production and management of DOE-owned SNF and associated facilities. (The Office of Civilian Radioactive Waste Management [OCRWM] is charged with managing the program for safe disposal of SNF produced by civilian nuclear power reactors.) Since its creation, the SNF management program has been addressing existing SNF vulnerabilities while planning for and working to ensure safe interim/long-term storage and eventual disposal of the SNF.

This paper discusses DOE's efforts to plan for and implement management of its inventory of SNF as represented by the DOE-owned SNF Program Strategic Plan issued in December 1994. There are a number of outstanding issues that must be resolved before progress toward disposition can be made. The paper presents an overview of issues, challenges, and options for the SNF program.

Fig. 1.

PROGRAM STATUS AND ACCOMPLISHMENTS IN FY1994

NEPA Activities

DOE currently is making key policy decisions under a process established by the National Environmental Policy Act (NEPA). DOE is preparing several EISs that will provide the specific information and analyses to help decisionmakers understand the potential environmental consequences of alternative policies and management approaches. DOE is committed to a comprehensive NEPA review process in making decisions on the storage, disposition, and, if appropriate, transportation of DOE-owned SNF. In a real sense, NEPA will drive DOE-owned SNF Program strategy.

In one EIS, titled "Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory (INEL) Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement," DOE is considering programmatic SNF management decisions on the appropriate means of managing existing and projected quantities of SNF for an interim storage period that could last until the year 2035. The EIS, which has been released in draft form, deals with the entire SNF Program, SNF at the Idaho National Engineering Laboratory, and foreign research reactor SNF. The final draft is scheduled for release in April 1995.

Several site-specific NEPA reviews will tier from this SNF EIS and be guided by, and consistent with, it. Sites currently planning or preparing EISs to analyze site-specific management activities include the Hanford site in Washington State and the Savannah River Site (SRS) in South Carolina.

Other SNF management decisions involve SNF from foreign research reactors that contain U.S.-origin enriched uranium. In April 1994, DOE issued an Environmental Assessment (EA) of Urgent-Relief Acceptance of foreign research reactor SNF after evaluating the potential environmental impacts on a number of ports of entry and transportation routes being considered for the potential receipt and shipment of small amounts of SNF from foreign research reactors to SRS. DOE reached a Finding of No Significant Impact for the proposed action. Based on the EA, DOE decided to accept a limited number of SNF elements containing uranium enriched in the U.S. from seven research reactors in Europe. A subsequent EIS, titled the "Proposed Policy for Acceptance of U.S. Foreign Research Reactor SNF EIS", will evaluate the potential environmental effects of establishing and implementing a policy to accept the return of spent fuel from foreign research reactors over the next 10 to 13 years. The Record of Decision for this EIS is scheduled to be issued in the summer of 1995.

Other site-specific analyses like the ones discussed above will be performed to address the environmental consequences related to relocating and managing SNF inventories in a range of configurations within the DOE complex. DOE will use these analyses and evaluations to implement its programmatic decisions on SNF management.

The SNF program will also be affected by activities being carried out by other DOE programs. For example, the Office of Civilian Radioactive Waste Management is planning NEPA reviews with regard to the suitability of the Yucca Mountain, Nevada,

site as a geologic repository and the utilization of a multipurpose canister development program.
The SNF Strategic Plan is closely connected to the NEPA process. However, it is not intended to prejudice decisions on programmatic alternatives under consideration in the Programmatic SNF EIS process or in follow-on NEPA reviews involving the implementation of programmatic decisions. The Strategic Plan addresses program issues and directions for the SNF program, but cannot present a specific path forward until the Programmatic SNF EIS process is completed.
Spent Nuclear Fuel Vulnerabilities
In August 1993, the Secretary requested that the Spent Nuclear Fuel Working Group -- made up of site personnel and participants from the cognizant Secretarial Offices, Operations Offices, the National Laboratories, and the Office of Environment, Safety, and Health -- assess the conditions of DOE SNF storage facilities. After studying the conditions at 66 facilities at 11 sites, the working group published the Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and their Environmental, Safety and Health Vulnerabilities (Volume I, November 1993). The Spent Nuclear Fuel Working Group found no conditions that required immediate action to prevent harm to the workers or the public, but did identify five DOE facilities and three burial grounds that warrant priority management attention to avoid unnecessary increases in worker radiation exposure and cost during clean up. The report also raised a number of issues and concerns about Department efforts to store SNF. For example, the absence of a clear path forward for the ultimate disposition of DOE-owned SNF has complicated DOE's efforts to maintain safe interim storage.
In February 1994, EM responded to the Working Group's report by publishing the first of a three-part "Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities." The Phase I Action Plan addressed 31 of the 33 high priority vulnerabilities and 48 of the lower priority issues. The Phase II Action Plan, released in April 1994, updated the Phase I Action Plan and reflected the resolution of a number of funding issues. Originally intended to address 44 (of the initial 106) vulnerabilities, the Phase II Action Plan actually covered 81 vulnerabilities. The completion of partial action plans identified in Phase II were addressed in the Phase III Action Plan released in October 1994. The Phase III plan constituted the final Headquarters report on all programmatic activities designed to respond to the vulnerabilities identified in the SNF Working Group Report, and represented the completion of the Secretary's initiative to assess the Department's SNF facilities. In keeping with EM policy, stakeholders were given the opportunity to review and comment on all of the action plans.
KEY ISSUES IN SNF MANAGEMENT
An essential element of the planning process for the SNF Program is the systematic identification and resolution of existing and potential issues. Until the NEPA process is complete, the SNF Program will work to address and resolve near-term issues and vulnerabilities, and to define requirements for interim storage with regard to design, location, and funding. This section discusses some of the major policy, management, and technical issues requiring resolution.
Should DOE-owned SNF be placed in the first geologic repository? The Federal Government is pursuing the development of deep geologic repositories for the disposal of civilian spent nuclear fuel and high-level radioactive waste. In 1985, the President accepted the Secretary of Energy's recommendation that defense nuclear wastes should be co-located with civilian nuclear waste. In 1994, the DOE General Counsel concluded in a legal opinion that the legislation empowering the first civilian geologic repository authorizes disposal of DOE-owned SNF in that repository, conditioned upon payment by DOE of a fee adequate to cover the full cost of disposal. The first repository is limited by law to 70,000 metric tons of heavy metal, and 10 percent of that has been allocated to defense materials. However, current projections of defense materials, DOE-owned SNF, and defense high-level waste show that the allocation is inadequate for all these materials. As yet, DOE has made no final decision regarding the specific strategy for the ultimate disposition of DOE-owned SNF.
Should new interim storage facilities be licensed by the NRC? Management of DOE-owned SNF until its disposal is expected to require new interim storage with a capacity to span the time period until the geologic repository is available. DOE is reviewing the technologies used by the commercial spent fuel generators to both

condition (or stabilize) fuel for storage and to design and operate storage facilities. Several of the commercial storage technologies under evaluation have been successfully licensed and regulated by the Nuclear Regulatory Commission (NRC). DOE SNF would come under jurisdiction of the NRC when it is disposed of, similar to DOE high-level waste, in a geologic repository. However, since the current statutory framework does not allow licensing by the NRC of new DOE interim storage facilities for DOE-owned SNF, a decision must be made regarding whether DOE should proceed toward satisfaction of NRC licensing requirements and reviews for these new facilities.

Should some DOE-owned SNF be regulated under RCRA? The Federal Facility Compliance Act of 1992, which addresses management of mixed wastes, raised questions concerning the applicability of Resource Conservation and Recovery Act (RCRA) waste management regulations to DOE-owned SNF, a fraction of which may potentially exhibit the characteristics of a RCRA hazardous waste. DOE-owned SNF is being evaluated in depth, following which results will be discussed with the Environmental Protection Agency and with the affected state regulatory agencies.

The SNF Program must resolve a number of other issues at various sites that will affect the path forward for ultimate disposition of SNF. One issue involves the path forward for N-Reactor fuel, the approximately 2100 metric tons of heavy metal of uranium SNF stored in the old and seismically vulnerable Hanford K-Basins facility located near the Columbia River. After consultations with stakeholders, DOE decided in November 1994 to place the SNF in overpacks and remove it from the K-Basins on an accelerated basis, stage it temporarily in a new interim storage facility, condition it into a dry form at a new conditioning facility, and return it to the new interim storage facility for dry storage pending its ultimate disposition. The packaging, transfer, and wet staging of the SNF are to be conducted under a K-Basins EIS; a Notice of Intent for this NEPA review is scheduled to be issued in February 1995. Other important issues concern the approach to conditioning DOE-owned SNF at SRS; the receipt of foreign research reactor SNF of U.S. origin for storage; and the need for new dry, passive, cost-effective storage facilities that meet current regulatory requirements to store SNF for a prudent interim period until a geologic repository becomes available.

SNF PROGRAM STRATEGY AND GOALS FOR FY1995

The SNF Program Strategic Plan captures the goals, vision, and strategies of the SNF program. The plan is divided into four sections: Mission, Vision, Situation Analysis, and Objectives. The mission statement describes the purpose of EM's SNF management program. The vision statement articulates the results of a successful SNF program; that is, how the SNF Program will look after achieving its long-term goals. The situation analysis briefly describes the current status of the program, discusses key issues and activities that will affect direction and progress, and identifies the key program stakeholders. The objectives section details the objectives that the SNF Program will strive to meet, the proposed strategies for meeting them, and success criteria that will be used to measure progress toward each objective. Figure 2 captures the broad activities that will guide the SNF program.

The mission of the EM SNF Program is to safely, reliably, and efficiently manage Department of Energy-owned SNF and to prepare it for disposal. In completing this mission, EM will protect the environment and the health and safety of workers and the public, while working with stakeholders and fully complying with applicable Federal, State, Tribal, and local laws, orders, and regulations.

The planning basis for the path forward for the ultimate disposition of DOE-owned SNF is to provide new interim storage pending placement of the SNF, along with vitrified defense high-level waste, in the first repository. This new interim storage will be designed and operated to national consensus standards as embodied by NRC requirements. DOE is committed to the responsible management of its SNF, whether under the Atomic Energy Act or RCRA regulatory regimes, based on the results of the evaluation of SNF process knowledge and negotiations with regulators. Figure 3 portrays the SNF program's vision of the path forward to ultimate disposition.

EM has established five top-level program objectives for the SNF management program, which constitute the overall program strategy for pursuing the path forward. These objectives are the broad goals the program will strive to meet on the way to achieving its mission. Attainment of these objectives will constitute the successful implementation of the SNF Program.

1. Establish an Effective Decisionmaking Process Involving Stakeholders

Because the SNF program is relatively new and is not expected to complete its mission until well into the next century, it must establish an effective decisionmaking process to ensure that program goals can be achieved safely, efficiently, cost-effectively, and in full compliance with all applicable environmental laws and regulations. An effective decisionmaking process identifies the decisions to be made, compiles and analyzes the information needed to develop a full range of alternatives, and empowers the right people to make decisions. The SNF program will utilize a systems engineering approach to determine requirements, define alternatives, and promote communication within the SNF program and between the SNF program and its stakeholders. By establishing a sound decisionmaking process now, the managers of the SNF program can define and develop program plans and directions that reflect inputs from many stakeholders but allow for timely decisions.

2. Assure Safe Existing Storage and Resolve Vulnerabilities for DOE-Owned SNF

SNF will be safely stored and managed in existing facilities, conditioned, and consolidated as necessary to alleviate environmental, safety, and health vulnerabilities. Radiation exposure to the work force and the public must be controlled to levels as low as reasonably achievable, and all SNF storage facilities must be managed to maintain environmental regulatory compliance and safety. The policy of the SNF program is to stabilize existing SNF storage facilities until the transition to interim storage is complete.

3. Achieve Safe, Secure Interim Storage of DOE-Owned SNF

EM must establish, operate, and maintain interim storage facilities to safely store SNF until it can be disposed of. This will require determining and implementing a safe, cost-effective, and technologically-appropriate approach to interim storage. DOE must address a number of issues, including the form the SNF will be stored in, planning and constructing modular facilities to lessen budgetary impacts in a given year, and the most

Fig. 2.

Fig. 3.

effective methods of packaging and transporting. EM must coordinate interim storage with known requirements for disposal to ensure that stored fuel is in an acceptable form for disposal. EM also is reviewing the potential role and extent of external regulation; oversight by the Nuclear Regulatory Commission is considered a step along the path forward to geologic disposal. In addition, EM must design, construct, and establish interim storage facilities in a manner that anticipates and facilitates the eventual decommissioning and transition to other uses of these facilities.

4. Prepare for Disposal

DOE is planning for the ultimate disposition of DOE-owned SNF in a geologic repository. However, decisions have yet to be made regarding a disposal site and the form in which the fuel will be disposed. These decisions are essential to achieving closure on the SNF program's mission. The SNF program will work with the Office of Civilian Radioactive Waste Management to establish and meet disposal requirements. Should some of the DOE-owned SNF be determined to be unacceptable for emplacement in its current form, EM will establish a program to produce an acceptable form. The SNF program will resolve all RCRA-related issues concerning DOE-owned SNF prior to disposal.

5. Transfer Surplus Storage Facilities

During and after the shipment of DOE-owned SNF from existing storage facilities to interim facilities and then to a disposal site, EM must plan and prepare for the transition of storage facilities no longer needed for SNF management to other uses.

These interrelated programmatic objectives constitute the top of the SNF program planning hierarchy. They will be used to guide and direct all SNF program planning activities, define operational or tactical objectives and project goals, set milestones and schedules, and establish indicators of performance. Managers and planners in the program will use them to develop and implement the plans and activities that will allow the SNF program to achieve its long-term goals.

The DOE SNF Program Strategic Plan goes on to discuss each of these strategic objectives in detail, briefly explaining them, setting out related strategies for meeting them, and establishing examples of success criteria for measuring program performance in implementing of the strategies. A comprehensive and more detailed discussion of strategies and success criteria is to be provided in the EM Spent

Nuclear Fuel Program Plan.
Key Goals for FY1995
During 1995, the SNF program will complete the Programmatic SNF EIS and the foreign research reactor EIS and issue Records of Decision (RODs). These ROD's will establish the program elements and provide the basis for developing the Program Plan, which is to include a master logic schedule. The Program Plan will provide the link between the policy/planning activities that occupied the early days of the program and actual implementation of the program as defined by the NEPA process in the field.
The SNF program will also finalize the licensing strategy and systems engineering approach that will help to integrate the SNF Program vertically between Headquarters and the field, and horizontally across the sites. The licensing strategy anticipates licensability reviews conducted by the NRC for new interim storage facilities. The systems engineering approach will provide a common basis for managing spent fuel across the complex as recommended by the Defense Nuclear Facilities Safety Board in Recommendation 94-1.
Planning for a dry storage demonstration project will also mature during 1995. This demonstration project will provide a test case for both the licensing strategy and the systems engineering approach. It will also enable DOE to investigate privatization and other procurement options and the applicability of commercial dry storage technology to DOE-owned SNF.
STAKEHOLDER PARTICIPATION
Key to resolution of the major issues and to ensuring the safe storage and disposal of SNF is the establishment of a sound planning process that involves internal and external stakeholders in vital programmatic decisions. The SNF Program Strategic Plan is to be used to inform stakeholders about the program and give them access to program planning and decisionmaking efforts. An effective decisionmaking process must include participation from all stakeholders.
The Environmental Management program is committed to working closely with its stakeholders at the local, state, and federal levels in planning and implementing its management efforts across the country. EM is developing the policies and methods by which stakeholders will help the SNF program develop the nature and scope of its activities, establish short- and long-term goals, strategies, and priorities, and participate in defining budget needs and allocations. By establishing an open and communicative relationship with its stakeholders, the SNF program can help ensure that its decisions and activities reflect the needs and desires of affected communities and organizations.
CONCLUSION
The SNF Strategic Plan presents a vision for the SNF Program: to safely, efficiently, and effectively manage the interim storage of DOE-owned SNF in preparation for its disposal in the geological repository. The five strategic objectives establish the broad scope of activities needed to achieve both near- and long-term success in meeting the challenge of effective and responsible SNF management. Over the coming years, EM will implement the strategies in this plan to address vulnerabilities, interim storage needs, and disposal requirements. Successful implementation of these strategies will mean that EM has accomplished the EM SNF mission and realized the EM SNF vision. DOE-owned SNF will be well on its way to safe, environmentally sound, and responsible disposal.

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MANAGING SPENT FUEL WITHOUT BUILDING BOMBS
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ABSTRACT
One of the many legacies which DOE must address in light of the Cold War's end is the management of spent fuel from its production reactors, foreign and domestic research reactors, and other sources. For decades, these fuels were reprocessed as

part of ongoing production operations. Now, though, a reassessment of the costs, risks, and options for spent fuel management is needed within DOE. As part of this reassessment, DOE should answer several fundamental questions: Who? What? Why? How? then Where?

INTRODUCTION

For decades, many of the spent nuclear fuel (SNF) management activities of the Department of Energy (DOE) were shrouded within the clouds of nuclear weapons production, which was given high priority throughout the Cold War. Fabrication, reactor operations, chemical processing, and other weapons manufacturing functions created vast volumes of radioactive and hazardous waste and environmental contamination. The costs and environmental burdens of storing and reprocessing spent fuel from research reactors and other sources were almost invisible when held next to the costs and consequences of weapons production activities.

Now that the Cold War is over, DOE is moving to create a new plan for managing SNF. First, the Department must address the lingering costs of bringing Cold War operations to an end. In some cases, conditions within the weapons complex have reached a deplorable state, and despite a halt to production activities, much work remains to be done to ensure safety. Second, DOE still has agreements to manage domestic and foreign research reactor spent fuel, naval fuel, and other materials. Without the cover of nuclear weapons production, the costs and burdens of handling this material should become more apparent. Third, DOE also has responsibilities in the realm of managing SNF from U.S. commercial power reactors.

The first two areas of responsibility affect what is broadly referred to as DOE-owned SNF. It is this class of material which we address below (though, several of our basic contentions are relevant to commercial SNF as well). DOE should reassess the costs, risks, and options for spent fuel management in a manner that reflects policies appropriate for the 21st Century. We recommend that this reassessment address several very basic questions: who, what, why, how, and where.

WHO?

Determining who is responsible for SNF management and associated costs is essential. Without Cold War production and in light of increasing federal budget pressures, costs for managing even relatively small quantities of SNF are becoming more visible than ever before. Moreover, spent fuel activities associated with operating research reactors, for example, must compete with ongoing programs necessary to pay the costs of past production activities (e.g., safely storing corroding fuel from past weapons production).

It's inadequate to approach this future planning with an assumption that DOE will continue its historic role, and DOE perhaps recognizes this at least in part. Related issues are considered in the ongoing Environmental Impact Statement (EIS) on Foreign Research Reactor SNF. For example, the EIS examines whether and when DOE will take title to the fuel, who will pay what costs, and similar policy considerations. DOE should apply this line of thinking to other aspects of its SNF management program. Also, the Department should reexamine the role of state and tribal governments, regulators, and concerned citizens in overseeing the program and formulating decisions.

Among the questions DOE should be asking are: Who should provide independent review and regulation of DOE's SNF activities? Who must concur in SNF decisions? Who should be responsible for the costs of which SNF activities? Who is responsible for long-term liability and management?

WHAT?

You don't have to look far back in history to find the time when DOE didn't have a good understanding of the condition of its SNF inventory and related facilities. While facility managers and some others may have been aware of parts of the picture, DOE did not have a clear, complex-wide image of the status of SNF at the end of the Cold War. Similarly, DOE had a poor understanding of SNF inventories and the condition of facilities in foreign countries that participate in the Reduced Enrichment for Research and Test Reactor program.

Coupled with this, DOE often failed to address problems in a timely manner - allowing some fuels to corrode, facility conditions to deteriorate, and political pressures to build. This is not to say that problems were never addressed, but it is clear that early action could have prevented many of the expensive and risky situations faced in the nuclear weapons complex today.

It wasn't until 1993 that things began to change. Through its SNF Vulnerability

Assessment, ongoing EIS's, and the development of a SNF Strategic Plan (the first in the Department's history), DOE is moving toward being able to clearly and fairly describe what comprises its SNF program. This is a vital first step, and despite other criticisms of DOE's actions regarding SNF and even weaknesses with this first step, the value of this effort should be recognized. Without the ability to clearly understand and articulate the conditions which exist, we doubt that DOE managers can effectively implement a SNF program.

In addition to completing this characterization of existing conditions, DOE needs to put in place a program to keep current its inventory and understanding of conditions within the complex. Undoubtedly this is recognized by DOE management, but we are concerned that it may prove more difficult than expected to implement. For years, DOE has struggled with poor internal communication. It often seems that each new study raises more questions than it answers. Time and again, we have seen DOE fail to consistently implement long range programs. The current confusion, for example, over the relationship between ongoing EIS's, the SNF strategic plan, and other decision making processes exemplifies DOE's continuing struggle to become focused, to give taxpayers a clear picture of its intentions, and to move forward. These types of problems continue, and - though they don't mean DOE is doomed to failure - they pose a serious challenge to the effective development and implementation of a long term SNF strategy.

WHY?

Once you understand who is responsible and what is to be managed, the next question is why or for what purpose. DOE should articulate the rationale behind its desire to do anything with spent fuel. This is a vital element in evaluating whether a proposed action is likely to accomplish the objective.

Frequently, the rationale offered is to prepare SNF for geologic disposal. This is inadequate, though. Indeed, why geologic disposal is desirable and whether it is possible should be open for discussion.

Many other questions need to be addressed as well, each targeted to a specific component(s) of DOE's SNF program. When non-proliferation concerns are asserted, DOE should explain why this is a concern, and why management options for SNF affect non-proliferation. For domestic research reactor spent fuel, why is DOE proposing transportation? Is it to alleviate safety concerns, to reduce the cost of research, or for some other purpose? Why does DOE desire to consolidate storage of all SNF of a given type at a single location within the U.S.? Does this make storage less risky, make it easier to pursue future processing activities, or accomplish some other objective? Why does DOE hold on to the notion that its SNF may not be a waste? Again, the goal of this questioning is to identify the problem which DOE is trying to resolve for the various types of SNF under consideration. During the Cold War, why was rarely asked. The driving force was weapons production, and it was often a relatively simple matter to process other SNF in step with production operations. Now, though, we need to ensure that there is an acceptable rationale for planning future SNF management activities and for evaluating the effectiveness of options selected.

HOW?

Once the problem, responsible agencies, and rationale have been established, it is possible to decide how to best manage spent fuel. Actually, we expect the decision will be a range of management options for the varied problems and rationales.

How is the problem best addressed? If spent fuel is in unsafe condition, how can its condition be stabilized? If the problem is the storage facility, rather than the fuel itself, how can storage problems be corrected? If the issue is non-proliferation, how can spent fuel management contribute to non-proliferation goals? Also, how can non-proliferation goals be balanced with community concerns about equity and local environmental, safety, and health risks? If the issue is domestic research, how should DOE's publicly funded activities relate to the needs of research facilities?

DOE is addressing some of these questions. In particular, the Department has focused substantial attention in the past year on identifying ways to stabilize SNF and improve the condition of storage facilities. Unfortunately, DOE's efforts have too often been short circuited by hasty leaps to a final solution when none is readily available or a clinging to historical practices such as reprocessing.

Indeed, the first reaction of many people in the DOE system is to continue reprocessing, with little change from the days of weapons production. This may be a

natural tendency - reprocessing is familiar to many DOE and contractor employees - but does it make sense for the 21st Century? Are there better, more cost effective options for SNF management? When we look at how to manage spent fuel, we should recognize the new opportunities provided by the Cold War's end. How can storage be improved? If some chemical processing is necessary, how can it be done without separating weapons-usable materials? How can vitrification and other technologies play a part in SNF management?

WHERE?

The location for transporting, storing, treating, and disposing spent fuel is usually the most controversial and high profile issue. It wasn't necessarily a surprise, therefore, that DOE's draft SNF Programmatic EIS seemed designed to address principally the question of where spent fuel should be managed. Perhaps the Department was inclined to attempt to resolve the most difficult question first, hoping that answers to other questions would then fall in place.

This is, however, the wrong approach. Before determining where SNF activities should occur, DOE should answer the questions posed above. The location should be selected based on its suitability for conducting the type of operation required to meet the identified needs. DOE should also be realistic in its assumption about the likelihood that SNF will be transported in the future to some "permanent" home, as well as its own ability to foresee future events.

CONCLUSIONS

DOE is taking steps to move spent fuel management into the post-Cold War era, but many of its steps appear hindered by the legacy of past actions and habits. It won't be easy to form a broad consensus, but such an effort is needed. To get there, we suggest DOE avoid the tendency to look first for a location to store SNF and instead take a deliberate approach to build the rationale for action and a plan which accomplishes the nation's needs.

We've suggested a series of fundamental questions to move us closer to that consensus. Our suggestions, though, are just a loose framework which identify many of the key issues still to be addressed. We hope that DOE is able to incorporate them into its strategic planning to produce better results.

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THE DEPARTMENT OF ENERGY FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL PROGRAM

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ABSTRACT

This paper presents an overview of the Department of Energy's (DOE) program for the acceptance policy for foreign research reactor (FRR) spent nuclear fuel (SNF). Since 1959, the United States, as part of the "Atoms for Peace" program, has supplied research reactor fuel (highly enriched uranium [HEU]) to foreign countries which agreed to forego the development of nuclear weapons. The U.S. accepted SNF containing HEU supplied by the U.S. from FRRs until the Off-Site Fuels Policy lapsed in 1988. A key U.S. nuclear nonproliferation policy objective is to eliminate this weapons-usable material from international civil commerce. In support of U.S. nuclear weapons nonproliferation goals, DOE is engaged in three key activities: the Reduced Enrichment for Research and Test Reactor (RERTR) program, initiated in 1978 to develop high-density, low enriched uranium (LEU) fuel to replace the use of HEU fuel in research reactors; the preparation of an Environmental Impact Statement (EIS), as required by the National Environmental Policy Act (NEPA), for a proposed policy to accept (over a 10 to 15 year period) FRR SNF containing enriched uranium supplied by the U.S. for interim storage in this country; and acceptance of a small number of SNF elements from FRRs needing urgent relief while the EIS is being prepared. The status and the technical issues associated with each of these activities will be discussed in this paper.

INTRODUCTION

For more than 50 years, the United States has played a leading role in international efforts to prevent the proliferation of nuclear weapons throughout the world. Proper management of spent nuclear fuel from foreign research reactors is a crucial part of the efforts geared toward achieving this goal since much of this spent nuclear fuel contains highly enriched uranium, which can be directly used in simple nuclear

weapons. U.S. activities have focused on preventing aggressive nations and terrorist groups from acquiring the nuclear materials necessary to make nuclear weapons. With the end of the Cold War and lessening of controls on nuclear materials in the former Soviet Union, there is worldwide concern that weapons-grade nuclear materials (highly enriched uranium and plutonium) could fall into the wrong hands.

A key element of U.S. nuclear weapons nonproliferation policy is to reduce international civilian commerce in highly enriched uranium. In support of this objective, the United States hopes to adopt and implement a policy to reduce the amount of highly enriched uranium in international commerce associated with the spent nuclear fuel from foreign research reactors, and to encourage foreign research reactor operators who use highly enriched uranium fuel to convert to high density low-enriched uranium fuel. The Department of Energy (DOE), through its foreign research reactor spent nuclear fuel program, will work with the Department of State as the agencies primarily responsible for developing and implementing a policy for the management of foreign research reactor spent nuclear fuel.

Background

Since 1945, every U.S. Administration has recognized that preventing the further spread of nuclear weapons must be a fundamental national security and foreign policy objective of the United States. The initial approach to nuclear technology was to classify all nuclear activities. However, the United States soon realized that it would be impossible to prevent other nations from acquiring nuclear technology. Consequently, since the 1950's, beginning with the "Atoms for Peace" program, the United States has provided peaceful nuclear technology to foreign nations in exchange for their promise to forego development of nuclear weapons. In addition, any nuclear technology provided would be subject to international safeguards and inspections to prevent diversion of materials or technology to nuclear weapons activities.

A major element of the "Atoms for Peace" program for peaceful nuclear cooperation, particularly in the early years, was the provision of research reactor technology and highly enriched uranium for use as fuel in the research reactors. Research reactors play a vital role in important medical, agricultural, and industrial applications, and also provide a tool for fundamental scientific research. For example, research reactors are a vital tool in cancer therapy and radioimmunoassay blood testing. Research reactors also have served as major training facilities in nuclear technology. For example, the research reactor operating in Austria is used by the International Atomic Energy Agency to train personnel who conduct international inspections of nuclear facilities worldwide.

The transfer of enriched uranium from the United States to other nations was usually supported by a bilateral research agreement for each foreign research reactor. Before 1964, these agreements provided for the lease of the enriched uranium, with explicit provision for the acceptance of the spent nuclear fuel by the United States. After 1964, most agreements provided for the sale of this material to the foreign nation.

After its use (irradiation) in a research reactor, the used (spent) fuel was generally accepted in the United States where it was reprocessed to extract the remaining uranium. In this way, control was maintained over the highly enriched uranium, which otherwise could be used in the production of nuclear weapons. The United States began accepting highly enriched uranium spent nuclear fuel from foreign research reactors in 1958.

After 1964, the operative policy under which the United States accepted foreign research reactor spent nuclear fuel containing uranium enriched in the United States became known as the "Off-Site Fuels Policy." This policy was delineated in a series of Federal Register Notices issued until 1987 and was incorporated into bilateral agreements with recipient countries. The term "Off-Site Fuels Policy" was used to indicate that the spent nuclear fuel had been irradiated at facilities not owned by the DOE. Under the "Off-Site Fuels Policy", the United States accepted, temporarily stored, and reprocessed spent nuclear fuel containing U.S.-origin highly enriched uranium. The rationale for the policy was to discourage the stockpiling abroad of spent nuclear fuel containing highly enriched uranium and to recover the fuel value of the highly enriched uranium remaining in the spent nuclear fuel.

In 1978, DOE initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program. This program was aimed at reducing the use of highly enriched uranium in civilian programs by promoting the conversion of foreign research reactors from

highly enriched uranium fuel to low-enriched uranium fuel. Research reactors are of particular interest in this endeavor because the major civilian use of highly enriched uranium is as fuel in nuclear research reactors. As a part of this program, the DOE developed low-enriched uranium fuel and worked with foreign research reactor operators to modify their reactors to run on such fuel. The foreign research reactor operators who converted to low-enriched uranium fuel did so in support of nonproliferation objectives, even though such conversions were expensive and generally resulted in reductions in the capabilities of the reactors and increased operating costs.

In 1988, the DOE's "Off-Site Fuels Policy" to accept highly enriched uranium spent nuclear fuel expired. At the end of 1992, the policy as it applied to the acceptance of low-enriched uranium spent nuclear fuel also expired. The "Off-Site Fuels Policy" was not immediately renewed because of the need to assess the environmental impacts of a new policy and to comply with the National Environmental Policy Act. Because the United States has not been in a position to accept highly enriched uranium spent nuclear fuel for six years (except for a recent "urgent relief" shipment of 153 spent nuclear fuel elements described later in this paper), many foreign research reactor operators will soon run out of storage capacity or face safety and regulatory issues associated with the presence of spent nuclear fuel at their sites (these facilities were not designed for long-term storage of spent nuclear fuel). The cessation of the United States' acceptance of foreign research reactor spent nuclear fuel associated with the expiration of the "Off-Site Fuels Policy" has undercut the perceived reliability of the United States as a partner in peaceful nuclear cooperation.

Program Objectives

As part of the United States objective of curbing the proliferation of nuclear weapons, DOE and the Department of State are committed to seeking a policy to reduce the amount of highly enriched uranium in international commerce associated with the spent nuclear fuel from foreign research reactors. By developing and implementing a proactive policy to manage foreign research reactor spent nuclear fuel, the United States will offer an incentive for foreign research reactors to continue in the RERTR program by using low-enriched uranium fuel instead of highly enriched uranium fuel. This proposed policy would affect only those foreign research reactors that have received fuels containing uranium enriched in the United States. The purpose of this policy would be to reduce the likelihood of diversion of U.S.-origin highly enriched uranium for use in nuclear weapons by reducing the amount of highly enriched uranium in international commerce.

By proposing a policy for management of certain foreign research reactor spent nuclear fuel, DOE and the Department of State have no intention of establishing a commitment to continue accepting such spent nuclear fuel indefinitely. Rather, the intent of the proposed new policy and the overall program is to remove U.S.-origin highly enriched uranium from international commerce while giving the foreign research reactors and their host countries time to convert to operation with low-enriched uranium fuel and make their own arrangements for disposition of their subsequent low-enriched uranium spent nuclear fuel. After the proposed spent nuclear fuel acceptance policy expires, the reactor operators and countries in which the research reactors are operating must be prepared to implement their own arrangements for disposition of their spent nuclear fuel.

PRIMARY PROGRAM COMPONENTS

Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

The Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement (FRR SNF EIS) will evaluate the potential environmental impacts that could result from the proposed action to adopt and implement a policy to reduce the amount of highly enriched uranium in international commerce associated with the spent nuclear fuel from foreign research reactors. The proposed action applies to aluminum-based and TRIGA (for Training, Research, Isotope reactors built by General Atomic) research reactor spent nuclear fuels and targets containing highly enriched uranium and low-enriched uranium of United States origin. Implementation of the proposed action would be performed under one of three management alternatives. Management Alternative 1 is to accept and manage the spent nuclear fuel in the United States. Management Alternative 2 is to adopt and implement a policy to facilitate the management of foreign research reactor spent nuclear fuel at one or more foreign

locations. Management Alternative 3 is a combination of components of Management Alternatives 1 and 2 and is referred to as the Hybrid Alternative. The Management Alternatives of the proposed action are portrayed in Fig. 1.
The foundation for the analysis presented in the FRR SNF EIS, is the evaluation of the components that comprise the basic implementation of Management Alternative 1. Since Management Alternative 2 and the No Action Alternative would not have any direct environmental impacts in the United States, they require only policy analysis in this EIS. Management Alternatives 1 and 3, however, would both have environmental impacts in the United States and the components of the basic implementation provide the parameters with which to analyze their potential environmental impacts in this EIS.
The detail of analysis provided for the basic implementation components is based on the fact that some variation of these components is utilized in each implementation alternative under Management Alternative 1, as well as in Management Alternative 3. In this way, analysis of the implementation alternatives as well as Management Alternative 3 can be tiered from the analysis of the basic implementation. In and of itself, the basic implementation of Management Alternative 1 is a viable implementation alternative for consideration under Management Alternative 1, along with the other implementation alternatives discussed below. However, the level of detail contained in the analysis of the basic implementation does not indicate any preference for this alternative. Rather, it merely eliminates the need to duplicate information later in the analysis.
A summary of components of the basic implementation of Management Alternative 1 is as follows:
a. Policy Period: The United States would accept qualifying spent nuclear fuel (U.S.-enriched low-enriched uranium and highly enriched uranium under specific circumstances) that is currently stored or generated during the ten (10) year period beginning when the policy takes effect. The policy would allow for an additional three (3) years of shipments for foreign research reactor spent nuclear fuel generated within the ten (10) year period, but which cannot be shipped in the period because of requirements for additional cooling, logistics, or unforeseen obstacles. Spent nuclear fuel would not be accepted for new research reactors starting operation after the starting date of implementation of the policy. Spent nuclear fuel would not be accepted from highly enriched uranium-fueled research reactors that could be converted to low-enriched uranium but for which the reactor operators have refused to convert.
b. Financing Arrangements: The United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing nations. Developing nations are those nations identified by the World Bank as other-than-high-income-economies nations. For developed nations, the United States would charge a competitive fee for all spent nuclear fuel management activities conducted.
c. Amount of Spent Nuclear Fuel to be Accepted: The United States would receive up to a fixed amount of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. The amount of foreign research reactor spent nuclear fuel that would be accepted is approximately 19.2 metric tons of heavy metal, from approximately 22,700 foreign research reactor spent nuclear fuel elements and is based on estimated inventories of foreign research reactor spent nuclear fuel currently stored or to be generated in the 10-year policy period. This amount is higher than originally estimated (12 metric tons of heavy metal, representing 15,000 elements) at the time of the 1993 Notice of Intent. The increase is due to a 3-year delay in the start date for the acceptance policy, resulting in additional inventory, plus an increase in the number of foreign countries that have indicated a desire to participate in this program.
d. Location for Taking Title: The United States would take title of the foreign research reactor spent nuclear fuel at the United States Territorial Waters Limit, or continental United States borders for shipment from Canada. Where DOE takes title would not have an effect on the environment. Title location of the spent nuclear fuel is relevant to questions that include the source and extent of liability for damage in the event of an accident outside the scope of Price-Anderson coverage.
e. Marine Transport: Marine transport of the foreign research reactor spent nuclear fuel would be accomplished by either or both chartered and/or regularly scheduled commercial ships. Chartered shipments may be on purpose-built ships or general

purpose commercial cargo ships. Regularly scheduled commercial shipments would be on general purpose commercial ships carrying other cargo at the same time.
f. Ports of Entry: The potential ports of entry are being identified through use of screening criteria that include topics such as appropriate experience, safe transit capabilities, adequate facilities, and population around the ports and along the routes to potential management sites. Several candidate ports of entry will be specified in the draft FRR SNF EIS. Three high population density ports are also being analyzed for potential impacts, these are: Elizabeth, NJ; Long Beach, CA; and Philadelphia, PA, although none of these ports are expected to be proposed ports of entry. This analysis is included only to demonstrate what the impacts would be for receipt through such high density ports.
g. Ground Transport: This component involves shipping of foreign research reactor spent nuclear fuel from the ports of entry (both seaports and Canadian border crossings) to potential DOE management sites. It could also include transport of spent nuclear fuel from one site to another within the United States. This would depend upon the availability or unavailability of certain sites to accept foreign research reactor spent nuclear fuel at the beginning of the implementation period. If a site were unable to accept the spent nuclear fuel, this could necessitate temporary receipt and management of the spent nuclear fuel at an available site and subsequent transport to another site. Both rail and highway capabilities are available to all ports and sites, except the Nevada Test Site, which lacks rail capabilities.
h. Management Sites: The potential management sites identified in this FRR SNF EIS are bounded by those sites identified in the Draft Spent Nuclear Fuel & Idaho National Engineering Laboratory Environmental Impact Statement (DOE, 1994) for foreign research reactor spent nuclear fuel. These sites are: the Savannah River Site; the Idaho National Engineering Laboratory; the Hanford Site; the Oak Ridge Reservation; and the Nevada Test Site. Of these sites, only the Savannah River Site and the Idaho National Engineering Laboratory have existing facilities that could store the spent nuclear fuel beginning in late 1995. Thus, a two phase approach to management would need to be undertaken, wherein either the Savannah River Site or the Idaho National Engineering Laboratory would accept either or both aluminum-based and TRIGA fuels during Phase 1 (approximately 10 years) and then once facilities are constructed or refurbished at the other sites, the spent nuclear fuel would be transported for management (approximately 30 years) to the Phase 2 sites. If either the Savannah River Site or the Idaho National Engineering Laboratory were chosen as the management site, then the two phase approach would not apply.
i. Storage Technologies: Under this component, DOE would receive and manage foreign research reactor spent nuclear fuel for a period starting in approximately late 1995, and continuing for 40 years until ultimate disposition. During the first few years, management would take place in existing storage facilities that use both wet and dry storage technologies. Thereafter, when construction of new facilities may become necessary, the storage technology proposed is dry technology.
Utilizing the components provided above, DOE and the will evaluate seven implementation alternatives for Management Alternative 1 in addition to the basic implementation described above. Each implementation alternative is comprised of the same components as the basic implementation, however, for analysis purposes, one of the components is varied. A summary of the seven implementation alternatives is as follows:
Alternative 1: Alternate Amounts of Spent Nuclear Fuel to be Accepted:
Under this alternative, there are three subalternatives from which to select. The first is to accept highly enriched uranium and low-enriched uranium from developing nations only. This equates to approximately 1.9 metric tons of heavy metal, representing about 5,000 elements. The second subalternative is to accept only highly enriched uranium, which equates to approximately 4.6 metric tons of heavy metal, representing about 11,200 elements. The third subalternative is to accept in addition to the amounts of spent nuclear fuel identified, highly enriched uranium and low-enriched uranium target materials used in Canada, Belgium and Indonesia for production of medical isotopes. Target materials are estimated to contain 0.6 metric tons of heavy metal, representing a uranium content of 620 typical elements. Under this subalternative, highly enriched uranium target material would only be accepted under conditions that encourage foreign research reactors to convert to the use of low-enriched uranium targets.

Alternative 2: Alternate Policy Durations:

There are two subalternatives that are being evaluated for this alternative. The proposed policy duration stated in the basic implementation is ten (10) years. Subalternative One proposes a policy duration of five (5) years. A five year period covers approximately 18,800 individual elements containing 13 metric tons of heavy metal. The policy would allow shipments and receipt of spent nuclear fuel to be made for eight (8) years starting from the effective date of the acceptance policy as long as the spent nuclear fuel was generated during the five-year period. The additional years would allow for cooling of spent nuclear fuel discharged late in the five-year period, logistics in arranging for shipment of the spent nuclear fuel and delays or other unforeseeable events.

Subalternative Two would allow for indefinite acceptance of highly enriched uranium and a ten (10) year period for acceptance of low-enriched uranium. The amount of low-enriched uranium and highly enriched uranium spent nuclear fuel is the same as stated in the basic implementation, approximately 19.2 metric tons of heavy metal, from approximately 22,700 foreign research reactor spent nuclear fuel elements.

Alternative 3: Financing Arrangements:

Under the basic implementation, the United States would bear the cost for management and transport of the spent nuclear fuel from developing nations and would charge developed nations a competitive fee. Three subalternative options are being evaluated. Under the first subalternative, DOE would subsidize developed and developing nations. In the second subalternative, DOE would charge all nations the full cost of accepting and managing the foreign research reactor spent nuclear fuel. Finally, in the third subalternative, DOE would subsidize developing nations as in the basic implementation, but charge the developed nations a full-cost recovery fee. The amount of spent nuclear fuel that would be accepted under this alternative would be the same as that identified in the basic implementation (19.2 metric tons of heavy metal, 22,700 elements) unless one or more foreign research reactor operators declined to participate in the program because of a higher fee.

Alternative 4: Location for Taking Title:

The location for taking title is relevant to questions of liability and regulatory authority. Thus, if DOE takes title at the site of the foreign research reactor, additional regulatory burden may be placed onto DOE due to the laws of a particular nation being imposed upon the owner of spent nuclear fuel. Taking title prior to shipment might impose upon DOE additional legal liability for damages not associated with a nuclear incident covered by the Price-Anderson Act. For this alternative, however, three variations for taking title were reviewed: taking title to the foreign research reactor spent nuclear fuel before shipment; taking title at the port(s) of entry; and taking title at DOE management site(s).

Alternative 5: Wet Storage Technology for New Construction:

The basic implementation allows for use of either dry or wet storage facilities during Phase 1 and construction of new dry storage facilities for Phase 2. This alternative calls for investigation of wet storage technologies for Phase 2, wherein water filled pools are used to store the spent nuclear fuel. Wet storage methods have been used historically by the nuclear industry and the management sites proposed herein to store a variety of spent nuclear fuel, including the aluminum-based fuels that make up the majority of the foreign research reactor spent nuclear fuel.

Alternative 6: Near Term Chemical Separation of Foreign Research Reactor Spent Nuclear Fuel in the United States:

Under this implementation alternative, near term conventional chemical separation would be conducted at either the Savannah River Site or the Idaho National Engineering Laboratory. Near term conventional chemical separation of foreign research reactor spent nuclear fuel at the other three proposed management sites would not be considered since the Oak Ridge Reservation and the Nevada Test Site do not have facilities in which such chemical separation could be conducted, and the facilities at the Hanford Site are either not suitable for the proposed operations, or are no longer operable. To consider chemical separation at these sites, the spent nuclear fuel would need to be stored in the United States for the period while a chemical separation facility at one of these sites was designed, a project-specific National Environmental Policy Act review performed, and the facility constructed and put into operation. Thus, near term conventional chemical separation at these sites

would not be a reasonable near term alternative.

Chemical separation activities at the Savannah River Site would be limited to aluminum-based spent nuclear fuel. In this case, the TRIGA spent nuclear fuel could, based on the Draft Programmatic SNF&INEL EIS, be transported to any one of the five potential management sites for storage. Transporting TRIGA fuel to the Hanford Site, the Oak Ridge Reservation, or the Nevada Test Site, however, would require new construction or refurbishment of an old facility. DOE believes this is unreasonable for such a small amount of spent nuclear fuel (1.0 metric tons of heavy metal). One of the reasons DOE might decide to chemically separate spent nuclear fuel would be to avoid the construction of such storage facilities. Therefore, under this alternative, the TRIGA fuel would either remain at the Savannah River Site or would be transported to the Idaho National Engineering Laboratory.

In contrast, the chemical separation activities at the Idaho National Engineering Laboratory could include both aluminum-based and TRIGA spent nuclear fuel. In this case, all the spent nuclear fuel could be transported to this site.

Alternative 7: Developmental Processing Technologies:

There is a wide variety of technologies that DOE could use to prepare the foreign research reactor spent nuclear fuel for ultimate disposition. These technologies could be applied at any of the five potential management sites and most would require the construction of new facilities. Potential environmental impacts from the applications of these technologies cannot be assessed at this time since development of the technologies is incomplete and no designs exist for facilities that might implement them. Nevertheless, DOE will discuss in the FRR SNF EIS the potential for conducting a research and development program that could lead to construction of such a facility. Examples include: Chop and Dilute; Chop and Poison; Melt and Dilute; Melt and Poison; Pyrochemical Process; Chloride Volatility Process; Glass Material Oxidation and Dissolution System; Dissolve and Dilute; Enhanced Extraction; and Dissolve and Poison. These are mentioned to provide some idea of the breadth of potential technologies that would exist for preparation of spent nuclear fuel for ultimate disposition.

For Management Alternative 2, DOE would seek to encourage and facilitate the management of foreign research reactor spent nuclear fuel overseas in a manner consistent with United States nuclear weapons non-proliferation policy. Two subalternatives are evaluated as implementation options for this Management Alternative. Subalternative One is to provide assistance to foreign nations with storage. DOE would provide assistance to ensure that appropriate storage technologies, regulations and safeguards were applied. This type of support would only be provided to those countries that are able to store their spent nuclear fuel in facilities in their own countries, as a step towards final disposition of the spent nuclear fuel. Ultimate disposition of the foreign research reactor spent nuclear fuel would still have to be arranged for at the conclusion of the storage period. In the meantime, foreign research reactor spent nuclear fuel containing highly enriched uranium would be stored in up to 41 countries around the world. Subalternative Two is to provide assistance to foreign nations with reprocessing of the spent nuclear fuel. Under this subalternative, DOE would encourage and assist foreign research reactors and reprocessors in reprocessing their spent nuclear fuel overseas in facilities operated under international safeguards sufficient to satisfy United States nuclear weapons nonproliferation concerns. The wastes resulting from this reprocessing would be returned to the country in which the spent nuclear fuel was irradiated. To comply with United States policy, agreements would have to be negotiated with one or more foreign governments to ensure compliance with United States nuclear weapons nonproliferation policy goals. Discussions are currently underway via diplomatic channels to explore the feasibility of meeting these conditions.

Under Management Alternative 3, the Hybrid Alternative, DOE would encourage the reprocessing of any foreign research reactor spent nuclear fuel at Western European reprocessing facilities (i.e., Dounreay, Scotland or Marcoule, France), as in Management Alternative 2, for those foreign research reactors that could accept back the reprocessed waste and DOE would accept and manage the rest of the foreign research reactor spent nuclear fuel in the United States as in Management Alternative 1.

Implementation of Management Alternative 2 would necessitate the negotiation of clear and binding agreements with those foreign nations capable of taking back

reprocessing wastes to ensure compliance with United States nuclear weapons nonproliferation policy. Overseas reprocessing will only be considered for aluminum-based spent nuclear fuel as the technology does not currently exist at any of the overseas facilities to reprocess TRIGA spent nuclear fuel. Thus, TRIGA spent nuclear fuel would be accepted in the United States and transported to the Idaho National Engineering Laboratory for storage until its ultimate disposition. The aluminum-based spent nuclear fuel that would be accepted in the United States would be chemically processed at the Savannah River Site. For the analysis performed on this Management Alternative, the distribution of the spent nuclear fuels by fuel type is consistent with the Draft SNF&INEL EIS Regionalization by Fuel Type Alternative.

In addition to the three Management Alternatives discussed above, the Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement is also considering a no action policy alternative, wherein the United States would neither accept foreign research reactor spent nuclear fuel containing uranium enriched in the United States, nor provide technical assistance or financial incentives for overseas storage or reprocessing. In this case, no foreign research reactor spent nuclear fuel shipments would be sent to the United States and no assistance to foreign nations for managing foreign research reactor spent nuclear fuel overseas would take place. Since the no action policy would only have environmental impacts outside the United States, these impacts are not addressed in the environmental impact statement.

Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel

Because DOE had not accepted any spent nuclear fuel containing U.S.-origin highly enriched uranium since 1988, several foreign research reactor operators began to run out of storage capacity and faced safety and regulatory issues associated with the presence of spent nuclear fuel at their sites. These reactor operators were faced with the decision to either shut down, or ship their spent nuclear fuel offsite for reprocessing. Neither of these options would serve the nonproliferation interests of the United States. At the urging of the Department of State, DOE prepared an environmental assessment that proposed to accept a small number of highly enriched uranium spent nuclear fuel elements at the DOE's Savannah River Site in Aiken, South Carolina. The purpose of the proposed action evaluated in this environmental assessment was to accept the minimum number of spent nuclear fuel elements necessary to maintain the status quo relative to the participation of the foreign research reactors with the Reduced Enrichment for Research and Test Reactors Program until the environmental impact statement discussed above could be completed and a decision made on whether or not to implement a new United States policy for management of foreign research reactor spent nuclear fuel.

The draft environmental assessment was made available to affected States and interested groups on October 18, 1993. Due to the numerous comments received from elected officials, private groups, and individuals, a revised draft of the assessment was released in February of 1994 for a 30-day public review and comment period. This revised draft included the results of DOE and Department of State team site visits to foreign research reactors under consideration for urgent-relief acceptance of their spent nuclear fuel. A reactor was identified as meeting the urgent-relief criteria if, because of the lack of near-term spent nuclear fuel storage space or presence of other problems, the reactor operator was likely to: 1) retreat from the conversion process to low-enriched uranium fuel, 2) reprocess spent nuclear fuel to avoid shutdown, or 3) shut down.

Public comments on the revised draft of the assessment were addressed in March of 1994 and the environmental assessment and an accompanying finding of no significant impact were released in April 1994. Subsequently, 153 spent nuclear fuel elements containing U.S.-origin highly enriched uranium were accepted from Denmark, Austria, Sweden, and The Netherlands in September of 1994. Subsequent shipments have been blocked by an adverse Court decision. This court decision is being appealed.

FY95 GOALS AND ACCOMPLISHMENTS (10/1/94 - 9/30/95)

The following goals were established for FY95 in relation to this environmental impact statement: 1) publish the Implementation Plan in October 1994; 2) issue the draft environmental impact statement shortly, hopefully by late March, 1995; 3) resolve comments stemming from the public comment period; 4) complete the final environmental impact statement by late summer; and 5) issue the Record of Decision

before the end of the year.
To date the first goal has been successfully accomplished, as the Implementation Plan was published in October 1994. At this point in time the other goals are on schedule and are anticipated to be accomplished as scheduled.
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OVERVIEW OF THE SPENT NUCLEAR FUEL PROJECT AT HANFORD
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ABSTRACT
The Spent Nuclear Fuel Project's mission at Hanford is to "Provide safe, economic and environmentally sound management of Hanford spent nuclear fuel in a manner which stages it to final disposition." The inventory of spent nuclear fuel (SNF) at the Hanford Site covers a wide variety of fuel types (production reactor to space reactor) in many facilities (reactor fuel basins to hot cells) at locations all over the Site. The 2,129 metric tons of Hanford SNF represents about 80% of the total U.S. Department of Energy (DOE) inventory. About 98.5% of the Hanford SNF is 2,100 metric tons of metallic uranium production reactor fuel currently stored in the 1950's vintage K Basins in the 100 Area. This fuel has been slowly corroding, generating sludge and contaminating the basin water. This condition, coupled with aging facilities with seismic vulnerabilities, has been identified by several groups, including stakeholders, as being one of the most urgent safety and environmental concerns at the Hanford Site.
As a direct result of these concerns, the Spent Nuclear Fuel Project was recently formed to address spent fuel issues at Hanford. The Project has developed the K Basins Path Forward to remove fuel from the basins and place it in dry interim storage. A systems engineering approach of identifying requirements and the functions necessary to address those requirements was utilized. Alternatives that addressed the requirements were developed and analyzed. The result is a two-phased approach allowing the early removal of fuel from the K Basins followed by its stabilization and interim storage consistent with the national program. The DOE approved the Westinghouse Hanford Company (WHC) Recommended Path Forward for K Basins fuel on November 9, 1994. The WHC Recommended Path Forward starts fuel removal from K Basins by December 1998 and completes it two years later in December 2000 or earlier. The fuel is taken to a new Staging and Storage Facility located in the 200 Area at the Hanford Site. The fuel is staged or held in temporary storage while the stabilization process is finalized and design and construction of the facility is completed. Following stabilization, the fuel is returned to the Staging and Storage Facility for dry, interim storage (up to 40 years) until its ultimate disposition is determined at the national level. The path forward would have the K Basins fuel placed in interim dry storage by the year 2006.
In parallel with developing the path forward for the K Basins SNF, the Spent Nuclear Fuel Project also identified a path forward for management of the other Hanford SNF inventories. This planning is being integrated with the deactivation or utilization planning for each of the facilities containing SNF, as well as the national plan for placing DOE's SNF into interim storage.
HANFORD SPENT NUCLEAR FUEL PROJECT BACKGROUND
Nuclear production reactors were operated at the DOE's Hanford Site from 1944 until 1988 to produce plutonium by irradiating uranium. The fuel from these reactors was then processed at the Hanford chemical processing facilities to separate and recover the plutonium for its intended uses. Additional fuel was irradiated at the Hanford

Site for research purposes or was shipped from off-site reactor facilities for study, storage, or recovery of nuclear materials at Hanford facilities.
The July 1990 Nuclear Weapons Stockpile Memorandum signed by the President of the United States stated that recovery of nuclear materials from Hanford SNF inventories was no longer required for national defense purposes or for future breeder reactor fuel. As a result, processing operations were phased out and legacy SNF inventories were left in a variety of facilities across the Hanford Site. Fuel irradiated at the Hanford N Reactor and single-pass production reactors, the Fast Flux Test Facility, and from the second core of the Shippingport Pressurized Water Reactor currently remain in storage on the Hanford Site, in addition to miscellaneous research and development program irradiated fuel materials.
In general, current SNF storage facilities at the Hanford Site were not intended for SNF storage for an extended period of time and are not adequate for continued storage until the means for final disposition is available (i.e., 40 years). In most cases, these facilities either have other missions or are being deactivated. The inadequacy of these existing facilities for continued SNF storage necessitates that plans to improve storage conditions at the Hanford Site, and ultimately implement interim storage, be identified and implemented in an expeditious manner.
The deficient conditions at the existing Hanford Site SNF storage facilities were identified in a number of internal and external reviews, including the DOE findings documented in the November 1993 report, "DOE Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and the Environmental, Safety and Health Vulnerabilities," (1). These reviews identified degraded fuel being stored in two deteriorating basins located near the Columbia River as requiring urgent action. In 1993 and 1994, Hanford Site stakeholders, regulators, and tribal governments also emphasized their desire for the DOE to relocate the fuel from the K Basins to an alternate location as safely and quickly as possible. These actions ultimately resulted in commitments for the expedited removal of fuel and sludge from the basins.
The Hanford Site's Spent Nuclear Fuel Project was formed in early 1994 to manage Hanford's SNF and to meet those commitments. The mission of the Spent Nuclear Fuel Project is to provide safe, economic, and environmentally sound management of Hanford spent nuclear fuel in a manner which stages it to final disposition. In completing this mission, all of the Site's spent nuclear fuel will be safely stored on site or transferred to appropriate off-site storage, consistent with national plans for all DOE-owned spent nuclear fuel, and the Project facilities prepared for deactivation or alternate use. When these conditions are met, the Project will be completed and the responsibility for management of the Hanford spent nuclear fuel will be turned over to an operating program.
HANFORD SPENT NUCLEAR FUEL STORAGE LOCATIONS AND INVENTORIES
DOE-owned spent nuclear fuel is currently located at several facilities on the Hanford Site as shown in Fig. 1. The facilities, their current primary missions, their fuel storage functions and quantities of spent nuclear fuel are as follows:
K East and K West Basins
The K East and K West Basins store irradiated defense production reactor fuel, primarily N Reactor irradiated fuel. These basins contain about 3,800 cubic meters (1 million gallons) of water each and were part of the K East and K West production reactor complex constructed in the 1950's. They were used to cool discharged fuel prior to chemical processing. After the K Reactors production mission ended, the basins were used as temporary storage capacity for N Reactor fuel while the PUREX processing facility was being refurbished and restarted. When the defense production mission ended in 1990 and PUREX operations were terminated in December 1992, part of the N Reactor fuel inventory remained in the K Basins with no means for near-term removal and processing. The K East Basin contains 1,145.8 metric tons of N Reactor fuel and 0.4 metric tons of single pass reactor fuel. The K West Basin holds 953.8 metric tons of N Reactor fuel and 0.1 metric tons of single pass reactor fuel.
The fuel at the K West Basin has been stored in lidded canisters and the basin environment is relatively clean and free of corrosion products. The fuel at the K East Basin is currently stored in open canisters and some have corroded extensively, releasing fission products to the basin water and producing a substantial quantity of sludge. The basins also have a history of leaking. In the late 1970's, K East leaked about 57,000 cubic meters (15 million gallons) of basin water and again in

1993, another 340 cubic meters (90,000 gallons) escaped. The suspected location of the leak is an unreinforced construction joint between the basin and reactor structures. These conditions lead to an extreme sense of urgency for removal of the fuel, sludge, and contaminated water as quickly as possible.

PUREX Plant

The PUREX Plant was operated to recover uranium, plutonium, and other nuclear materials from irradiated fuel for defense and research purposes. It too contained a small quantity of irradiated production reactor fuel when its operation was terminated. Some single pass reactor fuel is stored in the receiving basin at the facility and a small amount of N Reactor fuel resides on the dissolver cell floor. The total amount of spent nuclear fuel at PUREX is 2.9 metric tons. The PUREX Plant is currently in a program to place the facility in a safe, environmentally sound condition prior to decontamination and decommissioning. These plans include the transfer of the spent nuclear fuel to the K Basins to be managed along with the rest of the production reactor fuel.

T Plant

The T Plant was the first fuel reprocessing facility at Hanford. The processing equipment had been removed and the plant decontaminated. It now serves as a beta-gamma decontamination facility and provides solid waste

Fig. 1.

management services. The Shippingport Pressurized Water Reactor Core II is stored in the receiving basin at T Plant and contains 15.8 metric tons of fuel.

Fast Flux Test Facility

The Fast Flux Test Facility (FFTF) is a 400 megawatt sodium-cooled fast reactor that provided materials, fuel, and component testing capability for the U.S. breeder reactor program during the 1980's. The FFTF is located in the 400 Area at the Hanford Site. When its mission ended in 1993, 11.0 metric tons of reactor fuel cores and numerous fueled experiments remained in the reactor and in storage at the facility. The reactor has been shutdown and the facility is currently being deactivated.

308 Building

The 308 Building in the 300 Area of the Hanford Site was used for the manufacture and assembly of FFTF fuel and test assemblies. The building contains a small Training Reactor, Isotopics, General Atomic (TRIGA) reactor that was used as a neutron radiography facility to inspect fueled components. The 308 Building is being deactivated and the TRIGA reactor has been shutdown. Its core contains 0.02 metric tons of fuel which is currently stored in the reactor vessel.

325 Building

The 325 Building Shielded Analytical Laboratory in the 300 Area supports process demonstration and analytical chemistry requirements for a variety of DOE programs. Over the years, a variety of small spent nuclear fuel samples were stored initially as archive material and then accumulated while awaiting processing or disposal.

324 Building

The 324 Building, located in the 300 Area, is a shielded chemical processing laboratory used for the development of chemical processes from laboratory to pilot scale and for the examination and mechanical testing of irradiated specimens. Seven light water reactor fuel assemblies are stored in the facility. These assemblies were intended for test purposes but remain in the 324 Building only because there is no other storage location.

327 Building

The 327 Building is also located in the 300 Area. It provides shielded, ventilated, and specially equipped laboratories for physical and metallurgical examination and testing of irradiated fuels and structural materials. While the facility is central to the characterization of the fuel stored in the K Basins, its long-term future is uncertain. The spent nuclear fuel stored in this facility comes from a variety of DOE fuel testing programs that used the facility for destructive testing and metallurgical examination.

The combined spent nuclear fuel inventories at the 324, 325 and 327 Buildings total 2.3 metric tons. The total spent nuclear fuel at Hanford is 2,132 metric tons and represents about 80% of the total inventory at all DOE sites. The 2,100 metric tons of metallic uranium production reactor fuel currently stored in the K Basins is about 98% of the total spent nuclear fuel at the Hanford Site.

K BASINS PATH FORWARD

The Spent Nuclear Fuel Project has focused on resolution of the K Basins issues. A recommended path for resolving the safety and environmental concerns and providing for the safe interim storage of this material was developed. The recommendation culminates five months of engineering studies and evaluations focused on accelerated removal of fuel and sludge from the K Basins and its placement in a stable, dry storage configuration until final disposition is achieved in the future.
To arrive at the recommendation, systems engineering and risk-based decision techniques were utilized, in conjunction with a variety of technical and programmatic reviews that included independent assessments by senior experts from outside Hanford. The evaluation process included analysis of cost, schedule, regulatory and stakeholder drivers, and affected tribal values to assure a comprehensive, balanced treatment of the various alternatives. Results from these reviews and analyses were used to formulate a technical and regulatory strategy which optimizes within the alternatives studied. In developing the recommended path, a broad range of alternatives were considered including:
1. Containerization of the fuel and sludge in K East Basin and storage of these materials in the K Basins until facilities are available for the transition to dry interim storage.
2. Removal of the fuel and sludge from both basins at the earliest possible date to a newly constructed temporary wet storage basin that meets modern safety and environmental requirements until facilities are available for the transition to dry interim storage.
3. Expedited transition directly to dry interim storage based on a drying and passivation process.
4. Processing the fuel at a foreign reprocessing plant and providing retrieval and disposition of the sludge at Hanford.
5. Variations within and among the above alternatives.
The Recommended Path Forward utilizes fuel containerization, drying, passivation, and vault dry storage to achieve interim storage. Construction of the proposed vault storage facility is accelerated to allow early removal from the K Basins by staging of wet packed fuel and sludge while the more complex Stabilization Facility is constructed and brought to a fully operable state. The vault storage facility also serves the 40-year dry interim storage function as fuel and sludge are processed at the Stabilization Facility by drying and passivating and returned to the storage vault. The resulting strategy removes fuel and sludge from the K Basins by the year 2000 at a cost (through the year 2012) of approximately $1,150 million (unescalated) including operation and deactivation of Project facilities (including the K Basins).
The framework for the recommendation is a workable National Environmental Policy Act (NEPA) overlay which requires division of the Recommended Path Forward into two phases: the expedited response phase and the interim storage phase. The goal of the expedited response phase is to move the fuel and sludge into a new facility for temporary storage away from the Columbia River as soon as possible. The interim storage phase is structured to implement the Record of Decision (ROD) for a Hanford Site Spent Nuclear Fuel Management Environmental Impact Statement (EIS) which is compatible with the ROD for the DOE Programmatic EIS (2). The Recommended Path Forward is described in Ref. 3. The key elements of the Recommended Path Forward are shown in Figure 2 and described below.
Expedited Response Phase This phase, which would be evaluated as the preferred alternative in an interim action EIS, rapidly improves protection of the public, the environment, and Hanford workers. Fuel and sludge would be transferred from the K Basins to a newly constructed Staging and Storage Facility away from the Columbia River. The interim action EIS is justified by the urgent need to remove fuel and sludge from the K Basins. The preferred alternative is compatible with the DOE Programmatic EIS in that all options being evaluated in the DOE Programmatic EIS for management of spent nuclear fuel remain viable. Existing and modified facilities would be managed in accordance with DOE Orders. The new Staging and Storage Facility would be constructed and operated consistent with Nuclear Regulatory Commission (NRC) technical requirements.
In the expedited response phase, fuel and sludge in K-East and K-West basins would be packaged in large multi-canister overpacks (MCO). Modifications would be constructed at the K Basins to enable minimum fuel and sludge handling to load the MCO. The MCOs would be designed to store fuel and sludge in a wet or dry condition and would enable direct monitoring of fuel, sludge, and surrounding liquid and gas

spaces during the temporary wet storage stage. A line item project would be proposed to construct a shielded vault Staging and Storage Facility to receive, store, and monitor the MCOs. Upon receipt from K Basins, the MCOs would contain wet-packed fuel and sludge and would be stored until staged into the Stabilization Facility. Design criteria for the Staging and Storage Facility would consider other Hanford spent nuclear fuel in development of functions and requirements. The wet packed MCOs would be held in the Staging and Storage Facility only until the fuel stabilization process is available.

Interim Storage Phase

The second phase develops and constructs a fuel Stabilization Facility, based on a drying and passivation process. The fuel and sludge are dried and passivated in the MCO and recycled to the vault storage facility to be stored dry for up to 40 years. This phase is dependent on a completed ROD for the DOE Programmatic EIS and would be evaluated as the reference alternative in a Hanford Spent Nuclear Fuel Management EIS. New facilities constructed during the second phase would be designed to the intent of NRC technical requirements.

During the second phase, MCOs would be transferred from the Staging and Storage Facility to the newly constructed Fuel Stabilization Facility which would be co-located with the Staging and Storage Facility. Here, the fuel and unseparated sludge are dried and passivated to reduce the potential hazards associated with dry storage of the metal fuel. The size (annual throughput) of the Stabilization Facility would be balanced against operations costs to optimize plant size and processing duration since processing is not on the critical path to K Basins deactivation. Development of design criteria for♀

Fig. 2.

The Stabilization Facility will consider other Hanford spent nuclear fuel in development of functions and requirements. The Staging and Storage Facility would be redeployed for use as an interim storage facility until final disposition capability is available.

K BASINS PATH FORWARD NEAR-TERM OBJECTIVES

Major acquisitions will be initiated to support the K Basins Path Forward. These include the Staging and Storage Facility, the Fuel Transportation System, the Multi-Canister Overpacks and the Stabilization Facility. These acquisitions, except for the Stabilization Facility, will be necessary for completion of the expedited response phase of the path forward. Other activities will also proceed in parallel with the major acquisitions to improve the near-term safety posture at the K Basins. These actions are:

Installation of cofferdams to mitigate the consequences of a seismic event by isolating the basin from the anticipated leak site;

Establishing and maintaining Formal Conduct of Operations at the K Basins;

Completing essential systems recovery actions necessary for safe operations, such as electrical, water, fire protection, and maintenance systems improvements;

Reducing personnel exposure through improved dose reduction measures, particularly through source term reductions from cesium contaminated concrete basin walls;

Removing debris from the K East Basin such as unused canisters and discarded tools;

Implementing sludge management activities consistent with the path forward for sludge;

Treating and dispositioning K East Basin water;

Providing basin modifications and readiness for fuel removal.

Fuel and sludge characterization is in progress and technology development activities will be performed as necessary to support implementation of the major acquisitions and the other identified activities.

OTHER HANFORD SPENT NUCLEAR FUELS

To resolve vulnerabilities at other Hanford SNF facilities, and to support facility deactivations, a two-step approach will be utilized to attain safe, interim storage for these fuels. The first step for each fuel involves near-term actions to attain safe, economic pre-interim storage. The second step will be to place each fuel into interim storage as defined in the DOE Programmatic EIS ROD when completed.

Near-term actions for these fuels are as follows:

Most FFTF spent nuclear fuel will be transferred to dry storage casks at the 400 Area for pre-interim storage. The planning basis for pre-interim storage of non self-protecting (Class I by the year 2030) FFTF spent nuclear fuel is transfer to

the Plutonium Finishing Plant at the 200 Area in dry storage casks.

The 308 Building TRIGA reactor spent nuclear fuel will be transferred to the 400 Area storage pad for pre-interim storage in casks that can be qualified for off-site shipment.

The 324/325/327 Building light water reactor fuel inventories will be relocated to the 400 Area storage pad in dry storage casks to correct vulnerabilities at those facilities.

The PUREX Plant spent nuclear fuel will be transferred to the K Basins for consolidated management with the K Basins spent nuclear fuel.

The Shippingport Core II will continue to be managed at T Plant until transfer to interim storage.

The other miscellaneous spent nuclear fuels will remain where they are currently located until their disposition is defined by the DOE Programmatic EIS ROD.

CONCLUSION

In response to the identification of vulnerabilities associated with the storage of spent nuclear fuel at the Hanford Site, the Spent Nuclear Fuel Project was formed. The Project's mission is to provide safe, economic, and environmentally sound management of Hanford spent nuclear fuel in a manner which stages it to final disposition. In just a few months, the Project developed and recommended to the DOE, a path forward to place 2,100 metric tons of fuel currently stored in the K Basins (80% of the DOE's spent nuclear fuel) into dry interim storage. This plan will begin fuel removal from the basins in 1998 and complete that phase over the next two years. Planning for management of other spent nuclear fuel at the Hanford Site is also completed.

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BNFL APPROACH TO DOE SPENT FUEL MANAGEMENT: TAILORING TECHNOLOGICAL CAPABILITIES TO POLICY OBJECTIVES

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ABSTRACT

This paper explores the BNFL process alternatives for managing certain unique inventories of DOE spent fuel and rationalizes these alternatives from the perspective of technological maturity and preference and consistency with US policy objectives.

INTRODUCTION

BNFL has been supplying fuel cycle services to the commercial utility industry worldwide for decades. These services include uranium enrichment, conversion, and fuel fabrication on the "front end", and on the "back end", spent fuel shipment, storage, and reprocessing, and conversion of waste and byproducts to forms suitable for storage, disposal, and/or re-use, including the manufacture of mixed oxide fuel. As much as BNFL would like to think that the US offers a huge potential market opportunity for BNFL's reprocessing services, the reality is quite the contrary. US policies in the 1970s ended the US domestic reprocessing industry and set forth a path for direct disposal of commercial spent fuel, despite the fact that "recycle" seems to be the politically and socially correct choice for virtually every other US resource except uranium.

It just so happens that energy-rich countries like the United States can afford to make decisions on spent fuel management that other, energy-poor countries may view entirely differently. The fact is, BNFL would be the first to admit that either option -- direct disposal or reprocessing -- is entirely acceptable for most standard spent fuel forms and therefore both options should be able to enjoy

"peaceful co-existence." Furthermore, no nation should impose its solutions for the back end of its fuel cycle on other nations.
There are, however, certain unique inventories of spent fuel under DOE guardianship that may require special treatment, and it is these "exceptional cases" that are addressed in this paper.
Before discussing the essence of this paper, it is worth noting for the record the BNFL provides services to those customers that desire these services. BNFL is not in the business of policy-making, nor does it engage in coercion to maintain or augment its customer base. BNFL merely strives to provide practical, technically-sound options for those customers who might need these services.
In the case of DOE, BNFL has examined the inventories of DOE spent fuel and identified certain unique cases where BNFL technology and experience are particularly relevant to very unusual problems now facing DOE. The key, however, is for BNFL to be able to provide these services in a manner not inconsistent with US policy objectives.

US POLICIES RELATED TO SPENT FUEL MANAGEMENT

Fuel Cycle Policies

As already stated, the United States policy decisions in the 1970s resulted in cancellation of the US commercial reprocessing program and set forth a path for direct disposal of commercial spent fuel. However, this policy did not affect the government's nuclear weapons programs, which necessarily required reprocessing of government-owned spent fuel to separate weapons-grade plutonium in order to meet weapons stockpile requirements during the cold war.
But even the government nuclear weapons program eventually changed, due to the end of the cold war. Reprocessing facilities at DOE's Hanford and Savannah River sites were suddenly rendered obsolete, and were shut down for "maintenance" in the late 80s and early 90s, and have not operated since. In 1992, the Hanford Purex plant was placed in permanent shutdown and is now undergoing decommissioning. The fate of the Savannah River reprocessing facilities is in limbo, and subject to a National Environmental Policy Act (NEPA) review still underway.
In 1982, Congress passed the Nuclear Waste Policy Act -- later amended in 1987 -- which sets for the US government responsibilities for development of a permanent disposal facility for receipt of commercial spent fuel and government High-Level Waste (HLW) from DOE's reprocessing activities.
Ironically, the Nuclear Waste Policy Act did not address receipt and disposal of DOE-owned spent fuel, because at the time, it was assumed that all DOE fuels would be reprocessed, therefore, the only government material to be disposed of would have been in the form of vitrified HLW byproducts from the reprocessing activities.
Whether due to bad timing or lack of foresight, it turns out that DOE shut down its reprocessing facilities BEFORE it had completed the reprocessing of its rather unique inventories of spent fuel. As we all know now, DOE is grappling with a decision on how to manage the remaining 2700 metric tons of spent fuel, much of which is highly damaged or unstable, and certainly not anything like the commercial spent fuel for which direct disposal is well suited.
So to summarize, US government policies on spent fuel and HLW disposal are fairly well laid out in the direction of direct disposal, except for some "orphan" categories of DOE spent fuel that were not contemplated during the formulation and codification of US fuel cycle policy.

US Nonproliferation Policies

Overlaying the specifics of US fuel cycle policy are the much bigger demands of US nonproliferation policies. While the two are not unrelated, fuel cycle policy as discussed above is an internal, domestic policy, while nonproliferation policies are necessarily broader in scope, reflecting international policies, treaties, and objectives of a global nuclear community.
For discussion purposes, this paper will address the current nonproliferation objectives of the Clinton Administration, as articulated in a September 1993 White House Fact Sheet (1). In that statement, the US expressed its intentions to discourage reprocessing and plutonium use in the nuclear power industry worldwide, while, at the same time, not interfering with the legitimate commercial reprocessing activities underway in Western Europe and Japan.
Even though Western Europe and Japan are somewhat disappointed in the negative connotations of this US nonproliferation policy, they have so far been able to function within the constraints of these policies, despite occasional overzealous US

government "interpretations" and manifestations of these overall policy objectives.
DOE-HANFORD N-REACTOR SPENT FUEL
DOE identified a number of problems associated with its inventories of spent fuel in a DOE Spent Fuel Working Group (2) issued in December of 1993. To address these vulnerabilities, DOE is now undertaking a Programmatic Environmental Impact Statement (PEIS) (3) whose Record-of-Decision (ROD) is due out in June of 1995. Following close on the heels of this PEIS will be site-specific Environmental Impact Statements (EIS's) which will address the long-term management and disposition options specific to the inventories of fuel on each DOE site. These EIS's will identify the preferred options for stabilizing and treating each category of DOE spent fuel in a manner that will render the spent fuel effectively passive against future environmental threats and interactions (4). In some cases, this will mean simply doing nothing. In other cases, however, quite a lot will have to be done.
By far the largest and certainly the most unstable inventory of DOE spent fuel is the 2100 tons of spent fuel left over from the operations of the N-reactor at the Hanford site, now currently stored at the Hanford K-basins. This fuel would have been destined for reprocessing at the Purex plant, had the plant not been shut down in 1992. This fuel is an unusual, non-"standard" form of fuel, in uranium metal form (as opposed to the standard uranium oxide form), highly damaged and corroded, and requiring extensive chemical conditioning to render it stable for permanent disposal.
Knowing of this problem, the operators of the Hanford Plant contacted the only remaining operators of reprocessing facilities -- BNFL in England and Cogema in France -- to determine whether our respective facilities could handle this type of fuel.
It turns out that BNFL has extensive and very relevant experience in handling Magnox fuel from England's first generation of gas reactors, known as Magnox. This fuel is also metallic, and BNFL's policy has always been to store it for a minimal amount of time (less than 3 years), under water, until such time as reprocessing is undertaken. The reprocessing is done as much to stabilize the chemical form of the fuel for final disposal as it is for any recycle value attached to the fuel.
BNFL examined the situation with respect to the N-reactor fuel, and came to several conclusions:
1. The fuel was much more deteriorated than most of the fuel BNFL had ever handled, except perhaps for an inventory of highly damaged fuel from an Italian reactor customer, which BNFL successfully transferred into its own shipping casks, shipped, stored, and reprocessed. This experience demonstrates that BNFL can manage and process severely damaged metal fuel.
2. The severity of the damage to the N-reactor fuel is attributed to the excessive length of time the fuel had been stored at the K-basins (as long as 17 years), the lack of water chemistry control in the basins, and ironically, the oxygen deprivation and excessive hydriding associated with the containerized fuel stored in the K-West Basin. Some experts say the fuel is more accurately described as rubble, debris, and sludge, than it is spent fuel.
3. BNFL could indeed handle the fuel as well as the basin cleanup, all the way from providing the necessary number of Magnox shipping casks, supervising the loading of these casks, arranging for shipment by rail or truck to a sea terminal, where the fuel would be transferred to one of BNFL's dedicated ships for transport to BFL's storage and reprocessing facilities at its Sellafield site. With minor front-end modifications to either BNFL's Magnox reprocessing plant or its newly-commissioned THORP plant, the fuel could be safely reprocessed to render the material suitable for permanent disposal. Meanwhile, BNFL's success in cleaning up highly contaminated, sludge-filled spent fuel storage pools at Sellafield could be directly applied to the challenges of cleaning up the K-basins.
The operators and managers at the Hanford site, as well as technical program managers at DOE-Headquarters have shown obvious interest and enthusiasm in BNFL's technical capabilities. They were particularly impressed with the fact that BNFL could have all the fuel removed from the K-basins by the year 2002, which is DOE's commitment under the Hanford Tri-Party Agreement. Furthermore, all activities would be accomplished in compliance with applicable US, International, and UK regulatory and safety standards.
Sadly, those same DOE representatives became extremely pessimistic when it came to discussing actually putting the BNFL option into practice. This pessimism derived

from the perceived incompatibility of this option with current US nonproliferation policy objectives. For despite the fact that this was an extremely unique problem requiring a unique solution, and despite the fact that the reprocessing was to be undertaken for reasons totally unrelated to separation of plutonium from the fuel, and undertaken by the United States' closest and most trusted ally, these DOE representatives predicted that BNFL option simply would not pass the Clinton Administration nonproliferation policy litmus test.

ADAPTING BNFL TECHNOLOGY TO US POLICY OBJECTIVES

With that impasse in mind, BNFL experts set out to determine what, if anything, BNFL could do to meet the requirements of US policy of not producing any more separated weapons plutonium. After all, any successful provider of services must always be sensitive to customer needs, and BNFL is certainly no exception.

After some fairly rigorous examination of technology options available to BNFL, the following options were identified and presented to DOE representatives last May:

1. The chemical separation process could be interrupted just prior to separation of the uranium from the plutonium. The still-combined stream of uranium and plutonium would then be diverted to a nearby vitrification plant where, if proven to be feasible, the entire inventory would be vitrified in HLW glass logs suitable for permanent geological disposal. The intent here was to avoid separation of plutonium from uranium.

Due to the massive proportion of uranium as compared to plutonium in the unseparated stream, the practical result of this option would have turned out to be the generation of no less than 25,000 canisters of HLW, an amount that BNFL Transport Division would be more than happy to transport back to the United States. By contract, this compares to the normal return from such a quantity of spent fuel of this type of 450 canisters of HLW. Thus, this approach was deemed impractical, precisely because the large amounts of uranium remaining in a relatively minor amount of plutonium translated into a massive waste problem.

This difficulty lead BNFL to the conclusion that separation of uranium from the plutonium would be necessary from a practical waste-management standpoint -- not to separate out the plutonium, BUT TO SEPARATE THE MASSIVE AMOUNT OF URANIUM contained in the mixture (99% by volume).

2. Once having established the practical need to separate out the uranium in the process, the next challenge was how to deal with the plutonium stream in a manner not inconsistent with US policy objectives. So, for ideas, BNFL looked at what methods were under consideration in the US for disposing of OTHER inventories of excess weapons plutonium (the additional amount of Pu from N-reactor fuel represents only 6% of the total declared excess plutonium inventories). The National Academy of Sciences report on "Management and Disposition of Excess Weapons Plutonium" (5) recommended two preferred options, namely burning the plutonium in nuclear reactors as mixed oxide (MOX) fuel, and/or vitrifying the plutonium in HLW glass logs. These two options are the frontrunners in an ongoing PEIS on Disposition of Excess Fissile Materials, expected to be completed in the summer of 1996.

With respect to the first option of burning the plutonium as MOX fuel, there is no question that BNFL could treat the N-reactor plutonium in this manner, having done so for the rest of its customers, using a proven technology. In fact, with startup of BNFL's new Sellafield MOX Plant (SMP) in 1997, the proximity of the SMP to THORP is such that plutonium will never leave the co-located buildings. In essence, BNFL will be able to receive the N-reactor fuel at one end of THORP, reprocess it, and provide MOX fuel as the finished product out the other end. The resultant MOX fuel would have the equivalent energy of 10 million tons of coal.

With respect to the second option of putting plutonium into HLW vitrified glass, there is little or no experience worldwide with this option. In theory, adding the plutonium to the HLW glass mixture is feasible, providing that it is added in limited quantities (less than 5% by concentration) to preclude any deleterious effects on the stability of the glass form. The fact is, BNFL has no experience to date putting plutonium into glass precisely because the plutonium is purposely separated out for reuse. BNFL's HLW glass has negligible quantities of plutonium, by design. This is not to say that BNFL could not perform lab-scale tests to determine the technical feasibility of incorporating plutonium into HLW glass, and would be willing to do so, if the customer so desired. If the process proved feasible, this option would result in the production of some 2000 canisters of HLW glass, an amount substantially above the nominal 450 canisters for pure HLW glass, but still a

manageable amount to ship and dispose of.
BNFL believes that its options for stabilizing the N-reactor fuel via reprocessing, and its ability to manage the plutonium byproduct in a manner consistent with DOE's plans for disposing of other inventories of excess weapons plutonium, coupled with BNFL's flawless nonproliferation credentials and experience second-to-none in handling this type of fuel make it the only practical, feasible, and cost-effective solution. Nonproliferation concerns having been satisfied, the environmental economic consequences of NOT using the BNFL option are so dire as to defy logic.
THE DOE 'PATH FORWARD' PROPOSAL FOR INTERIM MANAGEMENT OF N-REACTOR FUEL
Putting logic aside, however, it appears that DOE has announced an interim "Path Forward" proposal (6) to remove and temporarily store the fuel on-site, at an estimated cost of $1.15 billion. Approximately half that cost is associated with the cleanup and decommissioning of the K-basins, which must be accomplished in any event. But the other half -- or roughly $600 million -- will be spent on a 40-year "interim" on-site storage facility for the spent fuel. The concept involves removing the fuel to multi-canister overpacks (MCO's), which would store the fuel wet until such time as a method can be developed and a facility can be built to convert the wet-stored material to dry storage, utilizing the same MCO's.
It is important to note that DOE has committed to meeting an NRC-equivalent safety envelope for the fuel drying process and the dry storage facility, even though the exact process for drying the fuel, rubble, and debris has yet to be developed or proven feasible, and even though NRC safety standards for drying and storing metal fuels don't exist.
According to DOE cost estimates, the as-yet-to-be designed "fuel stabilization facility" for drying the fuel and debris will cost $133 million in capital costs, and another $115 millon in operating expenses. These figures should be taken for what they really are -- cost guesstimates. All this for a facility that will not move the fuel one step closer to real, chemical stabilization that must eventually be undertaken no matter what. all this to "stabilize" fuel that is already so badly deteriorated that it is much better described as rubble, debris and sludge than it is fuel. All this to put off for 40 years onto someone else's watch a final decision that could be made today, but for an unfortunate misapplication of US policy. The really unfortunate aspect of this whole "Path Forward" option is that, after the 40 years of interim storage the facilities, the institutions, and the expertise to treat this fuel for permanent disposal may no longer exist. Meanwhile $600 million will have been wasted that could have otherwise been applied to a direct, technologically proven, final disposition route.
A CRYSTAL BALL FOR THE FATE OF N-REACTOR FUEL
BNFL has demonstrated flexibility, innovation, and sensitivity in developing options for DOE to effectively treat N-reactor fuel within the confines of US nonproliferation objectives. DOE's selection of a BNFL option will depend on political developments and NEPA analyses that have yet to be completed. BNFL will continue to provide technical expertise and experience in handling metal fuels as these deliberations unfold.
BNFL CONTRIBUTIONS TO OTHER US SPENT FUEL AND FISSILE MATERIALS MANAGEMENT PROBLEMS
If one acknowledges that BNFL technological solutions can be adapted to satisfy US nonproliferation objectives, as discussed above, then there are several other pressing but unique problems in the US to which BNFL can offer solutions. These solutions only need the political will and a "common sense" approach to become reality.
Briefly, here is a sampling of some other unique US spent fuel challenges begging for relief that are potential candidates for overseas solutions:
Foreign Research Reactor Spent Fuel
Under an Urgent Relief Environmental Assessment (7) DOE recently resumed its program to take back foreign research reactor spent fuel under its Reduced Enrichment of Research Test Reactors (RERTR) program. A formal EIS is presently underway to address the return of the remaining inventories of RERTR fuel under this program. There are approximately 15,000 subassemblies to be returned over the next 10 years. The original intent of the RERTR Program was to encourage operators of these research reactors to convert from high-enrichment cores, to low-enrichment, thereby reducing any proliferation concerns attached to the continued operation of these reactors.
The governor's office in South Carolina, where the fuel is to be shipped for

storage, has objected to his state becoming a storage site for all this RERTR spent fuel, and filed a lawsuit against DOE, the final outcome of which is yet to be determined.

A very practical, proliferation-proof solution to this dilemma that would still enable DOE to meet its policy objectives under the RERTR Program is under consideration in the full EIS. That solution involves reprocessing in England (Dounreay, or, assuming modifications could be made, at Sellafield), return of the high-enriched uranium, under full-scope safeguards, to the US where it could be blended down for return to the foreign research reactors as low-enriched fuel.

North Korean Spent Fuel

The Agreed Framework between North Korea and the United States includes a cessation in operation of the Democratic Peoples Republic of Korea (DLRK) reactor, short-term interim management of the fuel onsight, and eventual removal of the fuel from North Korea. This fuel is a metal fuel not unlike K-reactor fuel or Magnox fuel, and DOE officials have already acknowledged the fact that this fuel is unstable and will eventually require reprocessing. Again, BNFL could provide the casks, the ships, the reprocessing, and the return of plutonium to the US or a US-designated destination in a form unsuitable for weapons production (i.e., as MOX fuel or in HLW vitrified glass). DOE has already contacted BNFL to discuss these services, despite the fact that it is unwilling to move that same direction yet with respect to its own N-reactor fuel.

West Valley Fuel

DOE retained title to a small inventory (125 subassemblies) of commercial fuel at West Valley in New York, largely due to the fact that the fuel was damaged and could not easily be accepted back in the fuel ponds of its original owners. DOE is in the process of decommissioning the entire West Valley site, and had planned to ship the West Valley Fuel to the Idaho National Engineering Laboratory, up until the State of Idaho filed a lawsuit objecting to any future receipt of out-of-state fuel. The DOE has an agreement with the State of New York to have that fuel removed by the end of 1995 or otherwise be subject to a $600,000 per year penalty.

At DOE's request, BNFL has briefed the department concerning BNFL's capabilities to remove the fuel for reprocessing overseas. The same tailor-made management that would be applied to N-reactor fuel would be applied here, that is, return of the plutonium to the US in a non-weapons useable form.

The DOE Obligation to Accept Commercial Spent Fuel by 1998

Even though DOE denies the fact that it has a legal obligation to accept spent fuel from commercial nuclear utilities beginning in 1998 (8), there are a handful of utilities who are running out of on-site spent fuel storage capacity and are precluded for one reason or another from building additional on-site storage or shipping to another site. Several utilities face premature shutdown if DOE cannot take their fuel by 1998.

DOE has been unsuccessful in identifying, let alone being able to provide, any facilities for interim federal storage pending completion of a permanent geologic disposal facility. Even the proposed privatized Mescalero Monitored Retrievable Storage (MRS) Facility will not be completed until 2002, if it proceeds at all. BNFL could fill the gap for this limited quantity of "orphan fuel" until such time as DOE or the Mescalero develop an MRS, so as to enable those nuclear plants with limited on-site storage capacity to avoid premature shutdown.

Excess Weapons Plutonium Disposition

Although not strictly in the category of spent fuel, excess weapons plutonium disposition is a challenge that demands immediate resolution, especially with respect to the quantities of material in the Former Soviet Union. Western Europe has available, proven technologies and facilities, and operating history, that is directly applicable to the burning of this excess weapons material as MOX fuel. Western Europe also has industrial-scale experience with vitrification of HLW which, while not quite so directly relevant to excess weapons plutonium disposition, could be adapted to that goal if the US so desired.

CONCLUSION

Policy decisions that the US Government makes with respect to its domestic spent fuel management and disposal programs are certainly legitimate and of no consequences to other countries and their own, individual fuel cycle policies. However, there are certain unique inventories of US spent fuel that require alternative approaches.

BNFL has taken into account the sensitivities of US nonproliferation policies and unique technology problems to come up with solutions tailor-made to the "customer." It is now up to the "customer", the US Government, to acknowledge the legitimacy and practicality of these solutions in meeting US policy and program objectives.

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TECHNOLOGY INTEGRATION PLAN

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ABSTRACT

In 1992, the Secretary of Energy directed the Assistant Secretary for Environmental Management (EM) to develop an integrated, long-term, spent nuclear fuel (SNF) management program. In response, EM created the Integrated SNF Program to assess the U. S. Department of Energy (DOE) SNF and SNF storage facilities. As shown in Fig. 1 the Integrated SNF Program is responsible for life-cycle management of DOE SNF; that is characterization, processing, interim storage and preparation for disposal. In order to implement the Program, it was recognized that technology needs must be identified. A Technology Integration Program was formed to integrate the DOE complex-wide efforts for establishing timely, cost effective and consistent technical criteria for the development of technical solutions. The program is directed toward identification of: a) what activities need to be done, b) when they need to be completed, and c) what priority should be assigned to the various activities.

Fig. 1.

SCOPE OF THE PROBLEM

The DOE is responsible for approximately 2,700 metric tons of heavy metals (MTHM) SNF. In general the SNF is divided into five principle generation sources: Materials Production Fuels, Naval Nuclear Propulsion, Research Reactor Fuels, Specialty Fuels, and Special Case Commercial Nuclear Power Reactor Fuels. There are currently over 200,000 units (rods, elements, pieces, baskets, buckets, cans, etc.) of DOE SNF in storage at 29 different DOE facilities, universities, and private institutions in the United States. By identifiable characteristics, DOE SNF can be grouped into over 150 distinct fuels. For the purposes of the TIP the DOE-owned SNF is divided into 53 categories. These categories served as the starting point to determine the program technology needs.

The diverse inventory and its multiple locations complicate the development of an optimum SNF management plan driving the need to integrate technology development. In addition to such factors as enrichment and volume the SNF diversity, its physical integrity, and ES&H vulnerabilities all have a significant influence in determining the initial scope of the technology development program. Future increases when measured in terms of total fuel mass will increase the SNF inventory by 70%.

TECHNOLOGY INTEGRATION PLAN

The technology integration effort must be continuous, staying current with accomplishments and analysis of options. A DOE Spent Fuel Nuclear Fuel Technology Plan (TIP)* has been written containing extensive, preliminary data that is useful

in supporting some present and future planning activities. To be effective the plan must maintain program flexibility to minimize not only near-term expenditures, but life-cycle cost and environmental, safety, and health risks. The initial result from this effort indicate a clear need to continue and expand the in-depth analysis of the this effort. Technology development as defined in the TIP is "the scientific and engineering investigations required to ascertain basic principles of a process or concept."

The TIP contains technical input from the various DOE laboratories which are integrated with the programmatic goals and management philosophy identified in the new technology development plan for the DOE Office of Technology Development (EM-50). This comprehensive document focuses on major cleanup obstacles and presents a balanced set of plans are being developed to integrate all EM environmental research and technology development. The TIP will be a dynamic partner in this technology development process.

SYSTEMS APPROACH

Because of the distinct characteristics of the SNF fuels, preparation for interim storage and subsequent disposal should not be expected to follow the relatively simple pathway planned for commercial reactor SNF disposition. The selection of a management option must consider specific fuel type characteristics such as physical condition, chemical reactivity, and criticality safety issues. Some of the factors that could influence the selection of an SNF disposal technology for further development are:

- Can directly
- Dilute High Enriched Uranium with depleted uranium
- Shred, mix with glass in a geometrically safe can
- Reprocess, calcine waste, convert to glass-ceramic
- Cut; package in safe geometry
- Recycle metal/waste minimization
- Preparation of single waste form
- Dissolve, glassify, pour in commercial fuel can

In some cases, the evaluation of the proper SNF option for a particular fuel may lead to the conclusion that the SNF requires repackaging, or stabilization. Policy decision, economics and public acceptance must also be factored in the selection of the disposal pathway.

Figure 2 is a summation of the cost for identified tasks (funded and unfunded) through fiscal year 1998 (in 1995$). Potential for integration of tasks exists, but requires additional, detailed review to determine if all programmatic goals at each site can be accomplished in a timely manner after integration strategy is implemented.

Fig. 2.

In any event what is done today to manage the existing conditions and stabilize the SNF for interim storage must consider what the impact of these actions will be on the final SNF disposal. If direct disposal is possible, it would seem prudent to store robust SNF e.g., naval fuel, without further processing and, for the SNF that requires processing; process only once for both interim storage and disposal.

EXISTING STORAGE

In general most of the near-term activities are directed toward a stabilized storage posture for the DOE SNF. The technology development tasks in this area include activities related to SNF corrosion control, sludge characterization, characterization of degraded fuel, and facility conditions such as storage basin liquid leakage control. Obviously, these activities must receive priority since most of the activities are responding to vulnerabilities identified by the DOE Spent Fuel Working Group Report.* While much of the work is nonroutine and challenges the capability of the DOE laboratory complex most of the technology needed to accomplish these activities is in place. Innovative applications will be needed to resolve the existing storage vulnerabilities.

INTERIM STORAGE

Typical interim storage related activities are directed toward transitioning SNF into storage conditions that are adequate until a permanent repository is in place. Because most of the current storage practices, fuel designs, and facilities are not well suited for extended storage, many of the technology development tasks are associated with transition to dry storage, and repackaging of SNF, etc. Some level of fuel characterization will be required for interim storage. If, indeed the goal

is to stabilize the SNF such that the final fuel form may go directly to the permanent repository characterization may become critical from the standpoint of final performance form.

DISPOSAL

In general, there are two basic processes for preparing SNF for disposal; direct canning and chemical processing of the SNF to achieve an acceptable waste form. The first disposal option i.e., direct canning is straight forward although some development is needed in this area. Most of the SNF, currently stored in a wet environment must be dried before disposal. It may be difficult to dry the fuel sufficiently to prevent radiolytic and chemical production of hydrogen or fuel degradation over long periods of time. Some development work, although more of an application of existing technology, must be applied to canister configuration. The multi-purpose canister (MPC), for instance would allow near-term onsite storage, transportation, interim storage, and disposal without further handling of the SNF. Fuel characterization needs, criteria, and methodologies need to be developed foe SNF direct canning.

The second major area of development addresses the technologies needed to stabilize the SNF for interim storage. Existing chemical processing, although discontinued at this time, could assist with DOE SNF disposal. Aqueous processing is generally available. Solvent extraction processes for separation of fissile uranium and plutonium from aqueous high level waste are well developed and have been used for decades.

New processing technologies that may combine stabilization for interim storage and disposal preparation offer the additional advantage of separation of non-radioactive constituents into low level waste. These new technologies; Chloride Volatility Process, Electro Chemical Treatment, Glass Material Oxidation and Dissolution System, and various enhanced extraction technologies are in various stages of development.

EVALUATION OF TECHNOLOGY NEEDS

The TIP discusses the methodology for identifying the scope of the technology development needs for the DOE SNF and its various management options. This methodology resulted in an overview matrix that scoped individual technology development activities for each of the 53 SNF categories. There are 980 potential technology needs identified. Because of the magnitude of the potential problem each site was asked to present the SNF technology development programs (funded and unfunded) that they believed were justified by these SNF management issues. Technology development tasks that had a potential interface with other related programs were noted and discussed.

To help ensure that consistency and completeness were incorporated, the 49 funded tasks were compared to the potential technology needs to independently verify its need/justification. The funded tasks were also related to SNF categories to ensure that support was focused on problem SNF categories with issues that need early resolution. This comparative review showed good correlation between the tasks identified and the needs of the overview matrix.

As expected, the majority of tasks, Fig. 3 fall under interim storage. Perusal of the information shows little duplication in efforts for the wet storage, mechanical disassembly of fuels, properties of fuel for dry storage environment, and new technologies for disposal categories. However, there appears to be potential duplication of activities in both SNF characterization for dry storage and dry storage demonstrations.

Fig. 3.

CONCLUSIONS

The technology needs and development tasks presented in the TIP reflect roles and responsibilities that cross-cut DOE's organizational structure. Inputs, recommendations, and issues identified by these organizations were considered in developing the TIP. Additionally, in identifying the technology development needs specific to DOE SNF, consideration was given to integrating those efforts managed by organizations outside of EM-37. Some of the interfacing organizations are EM-50, RW, and DOE operations offices.

An SNF TIP has only been initially scoped. Although this effort has been greatly advanced, not all tasks required for a fully integrated plan are identified. The completeness of the data set does not allow a full picture of the ongoing technology development, and the needed development to be thoroughly assembled.

An overall time line, assessing when technologies will be needed, is difficult to derive. Such a tool is highly dependent upon assumptions such as repository opening date, acceptance criteria for DOE SNF, inventories to be emplaced in the first repository, long-term nonproliferation criticality issues for fissile material, the end of life of existing storage facilities, and possible advances that would allow SNF to be stabilized in existing storage configurations.

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A PATH FORWARD FOR THE GEOLOGIC DISPOSAL OF DOE-OWNED SPENT NUCLEAR FUEL

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ABSTRACT

The Department of Energy (DOE) Office of Spent Fuel Management has undertaken a program to ensure the safety of existing storage, provide for interim storage and prepare for ultimate disposition of DOE-owned spent nuclear fuel (SNF). DOE is responsible for managing approximately 2,700 metric tons heavy metal (MTHM) of DOE-owned SNF that is currently stored at DOE facilities. In the past, most DOE-owned SNF has been chemically processed to recover plutonium or uranium for the nuclear weapons program. However, with the phaseout of chemical processing throughout the DOE Complex, DOE-owned SNF must now be managed until its ultimate disposition, rather than for the few years until reprocessing, as originally intended.

Of the options available for ultimate disposition of DOE-owned SNF, geologic disposal is the primary option under consideration. There is a long history and numerous challenges associated with the planning for geologic disposal of SNF and high level nuclear waste (HLW). DOE faces additional challenges in its qualification of DOE-owned SNF for geologic disposal. In addition, institutional issues have recently surfaced with the introduction of additional material not originally considered in the repository program planning. Public stakeholders have expressed a keen interest both in the DOE-owned SNF program and the implications of its disposal on the repository program. DOE must also meet technical challenges to develop a coherent and cost-effective program for geologic disposal of the diverse DOE-owned SNF inventory, which in some cases is quite different than civilian SNF.

A number of efforts are underway to address the challenges associated with geologic disposal of DOE-owned SNF. The DOE Office of Spent Fuel Management is developing a memorandum that evaluates various options for introducing DOE-owned SNF into the planning for all other radioactive materials slated for repository disposal, identifies key issues and sensitivities, and proposes a path forward for geologic disposal of DOE-owned SNF. Another key effort is the formation of a joint Steering Group, consisting of the DOE Offices of Civilian Radioactive Waste Management (OCRWM) and Environmental Management (EM), to coordinate efforts in the geologic disposal of DOE-owned SNF.

BACKGROUND

The Department of Energy (DOE) is in the midst of a transition in its management of spent nuclear fuel (SNF) inventories. In the past, the liquid HLW resulting from the reprocessing of DOE-owned SNF was converted into a vitrified waste form for ultimate disposal in a geologic repository. Fissile material from the reprocessing of DOE-owned SNF was recovered for use in the nuclear weapons stockpile. However, the Secretary of Energy, in April of 1992, directed a phaseout of remaining reprocessing operations in the DOE complex, leaving an inventory of SNF to be managed along with the existing inventory of liquid HLW. Essential to the success of this transition in DOE-owned SNF management will be a new strategy to achieve ultimate disposition.

During the last four decades prior to 1992, DOE and its predecessor agencies reprocessed approximately 100,000 Metric Tons of Heavy Metal (MTHM) of SNF from various sources, including: DOE production reactors; the Naval Nuclear Propulsion Program; DOE, university, and other research and test reactors; special-case commercial power reactors; and certain foreign research reactors. Approximately

2,700 MTHM of SNF is what remains of the DOE-owned SNF inventory, and about 100 MTHM of additional SNF is projected to be received by DOE in the next 40 years.
The DOE Office of Spent Fuel Management was created in 1993 to develop a program to ensure the safe existing storage, provide for interim storage and prepare for ultimate disposition of DOE-owned SNF. Also in 1993, the DOE Office of Environment, Safety and Health issued its Spent Fuel Working Group Report. In this report, a number of vulnerabilities were identified regarding the current DOE-owned SNF inventory. One generic vulnerability identified was that, unlike HLW, there was no path forward for ultimate disposition of DOE-owned SNF.
The DOE now has a wide range of options available for the ultimate disposition of its SNF, and discussions have begun to determine if DOE-owned SNF will be disposed of in a geologic repository, along with civilian SNF. It is possible that some DOE-owned SNF will be processed to ensure efficient waste management. However, this paper focuses on the ultimate disposition of the DOE-owned SNF that remains in inventory at the time a geologic repository becomes available.
There is an established program and a long history associated with the planning for disposal of SNF and vitrified HLW. In 1955, the Atomic Energy Commission, a predecessor agency to DOE, requested input from the National Academy of Sciences (NAS) on the disposal of radioactive waste. In 1957, the NAS recommended that the best method of disposal would be placement in a deep geologic repository. Numerous studies were undertaken over the next 25 years to identify suitable geologic media for a SNF and HLW repository.
In a 1980 environmental impact statement (EIS), DOE evaluated a number of methods for permanent disposal of high-level radioactive waste and concluded that deep geologic disposal was the best method among the alternatives considered. In 1982, Congress passed the Nuclear Waste Policy Act (NWPA). All efforts to establish a repository for SNF and HLW were brought together under one Federal program: DOE's Office of Civilian Radioactive Waste Management (OCRWM). In 1983, DOE selected nine locations in six states for consideration as potential sites. Based on studies completed at these potential sites, the President approved three sites for intense scientific study, or characterization. However, in 1987 Congress amended the NWPA and directed DOE to study only one of the sites - Yucca Mountain in the state of Nevada.
At the present time, site characterization and engineering activities are being conducted to determine the suitability of Yucca Mountain as a repository site. A DOE position on technical site suitability is expected in 1998, and a final EIS on the development of a geologic repository is expected to be completed in the year 2000. This will support the DOE decision on site recommendation that will accompany a Site Recommendation Report to the President, as required by the NWPA. If the Yucca Mountain site is found to be suitable, and a site recommendation is made and approved by the President and Congress, DOE will submit a license application to the Nuclear Regulatory Commission (NRC). DOE's current schedule calls for the license application for construction authorization to be submitted by DOE to the NRC in 2001, construction to begin in 2004, and waste to be emplaced starting in 2010.
ISSUES
As DOE considers ultimate disposition of DOE-owned SNF in a geologic repository, a number of issues and constraints have become apparent. These issues and constraints lie in three general areas: OCRWM programmatic issues, institutional issues and technical issues.
Program Issues
The path forward developed by DOE for geologic disposal of its SNF inventory will need to take into account the existing constraints upon the OCRWM and EM programs. For example, the success of a path forward for geologic disposal of DOE-owned SNF is directly linked to the success of OCRWM's efforts to site and construct a geologic repository. Still more uncertainty is associated with the siting and construction of a second repository making it even more questionable as a viable alternative for ultimate disposal of DOE-owned SNF.
Even if OCRWM's efforts to site and construct a repository are successful, important programmatic issues exist regarding the space limitations for the first repository. Under the provisions of the NWPA, DOE is prohibited from emplacing more than 70,000 metric tons heavy metal (MTHM) in the first repository until a second repository is in operation. It is also possible that, upon completion of site characterization for the first geologic repository, a determination could be made that the capacity of

the repository is less than 70,000 MTHM. However, projections of waste inventories for geologic disposal are greater than 70,000 MTHM. Inventories of civilian SNF alone are expected to exceed 86,000 MTHM by the year 2030 (no new orders case). The current planning basis for the first repository anticipates the allocation of 10 percent of the available repository capacity for DOE vitrified HLW, with the first quantities of such waste planned for disposal in 2015. Inventories of DOE-owned spent nuclear fuel are anticipated to be approximately 2,800 metric tons heavy metal by the year 2035. While there is currently no consensus on a method of assigning metric ton heavy metal equivalence to high-level waste, it is anticipated that the total inventory of high-level waste alone will exceed the 10% OCRWM planning allocation. Assuming that 7,000 MTHM is available for DOE-owned SNF and HLW, there would likely not be enough space to accommodate all DOE-owned SNF and HLW without an expansion of the capacity of the first repository, or the opening of another repository. The current limitation on the first repository capacity also places importance on identifying the relative priority for disposal of DOE-owned SNF and HLW. As part of this process, a change in OCRWM's planning assumption would need to be made to accommodate a DOE decision to use all, or part of the 10% allocation to emplace DOE-owned SNF.

There are also program issues concerning the current schedule for the first repository. Currently, OCRWM is only authorized to investigate one repository and is not due to report to Congress on the need for a second repository until 2007-2010. For some of the DOE-owned SNF, it would be difficult to obtain the necessary data and analysis in time to be included in the license application for the first repository, whereas some DOE-owned SNF may be immediately available for repository emplacement.

Related to the space and schedule issues for the first repository is the need for new interim storage for some DOE-owned SNF. Many existing facilities for storage of DOE-owned SNF are old, and have environmental, safety and health vulnerabilities. In addition, the condition of some DOE-owned SNF is severely degraded and some may be unstable for handling. Therefore, DOE will have to make changes to its existing SNF management infrastructure in preparation for its ultimate disposition. In addition, there are numerous stakeholder concerns regarding interim storage of DOE-owned SNF that will have an important influence on determining the path forward for geologic disposal of DOE-owned SNF.

Institutional Issues

Resolving stakeholder concerns is a key element of the institutional issues regarding the geologic disposal of DOE-owned SNF. As part of the DOE effort to establish increased responsiveness to stakeholder concerns, the Office of Spent Fuel Management has spent significant resources and time interacting with stakeholders both through participation in the development of a number of environmental documents and through other forums for participation in the DOE-owned SNF program. A chief stakeholder concern, voiced at the recent public hearings for the Programmaic Spent Fuel Management draft Environmental Impact Statement is that storage of DOE-owned SNF at DOE sites, without a plan for ultimate disposition, may become de facto disposal of this material.

There are many other potential stakeholder concerns related to considering DOE-owned SNF for inclusion in the OCRWM program. The nuclear utilities have been dissatisfied with the progress made towards disposal of civilian SNF. Geologic disposal of DOE-owned SNF should be considered in such a way as to eliminate, or minimize, any impact on the repository program and disposal of civilian SNF. As part of its general opposition to the repository program, the candidate host State could oppose a path forward for repository disposal of DOE-owned SNF. This opposition may be further fueled by the perception that some DOE-owned spent nuclear fuel is "foreign waste". However, some stakeholders may favor a proposal for geologic disposal of foreign research reactor SNF, because it can assist in the accomplishment of the U.S. nuclear non-proliferation objective of removing U.S.-origin highly-enriched uranium from international commerce. In addition, foreign research reactor SNF would be only a very small percentage of the total SNF and HLW planned for repository disposal (less than one-twentieth of one percent).

Another institutional issue raised to date centered around the statutory authority for disposal of DOE-owned SNF in a geologic repository. Section 8 of the NWPA required the President to evaluate whether a separate repository for defense high-level waste would be required, or if defense HLW should be placed into a

civilian repository along with the SNF from civilian reactors. In 1985, the President approved the recommendation of the Secretary of Energy that defense waste should be co-located with civilian waste in a geologic repository. However, this recommendation did not explicitly address the ultimate disposition for DOE-owned SNF. Until 1992, the ultimate disposition of DOE-owned SNF was not an issue, because most was reprocessed. In November 1993, the Assistant Secretary for Environmental Management requested a legal opinion regarding the statutory authority for geologic disposal of DOE-owned SNF. In March 1994, DOE's General Counsel concluded that there is discretionary authority in Section 302(b)(4) of the NWPA for disposal of SNF by the U.S. government, conditioned upon execution of an intra-agency agreement and upon payment of a fee adequate to cover the full cost of disposal.

Technical Issues

As discussed previously, technical challenges are presented by the characterization needs for the DOE-owned SNF, diversity of DOE-owned SNF types, and lack of waste acceptance infrastructure. Characterization of DOE-owned SNF is a technical challenge that must be met in the near future. All characterization must be conducted under an approved quality assurance program based on NRC and OCRWM requirements. Much DOE-owned SNF has undergone degradation, and some may be unstable in handling, making characterization difficult. In addition, characterization facilities for DOE-owned SNF are few. Also, as most DOE-owned SNF was slated for reprocessing, much of the DOE-owned SNF inventory does not have traceability comparable to that of civilian SNF.

A characterization data base would be used to evaluate each fuel type to determine the extent of conditioning, if any, necessary to meet repository waste acceptance criteria (WAC). This data base would also be used to design engineered barriers and model long-term performance to ensure that Nuclear Regulatory Commission licensing requirements could be met. Some types of fuel would likely require substantial conditioning to meet repository WAC, requiring the development of new treatment facilities, and possibly new technologies. Other types of DOE-owned spent nuclear fuel, similar to civilian SNF, would likely meet repository WAC without further conditioning.

DOE-owned SNF consists of more than 90 types, with various enrichments, cladding, and fuel matrices. The numerous DOE-owned SNF types may appear to represent a challenge that would be very difficult to meet. However, commercial-type SNF represents 6.5%, and Naval Nuclear Propulsion Program SNF represents another 2%, of the DOE-owned SNF inventory. Both these DOE-owned SNF types should qualify for geologic disposal in a manner consistent with civilian SNF. Approximately 80% of DOE-owned SNF is Hanford N-Reactor SNF, which may have to be conditioned before geologic disposal. Thus, satisfactory qualification of three fuel types for geologic disposal would qualify approximately 90% of the DOE-owned SNF inventory for geologic disposal. The balance of the DOE-owned SNF inventory contains many fuel types that may need to be extensively conditioned, or even reprocessed, to resolve safety and environmental concerns.

Lack of a waste acceptance infrastructure, i.e. the resources necessary to determine the requirements for geologic disposal of DOE-owned SNF and how these requirements will be met, is another technical challenge. At present there is no approved waste acceptance process and no set of WAC for DOE-owned SNF. Requirements to guide qualification of DOE-owned SNF for geologic disposal can be found in the Code of Federal Regulations, 10 CFR 60 and 40 CFR 191 (currently under revision), and in the OCRWM Waste Acceptance Systems Requirements Document (WASRD). However, none of these sources are useful for detailed design of waste forms and waste packages. In addition, there are only two standard waste forms in the WASRD, civilian SNF and borosilicate vitrified HLW. Finally, the human resources necessary for ensuring waste acceptance of DOE-owned SNF are not yet in place. Personnel with substantial technical expertise, similar to the Technical Review Group for the Defense Waste Processing Facility, will need to be assembled to resolve this issue.

EFFORTS TO DEFINE A PATH FORWARD FOR GEOLOGIC DISPOSAL OF DOE-OWNED SNF

There are two primary efforts underway to address the challenges associated with geologic disposal of DOE-owned SNF. The first effort is the preparation of a memorandum that evaluates various options for integrating planning for DOE-owned SNF with the planning for all other materials slated for repository disposal, identifies key sensitivities, and proposes a path forward for geologic disposal of DOE-owned SNF. The second effort is the establishment of a joint Steering Group consisting of

the DOE Offices of Civilian Radioactive Waste Management and Environmental Management. This Steering Group will scope out the challenges related to integrating DOE-owned SNF into the repository program and identify near term priorities.
Memorandum on the Path Forward for Geologic Disposal of DOE-owned SNF
As part of the first effort, three planning options are under primary consideration for the ultimate disposition of DOE-owned SNF. Each option assumes emplacement in a geologic repository as the Department's proposed strategy for disposition of any SNF remaining in its inventory when a repository becomes available. No options for the ultimate disposition of DOE-owned SNF, other than geologic disposal, are under primary consideration by DOE at this time. The three options being considered would each integrate the Department's SNF into the program being conducted by OCRWM for the disposal of civilian SNF and establish the framework within which to evaluate DOE-owned SNF against the requirements for qualifying SNF for geologic disposal. All the options being considered are based on the assumption that the OCRWM program will accomplish its current program for constructing, licensing, and operating a geologic repository and that some DOE-owned SNF could be adequately characterized in time to be included in OCRWM's license application to the NRC. All options encompass these goals: assuring safe and environmentally sound existing storage of SNF, providing robust new interim storage facilities, as necessary, that are designed, constructed and operated to national consensus standards; and preparing all DOE-owned SNF for disposal in accordance with one of the three options addressed below.
Option 1. DOE-owned SNF and vitrified HLW would be authorized for disposal in the first repository. The total quantity of DOE-owned SNF and vitrified HLW would not exceed 10 percent of the repository capacity. Disposition of remaining DOE-owned SNF and vitrified HLW would not be decided until the DOE recommendation on the need for a second repository. In addition, under this option all DOE-owned SNF would be aggressively stabilized and prepared for repository disposal.
Emplacement priority for DOE material would be based on a systems approach that considers risk, cost, preparation time, readiness, and other factors. DOE-owned SNF that can be qualified in time may be included in the initial license application if it is of sufficiently high priority. The mix of DOE-owned SNF and vitrified HLW proposed for the first repository could be changed, if appropriate, by submitting a license application amendment after a decision on the need for a second repository. At this time, additional site characterization data and licensing experience could be incorporated into a decision on the disposition of any remaining DOE-owned SNF and/or HLW.
This option would address the lack of a path forward to ultimate disposition, by preparing for the geologic disposal of DOE-owned SNF. The option would also minimize impacts on the OCRWM licensing schedule for the first repository. Concerns over the impact of DOE-owned SNF on the queue for the first repository are minimized since there would be no change in the amount of DOE material proposed for disposal in the first repository. This option also would constitute a significant step towards satisfying stakeholders' concerns over de facto disposal of DOE-owned SNF at storage sites.
However, an amendment to any license application to change the mix of DOE-owned SNF and HLW in the first repository could add complexity to the licensing process for the repository. In addition, to the extent that DOE-owned SNF would displace some vitrified HLW that would otherwise have gone to the repository, storage of such waste would have to be maintained for a longer period. Although this may cause increased concern to stakeholders at the storage sites, almost all sites that store DOE-owned SNF also store HLW. Stakeholders would participate in any process for determining the relative disposal priority between DOE-owned SNF and HLW.
Option 2. DOE would work toward disposal of all of its SNF and vitrified HLW in the first repository, assuming that a request to Congress to remove the 70,000 MTHM limit will be granted.
This option would have all the advantages associated with Option 1. In addition, the focus of the disposal effort would be directed toward the first repository. Resolution of technical, regulatory, and institutional matters would be focused on meeting the requirements for this repository. However, it now seems unlikely that the information needed to include all types of DOE-owned SNF in the initial license application for the repository could be available by the current 2001 submittal date, although the licensing schedule could be delayed. This option would depend on

Congressional action to change the Act's restriction on the repository's capacity, an action that is uncertain. Furthermore, the candidate host state would likely oppose removal of the Act's limitation on the repository's capacity.
Option 3. All DOE-owned SNF would be deferred to a second repository.
This option would allow much of the HLW inventory to be disposed of in the first repository. The DOE Spent Fuel Management program could proceed, unencumbered by concerns related to integrating the program with the first repository activities. Also, any potential licensing or stakeholder issues arising from disposal of DOE-owned SNF in the first repository would be avoided. On the other hand, the option would not resolve the lack of path forward, as effectively as Options 1 and 2. Also, stakeholders would likely regard this option as further evidence that storage of DOE-owned SNF is, in effect, de facto disposal, given the substantial uncertainties associated with the authorization, timing, and availability of a second repository.
As progress is made towards defining the path forward for geologic disposal of DOE-owned SNF, a number of factors will play an important role. The option selected as the path forward for the ultimate disposition for DOE-owned SNF will seek to minimize impacts on the OCRWM's schedule for the first repository, while still achieving disposal objectives for DOE-owned SNF. The option selected will attempt to address stakeholder concerns regarding the program for the first repository. The option selected will also attempt to consider existing constraints of the OCRWM and EM programs. While the option selected may not be able to establish definitive timing for the disposal of DOE's SNF, it will attempt to provide DOE with the most viable option for disposal of DOE-owned SNF and constitute a significant step towards addressing stakeholders' concerns over de facto disposal of DOE-owned SNF at storage sites.
DOE-owned SNF Steering Group
The second primary effort DOE has undertaken to establish a path forward for geologic disposal of DOE-owned SNF is the establishment of a joint DOE-owned SNF Steering Group, consisting of EM and OCRWM, with other DOE offices participating on a consultative basis. The Steering Group will identify and propose resolution for technical, regulatory and programmatic issues related to geologic disposal of DOE-owned SNF and identify near term priorities. This joint Steering Group will also assist in directing DOE complex-wide planning and implementation activities related to preparation of DOE-owned SNF for geologic disposal, as well as develop disposal requirements specific to DOE-owned SNF.
A key function of the Steering Group will be to facilitate integration of the DOE-owned SNF program with the OCRWM repository program. Preparation of DOE-owned SNF for geologic disposal cannot be successful without close coordination between these two organizations. Numerous technical issues will have to be resolved jointly. Waste form specifications will have to be finalized for DOE-owned SNF. Waste package designs will need to be selected and developed. The DOE-owned SNF waste form and package will need to be integrated into the facility engineering for the repository. In addition, a decision will need to be made on whether to implement the OCRWM Multi-Purpose Canister (MPC) concept for DOE-owned SNF. DOE-owned SNF will also have to be integrated into the Performance Assessment for the repository. EM will also have to coordinate with, and possibly provide extensive support, to OCRWM as an EIS, and possibly a license application, for the first repository are developed and possibly submitted in the not too distant future.
Another notable effort of the Steering Group will be the development of an intra-agency agreement between EM and OCRWM to cover the terms of disposal of DOE-owned SNF. It has yet to be determined if existing fee methodologies are appropriate for calculating the disposal fee for DOE-owned SNF. The Code of Federal Regulations, 10 CFR 961, provides the terms for a standard contract for disposal of civilian SNF. There is also a Cost Allocation Methodology established for defense HLW, found in the Federal Register, 52 FR 161. OCRWM and EM, as part of the efforts of a joint Steering Group, have recently initiated discussions regarding an intra-agency agreement.
As the Steering Group addresses technical issues regarding the geologic disposal of DOE-owned SNF, it will draw upon the work of the DOE-owned SNF Technology Development program to help resolve these issues. This Technology Development Program is conducting a preliminary Performance Assessment (PA) to provide an initial evaluation of the performance of a number of waste forms, including a number

of DOE-owned SNF types, in various geologic media. The Technology Development Program is also examining how the DOE might implement an MPC concept for DOE-owned SNF, and potential new conditioning processes for DOE-owned SNF to ensure that repository WAC could be met.

SUMMARY

DOE is making progress towards resolving important issues regarding the ultimate disposition of DOE-owned SNF. There are a number of activities that DOE will undertake as it attempts to implement a strategy for the ultimate disposition of DOE-owned SNF. One of the key activities for the DOE will be to determine its strategy for ultimate disposition of DOE-owned SNF. Regardless of which strategy option is selected, EM will make efforts to integrate its programs for disposal of DOE-owned SNF and HLW and identify the emplacement priority for these materials, in consultation with stakeholders. Mechanisms, such as the DOE-owned SNF Steering Group are in place to resolve outstanding issues regarding geologic disposal of DOE-owned SNF. In summary, EM and OCRWM will work together to ensure that ultimate disposition of DOE-owned SNF is on a clear path forward; that DOE-owned SNF is disposed of in a technically sound manner; and that the candidate host state, nuclear utilities and other interested and/or affected parties are kept informed of progress made and consulted with during each step of the path forward.

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PERFORMANCE ASSESSMENT METHODOLOGY AS APPLIED TO THE GREATER CONFINEMENT DISPOSAL SITE: PRELIMINARY RESULTS OF THE THIRD PERFORMANCE ITERATION*

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ABSTRACT

The U.S. Department of Energy has contracted Sandia National Laboratories to conduct a performance assessment of the Greater Confinement Disposal facility, Nevada. The performance assessment is an iterative process in which transport models are used to prioritize site characterization data collection. Then the data are used to refine the conceptual and performance assessment models. The results of the first two performance assessment iterations indicate that the site is likely to comply with the performance standards under the existing hydrologic conditions. The third performance iteration expands the conceptual model of the existing transport system to include possible future events and incorporates these processes in the performance assessment models. The processes included in the third performance assessment are climate change, bioturbation, plant uptake, erosion, upward advection, human intrusion and subsidence. The work completed to date incorporates the effects of bioturbation, erosion and subsidence in the performance assessment model. Preliminary analyses indicate that the development of relatively deep-rooting plant species at the site, which could occur due to climate change, irrigated farming or subsidence, poses the greatest threat to the site's performance.

SITE DESCRIPTION

The Greater Confinement Disposal (GCD) facility is located at the Nevada Test Site in the Area 5 Radioactive Waste Management site at Frenchman Flat in Nevada (Fig. 1). Frenchman Flat is an alluvium filled basin, with maximum depth to bedrock of more than 450 meters. The site consists of twelve boreholes that are approximately 36.6 meters deep with 3 to 3.6 meter diameters. The disposal procedure consisted of filling the bottom 15.2 meters of the boreholes with waste then filling the remainder of the borehole with native sediments (1). In three of the boreholes the individual waste packages were separated by layers of probertite, a mineral which contains boron, a neutron absorber. The wastes are located within the unsaturated zone in the thick alluvial deposits of the basin, approximately 200 meters above the water table (2).

Fig. 1.

PERFORMANCE ASSESSMENT METHODOLOGY

Four of the twelve GCD boreholes contain transuranic (TRU) wastes and are regulated by 40 CFR Part 191 containment (191.13), individual protection (191.15) and groundwater protection (191.16) requirements (3). The purpose of this work is to provide the Department of Energy with enough information to determine if the site complies with these regulations. The compliance measure for the containment requirements is the calculated EPA sum. Due to the long regulatory periods (10000 years) and the complexity and uncertainty in the transport system, numerical models are required to simulate the release of the radioactive elements to the accessible environment.

The statistician George Box reminds us that no model can reproduce or predict exactly how a complex physical system behaves, but there are models that can calculate the probability of a future value lying between two specified limits (4,5). With the recent interest in the validation of groundwater flow models and the impossibility of that task (6,7) , it is important to recall Box's statement and consider which models will be the most useful in assessing the potential performance of a waste disposal site. The basic premise of this research is that the exact flow and transport system for the next 10,000 years at the GCD site cannot be predicted due to the uncertainty in the existing and future conditions at the site, but that it can be simulated and the maximum release rates bounded.

The performance assessment methodology applied to the GCD site is an iterative process of modeling and data collection (Fig.2). Sensitivity analyses of the performance assessment model are used to identify the most valuable data. In this case the most valuable data are those that minimize the uncertainty in the parameter values that cause the simulated releases to exceed the containment and protection requirements. In the first iteration of the performance assessment, the significant parameters and corresponding data needs were identified (2) and data were collected (8,9,10). The new data were used to reduce the uncertainty in the recharge rate (11), plutonium solubility (9) and plutonium adsorption coefficient (10). Additional modeling was performed using the revised parameter ranges along with refined models of plant uptake and erosion (12). As a result of the second performance assessment iteration, the conceptual model of the transport system was revised to one in which diffusion toward the ground surface provides the fastest pathway to the accessible environment. Due to the changes in the conceptual model, tortuosity and rooting depth became the most significant parameters.

Fig. 2.

The third iteration of the performance assessment involves the investigation of the diffusion pathway. This investigation includes an assessment of the combined effects of subsidence, bioturbation, erosion, plant uptake, upward advection and climate change on the site's waste-containment ability. It is assumed that all of these processes will occur, consequently they are included in the base-case model. In addition to the base case processes, other potentially significant disruptive events must be considered in the performance assessment. Scenario analyses have narrowed the suite of likely, potentially disruptive events to inadvertent human intrusion into the GCD boreholes by drilling and irrigated farming at the GCD site (13).

CONTAINMENT PERFORMANCE ASSESSMENT MODEL

The performance assessment uses a 1-D model of diffusive transport. The modeling is performed in a Monte-Carlo fashion with the uncertain parameter values generated randomly, from predefined probability density functions (pdfs), for each simulation using Latin hypercube sampling. There are a total of 40 uncertain parameters in the model. Details on these parameters and the distributions used to capture the uncertainty in the parameter values can be found in the documentation for the first two performance assessments (2,12). The results of the transport model are post-processed to calculate the EPA sum for each realization and to combine the results into a single complementary cumulative distribution function (CCDF) to assess the probability that the site will exceed the release limits specified in the containment requirements.

The performance assessment model is based on the following major assumptions:

- Transport occurs by liquid phase diffusion.
- The radioactive contaminants are sorbed on sediment surfaces.
- Individual isotopes are depleted and generated by radioactive decay.
- There is no downward, advective transport.
- The concentration of each isotope, in the liquid phase at the source, is maintained at the solubility limit for that element.

The concentration of each isotope, in the liquid phase, at the ground surface, is zero.

The above-ground concentration of the contaminants in plants is a function of the amount of the isotope at the rooting depth.

Erosion, bioturbation, subsidence and climate change will occur during the regulatory period.

After the institutional control period, there is a significant probability that people will live at or near the GCD site.

The consequences of the potentially disruptive future events and processes are being evaluated to determine the best method of incorporating these events and processes into the performance assessment. The consequence models for bioturbation and subsidence along with the details on how these consequences are incorporated in the performance assessment are given below. The climate change and human intrusion consequence models are still in development and are not included in this preliminary analysis.

Bioturbation Consequence Model

The bioturbation model is based on the following assumptions:

The contaminants are transported from the subsurface to the ground surface during burrowing.

The effective burrowing depth is uncertain, but can be bounded by the known burrowing depths for the animal and insect species found or expected at the site.

The effective depth is equal to the maximum depth of bioturbation.

All the contaminants at the maximum burrowing depth are transported to the ground surface.

As a result of these assumptions, bioturbation can be treated as a reduction in the pathlength from the waste to the accessible environment. For the preliminary analyses, bioturbation is modeled by reducing the depth of burial. The uncertainty in the maximum bioturbation depth that will occur at the site is accounted for by using a pdf to represent the probability of occurrence of each possible bioturbation depth and running the simulations in a Monte-Carlo fashion.

A literature search of studies on the existing fauna in the vicinity of the NTS indicates there are 63 different species that can be classified as burrowers. The burrowing animals include 1 owl, 14 lizard, 17 snake, 3 tortoise and 28 mammal species (mostly mice, squirrels, voles, rats and shrews). The burrowing depths of these species is generally less than 1 meter and with a maximum burrowing depth of three meters. In addition to these animals, there are numerous arthropod species that are known to burrow. The possible burrowing arthropods at the site include species of spiders, scorpions, cockroaches, ants and termites. With the exception of termites, the burrowing depths recorded for arthropods are generally less than one meter with maximum burrowing depths less than 3 meters. The literature search shows that the deepest burrowing fauna may be the termite. In general the termite is found with the food supply. In this case the maximum rooting depth would provide a limit for the termite burrowing depth. The rooting depths used in the simulations for existing conditions range from one to ten meters. However, there is evidence that some African termites burrow to the water table, down to 70 meters, to obtain water (14). The burrowing depths of the termite species at the NTS are unknown and the probability of establishing the African species at the GCD site is unknown. Preliminary analyses of the effect of bioturbation on the site's performance were conducted to determine the potential impact of deep burrowing species.

Subsidence Consequence Model

The following assumptions are made regarding the effect of subsidence on the transport system:

Subsidence will occur due to the compaction of the backfill, settling and compression of the waste containers in the boreholes and trenches.

Subsidence decreases the effective depth of burial.

The unconsolidated sediments surrounding the borehole can not maintain a steep slope or fracture due to the effects of weathering.

Subsidence in the borehole is followed immediately by side wall cave in that results in a slope of 40 degrees on the side walls.

Subsidence will occur immediately after the institutional control period.

Increased runoff into the subsidence depression results in increased plant growth.

As with bioturbation, subsidence is incorporated into the performance assessment by treating it as a decrease in the depth of burial of the waste. The subsidence depth

is an uncertain parameter value. Preliminary analyses were conducted to evaluate the potential effects of subsidence and bioturbation on the site's performance. The range of potential subsidence depths was estimated by assuming that the total collapsible void space in the boreholes was between ten and fifty percent of the volume of the borehole containing the waste (the bottom 15.2 meters). This may or may not be a conservative estimate. It is an arbitrary range selected for a preliminary analysis of the effects of subsidence and the combined effects of subsidence and bioturbation on the site's performance. Increased plant growth due to subsidence was not considered in the preliminary analysis. The subsidence depth and increase in plant growth as a function of subsidence will be bounded based on natural analog data.

PRELIMINARY RESULTS

The preliminary analyses were conducted to evaluate the effects of the uncertainty in the new model parameter values on the simulation results and to aid in the development of the performance assessment model. The parameter distributions for the existing hydraulic conditions were used in these simulations. The final performance assessment model will combine all of the significant future processes and disruptive events.

The results of the preliminary simulations incorporating different consequence models of bioturbation and subsidence are summarized in Table I. The only model that results in simulated violations of the containment standard is the one with a large range for the simulated burrowing depth. This model was designed to assess the potential significance of deep burrowing species like the African termite. The distribution selected for this simulation has a maximum burrowing depth of 19.2 meters, deep enough to reach the waste packages after erosion is included. The CCDF generated by 5000 realizations of this model is shown in Fig. 3. The last model listed in Table I combines the effects of erosion, bioturbation and subsidence. This model uses a smaller range of bioturbation depths that are more representative of the existing burrowing depths in the vicinity of the GCD site. The CCDF for the combined model shows that the site is likely to meet the containment requirements when this model is used (Fig.4).

TABLE I

Fig. 3.

Fig. 4.

SUMMARY

As a result of the first two performance iterations, the performance of the GCD site is being assessed using a diffusion transport model linked to a plant uptake model. Sensitivity analyses of the second performance assessment model indicate that plant uptake is the most important process and tortuosity and rooting depth are the most significant model parameters in determining how much of the contaminant reaches the accessible environment. The addition of subsidence and bioturbation to the performance assessment model result in simulations that produce a significant number of failures only when the bioturbation depths are allowed to exceed the existing burrowing depths. Subsidence, erosion and bioturbation are modeled as additive processes that reduce the depth of burial. As a result, these processes decrease the length of plant roots required to reach significant concentrations of the radioactive contaminants. The results of this modeling and previous sensitivity analyses indicate several modeling refinements that could significantly reduce the simulated release rates. These potential refinements include reducing the uncertainty in the minimum tortuosity value and maximum burrowing depth, and separating the bioturbation model from the calculation of the depth of burial. These changes are only necessary if the consequences of climate change, drilling and irrigated farming result in higher simulated release rates or if natural analogue studies indicate a significantly greater maximum subsidence depth than the maximum value used in the preliminary simulations. The current model of the site, for the existing hydrologic conditions with bioturbation, erosion and subsidence, indicates that the site is likely to meet the containment standards.

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USE OF A SCENARIO-DEVELOPMENT PROCEDURE TO IDENTIFY POTENTIALLY DISRUPTIVE SCENARIOS, GREATER CONFINEMENT DISPOSAL FACILITY, AREA 5, NEVADA TEST SITE

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ABSTRACT

The Greater Confinement Disposal (GCD) facility includes four boreholes that contain transuranic (TRU) waste. Presence of the TRU waste means that this facility must comply with the U.S. Environmental Protection Agency's Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Waste--Final Rule 40 CFR Part 191. To comply with the Containment Requirements of this rule, all potentially disruptive events and processes, and by implication all potentially disruptive combinations of events and processes (scenarios), must be identified for possible inclusion in performance assessments. A scenario-development procedure was adopted that identifies individual potentially disruptive events and processes and combines these events and processes to form scenarios. This procedure consists of five steps. Step 1 either creates, adopts, or adapts an initial list of events and processes. Step 2 classifies the events and processes using any of a variety of schemes for organizational purposes. This step may already be included in an adopted list. Step 3 screens the events and processes using well-defined criteria based on regulatory guidance to identify those events and processes that can be eliminated from scenario development without affecting the performance measure. Step 4 constructs all possible combinations of the events and processes surviving the screening process through the use of a logic diagram. Step 5

screens the scenarios using well-defined criteria to identify those scenarios that can be excluded from full-scale performance assessments without affecting the performance measure.

In using this procedure to identify potentially disruptive scenarios for the GCD facility, some modifications were made to the original procedure. To address the completeness issue of the list of events and processes to be screened, lists from 11 sources were combined and consolidated by eliminating duplications. The result was an initial list of 205 features, events, and processes (FEPs). Because of the diverse nature of the FEPs, additional screening criteria were developed. Screening of the FEPs identified four events for scenario development: exploratory drilling for natural resources, drilling withdrawal wells, irrigation, and subsidence. Recent environmental-isotope analyses of the vadose zone suggest that radionuclide transport from the boreholes to the water table by infiltration is not a feasible transport mechanism within the time frame of regulatory concern. For this reason, the event of drilling withdrawal wells was merged with exploratory drilling for resources. The descriptions of the remaining three events were modified slightly to aid in estimation of event probabilities and consequence analyses. The three events are: exploratory drilling for resources penetrates a TRU borehole, irrigation occurs at the Radioactive Waste Management Site (RWMS), and subsidence occurs at the RWMS.

Use of a logic diagram with these three events resulted in the construction of eight scenarios, including base-case (undisturbed) conditions. Screening these scenarios at this stage of scenario development was beyond the scope of this task. Based on the implementation assumptions, this scenario-development procedure produced a comprehensive set of mutually exclusive scenarios that are reproducible and auditable for use in GCD performance assessments.

DISPOSAL-SYSTEM DESCRIPTION

The Greater Confinement Disposal (GCD) facility consists of 13 boreholes (including GCD Test) augured into basin-fill deposits in Area 5 of the Nevada Test Site (Fig. 1). All of the boreholes are within the boundaries of the Radioactive Waste Management Site (RWMS) (Fig. 2), which is being used for the disposal of low-level waste. Each of the GCD boreholes is approximately 37 meters deep, and all but two boreholes are 3 meters in diameter. The remaining two boreholes are 3.7 meters in diameter (1). Four of the boreholes (Nos. 1, 2, 3, 4; Fig. 2) contain waste contaminated with transuranic (TRU) radionuclides (2). The TRU waste was placed in the bottom 15.2 meters of each borehole with probertite, in which the boron acts as a neutron absorber, used as backfill for this interval. The remainder of each borehole was filled with sifted, native material that had been removed during auguring.

Fig. 1.

Fig. 2.

Basin-fill deposits at the GCD location are approximately 506 meters thick, and the depth to the water table is approximately 244 meters. Total yearly precipitation at the GCD location ranges from 10 to 13 centimeters. Recent studies of environmental tracers in the unsaturated zone suggest that recharge is negligible to nonexistent at this location (3).

REGULATORY BASIS FOR SCENARIOS

Because of the presence of the TRU waste in four of the GCD boreholes, this facility will have to comply with the U.S. Environmental Protection Agency's Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste--Final Rule 40 CFR Part 191 (the Standard) (4). Performance assessments (PAs) are required to address the Containment Requirements (191.13) of this regulation. By definition in the regulation (191.12(q)), a PA includes the identification of processes and events that may affect disposal-system performance and estimates these effects. By implication, combinations of processes and events also must be included in the analyses in order to assure that no important disruptions of the disposal system are omitted from the analyses. These combinations of processes and events are referred to as scenarios.

DESCRIPTION OF THE SCENARIO-DEVELOPMENT PROCEDURE

If all combinations of all events and processes that might occur at a disposal site are constructed, the result is a larger number of scenarios than can possibly be analyzed (i.e., 2n, where n equals the number of events and processes). An alternative to identifying all scenarios is to only identify those scenarios that

may have a significant impact on the release of radionuclides from the engineered facility and/or the transport of radionuclides from the engineered facility to the accessible environment (i.e., the land surface and the geologic system more than 5 km from the repository boundary). A scenario-development procedure (5) that accomplishes this goal was developed as one component of the performance-assessment methodology (6,7) adopted by the PA Task of the GCD Project. Several assumptions are made in implementing this procedure:

only postclosure conditions are considered

only events and processes compromising the performance of the engineered facility and/or affecting radionuclide transport to the accessible environment are considered

regulatory guidance limits the severity of human-intrusion events that need to be considered

regulatory guidance limits the probability of occurrence of events and processes that need to be considered

the time frame for the scenarios is 10,000 years after disposal-system closure.

events and processes within a scenario are independent or dependence can be addressed during modeling

the sum of the probabilities of the scenarios developed by the procedure is equal to 1

These assumptions are incorporated into the scenario-development procedure as screening criteria and into the method of scenario construction.

The scenario-development procedure (5) contains five steps. In the first step, a list of events and processes that potentially can affect disposal-system performance is either created, adopted, or adapted. Cranwell et al. (5) contains a generic list of 27 events and processes that were considered to be a starting point in compiling a site-specific or geologic-medium-specific list of potentially disruptive events and processes.

The second step of the procedure classifies the events and processes using any of several schemes for organizational purposes and to help assure that no important events and processes have been omitted. One of the more useful classification schemes is to distinguish between naturally occurring, human-induced, and repository/waste-induced events and processes. Subdivisions within each category also may be useful. In an adopted or an adapted list, this step may already be incorporated in the list.

The third step of the procedure screens the events and processes using specific criteria based on guidance provided to the Standard to identify those events and processes that can be eliminated from scenario development without effecting the performance measure as determined in PAs. Three screening criteria were proposed: probability of occurrence, consequence, and physical reasonableness. Guidance to the Standard states that events and processes that have a probability of less than one chance in 10,000 of occurring in 10,000 years can be eliminated from consideration no matter what the potential magnitude of the consequence may be. The rationale behind the consequence criterion is that the elimination from analyses of any event or process that does not have the potential to affect radionuclide escape or transport simplifies scenario development by producing fewer scenarios and will not affect the calculated radionuclide releases in PAs. Screening out an event or process because of a lack of physical reasonableness is essentially the same as low probability, only the decision is based on logical arguments rather than a calculated (or elicited) numerical value.

Scenarios are constructed in the fourth step of the procedure by developing all possible combinations of the events and processes that survived the screening in Step 3. Combinations are developed through the use of a logic diagram. In the logic diagram, a decision is made at each branch as to whether the corresponding event across the top of the diagram occurs or does not occur in each path through the diagram (each path is a scenario). The sequence of events and processes within a scenario is not relevant to the definition of the scenario. Temporal relationships among events and processes may be addressed during modeling. As a result, each scenario is composed of a unique combination of occurring and nonoccurring events and processes. If none of the events and processes occur (the topmost branch in the diagram), the disposal system is undisturbed (base-case scenario). An example of a logic diagram is included in the section below demonstrating the use of this procedure for the GCD facility.

The final step of the procedure uses two screening criteria to identify which

scenarios should be included in performance-assessment analyses. Probability of occurrence is one criterion, using the same numerical constraints used to screen the events and processes. The other criterion is consequence, in the form of radionuclide releases to the accessible environment.

SCENARIO DEVELOPMENT FOR THE GCD FACILITY

One of the major concerns in scenario development is the possibility that events and processes that could have a significant impact on disposal-system performance are omitted from the original list of events and processes, thereby excluding these events and processes from scenario construction. To address this concern about completeness, scenario-development for the GCD facility (8) merged 11 lists from various sources. The sources for these 11 lists can be divided into four categories: generic lists from U.S. programs, site-specific lists from U.S. programs, generic lists from international programs, and lists from other national programs. The composite list contains 761 features, events, processes, and miscellaneous other listings, which for convenience are referred to as Features, Events, and Processes (FEPs). Consolidation of this master list by eliminating duplications and redundancies produced a revised list of 205 FEPs.

The motivation in the development of the events and processes list in Cranwell et al. (5) was to identify possible disruptions to the disposal system. Several of the merged lists were compiled with other objectives in mind (e.g., listing all conceivable FEPs, identifying pathways to man, etc.). Because of the diverse nature of these FEPs when compared to the Cranwell et al. (5) list, additional screening criteria were developed (Table I) to focus scenario development on identifying potential disruptions of the disposal system in order to address the Containment Requirements in the Standard.

TABLE I

Screening of the 205 FEPs in the revised list identified three events for scenario development: exploratory drilling for resources, irrigation, and subsidence over the Low Level Waste (LLW) trenches and/or the TRU boreholes. Exploratory drilling is a concern to disposal-system performance, because this event could result in the release of radionuclides directly to the surface if the borehole penetrates the waste in the boreholes or the diffusion halo surrounding the TRU boreholes. Both irrigation through the addition of water and subsidence through ponding of runoff are possible mechanisms for increasing the soil moisture in the vicinity of the TRU boreholes, thereby increasing the rooting depth of either crops or native vegetation. Using probability of occurrence or consequence modeling to screen these three events was beyond the scope of preliminary scenario development. The definitions of these events were modified slightly to be more specific on where the events occur for later use in probability estimation and consequence analyses. The revised wording results in the following three events: exploratory drilling penetrates a TRU borehole, irrigation occurs at the RWMS, and subsidence occurs at the RWMS. An event of drilling withdrawal wells originally survived screening, but the elimination of the transport pathway of radionuclides reaching the water table (9) eliminates concerns of withdrawal wells bringing contaminated water to the surface. The potential releases of radionuclides to the surface resulting from withdrawal wells being drilled into TRU boreholes were included by considering these boreholes to be exploratory drilling for water and therefore combined with the more generic event of exploratory drilling for resources penetrates a TRU borehole. If analyses of irrigation and subsidence indicate that the increased infiltration associated with either of these events can result in radionuclide transport to the water table, the event of drilling withdrawal wells will be reconsidered as a mechanism for radionuclide transport to the accessible environment.

The three events can be combined with the use of a logic diagram to form eight scenarios (Fig. 3). In this preliminary scenario development (8), no attempt was made to screen the constructed scenarios. Both probability estimation of and consequence analyses for the individual events in the scenarios are ongoing activities. Preliminary estimation of scenario probabilities (10) suggests that from the probability point of view, disposal-system performance is dominated by the base-case scenario (i.e., undisturbed conditions) and subsidence exclusive of the occurrence of the other two events. From the consequence point of view, no single scenario or small number of scenarios has been identified at this time through consequence analyses as dominating disposal-system performance. The dominant mechanisms of radionuclide transport to the accessible environment are human

intrusion by drilling and radionuclide uptake by plants. One or both mechanisms may be present in each scenario. Although the scenarios containing more than one of these transport mechanisms may seem more likely to result in greater consequences than scenarios with single mechanisms, the duration of occurrence of each mechanism in each scenario within the time period of regulatory concern may result in different conclusions.
Fig. 3.

CONCLUSIONS

Based on certain implementing assumptions, the use of a structured scenario-development procedure has produced a comprehensive set of mutually exclusive scenarios that are reproducible and auditable for use in PAs. Comprehensive refers to the fact that all possible scenarios are constructed from those events that survived screening. Mutually exclusive means that each scenario consists of a unique combination of events, which is essential when associating scenario consequences with scenario probabilities. The scenarios are reproducible in that an identical set of events will always produce the same scenarios. These scenarios are auditable in that the steps of the procedure are well defined, and the decisions made in implementing the procedure are documented.

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ABSTRACT

The Waste Isolation Pilot Plant (WIPP) in southeastern New Mexico has been constructed to be a repository for permanent geologic disposal of transuranic waste generated by the U.S. defense programs during the past 50 years. Assessment for 10,000 year performance of the repository is being carried out by Sandia National Laboratories under contract to the U.S. Department of Energy (DOE), to determine compliance with the EPA standards for long-term disposal of transuranic wastes (40 CFR 191). The assessment consists of postulation of scenarios for release of radionuclides from the repository to the environment, selection of conceptual models for movement of contaminants, collection of data for more than 300 parameters for waste and site characteristics and engineered barriers, and probabilistic consequence analysis to represent the results as Complementary Cumulative Distribution Functions (CCDFs). The compliance with the Containment Requirements of the standards shall be judged from the CCDFs. In addition, the standards contain Assurance Requirements, Individual Protection Requirements, and the Groundwater Protection Requirements. Many unresolved issues remain in every aspect of the WIPP performance assessment. To make the assessment more robust, these unresolved issues will have to be addressed.

INTRODUCTION

The Waste Isolation Pilot Plant (WIPP), located in southwestern New Mexico 40 km east of the city of Carlsbad, is planned to be a geologic repository for transuranic (TRU) radioactive waste, generated from the U.S. defense programs. The site for the project was selected in 1973 and the site characterization work began in 1974. Full-scale construction of the surface facilities and underground excavation began in 1981. By 1988, the facility was fully constructed including one out of the eight planned "waste disposal panels" of the repository.

The underground repository has been excavated at a depth of 653 meter in the lower part of a 600 meter thick geologic formation known as the Salado Formation, which as the name indicates, consists mainly of rock salt. Details of the geologic characteristics of the site can be found in Chaturvedi (1993) (1). The repository has been designed to hold contact-handled transuranic (CH-TRU) waste in 55 gallon mild carbon steel drums and steel boxes. The drums and boxes will be stacked in eight underground "panels" each consisting of seven "rooms". Each room is 91.5 meter long, 10 meter wide, and 4 meter high (300 ft x 33 ft x 13 ft) and will contain up to 6000 drums or "drum-equivalent" of CH-TRU waste. It is also planned to stack the drums and boxes in the panel drifts leading to the "rooms" and in the drifts leading to the panels. The repository has been designed to accommodate a total of approximately 850,000 "drum-equivalent" of CH-TRU waste. In addition, it is planned to emplace 7,500 canisters of remote-handled transuranic waste in horizontal boreholes in the repository rooms and drifts.

Even though the physical facility has been ready since 1988, waste shipment to WIPP has not yet begun because the decision to use WIPP as a permanent repository has not been made. This decision will be based on performance assessment, i.e. predictions of long-term behavior of the natural environment as well as the man-made waste containment systems. This paper briefly describes the performance assessment work that has been completed and outlines some significant issues that remain. Although the WIPP has to comply with the provisions of the Resource Conservation and Recovery Act (RCRA) and other regulations, this paper addresses primarily the issues related to assessing the compliance with 40 CFR 191 Subpart B (2).

The procedure for assessment of compliance with the EPA disposal Standards (2) is called "performance assessment" and is defined in the Standards itself at 40 CFR 191.12, as follows:

An analysis that:

1. Identifies the processes and events that might affect the disposal system;
2. Examines the effects of these processes and events on the performance of the disposal system, and

3. Estimates the cumulative releases of radionuclides, considering the associated uncertainties, caused by all significant processes and events.
Subpart B of the Standards were vacated by the First Circuit Court of Boston in June 1987 on grounds that they were less stringent than the Clean Water Act of 1971, and no explanation was provided by EPA for this discrepancy. The Standards were remanded to the EPA for revision and repromulgation. Shortly after this action, New Mexico entered into a modification to the Consultation and Cooperation Agreement with DOE to continue to evaluate WIPP against the vacated 1985 Standards because the technical requirements of the revised Standards were not expected to differ substantially. The EPA published the amended Standards in the Federal Register on December 20, 1993. The amendments were made only to those parts of the Standards in which the court had found faults; the bulk of the Standards remain unchanged.
The Sandia National Laboratory (SNL) published the results of the WIPP performance assessment annually from 1990 to 1992 (3, 4, 5). The EEG provided comments on these assessments (6). The DOE plans to submit an application for certification of compliance with these Standards to the EPA in December, 1996.
The EPA disposal standards (2) contains four "requirements", viz., containment requirements, assurance requirements, individual protection requirements and groundwater protection requirements. The performance assessment technique is used to assess compliance with all of these except the assurance requirements which are meant to provide additional assurance of containment of the waste within the repository.
The process of performance assessment consists of: 1) development of potential scenarios for release of radionuclides to the environment, 2) identification and selection of the most appropriate conceptual models, 3) obtaining data for each of the input parameters for the analysis, 4) selection or development of appropriate computer codes, 5) estimating the probabilities of various scenarios, 6) calculating the consequence of the scenarios, and 7) combining the results in the form of complementary cumulative distribution functions (CCDF's).
The following discussion identifies the unresolved performance assessment issues for each of these tasks.
UNANALYZED SCENARIOS
Formulation of scenarios is the bedrock of performance assessment. The WIPP performance assessment has considered only one disruptive scenario, drilling into the repository by future generations seeking minerals. In the 1992 Performance Assessment (3), two variations of this drilling scenario were analyzed. However, these analyses were incomplete because drilling was considered for only 1900 years (between 100 years after closure to 2000 years after closure) out of 10,000 year regulatory period. The following are some examples of scenarios that should be analyzed as part of the WIPP performance assessment.
Contaminated Brine Flows to the Surface
The human-initiated events scenarios analyzed by the WIPP performance assessment so far assume that the only material reaching the surface is cuttings from the drill-bits and some "cavings" from the annulus about the drill-bit in the waste disposal room. Brine flows from a brine reservoir underlying the repository to the surface should also be assumed. The consequences of this scenario could be significant.
Pressurized brine has been encountered in several boreholes surrounding the WIPP site and in the borehole WIPP-12 at the WIPP site, 225 to 250 meter below the repository horizon in the upper part of the Castile Formation (1). Varying quantities of brine was released to the surface during each of these encounters before the flow could be controlled. The performance assessment scenario should assume that any driller would face similar situation. Drilling-mud return flow would be expected to increase the effective radius of the borehole and bring waste to the surface in suspension and in solution. The DOE has agreed to analyze these additional scenarios.
Brine Slurry Release Scenario
A brine-slurry release scenario should be analyzed. A brine slurry might result from brine inflow from the Salado salt or intrusion into a Castile brine reservoir. Such a brine slurry could be under greater than hydrostatic pressure and thus have a force capable of driving some or all of the slurry to the ground surface. The possible implications of a brine-slurry filled room were first raised by Sandia National Laboratories in 1987 and were also evaluated in 1988 by EEG (7).

Subsidence

Subsidence could occur in the area overlying the WIPP some time after repository decommissioning. Subsidence can also occur from nearby potash mining. The 1992 Performance Assessment (5) identifies subsidence from mining of potash as a credible event but it has not been analyzed.

The potential for subsidence was discussed in the 1990 Preliminary Comparison (3). It was recognized that "subsidence could in turn conceivably affect the disposal system in three ways: by increasing hydraulic conductivity of the Salado Formation, by creating fractures through the Salado Formation, or by disturbing the surface drainage and groundwater flow in overlying units." The incorporation of the effects of subsidence into the performance assessment is still planned. In the 1991 Preliminary Comparison (4), an analysis of possible caving and subsidence over the waste storage areas from room closure concluded that no problems were likely to result for the waste disposal system. The maximum subsidence at the surface was calculated to be only 0.13 meter over an area of 1.54 x 106 m2. The affected area at the surface was determined by assuming an angle of draw of 35. It was further stated that if the Rustler-Salado contact residuum had (historically) lost about 400 meters due to dissolution without disrupting the confined water-producing Culebra and Magenta dolomite aquifers, subsidence should not be a problem.

No evaluation has yet been made of subsidence from potash mining. There are significant potash resources within and outside the WIPP site boundary and it is appropriate to consider subsidence effects from potash mining.

Gas Pressurization

Concerns about gas generation from the waste was the basis for the DOE's plans to conduct experiments with waste at WIPP. The performance assessment calculations, however, do not incorporate the potential fracturing of the repository leading to release of a fraction of the waste to the environment.

Climate Change

Geological effects of climate change, i.e. dissolution, subsidence, change in hydrological properties of subsurface strata, etc., should be considered in formulating release scenarios.

CONCEPTUAL MODELS

A conceptual model describes the way things are likely to happen. A combination of conceptual models is used in performance assessment. The selection of conceptual models, therefore, directly affects the outcome of the exercise. The WIPP project investigators have not yet settled on which conceptual models to use on a number of issues. The following examples from the WIPP performance assessment illustrates this point.

Salado Hydrology

The WIPP host repository rock, bedded salt of the Salado Formation, yields more brine than was anticipated (8). Three conceptual models have been proposed to account for this, viz., Darcy flow from far field; flow from the disturbed rock zone only, as a result of excavation; and squeezing of clay layers. The results of investigations conducted to date appear to favor the far-field Darcy flow model, but the Project has not yet decided which model is correct.

Contaminant Transport in the Culebra

The Culebra Member of the Rustler Formation overlying the Salado Formation is a 8 meter thick fractured dolomite bed which is the most transmissive zone overlying the repository and has been identified as the most likely pathway for transporting radionuclides from the repository to the accessible environment. The performance assessment has assumed dual porosity flow with matrix diffusion through the Culebra. However, a conceptual flow-model based entirely on channeling with no matrix diffusion also fits the hydrological field data. Sandia National Laboratory is conducting a 7-well field tracer test to try to resolve these questions. Additional laboratory studies may be necessary to establish matrix diffusion mechanisms. Until these issues are resolved, there is no basis to favor a particular conceptual model.

Chemical Retardation in the Culebra

The performance assessment has assumed chemical retardation during flow through the Culebra based on values derived through expert judgment of Sandia National Laboratory scientists. The judgments were based on values of retardation coefficient derived from powdered rock samples and assumption of corrensite clay lining the fractures, without sound experimental data. Credit for retardation may be taken if laboratory experiments on fractured core currently in progress and the field tracer

tests with sorbing tracers yield positive results indicating chemical retardation. However, there appears to be little reason to assume that corrensite clay lining is present in the fractures of the Culebra dolomite along the expected flow path.

DATA

Data are needed for two purposes - to show that the proposed conceptual models are correct and to obtain parameter values. About 300 distinct parameters were used by the latest performance assessment for WIPP (5). For many parameters, a range of values has been used for probabilistic consequence analysis. A narrow defensible range of values generally indicates high confidence in the understanding of that parameter. A broad range typically indicates correspondingly less confidence in those values and in the understanding of that parameter. For a robust performance assessment, therefore, one should strive to narrow the range as much as possible. This is generally achieved through well designed experiments and analyses of the results.

The data sets used in the WIPP performance assessment include parameters for natural (geologic) systems, source term based on waste characterization, and engineered systems to contain the waste. Since up to 70% of the WIPP capacity waste is yet to be generated and the characteristics of the existing waste are known mainly through "process knowledge", the challenge to the project is to develop an envelope that covers the performance assessment assumptions of the source-term and waste characteristics. The waste acceptance criteria will then be developed to guide the waste generators in shipping their waste to WIPP. This has not yet been achieved.

Experimental data have not been obtained to validate some conceptual models. This includes radionuclide retardation during postulated flow of contaminants through the Culebra aquifer, discussed in the previous section, and radionuclide solubility in the WIPP brine. Few relevant measured actinide solubilities exist. The WIPP performance assessment has so far used values obtained through subjective elicitation. The resulting values span a wide range. For example, the estimated plutonium solubility spans 12 orders of magnitude. When these wide, subjective estimates are used in Latin Hypercube Sampling of input values, non-conservative solubilities in consequence calculations may result. It is therefore essential to use the experimental values to limit the range. Where experimental values are not available, it is better to wait for completion of the experiment rather than use a non-defensible performance assessment for the compliance application.

CODES

Sandia National Laboratories has in hand all computer codes that are needed for performance assessment - 15 modeling codes and 35 utilities that assist in analysis. However, the major modeling codes may undergo significant revisions before final versions are used in compliance calculations. Two analyses not yet completed may require the development of additional computer codes.

The discharge of brine contaminated by radionuclides into members of the Rustler Formation is likely to cause changes in the density of water in the aquifers. This effect has not been analyzed and needs attention.

In the disposal rooms in the repository, complex processes are expected to occur in time. Brine will flow into the rooms. Gas will be generated by corrosion, microbial action, and a-radiolysis. Salt creep will close the room. However, these processes are highly coupled. Some elegant analyses coupling 2 of the 3 phenomena have been done. Until a fully coupled analysis is done, there cannot be assurance that less complete analyses suffice.

RESULTS

In the last performance assessment for the WIPP (5), USDOE noted that the calculated CCDF's were well below the allowable limits in the USEPA Standards. For the case of total release from repository/shaft barrier only, and a [0, 30] boreholes per km2 over 10,000 years sampled intrusion rate, the mean CCDF comes to within a factor of two or three of the USEPA containment requirement (5, vol. 4, Fig. 9-1, curve 1). This suggests several vectors of CCDF lie in the zone of violation of the containment requirement. This mean CCDF is not as conservative as it may appear because subjectively elicited solubilities are incorporated. The non-conservative basis of curve 1 in Fig. 9-1 is illustrated in Fig. 1. Using BRAGFLO-calculated brine flow from the repository up to the Culebra (70 vectors for both the E2 and E1E2 scenarios), all actinide solubilities at 10-3, 10-5 and 10-7 M, and the human intrusion rate samples uniformly between 0 and 30 boreholes/km2/10,000 years, the mean CCDF's are shown in Fig. 1, along with curve 1 from Fig. 9-1 (5, vol. 4). If

the extremely low subjectively elicited solubilities are not used, then the mean CCDF for the case of engineered barriers alone may not meet the containment requirement.
It is essential that the performance assessment analyses address these concerns as soon as practicable.
Fig. 1.
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53-4
CHECKING OUR PREMISES FOR REDUCING THE RISK OF INADVERTENT HUMAN INTRUSION
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ABSTRACT
Current regulatory assumptions about the risk of inadvertent human intrusion into a deep geologic repository view natural resources as a significant potential handicap for long-term containment. An argument can be made, however, that natural resources actually enhance a repository's defense-in-depth against inadvertent as well as intentional disturbance. Physical evidence and records of resource use have often survived hundreds, in some cases thousands, of years. We tread more carefully in areas with identified resources or a history of exploration and extraction than in virgin territory. Resource depletion diminishes an area's attractiveness to future prospectors and leaves in its wake disincentives or even barriers to human intrusion. Identified natural resources may therefore play a positive role in the long-term containment of dangerous wastes. Adding to this fresh perspective the ongoing reassessment of the biological effects of low-level radiation, we may streamline existing regulations, expedite disposal projects, and save taxpayers' money with no compromise on safety.
CONSTRAINING REQUIREMENTS
The Waste Isolation Pilot Plant (WIPP) has been designed and constructed as a deep geologic repository in salt to permanently isolate radioactive mixed waste from the biosphere. Before actual waste can be brought to the site, however, the U.S. Environmental Protection Agency (EPA) and other regulatory and oversight groups need to be reasonably certain that the repository will safely contain the waste for at least 10,000 years. Research results and model calculations to date show that the containment requirements will not likely be compromised if the repository is not disturbed during that time. Therefore, the principal concern remaining to be

addressed is the so-called inadvertent human intrusion scenario, i.e., breaching the repository by exploratory drilling or mining for natural resources. Such a breach might conceivably expose humans and the environment to radioactive and hazardous materials in concentrations exceeding regulatory limits.

To gain the confidence needed for compliance with the long-term containment requirements for radioactive waste (1), the EPA specifically prescribes the following:

> Places where there has been mining for resources, or where there is a reasonable expectation of exploration for scarce or easily accessible resources, or where there is a significant concentration of any material that is not widely available from other sources, should be avoided in selecting disposal sites. Resources to be considered shall include minerals, petroleum or natural gas, valuable geologic formations, and ground water ..... Such places shall not be used for disposal of the wastes... unless the favorable characteristics of such places compensate for their greater likelihood of being disturbed in the future (1) (italics added).

This regulation is commonly known as the resource disincentive requirement (2). The WIPP faces an additional resource constraint as the result of an agreement between the U.S. Department of Energy (DOE) and the state of New Mexico. In the Second Modification to the Consultation and Cooperation (C&C) Agreement (U.S. DOE and State of New Mexico, 1981, as modified), the parties agreed that:

> The DOE will not permit subsurface mining, drilling, or resource exploration unrelated to the WIPP Project on the WIPP site during facility construction, operation, or after decommissioning. This prohibition also precludes slant drilling under the site from within or outside the site (2).

UNDERLYING ASSUMPTIONS

Constraining resource requirements are based on the general assumption that selecting deep geologic disposal sites which also contain natural resources may "result in the denial of access to important raw materials" and "lead to future disturbance of the geological/hydrological system through exploration or production, including direct intrusion into the repository" (2).

Addressing these twin conflicts of resource denial and attractiveness, the EPA's Science Advisory Board (ESAB) recommended

> that EPA not preclude consideration of a potential site because natural resources are at or near the site, but rather should note that the presence of such resources is a highly unfavorable factor which should be included in the site evaluation (2) (italics added).

In response, the EPA agreed

> that automatically precluding a potential site because of one disadvantage is not desirable. At the same time, the Agency still believes that proximity to important or unique resources is a serious problem.. .. proximity to resources should be considered a serious disadvantage, but not an outright prohibition, for site selection (2) (italics added).

Consequently, "the EPA expects that sites with resources would be used only if it is reasonably certain that they would provide better overall protection than the practical alternatives that are available" (2). Applying similar assumptions the Environmental Evaluation Group (EEG), which provides independent technical evaluation of the project, avoids using the unfavorable italicized terms but agrees that: "Because the WIPP Site is in an area rich in oil and gas resources, the integrity of the repository is inherently subject to the drilling, production, and abandonment practices of the oil and gas industry" (3).

KEY ISSUE

As the term implies, the resource disincentive requirement is based on a negative assessment of natural resources vis-a-vis deep geologic disposal. Pertinent U.S. regulations are the most detailed and prescriptive of all national regulations (4). But their pessimistic view is by no means shared universally. Waste repositories in Germany, for example, are operating in idle and still active underground mines whose very reason for being is the presence or former presence of resources in economically recoverable concentrations and quantities. Once we decide to dispose of waste in existing or former mines rather than in special and fresh excavations, the choice of areas rich in natural resources actually becomes inevitable. This begs the question: "Is a disposal site containing natural resources or mines or both really more likely than areas without those features to be disturbed inadvertently?" The regulatory record does not offer any foundation for an affirmative response. And

that conspicuous lack of empirical data compels us to re-examine the underlying paradigm, because the opposite may be true. In that case, we would have to stand the current regulatory assumption on its head and contend that identified natural resources at a site selected for deep geologic disposal are more likely to discourage than to promote inadvertent human intrusion. Several lines of reasoning provide a solid basis for this revisionist assumption.

RESOURCE KNOWLEDGE

We generally accumulate more extensive and detailed geologic information about those areas known or suspected to be rich in natural resources than about those that are poor. Several decades of petroleum and potash exploration and production near Carlsbad generated a wealth of geologic data ready to be evaluated against the WIPP's site selection criteria. State and federal agencies maintained drilling and mining records sufficiently accurate to preclude unpleasant late surprises. The Department of Energy (DOE) took all the available information into account before choosing the best of several alternative sites. Without the knowledge gleaned from exploration and resource extraction records, the area might not have been identified as promising for waste disposal; but if it had anyway, all the required information would have had to be generated from scratch.

Inadvertent or, for that matter, intentional disturbance by drilling will more likely result from random or wildcat exploration in virgin territory than from delineation and development activities in an area whose geology and extractive history are fairly to very well known. Safety and liability concerns motivate entrepreneurs to either stay clear of previously mined areas or to proceed with special caution. And a history of depletion of natural resources in a given area certainly does reduce if not eliminate its attractiveness to future prospectors. The conclusion is therefore inevitable: The better the prospects for natural resources, the more we know about the geology, the more thoroughly we check for past mining, and the less likely will we stumble into the unexpected.

EXTRACTION EVIDENCE

All extraction of natural resources leaves some evidence. We have located and archaeologically investigated the 5000 year-old so-called King Solomon's mines at Timna (5). Ancient documents first mention the Wieliczka salt mine in Poland in 1044 (6). And our U.S. Dollar still commemorates the silver mines of Joachimsthal in Bohemia where the first "Thaler" was minted in 1518 (6).

Evidence of past mining or drilling, or both, on a potential disposal site itself may or may not disqualify that site from further consideration. It certainly does not everywhere outside the U.S. But be that as it may, such evidence immediately adjacent or around a designated disposal site may actually form a negative permanent marker. The site's distinctive "footprint" - an area without signs of any extractive activity, surrounded by indications of heavy activity - would remain permanently visible. Future generations would wonder, before digging on the site, why this oddly - shaped spot (Fig. 1a,b), certainly not a natural feature, had not been exploited together with the rest of the neighborhood. This reflection would lead to a search of available records; but even if all records had been lost, our descendants would likely remain wary. And that is, after all, what we want them to be. Ergo, evidence of resource exploration and extraction adjacent or around a site should not simply be presumed a negative attribute.

Fig. 1a.

Fig. 1b.

SUBSURFACE DISINCENTIVES

Extracting identified resources from the deep disposal site before decommissioning will by itself create physical disincentives to future repository penetration. At the WIPP, several potash seams lie in a zone from 125 to 240 m above the disposal level. One or more of these seams could conceivably be mined. If that were permitted, the mining pattern with the highest potential to deter inadvertent penetration should exactly overlie the disposal level room-and-pillar pattern. Under this arrangement, drilling on the disposal site would encounter either solid salt all the way (in the pillars) or hollow or at least disturbed intervals above the waste rooms (Fig. 2). Only the latter case concerns us. Here, the operator may face loss of drilling fluid circulation. That would give him an effective warning, because there is no natural reason or precedent for losing circulation in the Salado Formation salt. The warning could be amplified by marker materials such as cellophane shreds, chemical dyes, or engraved discs, left in the mined-out rooms, to

be brought to the surface by the drilling fluid if circulation were reestablished. Fig. 2.
TOLAN (7) suggested a further enhancement of this scheme. Subsurface barriers could be emplaced as backfill in mined openings above the repository. Layers of rubber tires, steel fencing, or baling wire would at least temporarily stop all drilling progress. Truck-mounted drill rigs cannot penetrate these obstacles while characterizing old land fills. "The rotary drill bit was not able to cut its way through the still pliable rubber tires... In the case of steel wire, the wire literally bound-up the drill bit and disabled it" (7). Resource removal with or without ancillary measures can therefore be a significant part of a repository's defense-in-depth (pun intended) against inadvertent human intrusion.

RADIATION HORMESIS

Most if not all current concerns about inadvertent human intrusion into a deep geologic repository for radioactive waste may become irrelevant if a paradigm shift already in progress continues and accelerates. U.S. and international radiation protection standards are at present based on the non-threshold linear hypothesis which assumes that all levels of radiation, regardless how minuscule, are harmful to organisms and should be avoided. Ample scientific evidence (8,9) contradicts this unscientific assumption. For example, if the lung cancer rate corrected for smoking is plotted against radon exposure, the reigning notion predicts a positive slope, while the data result in a negative one (10). "Exposed nuclear workers had statistically lower cancer mortality rates than unexposed control workers in nuclear plants" (11). Most researchers are still reluctant to conclude that low-dose radiation is beneficial, i.e., has hormetic effects; but a growing number are becoming convinced that the optimum dose of ionizing radiation is higher than the natural background, possibly by a factor of ten to fifty (12).
Sandia National Laboratories (SNL) performed deterministic calculations of long-term radionuclide transport behavior at the WIPP (13). In response to inadvertent human intrusion by drilling, the likely releases would result in small additional exposure to humans: from 1.3 times the annual natural background spread over 50 years, to 8 times the natural background for just one hour. These dose calculations were conservative because they took no credit for radioactive decay of the waste prior to being disturbed. We can therefore project that, as the concept of radiation hormesis gains ground, health physics concerns about inadvertent human intrusion will diminish and could conceivably vanish.

CONCLUSION

Upon even cursory examination, some of our principal premises about the effect of natural resources on the long-term integrity of a deep geologic repository are found wanting. Knowledge of the presence of resources and of their extraction, if applicable, actually decreases a site's potential to be disturbed. The more past and present resources we can identify, the less incentive remains for human intrusion-inadvertent or intentional. This recognition, coupled with the ongoing re-evaluation of the biological effects of low-level radiation, may ease the path to regulatory compliance for the WIPP as well as other deep geologic disposal projects.

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TOWARDS A QUANTITATIVE TREATMENT OF MODEL UNCERTAINTY IN THE PERFORMANCE ASSESSMENT OF HIGH-LEVEL RADIOACTIVE WASTE REPOSITORIES

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ABSTRACT

We address the problem of explicitly treating the uncertainty associated with the predictive models employed in a performance assessment. Model uncertainty comes from our inability to fully characterize the system itself and the physicochemical processes which govern it, as well as from deliberate simplifications which are inevitably introduced and affect the forecasting results of the model. In general, these uncertainties stem from gaps in current knowledge and understanding, and, as such, they are largely unknown. Moreover, this body of knowledge continuously evolves over time and, with it, so do our model representations of the physical systems. Therefore, no matter how adequate our characterization of model uncertainty is, it will always be limited by our current state of knowledge. We note that, for the models used in a performance assessment, their predictions cannot be observed over the time and space scales over which they are intended to be applied, so that model validation is generally of little significance. This paper presents and discusses two mathematical approaches to the quantitative treatment of model uncertainty, namely the alternate hypotheses and the reference model approaches. These approaches necessarily suffer from the limitations which inevitably affect the current body of knowledge and beliefs.

INTRODUCTION

Regulations for the long term disposal of nuclear wastes require that the repository system perform its containment and isolation functions for very long periods of time and for quite extensive spatial scales (1). Consequently, the analysis of the performance of disposal facilities relies on the application of predictive models and is inherently affected by the associated uncertainties. These uncertainties come from a variety of sources and are with regard to the future state of the disposal system, the models and codes used to simulate the relevant physicochemical processes and the numerical values of the parameters required to exercise the models (2). Inevitably, expert judgments need to be used, together with the available experimental data, to account for the uncertainties of a performance assessment.
The uncertainty associated with the models is often considered the most important one. However, traditionally, the majority of efforts have been directed towards the treatment of data and parameter uncertainty, while very little has been done to deal with model uncertainty. A model is a representation of a real system: as such, it is usually of approximate nature and uncertainties inevitably arise wherever there are possible alternative interpretations of the system and its phenomena, which are all plausible in light of the current knowledge of the system. It is important to note

that the model representation of the real system is necessarily dependent on the objectives of the particular analysis for which the model is to be used and the corresponding level of accuracy required (3).
Model uncertainty results from our inability to identify and fully characterize the physical processes and phenomena governing the system, so that there is not a definite single set of hypotheses describing the system. The uncertainties, stem from actual "gaps" in knowledge which can take the form of poor understanding of phenomena that are known to occur in the system, as well as complete ignorance of other phenomena actually occurring. Typical examples of this kind of uncertainties occur in a variety of scientific fields such as seismic hazard analysis, groundwater flow and transport, modeling of the biosphere for contaminant transfer, pharmacokinetic modeling, severe nuclear accident phenomenology. Reference 4, for example, reports a study of groundwater flow and contaminant transport, in unsaturated fractured tuff, for which the phenomena governing these processes are not fully characterized and different alternative descriptions arise. More precisely, the uncertainties in the fundamental assumptions on the actual mechanisms governing the groundwater flow and contaminant transport give rise, in this case, to six alternative conceptual models. In the medical field, Ref. 5 presents an analysis of model uncertainty associated with an investigation of the carcinogenic potency of chloroform. Once again, the uncertainty arises largely as a result of fundamental limitations in the understanding of the mechanisms by which chemicals induce cancer. This ignorance prevents us from defining appropriate measures of biologically effective dose and limits our ability to clearly specify the functional relationship between dose and response. Furthermore, it makes it difficult to extrapolate experimental results from animals to humans. Therefore, the situation is such as to inevitably lead to alternative interpretations of the key issues. In the arena of nuclear reactor safety, five different modes of reactor vessel breach have been considered at the Sequoyah Power Station, Unit I (6): an in-vessel steam explosion that fails the upper head of the reactor vessel and generates a missile of sufficient velocity to breach containment; a steam explosion adequate to fail the upper head, but not to damage the containment; a steam explosion that could fail the bottom head; ejection of a significant portion of the core debris at high pressure; melt-through of the vessel and gravity-driven pour of the core material. Clearly, each failure mode represents a distinct model based on a distinct set of phenomenological assumptions.
Model uncertainty can also arise due to our inability to fully characterize the system itself. This is typical of groundwater flow and contaminant transport problems in which the geology and hydrology of the site itself often cannot be adequately characterized, in the sense that the available data is not sufficient to uniquely determine the exact structure and properties of the soil (site characterization) and, consequently, the actual mechanisms of flow and transport occurring. Again, Ref. 4 provides, to some extent, an example of how a poorly characterized real system could make the development of a unique, defensible model a difficult task. Another example is given in Ref. 7, concerning a field experiment conducted at the New Mexico State University College Ranch, with the objective of studying water flow and tracer migration in shallow soils. A large amount of site characterization data was collected to assist in the interpretation of the experiment. As many as six hundred core samples and six hundred disturbed site samples were taken systematically over the experimental region. The post-test evaluation of the results indicated that despite the large number of soil samples, a thin layer of soil with distinct properties that permit transmission of water at low moisture contents was missed. It is quite clear that with this information missing, none of the many plausible models can replicate the resulting shape of the plume.
Finally, an inevitable source of uncertainty in models comes from deliberate simplifications that affect the forecasting results of the model. Typical examples of this situation are related to the dimensionality of the system representation and to the discretization of continuous space domains (parameterization). Reference 8 reports an example of the impact of different parameterization schemes, within the inverse problem of aquifer parameter identification for a two-dimensional unsteady groundwater flow. Another example is provided in Ref. 7 where three different models are depicted, which differ in the number of zones considered to capture the heterogeneities of the system. As previously noted, information is missing regarding a thin layer of soil which allows flow of water at low moisture content, so that

even the highly detailed 3621-zone model cannot replicate the bifurcation which actually occurs in the real plume. An important aspect of model uncertainty deriving from alternative parameterization schemes, is the inevitable trade-off which occurs with the uncertainty in the corresponding parameters, which are generally unknown and need to be reliably identified or inferred from measured data. As the dimension of the parameterization increases, the modelling inaccuracy in the system description typically tends to decrease but the reliability of the estimated parameters is also reduced. Reference 8 discusses this aspect and presents an approach to achieve the optimum dimension of parameterization.

Independently of its origins, three main issues arise when treating model uncertainty: i) identification of the uncertainties; ii) how can we feel confident that the uncertainty has been characterized to a sufficient level and iii) how do we quantitatively represent this uncertainty. These issues must be satisfactorily addressed by the modelers of a performance assessment.

The identification of possible sources of model uncertainty is of paramount importance and consists of making assumptions about the characteristics and behavior of the system, and recognizing processes and phenomena that are known to be uncertain. As model uncertainty is primarily due to insufficient understanding and knowledge of the system, it is very unlikely that this step of identification will lead to a complete and exhaustive set of sources of model uncertainty. Yet it should be performed as adequately as possible, in light of the existing body of knowledge. A systematic approach based on some sort of classification of the types of model uncertainty, as the ones previously discussed, might aid this task.

The second issue inevitably brings up the controversial, philosophical problem of model validation (3,9). However, regardless of the still largely unresolved ambiguity of the terminology and the lack of precise operational guidance, it seems fair to state that a complete model validation, in its strict scientific sense, cannot be obtained for performance assessment models because the predictive performance of such models cannot be observed over the time and space scales over which they are intended to be applied. However, it is important not to forget that what drives the whole performance assessment is actually the decision making process which is to be based upon it. In the model uncertainty example of Ref. 7, if the actual shape of the plume does not affect significantly the decision making process, then one may conclude that model uncertainty has been characterized to a sufficient degree. If, on the other hand, the final decision changes with the different, alternative parameterizations, then one must conclude that the uncertainty is not sufficiently characterized and further investigation is needed to resolve the matter. Therefore, it seems logical to attempt to characterize all the uncertainties inherent in the models of a performance assessment in light of their impact on the resulting decision. This seems to call for a somewhat different approach to model uncertainty and a methodology which aids in the assessment of the impact of the various model uncertainties on the decision making process. Furthermore, this way of approaching the problem is expected to ease the identification of those properties of the modelled system which necessitate further investigation, as indicated by the fact that the associated available and plausible alternatives lead to contrasting decisions.

Concerning the third issue, it calls for the development of mathematical models which enable to represent the uncertainty in a practical and useful way, according to the objectives of the overall performance assessment. A more detailed discussion of this issue is presented in the next section.

At this point, we emphasize that, no matter how adequately one addresses the problem of model uncertainty, and no matter how rigorous the mathematical treatment is, one will still not be able to place full confidence on the characterization of uncertainty thus achieved. This limitation is obviously due to the fact that these uncertainties originate from gaps in our knowledge and we actually cannot model what we do not know. As mentioned before, we might be leaving out of our model a process which actually occurs, or we might be applying a model under conditions for which it actually does not apply. Obviously, if we knew this we would perform our analysis differently; but we do not know it and, therefore, our characterization of the uncertainty cannot include such a thing. Furthermore, our understanding of the physical behavior of a system evolves over time, sometimes even quite drastically, and therefore new models might originate. These arguments lead to the conclusion that the characterization of the uncertainty that we can achieve is necessarily

limited. It is of fundamental importance that these limitations be clear to the analyst and explicitly communicated to the decision maker.
In the next section, we proceed with a discussion on some aspects of the mathematical treatment of model uncertainty and introduce, and compare, the alternate hypotheses and the reference model approaches (10).

QUANTITATIVE TREATMENT OF MODEL UNCERTAINTY

Consider the case in which N models are available for the description of the behavior of the dependent physical variable y=y(x), a function of the independent variable x. For simplicity we regard both y and x as scalar variables, the generalization to the multidimensional case being straightforward. A typical example of y could be the spatial distribution of the hydraulic head in an aquifer. We assume that the models provide an aleatory representation of the process of interest so that their predictive outputs are provided in the form of aleatory distributions for the quantity of interest y. Let Fi (y) = F(y|Mi ,ji), i=1,2,...,N, be the aleatory distribution for y provided by the available models Mi, containing the vector of parameters ji, i=1,2,...,N, which are all plausible in the light of the existing information and knowledge regarding the phenomenon under analysis. Given the available information, it is required to obtain a summary distribution which represents the best description of the behavior of y, in light of the objectives of the analysis. For this purpose, it is common practice to resort to the technical judgments of experts in the field. The experts are required to provide their opinions, within the framework of a given model for the aleatory distribution F(y) of the quantity of interest y.
Theoretically, there are two basic approaches to the problem of aggregating the results coming from multiple models (11). The first one focusses on the plausibility of the various models, i.e. on their likelihood to provide a correct representation of the real system. The other one directs its attention to the accuracy of the predictions of the models, which are considered as devices providing information helpful for the estimation of the quantity of interest. We refer to the first approach as the model-focused or alternate hypotheses approach and to the second as the prediction-focused approach. Both approaches are consistent with the Bayesian interpretation of probability and theoretically lead to the same results, as shown in Ref. 11. However, in practice the results obtained with the two approaches tend to differ due to the assumptions and simplifications introduced.
Given (M,j) e {Mi,ji}, where {Mi,ji} represents the set of models and corresponding parameter vectors, in the model-focused approach, the quantity of interest y(x) is obtained by looking at the likelihood with which each model M and vector of parameters j is "correct". The joint density function y(M,j) which expresses the analyst's beliefs regarding the numerical values of the parameters and the physical validity of the model assumptions can be expressed as y(M,j) = p(j|M)p(M), where p(j|M) is the probability distribution function (pdf) of the parameter vector j, conditional upon model M, and p(M) is the probability expressing the analyst's confidence in the set of assumptions underpinning model M. By applying the theorem of total probability, we can assess the unconditional aleatory distribution F(y) as Eq. (1)
The alternate hypotheses approach, (Eq. 1), is intuitive and consistent with the theory of probability and allows for Bayesian updating of its terms, upon collection of additional evidence. However, the interpretation of the quantity p(Mi) as the probability that model Mi is correct, or that its underpinning assumptions are valid, has raised much controversy from both a philosophical and a practical point of view. Although other interpretations have been proposed, two key issues are of fundamental importance here. First, it must be clear that probability is interpreted as a degree of belief: we accept the primitive notion of likelihood and use probabilities simply as numerical measures of our beliefs regarding the likelihood of a certain event. Second, as previously noted, the conditions under which we can declare that a model provides a good approximation to the real behavior of the quantity of interest depend necessarily on the objectives of the particular analysis and the corresponding level of precision required, so that we should re-write the model probabilities as p(Mi|O), where the objectives O are expressed explicitly or, as is often the case, implicitly ("everyone knows about them"). The dependence of the analysis on its objectives is a concept familiar to all analysts, so that we continue our presentation without explicitly expressing this dependence. The two issues just pointed out should provide insights on the meaning to be given to p(Mi)

and its assessment in practice. We also note that the average F(y) (over all epistemic uncertainties) is not necessarily the only information that a decision maker would rely on. The full distribution, which contains the complete information on the assessed uncertainty but still leaves unsettled the question of how a decision ought to be made, should in general be reported to the decision maker who may then choose his own summary measures upon which to base the decisions.
The alternate hypotheses approach is based on two fundamental assumptions: mutual exclusiveness and collective exhaustiveness of the set of models. While the first assumption can be often accepted in practice, the second one is harder to accept for it requires that a perfect model not only exist but it also be one of the N models considered. In general, the complexity of the phenomena is such that the list of plausible models considered is necessarily incomplete. Moreover, as noted earlier, progress in understanding the physical laws underpinning the process under analysis and the increasing computational capabilities are such that models evolve in time. One possible modification of this approach, which addresses the problem of exhaustiveness, demands that p(Mi) be considered as the probability of model Mi providing the best approximate description of the process within the objective of the analysis, and then allows for a correction factor in Eq. 1 which takes into account the possible existence of other descriptions not considered within the N models. In other words we would write
Eq. (2)
where the superscript o stands for other. Although this formulation seems to resolve in principle the problem of exhaustiveness, it still leaves open many issues from the practical point of view regarding the definition of the alternate hypotheses Mo and the assessment of Fo(y) , together with the associated probability p(Mo). Notwithstanding these difficulties, this approach, in the form of Eq. 1, has been used in the past in various fields of application supported by a somewhat implicit assumption of exhaustiveness. A typical example of application of this approach is given in Ref. 5. In this work, the concept of probability trees is used to decompose the problem into its underlying physical assumptions, and expert judgments are formally elicited to assign probability weights to the different assumptions constituting the branches of the tree. Distributional characterizations of cancer risk from exposure to chloroform are constructed by computing the risk estimates for all possible models represented in the tree as combination of branches, each complete pathway through the tree corresponding to one model. Each expert's distributional characterization of risk is then computed by means of the weighed sum of Eq. 1 in which the output of each model is weighed by the probability that the expert assigned to that model, this latter probability being obtained by multiplying the probabilities assigned to each branch along the complete pathway in the tree. This approach was used to construct a risk distribution for each expert participating in the project. The set of distributional risk characterizations obtained from the experts can then be combined in a composite risk measure reflecting the views of the entire group of experts. Similarly, in the reactor safety example of Ref. 6, expert judgment was used to assess how likely each mode of vessel breach is, i.e. p(Mi).
Concerning the prediction-focused approach, a simple framework previously employed is provided by the so called reference model approach which focuses attention on one single model of the process under study and aims at appropriately modifying its output so to allow for the inherent uncertainty in the model description (10). The appropriate modification is to be assessed with the available information which in most practical situations relies on expert judgments regarding the validity of alternate hypotheses and the reliability of the associated models.
Suppose that one single model M* is selected as reference to describe the process. Even though this model may represent the best and most advanced technology, there still are significant uncertainties associated with it. In other words, this model alone does not meet the objectives of the analysis, in a sense that its output gives only an approximate description of the real process and a modification is in order to account for the associated uncertainty. To formalize this situation, we introduce a factor D*=D*(x), which may be additive (D*a) or multiplicative (D*m), so that our assessed value for the quantity y can be written as
Eq. (3a)
Eq. (3b)
The factor D* is in general unknown and the uncertainty associated with it can be

represented in the form of a distribution g(D*). It is important to note that the uncertainty in D* could be of the epistemic type only or both epistemic and aleatory. In the first case D* simply represents the systematic bias of the model prediction and the uncertainty in its numerical value is strictly related to our lack of knowledge which could in principle be eliminated with a single observation of y. In the second case, the model bias D* itself exhibits aleatory variability, due to some random effects which have been neglected in the model M*. In this case, a single observation cannot eliminate the uncertainty and actually a sequence of observations, repeated under seemingly identical experimental conditions, would lead to different values. D* is then described by an aleatory distribution whose parameters are uncertain due to the epistemic uncertainty in their values. For example, in the additive scheme the distribution F(y) is given by,

Eq. (4)

where z(y*,Da* ;j) is the joint aleatory distribution of y* and D* with parameters jand w(j) is the epistemic distribution representing the state-of-knowledge uncertainty in the parameters j. A corresponding expression can be derived for the multiplicative version of the approach.

The general formulation of model uncertainty provided by Eqs. 3a-b has also been employed in other contexts. In the fire risk assessment area, the actual time to damage Td of an object can be considered to be the product of its deterministic reference model (drm) prediction, Td,drm, and a correction factor D which accounts for the inadequacy of the reference model, viz. Td = Td,drm D. In earlier applications, this factor was treated as deterministic with an epistemic probability distribution function reflecting the analysts uncertainty regarding the amount of systematic over-or underestimation of the damage time by the deterministic reference model (12). In a subsequent analysis, it was recognized that D itself may be an aleatory variable and its aleatory distribution was introduced in the form of a lognormal with parameters m and m (13). The epistemic uncertainty is in this case described by a pdf over the parameter vector (m,s), which can be updated using Bayes' theorem.

In seismic risk analysis, a similar formulation is applied for estimating the model uncertainty of predicted ground motion for future earthquakes (14). In this case, the data coming from recorded earthquakes can be exploited to quantify the goodness-of-fit of simulations of the recorded earthquakes by considering the differences in the response spectra of the observed and simulated ground motions. The natural logarithm of the average horizontal spectral acceleration, lnSAij for the j-th station and the i-th earthquake is given by lnSAij (f) = lnSAcij (f) + m(f) + eij(f) where f is the frequency, superscripts o and c refer to observed and calculated quantities, m(f) is the model bias and eij(f) is the error term, assumed to be a normally distributed random variable with zero mean and variance equal to m(f). In this case study, SAcij(f) represents the output of the model and it is corrected by a multiplicative factor Dijm(f) = exp[m(f) +eij(f)], to account for the inaccuracy of the model description. The correction factor Dijm(f) contains both aleatory and epistemic uncertainty, since it is meant to account for "modelling" uncertainties (differences in the actual physical process and the numerical simulation), as well as "random" uncertainties (detailed aspects of the earthquakes source and wave propagation that cannot be modelled deterministically on the basis of the current state of knowledge). In this particular application, the availability of the data allows one to resort to classical statistical techniques for the assessment of modelling, as well as parametric, uncertainty. In a more general case, the definition of these uncertainties will result from all available data, including information provided as experts' opinions, especially with regards to model uncertainty.

Finally, we conclude this section by comparing the two approaches presented so far, in the case of deterministic models providing predictions of the quantity y. For simplicity, we refer to the case with no parameter uncertainty, although the considerations that will follow can be easily generalized to the case with parameter uncertainty. In this case, Eq. 2 can be re-written as

Eq. (5)

We now observe that Eqs. 5 and 3a provide a very similar formulation for the model uncertainty. Intuitively, it is easy to see that the quantity yo p(Mo) in Eq. 5 has the same basic objective of the correction factor Da* in Eq. 3a and that is, to account for the inaccuracies and inadequacies possibly present in the description

provided. Also, it appears reasonable that this quantity contains uncertainty which could be epistemic or both epistemic and aleatory, and the same considerations made for D* apply. Hence, if we are willing to consider our single model M* as actually a "supermodel" which appropriately combines the output predictions of N submodels as in the summation term on the right-hand side of Eq. 6, and if we agree on accounting for possible "leftouts" through an additive correction factor Da* = yo p(Mo), then Eqs. 5 and 3a are formally identical.

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STRATEGY UTILIZED FOR ASSESSING BASELINE RISKS TO HUMAN HEALTH FROM K-65 AND METAL OXIDE RESIDUES STORED AT THE FERNALD SITE*

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ABSTRACT
The U.S. Department of Energy (DOE) is responsible for cleanup activities at the Fernald Environmental Management Project (FEMP) site in southwestern Ohio. The 425-hectare (1050-acre) site consists of a former 55-hectare (136-acre) Production Area, an adjacent Waste Storage Area and various support facilities. From 1952 until 1989, the FEMP processed uranium into metallic "feed" materials for other DOE facilities in the nation's defense program. In accordance with the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), the FEMP site is currently listed on the National Priorities List (NPL). To facilitate an expeditious cleanup effort, environmental issues associated with site cleanup are being managed under five operable units.
This paper summarizes the risk assessment strategy employed to determine baseline human health risks associated with K-65 and metal oxide residues currently stored in Operable Unit 4. The K-65 and metal oxide residues were generated during the 1950s as a result of the extraction of uranium from uranium-bearing ores and concentrates. These residues are currently stored within Operable Unit 4 in concrete silos. Silos 1 and 2 contain approximately 6,120 cubic meters [m3] (8,005 cubic yards [yd3]) of K-65 residues, while silos 3 contains approximately 3890 m3 (5,080 yd3) of cold metal oxides. These concrete silos are beyond their design life and require remedial action. The risk assessment conducted for Operable Unit 4 constitutes the first detailed human health risk assessment to be approved by the Environmental Protection Agency (EPA) for the CERCLA clean-up effort at the FEMP Site. This paper discusses the FEMP's use of a Risk Information Quality Objective process in concert with the traditional risk assessment approach to determine baseline risk to human health and the environment posed by Operable Unit 4. A summary of the baseline risks to human health is also presented.
DESCRIPTION OF PROBLEM
As part of the process of developing remedial action objectives for Operable Unit 4, an assessment of the baseline risks posed to human health and the environment, by the contents of the silos and the contamination in the surrounding soil, was conducted. Given the concern about the stability of the silos, and the fact that Silos 1 and 2 contain radium-bearing residues (also referred to as K-65 residues), and Silo 3 contains thorium oxides, the evaluation of the potential risk, to human health and the environment associated with these materials was necessary. Extensive sampling conducted during the remedial investigation (RI) revealed that the major radionuclides in the K-65 residues in Silos 1 and 2 include in excess of 3,700 Curies (Ci) of Radium (Ra)-226, 600 Ci of Thorium (Th)-230, 1,800 Ci of Lead (Pb)-210, and more than 28 metric tons of uranium. (1) The K-65 residues also contain significant amounts of barium, lead, arsenic, and polychlorinated biphenyls (PCBs). The K-65 residues also generate several million picocuries per liter (pCi/l) of radon (Rn)-222. Silo 3 metal oxides contain significant concentrations of radionuclides from the uranium decay series with the predominant radionuclide being Th-230 in excess of 450 Ci. The metal oxides also contain several metals including arsenic, cadmium, chromium, and selenium. Radionuclide contamination, consistent with the contents of the silos, was also detected in the Operable Unit 4 surface soil, subsurface soil, and perched water.
BASELINE RISK ASSESSMENT APPROACH
The baseline risk assessment evaluates the risks to human health and the environment in the absence of remedial action. The Risk Assessment Work Plan Addendum established the Operable Unit 4 and the site-wide baseline risk assessment approach for establishing constituents of concern (COCs), developing exposure scenarios, and conducting toxicity assessments. (2) In order to define the level of quality for the required risk information and also to ensure that the risk information developed was sufficient to effectively evaluate a wide range of remedial alternatives, a Risk Information Quality Objective (RIQO) strategy was formulated by the Operable Unit 4

risk assessment project team. The RIQO process is a very structured approach similar to the data quality objective (DQO) process traditionally used for establishing data collection quality objectives for field sampling programs. (3) The RIQO approach enables the establishment of clear objectives, decisions, impacts, and uncertainties for the overall remedial investigation and baseline risk assessment. This process also helped to establish the bases for discussion and negotiation of key issues with the EPA. As a result,mutual agreement on the risk assessment approach was reach between the site and the EPA. This enabled Operable Unit 4 to effectively determine: 1) whether action is necessary (baseline risk assessment), 2) what action is necessary to reduce risks (feasibility study risk assessment), and 3) the contribution of the residual risk to the entire site (comprehensive response action risk evaluation [CRARE]) (4).

DEVELOPMENT OF RISK INFORMATION QUALITY OBJECTIVES (RIQOs)

Based on the established end use for the risk information, the RIQOs helped to define the level of quality required through the following process:

Step 1.

Problem statement A clear statement of the area of concern (e.g., human health risk, compliance with Applicable Relevant and Appropriate Requirements [ARARs], nature and extent of contamination, contaminant fate and transport, etc.) and an evaluation of the practical limitations imposed by the data collection process was first established.

Step 2.

Identification of a decision that addressed the problem Once the stated concern was established, a decision/question was formulated which enabled the development of a list of alternative actions that addressed the problem. Examples of the types of decision/questions formulated include:

What contaminants are present both in the silos and within the environmental media and at what concentrations?

Where are the contaminants located within the soils surrounding the silos?

What is the migration potential of the various contaminants?

What are the pathways of exposure to people and the environment from these contaminants?

Step 3.

Identification of inputs that affect the decision The next step involved the identification of the specific variables or characteristics to be measured or investigated,in addition to any other information needed to make the decision (e.g. aquifer flow characteristics, or wind speed frequency and direction).

Step 4.

Specification of the domain of the decision This part of the process required a detailed description of the boundaries of the decision including spatial and temporal considerations and in particular those critical to ascertaining the impact of contaminant fate and transport on future land use scenarios.

Step 5.

Development of logic statement The logic statement discussed how the risk information was to be used in the decision process. For the Operable Unit 4 the end use of the risk estimates were used to determine the magnitude of the risk to sensitive receptors and also as a basis for evaluating the optimum approach to risk reduction. The logic statements included those used to screen COCs and to evaluate the upper end of the risk estimates for determining when action is needed and within what media.

Step 6.

Establishment of constraints on uncertainty This step involved placing constraints on uncertainty. Objectives for controlling decision errors were stated as limits on the acceptable probability of making an incorrect decision on the basis of the study findings. The limits on uncertainty were based on careful consideration of the consequences of incorrect conclusions.

Step 7.

Optimize design for obtaining risk information This step addressed mechanisms for optimizing the evaluation, collection, and presentation of the risk information by identifying the most efficient way one can be expected to achieve the desired results given the constraints on uncertainty.

RESULTS OF ASSESSMENT OF OPERABLE UNIT 4 BASELINE RISKS TO HUMAN HEALTH

Determination of Constituents of Concern
The primary source terms for Operable Unit 4 were the contents of Silos 1, 2 and 3, and the contamination in the surrounding surface soil, subsurface soil, and the earthen berms surrounding Silos 1 and 2. The RIQO process was used to first identify constituents of interest. Two statistical tests were then used in sequence to identify COCs: a "location" test (student's t-test to compare the mean of site-related data with the mean of the background data, or a Wilcoxon Rank Sum [WRS] test or Mann-Whitney U-test, a direct corollary to the WRS, to compare the two distributions of rank ordered data), followed by a "95th Percentile Test." The 95th Percentile Test was used to determine if any sample measurement (not the mean, upper confidence limit [UCL], or any other statistical parameters) for a given constituent exceeded the upper 95th percentile for the background concentrations. If so, the test indicated that the site has at least one relatively high concentration and that the constituent should be considered a COC. If either test rejected the null hypothesis, i.e., the distribution of measurements at the site appears to be shifted to the right (to higher measurements) of background, the constituent was considered to be a COC. The constituent was not included as a COC only if both tests indicated that there was not a "significant difference" between the two distributions. Constituents were omitted from the list of COCs if they were: 1) common laboratory contaminants found in concentrations less than 10 times the blank concentrations; 2) essential elements (eg. sodium, magnesium, iron etc.) and known to be non-toxic; 3) chemicals that are ubiquitous in nature (eg. silicon, chloride, etc.) and were inappropriate for hazard analysis; 4) chemicals found at very low concentrations (<1 part per million [ppm]) and known to be non-toxic; 5) chemicals that are identified only as a chemical group (eg. total organic carbon, chlorinated hydrocarbons, etc.) and cannot be properly addressed in a risk assessment; or 6) chemicals that are from off-site anthropogenic sources (autos, local factories, etc.) unless they presented a significant risk.
Exposure Assessment Scenarios
Three land-use scenarios and two source-term scenarios were developed. The land-use scenarios established 1) current land-use without access controls, 2) current land-use with access controls, and 3) future land-use without access controls. No remedial actions were assumed to have been taken, and no members of the public establish residence within the boundaries of Operable Unit 4 for the first land-use scenario. Potential receptors included an off-property resident farmer, a trespassing child, an on-property worker (groundskeeper), and an off-property user of surface water.
The second land-use scenario is similar to the first, except that, it was assumed that the site access restrictions currently provided by DOE are maintained. This scenario further assumed that DOE maintains a site-specific health and safety program to ensure that workers and visitors are properly protected. Potential receptors under this scenario included an off-property resident farmer, a trespassing child, and an off-property user of surface water.
The third land-use scenario included exposure routes that require development time, such as establishing a home and farm where members of the public were assumed to have established residence within the Operable Unit 4 boundaries. Access controls were assumed to be absent and again, no remedial actions have been taken. Hypothetical receptors under this scenario were a reasonable maximum exposure (RME) on-property resident farmer, a central tendency (CT) on-property resident farmer, an on-property resident child, an off-property resident farmer, and an off-property user of surface water.
In addition to the three land-use scenarios, two source-term scenarios were established: the current source-term scenario and the future source-term scenario. The current source-term scenario considered the silos as they exist today. The future source-term scenario assumed complete structural failure of Silo 3, resulting in the spread of its contents to Operable Unit 4 surface soil, and the collapse of the Silo 1 and 2 domes, consequently exposing their contents to the elements and increasing the leaching of their contents by precipitation.
Under the current land-use scenario without access control and under the future land-use scenario, risks were calculated using both the current source-term and the future source-term. Under the current land-use with access control scenario, the future source-term does not apply; the assumption was made that under institutional control of DOE, measures would be undertaken to maintain the current configuration

of the silos and implement mitigative action in the event of silo failure. Thus, under the current land-use with access control scenario, risk was calculated only for the current source-term.
These land-use/source-term/receptor scenarios provided the framework for conducting fate and transport modeling using the exposure pathways and transport mechanisms mentioned above. Exposure point concentrations for all of the COCs were then established. These exposure point concentrations established the amount of each COC to which human receptors could potentially be exposed. Figure 1 illustrates the conceptual model developed to represent the potential exposure pathways and routes for human contact with Operable Unit 4 Silo material. A similar conceptual model was developed for soils, berms, and other environmental media.
Fig. 1.
Toxicity Assessment
Intakes calculated in the exposure assessment were used in conjunction with EPA cancer slope factors to determine the incremental lifetime cancer risk (ILCR). Table I provides a summary of the cancer risks for Operable Unit 4. In accordance with CERCLA guidance, risks greater than 1x10-4 to 1x10-6 were considered unacceptable. Toxicity data were taken from the Integrated Risk Information System (IRIS)5 and the updated Health Effects Assessment Summary Table (HEAST). (6) For chemical toxicants, risk was characterized using dose thresholds or reference doses (RfDs). These values are also developed by the EPA to indicate the potential for adverse health effects from exposure to chemicals exhibiting non-carcinogenic effects. To determine if the exposure levels of Operable Unit 4 constituents may cause adverse health effects, the estimated intake of a particular constituent (calculated from the exposure assessment) is compared to the RfD. If the ratio of estimated intake to the acceptable intake is greater than 1, the site-related intake may cause toxic effects. This ratio is called the Hazard Quotient (HQ). When HQs for multiple COCs are summed, the resultant value is the Hazard Index (HI). Table II summarizes the hazard indices for Operable Unit 4.
TABLE I
TABLE II
Assessment of Uncertainty
Recognizing that uncertainty is a factor throughout the exposure and toxicity assessment process, a qualitative assessment of the uncertainty was done. The sources of uncertainty examined included the analytical data, the values of input variables for the models, the accuracy with which the models represent the actual environment or biological processes, the manner in which the exposure scenarios were developed, and the high-to-low dose and interspecies extrapolations for the dose-response relationships. Table III presents a summary of this qualitative uncertainty assessment and the potential impact and resultant bias imposed on the Operable Unit 4 baseline risk assessment.
TABLE III
CONCLUSIONS
Given the RIQO logic process and the uncertainty in the risk information developed, the results were found to be valuable in supporting the risk management decisions for Operable Unit 4. The risk information was then carried forward to support the development and evaluation of alternatives in the feasibility study. The RIQO logic process provided the basis for ensuring that the risk information carried forward was sufficient to enable site-wide risk management decisions. The methods, models, and cross-communication between operated units on risk information, set the stage for developing cleanup priorities for the site as a whole.
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Session 54 -- Uranium Mining and Remediation
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54-1
INVESTIGATION OF RADIOACTIVE CONTAMINATED SITES FROM URANIUM AND OTHER MINING ACTIVITIES IN THE NEW FEDERAL STATES OF GERMANY
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ABSTRACT
The Federal Office of Radiation Protection, BfS, is sponsoring the project "Radiological Registration, Investigation and Assessment of Mining-Related Contaminations from the Past - Register of Contaminated Sites", for which GRS (Gesellschaft fr Anlagen- und Reaktorsicherheit) is the main contractor. 5649 uranium ore and other mine-related objects were identified initially in 34 areas, totalling approx. 1500 km in the states of Saxony, Saxony-Anhalt and Thuringia. Following registration in the A.LAS.KA. data bank, 5352 of the objects occupying 38 km of the total area were assessed with respect to local dose rates and classified within three groups: Those considered to be irrelevant in the sense of federal task; those possibly relevant because of fulfilling radiological assessment criteria, and those possibly relevant due to the exceeding of other, non-radiological criteria. In all, about 60% of the original object area remains to be investigated.
TASK
Since 1991, on behalf of the Federal Office of Radiation Protection (Bundesamt fr Strahlenschutz, BfS), the Gesellschaft fr Anlagen- und Reaktorsicherheit (GRS) has been the leading organization working on the Project "Radiological Registration, Investigation and Assessment of Mining-Related Contaminations from the Past - Register of Contaminated Sites". Based on information about areas with remains from uranium ore mining and other past mining activities and on aerogamma measurements of the 1980s in the states of Saxony, Thuringia and Saxony-Anhalt, 34 suspected areas totalling 1500 km were defined. The aim of the first project phase was to register and secure all existing information and data about the facilities and remains (the so-called "objects") and to process them for inclusion in a data bank, after which they would be evaluated. The first result (1, 2) was that after the evaluation of the information and documentation from the various sources, more than 5000 objects - heaps, disposal plants, shafts etc - were registered, and a large amount of object-specific data was stored in the data bank. Moreover, areas with high gamma radiation intensity ("aerogamma abnormality areas") not associated with any previously known objects were identified. On-site inspections revealed that the data were incomplete and in part not up to date, thus requiring renewed examination; in general, the data sufficed neither for a characterization of the objects nor for a reliable radiological assessment.
VERIFICATION
In the "verification phase" that followed and which was completed in 1994, all existing data were checked on site with the help of experienced local expert companies, and the data for each object were supplemented by scoping and representative measurements of the local dose rate. Moreover, the aerogamma abnormality areas not associated with mining-related objects were investigated in order to find the reason for the high level of gamma radiation. In addition, the information required for the radiological assessment was compiled over the use of the objects and their environment as well about existing protected objects. The kind of objects and the situation within the suspected areas are shown on digitalized topographical maps on a scale of 1 : 10 000. The know-how of local and regional authorities was considered in the investigations. The assistance given by their representatives proved to be of much value.

A total of 5352 objects covering a total area of approx. 38 km were verified (3). Table I shows the distribution of the various kinds of objects.

WISMUT PROPERTIES

The 297 objects located on properties of the WISMUT GmbH were investigated by the company itself in preparation for making decisions about ecological restoration measures (Environment Register). The procedure followed during these investigations was basically the same as with the Register of Contaminated Sites; thus it was possible to integrate the objects into the data bank.

ASSESSMENT AND CLASSIFICATION

A radiological assessment of the 5352 objects was carried out on the basis of the criteria recommended by the German Radiation Protection Commission (SSK) (4) for areas that were used for storing heaps, and which were then used for other purposes, as well as for the areas that were contaminated by uranium ore mining and which were then also put to different use. In this context, a decision-tree was developed which, in addition to the local gamma dose rate output of 300 nSv/h for the consequent area use, also employs the relation of the local gamma dose rate that is measured on the object and in its surroundings as differentiation criterion. Furthermore, the size and volume, as well as any covering of the object, is considered (5, 6, 7, 8).

As a result of the assessments, the objects were classified in three groups. The objects in Group 1 are considered to be not relevant in the sense of the federal task. Their local dose rate measurements do not differ significantly from those of the surrounding area. The objects in Group 2 are considered to be possibly relevant in the sense of the federal task because of fulfilling the radiological assessment criteria. Objects in Group 2a were regarded as possibly relevant in the sense of the federal task since these objects were found to exceed other, non-radiological criteria.

1017 Shafts and galleries were not included in the classification due to the fact that they were investigated separately.

RESULTS

After the verification program was completed, there were registered 5649 objects in the 34 suspected areas - including the 297 WISMUT properties. The relevant parameters and information are included in their entirety in the data bank "A.LAS.KA." (acronym of the German term 'Altlastenkataster', which means Register of Contaminated Sites) (9). Thus, detailed and up-to-data data on all objects in the suspected areas, as well as a computer-aided possibility of identifying them on digitalized maps, now exists (10).

Beyond the defined suspected areas there are some indications of additional objects which are to be included in future investigations, if there is cause for concern due to the geological conditions or following from the findings of uranium prospecting.

The radiological assessment and classification of the 4335 objects (Fig. 1) shows that according to the current state of the project, 65% of the objects (Group 1 plus shafts and galleries) are irrelevant for the radiation exposure in these areas and are therefore not examined further.

The remaining 35% of the objects cover an area of approx. 34 km; this equals 89% of the total of all object areas originally considered. Compared with the number of remaining objects, the reduction in the size of the object areas that have to be further investigated is small. This results from the majority of Group 1 objects being very small heaps dating back to medieval mining activities. However, the area for future investigations is reduced by another approx. 11 km since investigations have already been carried out within the framework of other projects at 6 mill works in Saxony-Anhalt (approx. 5 km) and in a mining area in Saxony (approx. 6 km). There is, therefore, the need for further investigation of an area of approx. 23 km - approx. 60% of the original object area - in the form of measurement programs.

These site-specific measurement programs have already been carried out in Helbra (Saxony-Anhalt), Dittrichshtte (Thuringia) and in Lengenfeld, Aue- Hakenkrmme, Freital and Gottesberg/Schneckenstein (Saxony). These programs determine the kind, extent and magnitude of the possibly radioactive contaminations. As to the currently continuing measurement programs (Annaberg-Buchholz and Johanngeorgenstadt in Saxony) (7), and the ones planned for the future, the results of the verification program and classification of objects form an important basis for the definition of the investigation areas.

The causes could be identified for most of the aerogamma abnormalities, unassociated

with mining-related objects, that cover a total area of 22 km. To establish the causes in that small region of the aerogamma abnormality areas where so far no causes could be identified, further investigations are planned.

SUMMARY AND OUTLOOK

The results of the verification show that in the mining areas of Saxony, Thuringia and Saxony-Anhalt there is no large-scale radiation exposure exceeding natural regional levels. Investigations are therefore only necessary for small segments of the originally defined suspected areas. The kind and extent of the investigations required are determined and optimized on the basis of the verification results. During the course of the activities, furthermore, it turned out that there were also some individual objects related to uranium ore mining and other past mining activities found outside the defined boundaries of the suspected areas. These mainly concern areas in Saxony-Anhalt and Saxony, especially the 'Erzgebirge' region and mid-Saxony. An examination is currently going on as to whether or not these objects have to be included in the investigations.

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54-2

REGISTER OF CONTAMINATED SITES - A PROJECT FOR RADIOLOGICAL INVESTIGATION AND ASSESSMENT OF CONTAMINATED SITES IN THE NEW FEDERAL STATES OF GERMANY

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ABSTRACT
According to the Radiological Protection Precaution Act, the investigation of the radioactivity in the new federal states of Germany, resulting from mining activities in areas of increased natural radiation, was defined as a federal task. The Federal Ministry of Environment, Nature Conservation and Nuclear Safety (BMU) gave the responsibility to Federal Office for Radiation Protection (BfS). For the implementation of the necessary tasks the project "Radiological Registration, Investigation and Assessment of Contaminated Sites from Past Mining Activities (Register of Contaminated Sites)" was initiated.
On the basis of the obtained results from paper work, site inspections and targeted measurement programs, recommendations founded on radiological evaluations can be given with regard to the need for ecological restoration of the investigated areas and objects.

FEDERAL TASK
When Germany was reunified four years ago, the Federal Government found itself confronted in wide areas of the new federal states with the legacies from uranium ore mining. The exact kind, number and distribution were unknown, and the population in these areas was deeply worried about the possible negative effects these legacies might have on their health, on the environment and on the economy. The Unification Treaty (1) as well as the correspondingly amended German Radiological Protection Precaution Act (2) therefore fixed the urgent need for the Federal Government to clarify the situation as follows:

"The investigation of environmental radioactivity stemming from mining in areas of natural radiation is the task of the Federal Government. The authority responsible is the Federal Office for Radiation Protection."

Although the written text of the law does not specify explicitly the kind and the scope of the governmental activities which the Federal Office for Radiation Protection (BfS) has to implement, they can be derived from the Radiological Protection Precaution Act:

1. In the legal sense of radiation protection, the task is to investigate an existing situation which can only be changed by intervention. No binding limit values of any legal regulations concerning prospective, plannable radiation protection measures can be applied directly; instead, they have to be replaced by reference levels.
2. The main focus of the investigations is to be on those mining legacies (hereafter referred to as "contaminated sites") where the parties that are responsible have ceased to exist or cannot be held responsible any longer. Any mines in operation and companies of the uranium mining industry (Wismut GmbH) fall under the supervision of the state authorities in their function as regulatory bodies for radiation protection.
3. The federal task is not only limited to the sites contaminated by uranium ore mining and processing but also includes sites of other mining activities that are suspected to be contaminated by radioactivity.
4. The federal task is to perform an investigation of the radiological situation over a wide area and to consider it as a national problem. It is thus not only limited to a mere analysis but also includes an assessment that is as thorough as it is required for a corresponding radiological classification of areas that are suspected to be contaminated and of relevant mining-related objects (buildings, plants, heaps, areas etc. will hereafter be referred to as "objects"). The task is also oriented on the need, aims and priorities of action. The decisions about ecological restoration and its objectives, the preparation of appropriate projects and expert analyses as well as the actual performance of ecological restoration measures are not part of the federal task. The tasks of the authorities that are responsible for carrying out radiation protection measures within the framework of ecological restoration are not affected by the government task.
5. Furthermore included are the tasks that are related to the methodology of the investigations as well as to the documentation and information about the mining-related environmental radioactivity.

FEDERAL PROJECT

Organization

In mid-1991 the BfS, financed and supervised by the Federal Ministry of Environment, Nature Conservation and Nuclear Safety, initiated the project "Radiological Registration, Investigation and Assessment of Contaminated Sites from Past Mining Activities (Register of Contaminated Sites)" to deal with the federal task. The Gesellschaft fr Anlagen- und Reaktorsicherheit (GRS) was commissioned by the BfS to coordinate the sub-contractors (mostly regional companies) that are involved, to develop and maintain the central data bank about mining-related environmental radioactivity, and to perform itself some technical work within the framework of the project. The German Radiation Protection Kommission (SSK) is taking part in the scientific assessment of the situation; it is constantly kept up to date with the results of the investigations. Regional, district and local authorities are involved wherever necessary. The investigation results are made available to them for decision-making. In all, this form of organization of the project, which is exceptional in its kind, scope and importance, has proved to be effective and has led to close cooperation among the parties involved.

Structure of the Task

Before work was begun, a project structure plan was worked out (Fig. 1) which served as a guideline for the working progress and the coordination needs with the sub-contractors as well as for drawing up a time schedule. Beside the commissioning and controlling of the sub-contractors it was the task of GRS to build up the data bank (ALASKA) and to supervise the collection of data as an expert consultant.

Fig. 1.

The data that were registered during the historical search mainly came from the evaluation of files and maps. A verification phase from 1992 to 1993 was meant to ensure that the data for all areas and objects were comprehensive, correct and up to date. In this context all the data about the suspected areas and objects were checked by interviewing contemporary witnesses and by local inspections and measurements.

In a number of cases, however, the existing data base after the verification phase is not sufficient for a final radiological evaluation of individual sites. Consequently, measurement programs were planned to be carried out from 1993 onwards. Three pilot projects were already carried out in 1991 and 1992 parallel to the securing of the already available data; they serve as models for site-specific measurement programs. The aim of the pilot projects was, alongside an initial site evaluation, to determine the necessary extent and content of a measurement program and to gather experience from the investigation methods and measurement techniques.

Time Schedule and Finances

A time schedule was developed for the planned duration of the project, and the necessary financial means were calculated. Figure 2 shows the current state (end of 1994) of the work performed and the financial means that have been used up so far.

Fig. 2.

RESULTS

The objective related to the federal task of clarifying the radiological situation in the mining areas on a large scale and the available resources of manpower, finances and time required a staggered procedure from the start. This is true of both the registration and the assessment procedure. The principle aim is to identify as early as possible those areas and objects that were not influenced by mining or which are irrelevant from the point of view of radiation protection and can therefore be excluded from the project's investigations, leaving only those areas and objects which still have to be investigated further to clarify the situation.

Definition of "Suspected Areas"

To start with, the areas had to be defined in which mining-related contaminations with higher than normal radioactivity were known or suspected to exist. These were mainly the areas involved in the uranium industry, but also some disused mining areas of the Erzgebirge mountain region (copper slate and black coal). Other areas that were included were those with abnormal gamma radiation levels which had been detected in the 1980s by measurements from the air.

Based on these activities, 34 "suspected areas" totalling up to 1500 km were defined in Saxony, Thuringia and Saxony-Anhalt as priority investigation areas. Concerning a radiological assessment this means that radiologically relevant objects and contaminated stretches of land outside these areas will, if ever, only have to be

expected in very few places and will have a considerably lower level of contamination.

Evaluation of Existing Documentation

The existing documentation, especially for the suspected areas, was obtained from the authorities, companies and other institutions of the former East Germany and was evaluated with regard to the information that was relevant for the project's objective. The result was that approx. 4800 mining-related objects, of which an approx. 4000 were heaps, were registered; their data (coordinates, size a.o.) were collected in the specially created data bank as far as they were available. The registered objects covered an estimated total area of approx. 60 km and the areas with abnormal gamma radiation levels one of approx. 170 km.

The radiological assessment in this project stage led to the result that for the areas outside the objects there were no indications of any mining-related and thus also of no radiologically relevant contaminations, so that these areas could therefore be assumed to be not radiologically contaminated.

Verification

During the following project stage, which was mainly carried out in 1993, the registered objects in the suspected areas were inspected on site, and scoping measurements of the local dose rate of the gamma radiation were performed on these objects and their immediate surroundings ("verification"). The verification phase yielded the following results:

- detailed and up-to-date data about all suspected sites and objects,
- details about the situation and the size of the areas and objects,
- a supplementation of the data bank by additional objects that had not been registered by the historical search,
- comprehensive photographic documentation for all investigated sites
- site-specific results of the radiological control measurements, and
- indications to further objects outside the suspected areas, which are included in the further investigation if they prove to be relevant, e.g. due to geological conditions.

The data bank entries were corrected and supplemented if required. A total of 5352 objects within the suspected areas were investigated with a total area of about 38 km.

Radiological assessment was carried out on the basis of criteria that were derived from SSK-principles and recommendations (3). The verification results as well as the kind, scope and application of the criteria derived from the verification for the radiological assessment of the contaminated sites are explained in another paper (4).

The results of the project stage are handed over to the governments of the affected federal states in the form of the ALASKA data bank whose data and user program have been brought up to date; it is explained to the authorities and comes complete with digitalized maps, with the objects and abnormality areas marked on them, and a summary report. After that, district administrations will be given the material as well as advice and guidance. Information brochures were produced for the general public.

Measurement Programs

After the large group of objects, which as a result of the verification have been classified as "not radiologically relevant", has been excluded, the remaining objects are investigated by considering the local relevance that has been allocated by the regional administrative bodies. The horizontal and vertical expansion of the contaminated areas is defined exactly in detailed measurement programs ("basic measurement programs"); the kind, concentration and inventory of radioactive materials is determined and the influence of the radiation exposure on these areas and their environment is estimated in order to achieve a precise radiological classification of the objects for an assessment of the problem by the Federal Government as well as for a basis for decision-making for state, regional and local governments and for other land owners. These investigations have been going on since 1993. The programs rest on the experience gathered during three previous pilot projects that were carried out at representative contaminated sites. Measurement programs of this kind were already finished in the areas Aue/Hakenkrmme, Freital/Gittersee, Dittrichshtte, Hettstedt and Gottesberg/Schneckenstein and are still going on near Annaberg-Buchholz and Johanngeorgstadt (Fig. 3). Further programs are being prepared. All relevant sites are to be analyzed and assessed by

1996/97. All measurement data are put into the ALASKA data bank.
Fig. 3.
On the basis of the already mentioned recommendations an assessment scheme was developed which helps defining the objects and areas that have to be considered as "radioactive contaminated". It follows from the results of the measurement programs that only a small part of the investigated areas actually has to be regarded as radioactive contaminated. Remediation measures for these partial areas will have to be initiated for reasons of radiation protection if the result of a site-specific investigation shows that this object, in its current state and under consideration of its actual use, causes an annual radiation exposure of more than 1 mSv. For the estimation of the exposure the BfS has drawn up the calculation basis with generic parameters which comply with the specific character of the problem through adaptation to official guidelines.
In the course of the investigations some limited areas were detected that had considerably high contamination levels (e.g. local gamma dose rate 1 mSv/h and higher) so that immediate measures must be considered. In such cases the state authority responsible for the implementation of radiation protection measures is informed immediately.
SUMMARY AND OUTLOOK
In the starting phase of the project "Radiological Registration, Investigation and Assessment of Contaminated Sites from Past Mining Activities (Register of Contaminated Sites)" decisive progress could be made in the estimation of the radiological situation in the uranium ore and other mining areas. After a historical investigation the existing data were initially checked as to their correctness and completeness in a verification phase by means of on-site investigations carried out by local firms. This was done in order to put the investigation need in defined terms. In a parallel step, an examination of the available information is carried out in cooperation with the state environment ministries and the local decision-making bodies. However, this state of knowledge does not suffice to determine the actual need for ecological reconstruction of the areas and sites of former uranium ore and other past mining activities. For preventive public health protection reasons and in order to support the economic development in the regions concerned, site-specific measurement programs were carried out.
On the basis of these measurement programs, recommendations founded on radiological evaluations will be given with regard to the need for ecological restoration of the investigated areas and objects.
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54-3
ABANDONED URANIUM MINE RECLAMATION FOR DOE URANIUM LEASE MANAGEMENT PROGRAM
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ABSTRACT
The U.S. Department of Energy Grand Junction Projects Office has developed a process for the reclamation of abandoned uranium mine sites located on lands under its administrative jurisdiction. This process allows mine sites to be reclaimed in a cost-effective manner while maintaining a practicable approach to the presence and cleanup of naturally occurring radioactive materials resulting from past mining activities.

OVERVIEW

The U.S. Department of Energy (DOE) Grand Junction Projects Office (GJPO) currently administers 43 uranium lease tracts located in southwestern Colorado (38), southeastern Utah (4), and northern New Mexico (1) consisting of approximately 10,000 hectares of land (see Fig. 1). These lands were withdrawn from the public domain, from 1948 until 1954, by the U.S. Atomic Energy Commission (predecessor agency to the DOE) to develop a source of domestic uranium ores for defense purposes. Subsequent to withdrawal, these lands were included in the Uranium Leasing Program. All of the mines were abandoned and many of the mine portals were temporarily closed when the Uranium Leasing Program ended in 1962. However, little else was done to reclaim the environmental disturbances resulting from the mining activities that occurred during this initial leasing period. As part of the present leasing program, DOE-GJPO adopted a policy for the reclamation of these undesirable environmental conditions.

RECLAMATION PROCESS

DOE-GJPO developed a process for the reclamation of abandoned uranium mines located on lands under its administrative jurisdiction. DOE's primary concerns for the reclamation of abandoned uranium mine sites are

To eliminate all physical safety hazards that have resulted from previous mining activities, including all mine openings and portals and all surface depressions that contain severe vertical drops of more than 1 meter.

To recontour the areas of disturbance to blend in with the natural topography surrounding the site while allowing as much vegetation regrowth to survive, as practicable.

To redirect all storm water away from the immediate areas where mine openings have been closed to eliminate the possibility of water flow and erosion into the mine workings and to contain and control all storm water that contacts the site.

To reduce the potential for the general public's exposure to radiological materials, as much as practicable.

ELIMINATE ALL PHYSICAL SAFETY HAZARDS

DOE-GJPO will use methods approved by the U.S. Bureau of Land Management (BLM) for permanent closure of all mine openings and portals (shafts, adits, inclines, and vent holes). These methods include man-made bulkheads or backfilling with rock (waste-dump materials) from the mine site. Selective placement of material within the mine opening and portals, according to size, will enhance slope stability and preclude future sloughing of the materials after closure is completed. After closure, all depressions surrounding or immediately adjacent to the mine openings and portals and all shallow open pits with severe vertical drops will be backfilled with additional waste-dump materials located at the site and then will be slightly mounded to preclude future subsidence.

RECONTOUR AREAS OF DISTURBANCE

After the mine openings and portals are successfully and permanently closed, all areas of disturbance will be recontoured to blend in with the natural topography, as practicable. This recontouring includes

Reducing the slopes of the remaining waste-dump materials to less than a 3 to 1 ratio, while providing a basin effect on the top of the waste-dump materials to collect and contain all storm waters that contact the site.

Removing waste-dump materials from the existing natural drainages, as practicable, and eliminating the potential for these materials to enter the drainages in the future.

Recontouring the immediate areas to provide an undulating surface that closely represents the natural surrounding topography.

Any area of disturbance that has revegetated naturally during the past several decades will not be redisturbed unless physical hazards exist that must be addressed. In either case, as much of the existing vegetation regrowth will be left (as practicable) during the reclamation efforts. Any topsoil that was previously stockpiled on the site or is in the immediate vicinity of the site will be spread across the disturbed areas to promote revegetation. If topsoil has not been previously stockpiled or is not available within the disturbed area, it will not be mined at another location and hauled in. A second area of disturbance will not be created to enhance the reclamation efforts or success of the initial area. All disturbed areas will then be reseeded with a BLM-approved mixture of native grasses and shrubs developed for the specific climatic conditions.

REDIRECT ALL STORM WATER

Several efforts will be undertaken to control the effects that storm waters have on the abandoned mine site. First, all mine entrance locations will be backfilled sufficiently to provide positive drainage away from the entrance area. Second, the waste-dump materials will be recontoured to create a basin on top of the waste-dump materials that will collect and contain all storm waters that come in contact with it. The basin will be constructed to create minibasins within the major basin feature to promote individual collection points that could enhance the revegetation efforts. Storm waters collected in the basins will be allowed to evaporate from or percolate through the basin. Third, all major drainage features leading onto the mine site will be diverted around the site or stabilized in place to reduce the overall effects that major storm-water events could have on the reclamation efforts and the abandoned site in general. The last step will be to place silt fences or other suitable devices within and across the drainages leaving the mine site to preclude the transport of sediments from the site to downstream locations.

REDUCE POTENTIAL EXPOSURE TO RADIOACTIVE MATERIALS

All abandoned uranium mines have naturally occurring radioactive materials (NORM) associated with them in the form of 1) low-grade materials that are uneconomic to transport and mill or 2) residual ore-stockpile materials. Outcrops of uranium-bearing formations often exist at the mine sites and represent the initial uranium discoveries that led to the mines' development. These radioactive materials are a potential source of radiation exposure to the general public and the environment. However, it is often cost prohibitive, if not entirely impossible, to eliminate these NORM sources from the environment.

It is DOE's policy to minimize the potential exposure to the general public resulting from mining-related radioactive materials. During reclamation activities on DOE lease tracts, the following measures will be taken to reduce or limit the potential for exposure to NORM:

A cursory radiological scan of the entire mine site will be conducted during the preliminary reclamation activities to identify those areas of greatest radiological concern.

All materials within the areas of greatest radiological concern will be used for the initial backfill of the existing mine openings and portals to place these radioactive materials below ground surface, leaving less-radioactive materials exposed at the ground surface.

Additional scans will be conducted during the entire reclamation process to locate other areas of radiological concern for control or burial. After final recontouring is completed, the potential exposure from these areas of radiological concern will be further minimized by covering the disturbed areas with any immediately available topsoil.

STAKEHOLDER INVOLVEMENT

Numerous Federal and State agencies are designated as potential stakeholders in the reclamation of the abandoned uranium mine sites located on the DOE lease tracts:

U.S. Department of InteriorBLM
- Colorado State Office, Denver, Colorado
- Grand Junction District Office, Grand Junction, Colorado
- Uncompahgre Basin Resource Area Office, Montrose, Colorado
- San Juan Resource Area Office, Durango, Colorado
- San Juan Resource Area Office, Monticello, Utah

U.S. Department of InteriorBureau of Mines, Denver, Colorado

U.S. Department of InteriorNational Park Service (NPS); Mining and Minerals Branch, Denver, Colorado

U.S. Department of AgricultureForest Service (USFS)
- Manti-La Sal National Forest, Price, Utah
- Manti-La Sal National Forest; Monticello District Office, Monticello, Utah

U.S. Environmental Protection Agency (USEPA)
- Radiation Studies Division; Radiation Studies Branch, Washington, DC
- Region 6, Dallas, Texas
- Region 8, Radiation Programs Branch, Denver, Colorado
- Region 9, Radiation Programs Branch, San Francisco, California

Colorado Department of Public Health and Environment; Radiation Control Division

Colorado Department of Natural Resources; Division of Minerals and Geology

New Mexico Environment Department

New Mexico Energy, Minerals, and Natural Resources Department; Mining and Minerals Division; Mining Act Reclamation Bureau
Utah Department of Environmental Quality; Division of Radiation Control
Utah Department of Natural Resources; Division of Oil, Gas, and Mining
Wyoming Department of Environmental Quality

A copy of DOE's proposed reclamation process was transmitted to each agency, along with a solicitation for review comments. To date, comments have been received from five agencies:

Wyoming Department of Environmental Quality
Colorado Department of Public Health and Environment
New Mexico Environment Department
BLM, Grand Junction Resource Area Office
USEPA, Radiation Studies Branch

The majority of comments received focus on two subjects: 1) endorsement of DOE's intentions and the general aspects of the proposed process and 2) the continuing concern for adequate protection of the general public and the environment from the effects of residual radioactive materials at the mine sites.

FIELD IMPLEMENTATION AND REVIEW

DOE recently completed reclamation activities on the first phase of abandoned mines to be reclaimed by this process. These activities include work at multiple shallow subsurface mine sites (less than 30 meters below ground surface) and shallow surface mine sites (less than 6 meters deep) on four lease tracts located in southwestern Colorado near the communities of Slick Rock (C-SR-14A, C-SR-16, and C-SR-16A) and Uravan (C-CM-24).

The lease tracts are accessed by Colorado State Highway 141 and several county roads. The regional topography consists of relatively flat-lying mesas that are broken up by the canyon features of the Dolores and San Miguel Rivers and their respective tributaries. The region is primarily vegetated with stands of pinyon pine and Utah juniper intermixed with open grass- and sagebrush-covered parks. The region has an arid climate with an annual precipitation of 30 to 40 centimeters. The primary land use in the lease-tract areas historically has been mining and grazing, although some dryland agricultural crops are grown adjacent to Lease Tracts C-SR-16 and C-SR-16A. Hunting is the only recreational use of these lands. The mine sites within the lease tracts are located at elevations between 1,700 and 2,200 meters above mean sea level. The uranium-bearing formation of interest in the region is the Salt Wash Member of the Morrison Formation. By the very nature of the geologic, hydrologic, and climatic conditions of the region, none of the mines associated with these lease tracts encountered groundwater; consequently, groundwater was not an issue in the reclamation of these mine sites.

The mine sites on Lease Tracts C-SR-14A and C-SR-16A were all similar because the ore horizon or zone of mineralization was shallow, if not exposed at the surface, and was first accessed by open pits. This near-surface occurrence of ore led to extensive surface workings. Later, as the ore-bearing formation dipped further below grade, adits leading from the bottom of the open pits were used to mine the ore. The overburden material and associated mine waste were scattered throughout the different mine sitessome in large dumps, others in small piles from single ore carts. The permanent closure of these mines and the subsequent recontouring activities were achieved with conventional construction equipment (front-end loaders, backhoes, and bulldozers). Mine sites with underground workings were left slightly mounded to prevent storm water from eroding into the underground workings. The shallow open pit areas were left slightly bowled to contain and control storm water.

Several of the mine sites on Lease Tract C-SR-16 were also similar because the ore horizon or zone of mineralization at each site was entirely below grade and was accessed by shallow, gently sloping inclines. These mine sites had mine-specific waste dumps. Two of the mine sites were somewhat unique because they had ventilation shafts. The permanent closure of these mines (including the ventilation shafts) and the subsequent recontouring activities were achieved relatively simply with conventional construction equipment. These mine sites were left slightly mounded to prevent storm water from eroding into the underground workings.

One mine site on Lease Tract C-SR-16 was distinctly unique from all of the other mine sites because it is located along and under the rim of Summit Canyon. This mine site consisted of eight adits located approximately 8 meters below the canyon rim.

During past mining operations, the mine waste materials were deposited along the outer edge of the mine-site access road that traversed along the south slope of the canyon to the different mine adits. The large boulders found above and adjacent to the mine portals along the canyon rim were considered to be extremely unstable, and worker safety became the primary concern during the reclamation of this site. These unstable conditions prompted the use of explosives to close the portals. Vertical holes were drilled along the rim above the portals, and angle holes were drilled into the adit walls from an area directly in front of the respective adits. The holes were then loaded with explosives and detonated. This method was successful at seven of the eight adits. However, the operation had to be repeated for successful closure of the eighth and largest adit. The entire rim area was then recontoured with materials that were available along the surface of the rim. A berm was constructed above and along the rim to divert all storm water away from the reclaimed area. The mine waste dumps were left intact to act as a berm along the canyon wall to control storm-water runoff and limit further erosion of this material into the canyon.

The mine site on Lease Tract C-CM-24 had two unique featuresit was the oldest of the abandoned mine sites reclaimed, and the ore horizon or zone of mineralization was at considerable depth. Mine access consisted of a small, but relatively steep incline that had been closed previously but was now showing signs of subsidence. The incline was permanently closed and recontoured with a conventional backhoe. To prevent storm water from eroding into the underground workings, the mine site was left slightly mounded. The rock-waste dump at this mine site contained substantial vegetation regrowth and was left untouched.

During this reclamation project, a total of 40 mine portals (adits and inclines) and three ventilation shafts were permanently closed with methods approved by the BLM and the State of Colorado (Department of Natural Resources, Division of Minerals and Geology). Approximately 38,000 cubic meters of material (mine waste, overburden, and topsoil) were backfilled into the depressions and open pits and recontoured to blend in with the surrounding topography. After the mine closures and recontouring were completed, the entire areas of disturbance (approximately 10 hectares) were reseeded with a seed mixture, approved by the BLM, that was distributed by mechanical and hand-held broadcast seeders.

Following the successful completion of reclamation activities, an invitation was issued to each of the stakeholder agencies to tour the associated lease tracts and review the results. Several agencies (listed below) participated in the on-site review of reclamation activities. Several other agencies expressed interest in reviewing the reclamation process results but stated that they would be unable to attend.

- BLM, Montrose District Office, Montrose, Colorado
- BLM, Uncompahgre Resource Area Office, Montrose, Colorado
- BLM, San Juan Resource Area Office, Durango, Colorado
- NPS, Land Resources Division, Mining and Minerals Branch, Denver, Colorado

Tour participants visited each mine site and were given a general overview of site activities. The unique situations encountered at the different sites were also discussed. Following the tour, all participants concluded that the DOE's process was practicable for the respective mine sites reclaimed.

CONCLUSION

DOE administers 43 uranium lease tracts with a land mass of 10,000 hectares that collectively contain a maximum of 200 abandoned and active uranium mine sites. In contrast, BLM, USFS, and other Federal and State agencies administer millions of hectares with a multitude of similar uranium mine sites. Currently, no regulations require the reclamation of most of the old, pre-law mine sites on Federal and State lands. However, the physical and environmental liabilities associated with these mine sites may prompt the appropriate administrative agency to address these reclamation issues. DOE believes that the aforementioned process provides a practical and cost-effective approach to abandoned uranium mine-site reclamation.

54-6

NATURAL FLUSHING STRATEGY: A RISK-BASED APPROACH TO MEETING GROUND WATER CLEANUP STANDARDS

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ABSTRACT
Developing and implementing a cost effective and community acceptable ground water cleanup strategy can easily become an insurmountable task. The primary challenge of any ground water cleanup program involves adequate actions that lead to protection of human health and the environment. Depending on the cleanup strategy selected, approaches that eventually lead to a cleanup goal can vary widely with regard to cost and many approaches result in little actual risk reduction.
An alternative to typical ground water cleanup approaches is "natural flushing". Natural flushing is a provision identified by the U.S. Environmental Protection Agency (EPA) for the U.S. Department of Energy's (DOE) Uranium Mill Tailings Remedial Action (UMTRA) Project. The EPA has issued final regulations to correct and prevent contamination of ground water beneath and in the vicinity of inactive uranium processing sites by uranium tailings (60 FR 2854, January 11, 1995). The regulations apply to tailings at 24 locations that qualify for remedial action under Title I of Public Law 95-604, the Uranium Mill Tailings Radiation Control Act (UMTRCA), on November 8, 1978 (42 U.S.C 7901 et seq.). They provide that tailings must be stabilized and controlled in a manner that permanently eliminates or minimizes contamination of ground water beneath stabilized tailings, so as to protect human health and the environment. They also provide for cleanup of contamination that occurred before the railings are stabilized (Fig. 1).
Fig. 1.
A hypothetical model uranium mill tailings site was created to assess the natural flushing strategy in a generic convention. The basic assumptions of the model are 1) the milling process operated for 30 years; 2) the tailings and associated contaminated materials have been relocated; 3) a significant ground water plume exists on-site and down gradient of the site; and 4) the site is located on shallow alluvial deposits that discharge to a nearby river. A rudimentary solute transport analytical model was used to predict if the site would flush to the standards within 100 years.
BACKGROUND & PURPOSE
Natural flushing can be a viable alternative in situations where water use and ecological considerations are not negatively impacted and cleanup will occur within a reasonable time. Further, institutional controls, when enforced by a local, state, or federal government entity can be relied upon for periods of time up to 100 years. Adequate safeguards are provided in the standards to prevent this alternative from being used to circumvent active cleanup of water that is foreseen to be used as drinking water by nearby populations.
Under conditions where active ground water remediation for completely achieving compliance with the standards is impracticable from an engineering perspective, environmentally damaging, or excessively costly, the natural flushing strategy is viable, if contaminated ground water can meet the standards within an acceptable time frame (less than 100 years). This strategy should be considered for sites where the attainment of the ground water cleanup standards can be met through partial or complete reliance on natural processes and no use of the ground water as a source for a public water system exists or is projected. The strategy must also consider the beneficial use of ground water, such as agricultural and industrial uses (Fig. 2).
Fig. 2.
The UMTRA regulations establish a time limit on the extension of the remedial period to limit reliance on the use of institutional controls to manage public access to contaminated ground water. The use of institutional controls is permitted for the natural flushing strategy only when the remedial period can be demonstrated to be less than 100 years. The application of institutional controls would be considered only when enforced by a permanent government entity that can be relied on for periods up to 100 years to provide adequate safeguards to nearby population centers. Natural flushing has the highest potential for success as a cleanup strategy when the contaminated ground water discharges to a surface water body that will not be adversely affected by the contamination.
NATURAL FLUSHING AS A CLEANUP STRATEGY

Using the natural flushing strategy, the hypothetical model site for the flow regime assumes the aquifer's ability to transport solutes is sufficient with respect to time to dilute the source concentrations to below an applicable standard under 40 CFR, 192.

A typical model site for natural flushing will have many or all of the following hydrogeologic attributes: 1) an alluvial depositional environment; 2) high saturated hydraulic conductivity; 3) high effective porosity; 4) matrix diffusion being a minor component to solute transport rates; and 5) the predicted plume within an area not projected for municipal water supply. The dilution phenomenon is a result of advective transport and hydrodynamic dispersion. Advective transport is the process by which solutes are transported by the bulk motion of the flowing ground water at a rate equal to the average linear velocity of the ground water. Coupled to the advective transport phenomenon is hydrodynamic dispersion. This results in the solute spreading out from the path that would be expected solely due to advection, Freeze and Cherry, (1979).

This spreading phenomenon occurs because of mechanical mixing during fluid advection and because of molecular diffusion due to the thermal-kinetic energy of the solute particles. These dilution processes and others are what natural flushing is based upon.

In addition to the processes of dilution and dispersion, geochemical attenuation processes also contribute to the efficacy of natural flushing. Chemical processes can affect the rate at which solutes are transported in the flow regime. Sorption is a well known geochemical phenomenon that includes adsorption and desorption of soluble ions to receptor sites on the surface or between layers of certain fine-grained aquifer matrices. Generally, ion exchange between soluble ions in a uranium mill tailings contaminated aquifer and the aquifer matrices is primarily limited to cation exchange.

There is no universal accepted theory of solute transport that encompasses all of the above phenomena of transport processes that can be used to predict the time rate of change in concentrations (natural flushing) of a contaminated aquifer. But, under certain assumptions, the three-dimensional, unsteady-state, homogeneous, aniotropic, mass transport equation using constant, uniform velocity and dispersion coefficients of a miscible contaminant subject to sorption, first order decay and injection sources R.W. Cleary, (1992) may be expressed in the cartesian coordinate system as: Eq. (1)

In summary, to mathematically derive the solute transport phenomenons for a given flow regime is very complex and almost always has to be represented by averages or ranges, acknowledging some degree of uncertainty. Therefore only basic mathematical assumptions and the construction of a simple analytical mass transport model is used to illustrate natural flushing at a hypothetical uranium mill railings site. In recent years much work has been done on the theories of mass transport in response to the great interest in problems of ground water contamination. One of the outcomes of this has been the development of what is essentially a new branch of subsurface hydrology, where the flow of fluid and solutes is treated by statistical models that can account for the role of varying hydraulic conductivity that accompanies aquifer heterogeneity Fetter, (1992).

As a preliminary evaluation, the analysis of the hypothetical model does not account for aquifer heterogeneity. It must be noted that this phenomena can significantly affect solute transport predictions.

ANALYSIS OF A MODEL SITE: IS IT A CANDIDATE FOR NATURAL FLUSHING?

In order to apply the natural flushing strategy to a given contaminated site, the flow regime should be well understood with regard to solute transport. The most direct and simple way to illustrate the effects of advection, dispersion, first order decay and sorption is through a one-dimensional mass transport analyses in which the results are plotted as breakthrough curves. Breakthrough curves typically plot normalized concentration (C'/Co, where Co is the maximum concentration) vs. aquifer pore fluids. The pore fluids may refer to a laboratory column or an aquifer. For the purposes of this illustration, consider saturated aquifer conditions in which constituents of concern have leaked into the groundwater from the milling and tailings seepage that occurred over a thirty year period.

To illustrate the application of the natural flushing strategy to a model site, a hypothetical uranium mill tailings processing site is described. To typify the model site as representing a Title I UMTRA processing site, common site conditions and

attributes are described. Conditions and parameters described in the generic model are similar to seven UMTRA sites. This screening tool is directly applicable to most of the UMTRA sites targeted for natural flushing. The tailings and other materials such as wind-blown and waterborne tailings, and contaminated soils have been removed and relocated to an engineered disposal cell. This eliminates the most significant source for continued ground water contamination. In addition, relocating the source-term eliminates any potential for transient drainage effects. Some finite amount of transient drainage will occur under unsteady state conditions at some UMTRA sites where the tailings have been encapsulated in-place or on-site. Often times the tailings exist in highly-saturated conditions. Transient effects are the direct result of seepage out the bottom of the cell and consolidation due to loading.

The seepage velocity during the operation of the mill was assumed to be substantial over the life of the operation. The tailings have been removed and the contaminated soils in the unsaturated zone below the footprint of the former tailings pile have also been removed to the water table or assumed not to be a significant secondary source-term contributor.

At the model site, tailings were placed on unconsolidated alluvial deposits and were exposed to the environment for 30 years. The alluvium is 25 meters thick and overlies bedrock characterized as an aquitard. For the purpose of this illustration, the alluvial system is assumed to be homogenous and isotropic. However, in real flow regimes this assumption can lead to erroneous predictions. The unsaturated zone is relatively insignificant, as the thickness is less than 1.5 meters and the wet tailings most likely created a ground water mound beneath the footprint.

The historical seepage that has exfiltrated out of the hypothetical tailings pile can be estimated based on a water budget for the model site and can be expressed as the following generic expression:

Qin = Qout + Qret,

where;

Q in is the amount of water from the processing activities and tailings, plus precipitation that has deep percolated over 30 years

= 9.1 x 105 m3;

Qout is the seepage that has exfiltrated out the bottom of the tailings pile over 30 years

= 8.9 x 105 m3;

Q ret is the volume of water retained in the tailings under gravitational forces

= 1.8 x 104 m3.

The Qin is the cumulative contribution from milling effluent and deep percolation as a result of precipitation. The total volume of milling water discharged with the tailings was approximately 4.6 x 105 m3. Average annual precipitation is 30 centimeters per year. Only 10 percent of the precipitation deep percolates. The area of the tailings pile is 50 hectares. The volume of precipitation that has deep percolated through the tailings footprint over 30 years equals 4.5 x 105 m3 The total volume of dry tailings equals 1.2 x 106 m3. The void space of the tailings equals 40%. The total volume of pore water in the railings, assuming 100% saturation and neglecting evaporation and any transpiration equals 4.6 x 105 m3. On the basis on published data, the amount of water retained beyond the pull of gravity, (Qret) if the saturated tailings pores is estimated to be 40% by volume based on grain size of the tailings. After the milling process had ended, it is estimated that 60% of the pore water can be expected to drain out, leaving 40% as (Qret), which equals 1.8 x 104 m3. The historical seepage discharge (Qout) over the 30 year period equals 8.9 x 105 m3.

The cross-sectional area through which the historical seepage discharge has occurred at the model site can be approximated by the following generic calculation:

A = Qout/[Kh x (H/L)], where;

Qout = past seepage discharge over 30 years = 8.9 x 105 m3

(3 x 104 m3/yr)divided by the horizontal saturated hydraulic conductivity Kh

(1.67 x104 m/yr) times the horizontal hydraulic gradient H/L = 0.0025

A = 7.1 x 102 m2.

If this cross-sectional area includes the entire 25 meter thick saturated zone and if the area is rectangular in shape, the width will be approximately 28 meters. The seepage flux was estimated to be 3.0 x 104 m3/year, which would produce a volume of seepage discharge of 8.9 x 105 m3 over 30 years. Assuming the model site's uppermost

aquifer has an effective porosity (Ne)of 20 percent (based on core analysis), and the seepage discharge travels with the natural ground water average linear velocity (209 m/yr), neglecting dispersion and retardation, the volume of contaminated aquifer at 30 years would be 8.8 x 105 m3. The latter value is equivalent to one aquifer pore volume. This is without regard to concentrations, assumes no mixing and a non-reactive contaminant.

To estimate how long it would take to flush one pore volume of contaminated aquifer, the following generic equation can be used for the model site:

Tpv = Vn/Q, where:

Tpv = time to flush one pore volume;

Vn = volume of contaminated aquifer and n is effective porosity

Q = Darcian strip discharge = cross sectional area at discharge point times hydraulic conductivity times horizontal gradient divided by effective porosity.

Tpv = 6.2 years* to flush one contaminated aquifer pore volume. * based on advection only

Advection is mathematically described by the following term: (V). Advection involves transport by bulk movement and is the primary mechanism responsible for contaminant migration in aquifers. The average linear velocity (Vs) of the slug of contaminants is the (horizontal hydraulic conductivity) 16,680 m/yr (45 m/day) times (horizontal hydraulic gradient) 0.0025 divided by (effective porosity) 0.20. If no longitudinal dispersion occurs, the seepage water would move downgradient at an average linear velocity (Vs) of 209 m/yr.

Dispersivity (aL) is the phenomenon of contaminants spreading out and attenuating in concentration due to molecular diffusion and stratification at the microscopic and macroscopic levels R.W. Cleary, (1992). At the microscopic level there is a distribution of velocities in the pore spaces due to frictional resistance caused by the solid surfaces of the porous material. At the macroscopic level there are large-scale heterogeneities which usually are the primary cause of contaminant spreading. Chemical species may also move from higher to lower areas of concentration within and between pore spaces due to molecular diffusion. Neglecting molecular diffusion, based on negligible effects due to relatively fast moving flow systems, the dispersion coefficient is a linear function of the average linear velocity. Dispersion values in porous media is currently under much research.

Longitudinal coefficient of dispersion (Dxx)can be expressed as:

Dxx = aLV, where aL= 0.1X

This is known as the one-tenth rule by some researchers, Lallemand-Barres and Peaudecerf, (1978). They were the first to study published values of dispersivities as a function of the distance traveled. The distance variable (X)could be measured from the source release i.e., railings pile and an observation well. Thus, dispersivity is a function that is related to scale and other considerations. For example, the greater the distance over which dispersivity is measured, the larger the value becomes. However, to conserve mass for an analytical solution, a deterministic dispersivity is selected for the model site. The dispersivity is chosen to be 10 m. The longitudinal dispersion coefficient (aLV) for the model site is 2090 m2/yr.

Many abiotic and biodegradable and radioactive reactions decay according to first order kinetics given by the following expression:

dC/dt = -KC

where K is the first order decay constant. It is equal to 0.693/T, where T is the half-life of the contaminant. Natural uranium with a half-life of 4.5 billion years is the mobile radioactive contaminant of most concern at uranium mill tailing sites. With such a long half-life, first decay reactions are considered insignificant to the other mass transport processes. For the model site, decay reactions are not accounted for.

Sorption (to account for adsorption and absorption) is the partitioning of a chemical species between the liquid and solid phase. The result of this partitioning from a modeling viewpoint is a phenomenon called retardation in which the effective velocity of the chemical species is less than that of the uncontaminated ground water. Retardation is characterized by a parameter called the retardation factor. Under equilibrium conditions a sorbing solute will partition itself between the liquid and the solid phases according to the value of Rf. The fraction of the total

contaminant mass contained in the aquifer which is dissolved in the solution phase can be calculated from the following expression:

Fd = 1/Rf

The "distribution coefficient" (Kd) is a laboratory determination of the amount of solute left on a soil sample after it has been mixed and allowed to reach equilibrium with the soil. It can be expressed by the following relationship:

Kd = C'/Co, where

C' = mass of solute per unit volume of dry bulk solid

Co = mass of solute per unit volume of liquid

Linear equilibrium adsorption isotherms are commonly used to describe the sorption of contaminants on the solid phase surfaces. The common unit for Kd is ml/g. If Kd for a given contaminant species equals zero, there is no sorption occurring. This is the case for some common contaminants related to uranium mill tailings. Contaminant species such as sulfate and nitrate are both anionic solute species that generally have a very low Kd and is the reason why these analytes are considered good indicator parameters or ground water tracers because they are transported at the same average linear velocity as the ground water. Note, biodegradation processes due to microbial reduction can be a significant process affecting concentrations of nitrate and sulfate.

If the distribution coefficient has been determined, (this can be done through laboratory batch leach experiments for each constituent of concern) and, the bulk density and effective porosity have been calculated, the retardation factor for saturated conditions can be expressed by the following equation:

Rf = (1 + Kd rb /ne),

where:

Rf = retardation factor;

Kd = distribution coefficient; varies with contaminant

rb = dry bulk density = 1.6 g/cc;

ne = effective porosity = .20.

For the model site, the bulk density was calculated by laboratory analysis. The bulk density equals 1.6 grams/cm3. The effective porosity was taken from the literature and is equal to 20 percent. For a non-sorbing species such as nitrate, Rf equals 1.0 while a sorbing species such as radium will have an Rf greater than 1.0. The solute (ion) velocity (V*) can be calculated by the following expression:

V* = V/Rf,

where:

V = the seepage velocity or average linear velocity.

For the model site, a list of common contaminants associated with uranium mill tailings has been developed. Associated with each contaminant is a Kd. If the contaminants have an EPA Maximum Concentration Limit (MCL), then it is listed, if not, the statistical background concentration must be determined.

TABLE I

Choosing a one-dimensional analytical transport solution for the conceptual model involved finding a solution for a point source. Because the tailings, in near saturated conditions, leaked contaminants into the ground water for 30 years, and then were subsequently removed, a continuous source solution would not be appropriate. A contaminant transport equation from Domemico & Schwartz, (1990) was used to estimate the time it would take to dilute to the MCL (for a given contaminant) at the centroid of the plume.

The equation does not account for retardation and therefore is very conservative.

Eq. (2)

t = time in years

Co = original concentration in mg/l

Vo = original volume in meters3

Co x Vo = mass of the discharge

Dxyz = coefficient of hydrodynamic dispersion in meters2 per year

Cmax = MCL for the contaminant

With this equation, an idealized three-dimensional point source spill, spreading occurs in the direction of flow, and the maximum concentrations occurs at the center of the "cloud," that is, where Dy = Dz, and is an order of magnitude smaller than Dx. Going to the contaminant list table and choosing uranium as the contaminant of concern and substituting into the equation results in an estimate of approximately 40 years for the aquifer to flush to the MCL for uranium at the centroid of the

plume. Note that this estimate does not account for retardation. Trying another contaminant for testing the potential for the model site to naturally flush to the standards within 100 years, nitrate is chosen. Plugging nitrate, which has a non-reactive retardation factor of 1, into the equation results in approximately 48 years for the model site to flush to the MCL.
For Uranium:
Eq. (3)
For Nitrate:
Eq. (4)
CONCLUSIONS FOR SELECTING NATURAL FLUSHING AS A GROUND WATER COMPLIANCE STRATEGY
The hydrogeologic attributes of the model site make it a strong candidate for natural flushing. However, this analytical exercise is only a screening tool for selecting a candidate site. Should other screening techniques also support the selection of natural flushing, detailed characterization, monitoring, and numerical modeling must be implemented.
Detailed characterization should be extensive enough to understand the gross system heterogeneities. Following the placement of a monitoring-well network, a period of "calibration monitoring" would establish a better understanding of the dynamic system with regards to seasonal variations. The number of monitor wells in the network should be minimized, yet provide sufficient coverage to delineate the areal and vertical extent of plume, and understand aquifer parameters and boundary conditions. A three year period of "calibration monitoring" might be sufficiently long to predict that the model site continue as a candidate for natural flushing. During the period of detailed characterization and monitoring a numerical simulation of flow and solute transport would be necessary to more accurately understand aquifer conditions and flushing predictions. Because the system uncertainties will never be reduced to zero, geostatistics should be incorporated into the modeling task. The results of the modeling would be most likely be presented as probabilities rather than as deterministic values.
At this point, a remedial action plan can be submitted to the regulatory agencies for review. Assuming the site has no technical deficiencies in support of the natural flushing strategy and the regulators and public agree with the compliance approach and institutional controls, a period of verification and compliance monitoring will be required. Most scientists and regulators agree that ground water models cannot be validated. Therefore, modeling results by themselves usually are not a sufficient demonstration that the site will flush as predicted and long-term compliance monitoring will most likely be required by the regulators.
Compliance monitoring needs to be optimized to account for areal and vertical coverage of the contaminated zone for the constituents of concern. A network of wells set on a grid coupled with quarterly sampling for a full suite of analytes is not an optimized compliance monitoring program. In such a case, it might be more cost effective to actively remediate the plume with an engineered remedial strategy. However, if the optimized compliance monitoring program provides adequate coverage with a minimal of wells, and the sampling frequency and duration is reasonable, coupled with a condensed list of contaminants to be sampled (based on risk), then natural flushing becomes a viable and cost effective strategy.
The DOE UMTRA Project has targeted seven sites for natural flushing. Detailed characterization is beginning at some of the sites along with initial modeling for screening purposes. The DOE continues an aggressive public outreach program for the UMTRA Ground Water Project to explain the pros and cons of natural flushing and the other ground water compliance strategies to the involved stakeholders.
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EVALUATING THE ACCEPTABILITY OF SOIL TREATMENT METHODS IN THE DEPARTMENT OF ENERGY'S FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM: A STAKEHOLDER ANALYSIS*

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ABSTRACT

Can radiologically contaminated sites be treated to permit unrestricted public use? Can treatment methods satisfy the concerns of federal and state regulators who must ensure that residual radioactivity is low enough to permit site access and beneficial reuse of de-contaminated soil? Are these methods potentially harmful to cleanup workers or the public? And, if so, can they be made acceptable to people who live, work, or own property near these sites? This article examines the acceptability to stakeholders of two sets of options for treating contaminated soils in the U. S. Department of Energy's Formerly Utilized Sites Remedial Action Program (FUSRAP): immobilization techniques designed to prevent the further migration of contaminants, and volume reduction methods that reduce the amount of contaminated material for disposal. These selected options, originally developed for treating hazardous waste, recovery of heavy metals in the mining industry, or treating low-level radioactive waste, are potentially effective for treating radionuclide-contaminated sites. Treatment method vendors, cleanup contractors, federal and state regulators and other recognized experts--stakeholders affected by treatment--were interviewed on the technological maturity, public and regulatory acceptability, and environmental and public health impacts of these options.

BACKGROUND

FUSRAP sites comprise a diverse array of government- or privately-owned sites in the continental U. S., most of which were used by two of DOE's predecessors, the Manhattan Engineer District (1942-1946) and the Atomic Energy Commission (1946-1973), for processing uranium and thorium ores and storing radiological concentrates and residues. Activities at these sites contaminated equipment, buildings and soils with Uranium-238 or Thorium-232 and by-products of their decay (radon and thoron). Many sites are located in densely populated areas where the public is concerned that wind and rain threaten them with exposure to contaminants through inhalation or ingestion. In some instances, contractors have used contaminated soils in roadbeds or building foundations.

Congress initiated FUSRAP in 1974 to decontaminate and restore these and other sites to a condition suitable for human use. FUSRAP's objectives are to clean up sites by undertaking actions that reduce radiation exposure to the public. This is usually done by excavating and disposing contaminated material and certifying sites acceptable for unrestricted public use. Cleanup has been undertaken at nearly 400 sites. FUSRAP now encompasses 46 sites in 14 states.

Interim Storage, Disposal, and Transport

At sites with large volumes of excavated material, the most common management method for contaminated soils is interim storage. Interim storage facilities consist of large, compacted soil piles covered by fabric or plastic membranes (covers) secured at the sides. These piles may have separate cells for certain contaminants (e.g., radium) and they are monitored by health physicists (1). Historically, storage facilities have been susceptible to stormwater runoff and wind erosion, leading to the contamination of vicinity properties. Moreover, plastic membrane liners wear with age, can become damaged by severe weather, or act as shear planes during seismic events (2). In most instances, they fail to meet CERCLA and NEPA requirements.

Offsite disposal of contaminated soils is also problematic. It may be impractical due to the large volume of contaminated material involved. Moreover, while contaminated soils could be disposed of in Low-level Waste compact disposal facilities, no new disposal sites have been opened and considerable uncertainties surround the disposal capacity of new facilities and their availability (3). Offsite disposal may also pose transportation risks to worker and public health and the environment. Truck or train accidents can rupture barrels of contaminated soil, exposing drivers and the public to radionuclides. Spilled soils may wash off roads or railroad tracks and enter creeks or other waterways. Moreover, some chemical treatment techniques for contaminated soils require shipping acids and solvents to

contaminated sites. Spills of these materials may be hazardous to workers and the public. The risks of transporting contaminated soils vary depending on the distance of a disposal facility from a FUSRAP site and the number of shipments required (4). In short, problems with offsite disposal and interim storage have led to efforts to develop treatment methods which can decontaminate or limit the spread of site contamination, save money, and permit reuse of sites and soils.

FUSRAP TREATMENT METHODS: OVERVIEW

Volume reduction methods under investigation for FUSRAP include physical separation, soil washing and solvent flushing (removing contaminants by chemicals). These methods have as their objective the beneficial re-use of treated material through the physical removal of contaminants by water, surfactants or chelating agents (catalytic chemicals that bind contaminants with metals to make it easier to remove them from soils). They are sometimes referred to as "treat-and-put-back" approaches because the remaining residues can then be disposed on- or offsite. Bioleaching and biosorption are recently introduced volume reduction methods.

Immobilization methods include solidifying contaminated material through vitrification (a process that isolates contaminated soil in an epoxy or glass-like enclosure); dewatering (installing drains and pumps to achieve a dry environment around contaminants); grouting and other stabilization or solidification techniques that "cement" contaminants in place; chemical fixation techniques that employ heat or chemical neutralization to block further contamination of a site; bioremediation, the use of organisms to degrade organic contaminants in soil that are mixed with radionuclides; and, brickmaking (the extrusion of contaminated material into bricks).

Distinctions among all of these potential FUSRAP soil treatment methods are important because why, how, and where contaminated soils are treated partly shapes options for future site use. In addition, different methods have different impacts. Table I depicts principal treatment methods under consideration in FUSRAP. This table is a synthesis of U.S. EPA categories for treatment, other expert judgments, and our attempt to provide a coherent scheme based on radionuclide site concerns (5).

A treatment method that immobilizes contaminants within a site may have vastly different impacts upon the future use of a site than one that excavates contaminated material, treats it, and then transports the remaining residue to an offsite disposal facility. For example, an "in-situ" immobilization technique such as vitrification that effectively traps radionuclides in perpetuity might mitigate public concerns about radon gas venting through the soil and possibly threatening public health in nearby communities. However, because it literally melts the soil under the site, it might also prevent underground repair work on utilities, foreclosing site use by residential or commercial developers who desire access to water, sewer, or electrical power (6). By the same token, chemical treatments might destroy organic compounds in soil, lessening its fertility, and prohibiting its later use for farming or gardening.

Likewise, an "ex-situ" volume reduction method such as soil washing might generate large volumes of clean soil that can be used as fill dirt or ground cover. However, if there are no acceptable standards for determining how "clean" the soil must be to permit its beneficial reuse as, say, foundation fill for a housing development or highway, reuse might not be acceptable at a site intended for unrestricted public access (7). Furthermore, unless their characteristics are similar to the original waste stream, the remaining contaminants from soil washing--concentrated in volume and diluted in form--might be difficult to dispose offsite until solidified. These contaminants may pose a hazard to remediation workers responsible for treating, packaging, and shipping residues offsite.

Treatment methods also have different implications for future site use. For example, some forms of soil treatment are likely to be used in conjunction with the partial excavation and removal of FUSRAP soils (8). The idea behind partial excavation is to remove only those contaminants close to the surface, or on the exterior of buildings, that pose the greatest threat to human health. If partial excavation is combined with immobilization to ensure that remaining contaminants are stabilized, a site may then be used by the public. However, deed restrictions might be required to ensure that remaining underground contaminants remain permanently undisturbed, and to relinquish DOE from future liability if the site is improperly used by subsequent owners.

The environmental and societal impacts of treatment are site-specific. As a consequence, so too is their acceptability.
Some residents adjacent to FUSRAP sites in New Jersey, for example, contend that contaminated soil storage sites within eyesight of their homes has reduced property values, made it difficult to sell their houses, and threatened the economic stability and future development of their communities. This perception, in turn, has generated stigma--the belief that future economic opportunities may be lost because businesses will be deterred from moving to these communities out of fear they will become permanent storage sites for radioactively contaminated material (9).
The experience of other DOE remedial action programs suggests that stigma may have some effect on the willingness of local communities to accept treatment. In Richland, Washington, for example, workshops have been held with local community officials and others over the possible use of remedial action technologies to remove volatile organic compounds (VOCs) from contaminated soils at the Hanford site. These workshops reveal that the reputation and credibility of DOE as a developer of technology, as well as the desire to protect jobs and property values, affects public perceptions of these technologies' effectiveness and of the motives of cleanup technology protagonists (10).
Treatment method acceptability may also be time-dependent. For example, at the St. Louis FUSRAP site, an interim storage facility, coupled with site access restrictions, fully complies with regulations designed to protect worker and public health, if future uses of the site are restricted to transient, recreational uses. However, contemplated changes to land use could make it impossible to comply with current regulations without innovative methods of remediation. The following passage from the draft St. Louis Site Feasibility Study illustrates this dilemma:
Under current (i.e., existing use) scenarios, the St. Louis site is within the acceptable range EPA has specified for protection of human health for . . . members of the general public. . . . Under future risk scenarios (however), potential cancer risks are higher for . . . the public. . . . Accordingly, the overall objective of remedial action . . . is to eliminate or minimize the potential future health risks posed by contamination under assumed future changed use of the site (11).
Finally, a principal incentive for treatment is cost-savings. Current DOE estimates to complete remediation of FUSRAP sites project the expenditure of approximately $2.5 billion between now and 2016 for planning, environmental compliance activities, development, operation, closure of storage/disposal sites, and other activities. A 1986 projection pegged cleanup to be completed in 2001 at a cost of $700 million for disposal, while an earlier estimate pegged cleanup completion in 1995 at a cost of $300 million (12). Cleanup technology costs are difficult to accurately estimate. Most of the factors driving costs are site-specific and require field-testing in different soil types to determine if they can reduce volumes of contaminated material; halt the spread of waste material in groundwater and soil; can be used in conjunction with other methods or under different combinations of approaches (e.g., partial as opposed to total excavation); and produce concentrated hazardous waste residues that pose special management problems (13).
DETERMINING STAKEHOLDER ACCEPTABILITY: METHODOLOGY
A review of EPA, DOE national laboratory, and Nuclear Regulatory Commission reports has identified several bench-scale (experimental), pilot-scale (field-tested), and full-scale soil treatment technologies as well as salient criteria for their evaluation. A literature review also identified domestic and foreign vendors who offer treatment services. A survey was then drawn up which asked respondents to rate 5 promising volume reduction and 6 immobilization methods under consideration for radionuclide-contaminated sites. Eight criteria were used to evaluate the former while only the first six were employed for the latter (beneficial soil reuse and transportation are not factors in immobilization):
Technology maturity (e.g., ease of scale-up/demonstration experience/impact of lack of cleanup criteria).
Applicability to a range of thorium/uranium-contaminated sites.
Public acceptability.
Ability to reduce contamination or exposure to workers and nearby residents.
Likelihood of producing hazardous emissions or difficult to dispose residues.
Ability to obtain regulatory approval.
Ability to permit beneficial reuse of soils, and
Minimization of transportation risk.

A 1 to 5 scale was used for rating each method's performance on these above criteria with 1 representing the lowest appraisal of the technology and 5 the highest. Respondents also offered a wide-range of opinions on future use issues, standard-setting criteria, and worker and public impacts of treatment.
Over 100 DOE site remediation contractors in charge of cleaning up contaminated sites, vendors who develop and sell treatment methods, federal and state regulators, and other authorities were queried. Respondents were selected from participants at recent soil remediation conferences. Key individuals and organizations were identified as having expertise and interest in soil treatment issues. Care was exercised to select a broad-range of participant types (e.g., vendors, treatment contractors, federal and state regulators). Opinions on the impacts of treatment on additional stakeholders were also solicited, including residential and commercial property owners, the public, and those living along transport corridors traversed by waste material destined for offsite disposal. Comments were also provided on the importance of cleanup standard-setting criteria for hastening the broad introduction of treatment.
The survey was partly inspired by an earlier EPA study (14) which convened a small group of experts to evaluate the performance (i.e., reliability and effectiveness) and commercial potential of several technologies considered appropriate for treating radionuclide-contaminated Superfund sites. These technologies included physical separation, soil washing, chemical extraction, vitrification, land encapsulation (i.e., permanent onsite storage through capping), solidification, and mine disposal of uranium mill tailings. While that study employed a numerical rating system similar to ours to assess reliability, effectiveness, and R&D maturity, it did not encompass more recent technologies; nor did it attempt to rate public and regulatory acceptability, emissions, or residue-generation impacts of technologies. In addition it did not rate prospects for beneficial reuse or transportation risk. EPA concluded that soil washing and chemical extraction bench- and pilot-scale studies show some commercial promise but require further study and additional field testing at contaminated sites. It also recommended that priority be given to the design and performance of realistic treatability studies, including the sequential employment of multiple treatment technologies.
FINDINGS
While various in- and ex-situ soil treatment technologies show promise for cleaning up uranium and thorium contaminated sites, survey respondents contend that none is a panacea for removing or stopping the spread of contamination or for permitting beneficial re-use of sites or of cleaned soils. Moreover, none is without risk and each has shortcomings ranging from limited, inconclusive field experience to limited application to certain sites. Some may also impose potentially adverse impacts on the public, workers, and the environment (e.g., volume reduction may concentrate contamination in a less stable medium). Perhaps the most significant finding is that, despite divergent perceptions and varying "stakes" in treatment effectiveness, a general consensus emerged around these core conclusions.
Volume Reduction Findings
As shown in Table II, respondents concur that radiologically contaminated soils can be treated through volume reduction technologies. In varying degrees, these technologies can permit future use of contaminated sites and beneficial re-use of soils. Physical separation and soil washing are the treatment methods most favored by respondents for accomplishing these objectives. Vendors, remediation contractors, and regulators feel soil washing with offsite disposal is safer than soil washing with onsite disposal.
Most respondents believe the public is very concerned about decision maker assurances that contaminated soil will be removed from nearby neighborhoods and that "clean" soils can be beneficially re-used. Physical separation and soil washing with offsite disposal are viewed as most likely to accomplish these objectives. Most respondents do not favor solvent flushing due to its potentially adverse environmental and social impacts. Most also feel that bioremediation methods are still in an experimental stage of development for use at radionuclide contaminated sites. If successfully developed, however, they are likely to curry favor with the public due to the potential for greater safety over more established alternatives. While solvent flushing and bioremediation are not regarded with as much favor with regulators as with vendors or contractors, the former are unsure about the potential impacts of all methods.

Survey respondents generally feel that volume reduction methods have been concentrated at bench- and pilot-scale (i.e., demonstration stage) levels--not at full-scale levels adequate for investing confidence. While respondents have fairly high confidence in physical separation and soil washing, they are concerned about the impacts of aggressive chemical or biological methods that damage the soil matrix, leave toxic residues in their wake, fail to satisfactorily remove radionuclides, and minimize productive future site uses. Regulatory acceptability is generally tied to perceived public and worker acceptability and consensus over cleanup standards and future site use. Vendors and contractors tend to regard physical separation and soil washing as mature, adequately tested, and broadly applicable technologies.

Immobilization Findings

Immobilization technologies displaying promise, according to respondents, include site dewatering and stabilization/solidification. Dewatering's long-term effectiveness, however, is viewed more cautiously. As shown in Table III, stabilization/solidification is thought to be the safest immobilization technique due to its reliance on chemically inert means of isolating wastes (e.g., grouting contaminants in place). It is preferred over vitrification, which risks exposing workers to toxic vapors, and brickmaking, which compels workers to handle contaminated soils.

Comments by respondents suggest the need to employ conservative assumptions about contaminant solubility and mobility in any evaluation of immobilization. As a result, respondents appear unsure about the long-term effectiveness of these methods, except for sediment dewatering and contaminant fixation/stabilization. Comments also indicate concurrence among respondents over the availability of effective techniques to prevent worker and public exposure to hazardous emissions from treatment (e.g., protective clothing, adequate ventilation, toxic gas collection systems).

Vendors and contractors generally agree about the relative maturity, site applicability, and worker and public risks of dewatering and brickmaking. Both rank vitrification and bioremediation lower than other immobilization methods. However, vendors (who are closer than contractors to the technical developments in treatment) appear to have somewhat more faith in the ultimate viability of bioremediation.

Dewatering appears to be the preferred technology for immobilization. However, it is perceived to be the least effective for accomplishing long-term reduction of potential exposures, probably because--as several commenters indicated--water management practices are unlikely to remain effective for thousands of years.

Bioremediation appears to be the least favored immobilization approach. While biological methods are generally considered to be feasible treatment methods for organic contaminants, they are viewed as having limited utility for radioactive materials. Some success has been achieved through the use of bacteria for reducing or oxidizing contaminants which can obtain energy from the electronic transitions. These changes in oxidation states impact the mobility of contaminants. However, according to respondents, this characteristic of contaminants is not given enough consideration in risk assessments to make it possible to draw confident judgments.

CONCLUSIONS

There are notable differences in perceptions of public and regulatory acceptability of cleanup technologies by category of respondent. Contractors have greater confidence in the public and regulatory acceptability of vitrification than do vendors. Vendors, in turn, have greater confidence in the public and regulatory acceptability of dewatering and brickmaking. Satisfying the concerns of federal and state regulators, in almost all cases, is problematic. Regulators want proof of treatment effectiveness, prefer time-tested technologies, and are concerned about residual contamination and management of wastes generated by treatment. They are reluctant to endorse "new" or untested technologies.

Based on respondent input, overall treatment method suitability is dependent on resolution of several issues. These include: getting stakeholders to agree upon future site use; developing specific cleanup criteria appropriate to that desired use; negotiating target cleanup levels; determining the type and concentration of contaminants at a site; and adopting realistic, viable cleanup schedules. Treatment is more likely to be judged suitable if it reduces contamination to safe levels without exposing site workers to excessive risk; prevents the spread of contamination to other parts of a site or to offsite properties; complies with

federal, state, and local regulations designed to protect health and safety (especially with state regulations pertaining to land use); is easily understood by the public; and, ensures safe, final disposition of contaminated material.
Finally, according to respondents, there are many risks from treatment that must be resolved to ensure acceptability. Risks to cleanup workers during treatment include exposure to fugitive dust, toxic gases (from vitrification), chemicals, solvents, and waste residues; and the dangers associated with working near heavy machinery. Also, while treatment can satisfy property owners worried about land values, it must ensure minimal site disruption and satisfy future use aspirations. Related to this point, the ability to beneficially re-use soil, while having considerable economic advantage, requires agreement over cleanup criteria to establish "how clean is clean." Cleanup criteria affect future site use, cleanup cost, and the likelihood of 'scaling-up' a treatment method from pilot-scale to full-scale application at a given site. Agreement over cleanup standards, in turn, requires evaluation of potential doses from various pathways. The good news is that these issues are subject to negotiation within the confines of NEPA and CERCLA requirements (15).
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Session 55 -- Managing Health and Safety Successfully
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55-3
SAFETY ASSESSMENT AND SURVEILLANCE OF DECOMMISSIONING OPERATIONS AT DOE'S NUCLEAR FACILITIES
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ABSTRACT
A description is provided of a systematic approach currently being developed and deployed at the Department of Energy to obtain assurance that post-operational activities at nuclear facilities will be conducted in a safe manner. Using this approach, personnel will have available a formalized set of safety principles and associated question sets to assist them in the conducting of safety assessments and surveillance. Information gathered through this means will also be analyzed to determine if there are any generic complex-wide strengths or deficiencies associated with decommissioning activities and to which attention should be drawn.
INTRODUCTION
The U.S. Department of Energy (DOE) is committed to achieving the stated goal of cleaning up its complex and bringing all sites into compliance with applicable environmental regulations by the year 2019. Many facilities have already ceased operational activities and are in the transition to the post-operational phase of the facility life cycle. The final stages will involve, to various degrees, decontamination, dismantlement and decommissioning. Included in the complex are facilities that were designed and operated several decades ago and for which little consideration had been given at that time to the eventual need for decommissioning. The act of decommissioning a nuclear facility is a relatively recent concern, both within the DOE and in the commercial nuclear field, and only a limited set of decommissioning-specific standards and requirements have been issued. This has complicated the task of performing safety assessments and surveillance over these activities which must, nevertheless, be conducted in a manner that ensures the health and safety of the workers on the project, other site employees and the general public. In order to facilitate this, the DOE is employing a systematic approach developed specifically for use in evaluating the safety of the activities encountered while facilities are in the post-operational status. This approach is designed to provide the safety assessment personnel with a framework on which they can base their technical judgement, to assure a consistent approach to safety assessment of D&D operations and to enable the systematic collection of data during the post-operational part of a facility's life cycle.
METHODOLOGY
The approach adopted is founded on a set of decommissioning-specific safety principles from which has been derived a formalized surveillance plan and question set for use in the field. Both the principles and the surveillance plan and question set are directed predominately at facilities that could pose nuclear and/or radiological hazards. Consequently, the prime focus is on activities related to those hazards and the actions necessary to protect the health and safety of the workers and the general public from them. Protection against other types of hazards, such as chemical or construction/demolition hazards, has not been addressed in detail, except where a potential exists for these other hazards to affect the nuclear or radiological safety. However, in limiting the present approach to nuclear and radiological hazards, the importance of these other hazards to overall safety is not being ignored. In fact, experience to date indicates that non-nuclear hazards

are a leading source of accident and injury in the post-operational phase and a need exists to examine these other hazards in this specific context.

Safety Principles

The safety principles that underpin the evaluation process express both the fundamental philosophy that should guide a decommissioning operation in its entirety and the precepts applicable to the various facets of the operation. In this respect, the formulation and compilation of the principles owe much to those used by the Nuclear Installation Inspectorate branch of the Health and Safety Executive in the United Kingdom when evaluating the safety of the design, construction and operation of nuclear plants (1). The contents of a preliminary draft version of the decommissioning-specific safety principles have been described previously (2).

The safety principles can be roughly divided into two types: a group of fundamental principles and numerous groups of principles to be applied to specific areas. The fundamental principles are derived from internationally-accepted principles developed to ensure the conduct of safe operations at all times at any type of nuclear facility. Their roots lie in the recommendations of the International Commission on Radiological Protection (3), and the principles themselves address the need to keep radiation doses beneath the statutory limits and, in addition, as low as is reasonably achievable (ALARA). The fundamental principles also emphasize the importance of accident prevention and the minimization of the radiological consequences from any accident that might occur.

The fundamental principles are general in nature; translation of their intent into actual operations requires more specific direction. Thus other sets of principles were developed to address the various facets associated with decommissioning operations. These other, so-called "second tier," sets of principles have been built on the fundamental principles, taking note of relevant rules and regulations, and of associated statutory limitations. Beyond that, they are not intended to be prescriptive nor to be viewed as technical specifications. These second tier principles are presently grouped under three headings: "Safety Analysis," "Engineering Practice" and "Waste Management." The groups themselves contain several subsets to deal with certain individual subject areas. The "Safety Analysis" principles are classified according to whether they are intended to cover normal decommissioning operations or off-normal and accident conditions. The "Engineering Practice" principles are comprised of a set of basic principles that have application across the full spectrum of decommissioning engineering practice and several sets of principles that address more specialized topics (codes and standards, equipment, human factors, radiological protection, hazard analyses, etc.). The "Waste Management" principles deal mainly with the need to reduce the risk of radiation exposure and contamination during the generation, storage, handling and transport of radioactive waste. In particular, emphasis is placed on the need to minimize the quantity of waste as far as is reasonably practicable.

It is important to note that this adoption of a set of safety principles does not represent a radical breakaway from past methods of conducting safety assessments of operations at nuclear facilities. Rather, the set of principles represents a formalized statement of the criteria already in use by personnel conducting the assessments. In addition, the principles will be under continual review to ensure that they remain appropriate and applicable in the light of ongoing developments in relevant scientific and technical fields. In this context, feedback will be sought from personnel in the field and comments incorporated into revised principles as appropriate.

Surveillance Plan and Question Set

The term "surveillance" is used here to define the oversight activities applicable to a facility's many post-operational life cycles phases. As such, it has a broad context and should not be confused with the more narrowly-defined surveillance and maintenance phase. The purpose of the plan is to establish responsibilities and methods for conducting reviews and to provide a means for systematically collecting data on facilities that are in post-operational status. The activities to be performed under the surveillance plan fall under three principal categories: document reviews, personnel interviews and on-site observance and inspection.

The approach initially considered (4) was modified in the light of peer review but the focus has remained unchanged - the implementation of the safety principles. This implementation involves a graded approach consistent with the nature and magnitude of the hazards perceived to be present. Inherent in this approach is that the degree

of applicability of the principles will change as decommissioning operations proceeds. In most cases it is anticipated that the nuclear and radiological hazards will decrease but, because of the uncertainties associated with the post-operational condition of any facility, it is conceivable some conditions may be encountered that will bring about a heightened awareness of hazards and an associated increase in safety concerns.

The basic plan for conducting the surveillance activities involves partitioning a decommissioning operation into specific functional areas, such as management, radiation protection, packaging and transportation, fire protection, emergency planning, training and quality assurance. Standardized question sets, derived from the safety principles, have been developed for each area. Safety assessments can be conducted on an overall functional basis using a limited number of questions from each functional area, or, if deemed necessary, on a more detailed basis using most or all of the questions in a functional area. An additional option is to conduct the evaluation on a principle basis, in order to verify that the intent of a specific principle is being fully met. The question sets will also be subject to the same iterative review being applied to the principles in order to confirm their currency in the light of developments in all aspects of the decommissioning process.

In tandem with the evolution of the question set is the development of a guidance document for application of the principles in practice. This document is being designed to provide background information on each principle and give examples of how it might be applied in specific instances. This type of guidance is thought to be preferable to a standardized question set in some instances where engineering and scientific judgement play a prominent role (for example, in evaluating the content of a safety analysis report). However, the use of question sets will be more appropriate in other types of assessment, such as confirming that procedures are being used correctly.

In addition to providing assurance that the decommissioning procedure has been planned and is being conducted in a safe manner, the data generated through the post-operational facility surveillance plan will be evaluated and analyzed from a complex-wide perspective. The results of such analyses will be used to detect DOE-wide generic strengths and weaknesses (for example, by indicating the need for standards and training in specific technical areas) and thus be of assistance in expediting subsequent decommissioning projects.

Because of the evolutionary nature of the project, input is being sought from all parties concerned, from those responsible for preparing policies and standards to those charged with conducting the actual D&D operations. This kind of interactive communication is being strongly encouraged to ensure that the safety principles and question sets are appropriate, that there is general agreement and understanding of the methodology and that its application will be feasible and practical.

When finally developed and tested, it is planned to investigate the extension of this methodology to other aspects of post-operational activities at facilities, such as mission transition and environmental restoration. Eventually, it is envisioned that it could be adopted for surveillance of all phases of a facility's existence, including design, construction and operation of new facilities. Additionally, although the current plan is focused on nuclear and radiological hazards, the methodology and framework described can be applied generically to address other conditions, such as chemical and conventional industrial hazards.

SUMMARY

The DOE is developing and deploying a systematic approach to obtaining assurance that post-operational activities (specifically, decommissioning) at nuclear facilities will be conducted in a safe manner. Personnel will have available a formalized set of safety principles and associated question sets to assist them in the conducting of safety assessments and surveillance. The information gathered through this means will also be analyzed to determine if there are any generic complex-wide strengths or deficiencies associated with decommissioning activities and to which attention should be drawn.

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55-4

HAZWOPER TRAINING AT THE DEPARTMENT OF ENERGY: A PARTNERSHIP BETWEEN GOVERNMENT, CONTRACTORS AND LABOR ORGANIZATIONS TO PROTECT WORKERS' SAFETY AND HEALTH

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ABSTRACT

As the Department of Energy's (DOE) mission has shifted from weapons production to environmental restoration, the site worker will be exposed to new operations and hazards while conducting restoration operations. Protection of the health and safety of all workers is a Secretarial priority as well as a goal of Environmental Management (EM). To conduct restoration operations in a manner which protects worker health and safety, all workers must be trained to identify hazards and the appropriate way to safely conduct operations. All workers engaged in hazardous substance response or emergency response, a common element of restoration operations, are required to meet the Occupational Safety and Health Administrations' (OSHA) regulation requirements 29 CFR 1910.120 and the EPA Hazardous Waste Operations and Emergency Response (HAZWOPER) training requirements (40 CFR 300.150). Congress recognized this need and authorized the Secretary of Energy, through the National Defense Authorization Act for Fiscal Years 1992 and 1993, to award grants for training and education for persons engaged in hazardous substance response or emergency response at DOE nuclear weapons facilities and further authorized $10 million to carry out this mandate.

The Office of Environmental Restoration was charged with developing a program to meet the HAZWOPER training requirements and address the specific needs across the complex. The goal of the training program has been to provide site specific quality training to workers in a timely and cost effective manner. The first grantees to receive awards for this program were comprised of labor organizations and universities with prior experience in conducting joint labor-management HAZWOPER training programs. Training is provided at the site to address site-specific needs and is available to DOE and contractor employees, regulatory agency personnel, state, local and Tribal government officials and local emergency responders working in hazardous substances and emergency response operations at DOE sites.

A partnership between government, contractors and labor organizations works to ensure that the training is timely and site specific. A cornerstone concept of this program is workers training workers, such that the workers who are familiar with performing a given task in a hazardous environment are training other workers to perform that task in a manner protective of their health and safety. Added benefits of this partnership include the fostering of close cooperation between management and workers, improving efficiency and quality of training, improving the ability to address worker concerns, and empowering all stakeholders to address site-specific safety and health needs.

INTRODUCTION

Protecting worker health and safety is a priority of the Secretary of Energy and is a primary goal of the Office of Environmental Management (EM). As the Department of Energy's (DOE) mission has shifted from weapons production to environmental restoration, the site worker will be exposed to new operations and hazards while conducting restoration activities, many of which will be associated with potential exposure to hazardous substances and wastes. To provide protection to workers' health and safety, all workers at DOE sites engaged or potentially engaged in environmental restoration activities, including hazardous substance response or emergency response, are required by the Comprehensive Environmental Response, Compensation and Liability Act of 1980 (CERCLA) (42 USC 9604(f) and 9651(f)) and DOE

Orders 5480.4 Environmental Protection, Safety, and Health Protection Standards and 5483.1A Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned, Contractor-Operated Facilities, to meet the requirements of the Occupational Safety and Health Administration's (OSHA) regulations 29 CFR 1910.120 and the EPA Hazardous Waste Operations and Emergency Response (HAZWOPER) training requirements (40 CFR 300.150). Congress recognized this need and authorized the Secretary of Energy, through the National Defense Authorization Act for Fiscal Years 1992 and 1993, Section 3131, to award grants for training and education for persons engaged in hazardous substance response or emergency response at DOE nuclear weapons facilities. For purposes of Section 3131, the term "hazardous substance" in addition to its definition under CERCLA includes radioactive waste, mixed radioactive and hazardous waste. The Office of Environmental Restoration, a separate office within the Office of Environmental Management, was charged with developing a program to meet the HAZWOPER training requirements and address the specific needs across the complex. The Regulatory Integration Division, within the Office of Environmental Restoration, has been tasked with program implementation.

The goal of the training program has been to provide site specific quality training to workers in a timely and cost effective manner. A partnership between government, contractors and labor organizations is in place to meet this goal. A cornerstone concept of this program is workers training workers, such that the workers who are familiar with performing a given task in a hazardous environment are training other workers to perform that task in a manner protective of their health and safety, for example electricians training electricians. Added benefits of this partnership include the fostering of close cooperation between management and workers, improving efficiency and quality of training, improving the ability to address worker concerns, and empowering all stakeholders to address site-specific safety and health needs. Training is available to DOE and Contractor employees, regulatory agency personnel, state, local and Tribal government officials and local emergency responders working in hazardous substance response and emergency response operations at DOE sites.

AUTHORIZATION FOR PROGRAM

Training Grant Program

The National Defense Authorization Act for Fiscal Years 1992 and 1993 (42 USC 7274(d)) authorized the Secretary of Energy in Section 3131(a)(1)(A)-(B) to award grants:

" (a)(1)(A) to provide training and education to persons who are or may be engaged in hazardous substance response or emergency at Department of Energy nuclear weapons facilities; and (B) to develop response curricula for such training and education."

Initially, the grant program was authorized for $10 million in 1992 to develop training curricula and to provide training and education.

The Secretary was further authorized in Section 3131(a)(2)(A)-(B) of the National Defense Authorization Act to award the training grants to non-profit organizations demonstrating capabilities in:

"(a)(2)(A)(i) implementing and conducting effective training and education programs relating to the general health and safety of workers; and (ii) identifying, and involving in training, groups of workers whose duties include hazardous substance response or emergency response."

Preference in grant awards is to be given to employee organizations and joint labor-management training programs that were grant recipients under Section 126(g) of the Superfund Amendments and Reauthorization Act of 1986 (SARA) (42 USC 9660(a)).

Any organization receiving a grant under this program is required to be in conformance with DOE Orders relating to employee safety and health, including but not limited to DOE Orders 5480.4 and 5480.11 Radiation Protection for Occupational Workers, when carrying out training, education or curricula.

Enforcement of Employee Safety Standards

Civil penalties may be assessed by the Secretary against any DOE contractor employing persons engaging in hazardous substance response or emergency response at DOE nuclear weapons facilities for failure to provide training for those activities or to certify that the employees are adequately trained for performing such response actions in accordance with DOE Orders on employee safety training. These penalties may not exceed $5,000 for each day in which a violation occurs. The Secretary is required to develop regulations to carry out the provisions of Section 3131(b).

Funding
For fiscal years 1992 and 1993, $10 million was authorized and appropriated for each year. In 1994 Congress authorized $11 million. The funding for fiscal years 1995 and 1996 will remain level at $10 million. Funding is through the EM budget, however training is not limited to EM employees or contractors.
GRANTS ADMINISTRATION
Selection of NIEHS as Grants Administrator
Under Section 126(g) of SARA the National Institute of Environmental Health Sciences (NIEHS) developed and administers a HAZWOPER training program in consultation with the Environmental Protection Agency (EPA). DOE evaluated the program developed by NIEHS for suitability of adaptation to DOE program and training needs. It was determined that the program would be suitable to adapt to DOE. In an effort to rapidly move to the program implementation stage and to leverage program resources, DOE considered the suitability of NIEHS as grants administrator for the DOE program. Based upon a review of the NIEHS program, DOE entered into an agreement with NIEHS to award and administer the grants and to adapt the HAZWOPER program to meet the DOE needs.
Administration Responsibilities
As grants administrator, NIEHS has several responsibilities. The first responsibility was to establish technically proficient training materials and to ensure that such materials addressed DOE's special needs specifically with regard to radioactive and mixed waste. They were to make announcement of grant availability, develop criteria for judging grant applications in conjunction with DOE, and finally, to make grant awards within the specified criteria for selection. They have an ongoing responsibility to ensure the quality control of all course content and presentations to maintain consistency across the complex.
Selection of Grantees
In 1993, NIEHS with DOE concurrence selected seven grantees from the grant applicants. All grantees met the requirements specified in Section 3131 of being non-profit organizations that were employee/labor organizations and/or experienced in worker health and safety training. The initial grantees were: University of California at Los Angles Labor Occupational Safety and Health Program (UCLA); University of Medicine and Dentistry of New Jersey (UMDNJ); International Union of Operating Engineers (IUOE); Oil, Chemical and Atomic Workers Union (OCAW); International Chemical Workers Union (ICWU); Laborers Associated General Contractors Education and Training Fund (L/AGC) (also teamed with the International Brotherhood of Teamsters); and United Brotherhood of Carpenters and Joiners Health and Safety Fund (UBC).
TRAINING
Training Objectives
There are several objectives that DOE desires to see met by the HAZWOPER training program. First and foremost, is to ensure that all workers in DOE nuclear weapons facilities that participate in hazardous substance response and emergency response receive quality safety and health training. It is an objective that all training offered in the HAZWOPER program be of the highest quality to protect life and health.
The training must also be geared to site specific needs and worker specific needs. Through the cooperation of management and workers, site specific needs can be identified based upon site configuration, geography, and planned work activities. The worker specific needs are addressed by workers training workers. A worker who is experienced with performing a certain required task under hazardous conditions is best suited to understand the constraints of the situation and the task requirements. For example, an electrician who has performed wiring tasks in Level A personal protective equipment would be best suited to train another worker who will perform a similar task under similar conditions.
A final objective is to train as many persons as possible in a cost effective manner through providing training locally. This reduces the cost of training through reduced travel costs, making funds available for additional training. It also reduces time away from the job allowing more persons to be trained. Local training also allows for site specific needs to be identified and more easily incorporated in the training.
Training Scope and Course Elements
The scope of the training covered in the curriculum is the worker protection

requirements for 40 hour training contained in 29 CFR 1910.120, the HAZWOPER standard. Additionally, the training contains modules on radioactive and mixed waste. The courses meet the requirements of Minimum Criteria for Worker Health and Safety Training for Hazardous Waste and Emergency Response, NIEHS, December 1991. All courses meet the minimum criteria delineated above. Examples of training course elements covered are:

Personnel responsible for site safety and health,
Safety, health, and other hazards present on-site,
Use of personal protective equipment (PPE),
Work practices to minimize risks from hazards,
Safe use of engineering controls and equipment on-site,
Medical surveillance requirements, and
Recognition of symptoms and signs indicating overexposure to hazards.

While the elements listed above are only representative of the course content, it is the intent that the course elements are such that the worker is prepared to recognize and analyze the hazards that may be present, take appropriate action and thereby protect their own and other workers health and safety.

Training Accomplishments

Training has been offered across the DOE complex primarily through the DOE Operations Offices and Headquarters. It should be noted that the HAZWOPER grantee training is not the only HAZWOPER training available in the complex and in fact represents only a portion of the training available. It was intended as a supplement to the existing training to assist DOE and its contractors in meeting the requirements of 29 CFR 1910.120 and DOE Orders. Further, contractors not in compliance will be faced with civil penalties. In Fig. 1 below, the number of workers trained by the Operations Offices is depicted. The total number of workers trained to date is 7468, with the Oak Ridge Operations having trained the greatest number of that total at 2594 workers.

Fig. 1. Number of workers trained by operations office- *Sept. 1, 1993-June 30, 1994.

These training statistics are undergoing further analysis to determine the reasons for the differences in workers trained by Operations Offices. The information will be used to increase the success of all the Operations Offices and to achieve full worker training at all sites.

CURRENT AND FUTURE PLANNED ACTIVITIES

New Grant Awards

New training grants will be issued in Fiscal Year 1995. The source selection process has already been initiated by NIEHS with a request for applications. Selections will be made in the last quarter of Fiscal Year 1995 in order to start the Fiscal Year 1996 program.

Advanced Notice of Proposed Rulemaking

The Secretary of Energy issued an Advanced Notice of Proposed Rulemaking on August 26, 1994 in the Federal Register (59 FR 44139), Enforcement of Employee Safety Standards at Nuclear Weapons Facilities. The purpose of the notice was to invite public comment on DOE plans to implement Section 3131(b) of the National Defense Authorization Act of 1992 and 1993, enforcement authorities to assess penalties against DOE contractors failing to provide training to employees who are assigned to hazardous substance response and emergency response duties or who fail to certify that the employees are adequately trained.

DOE particularly noted several matters of interest for comments. The matters of interest concern: training documentation content and format, including requiring wallet-sized photo identification documenting training; employer certification of employee training adequacy; the structure of an enforcement program to provide incentives to DOE contractors to comply; the development of criteria for assessing adequacy of contractor training programs; defining "hazardous substance" to include radioactive waste, mixed waste, hazardous waste and substances defined in Section 101(14) of CERCLA; defining "nuclear weapons facility"; and appropriate enforcement procedures, including integrating these procedures with the DOE program on nuclear safety in 10 CFR 820. Comments were due on October 25, 1994. The Office of the Assistant Secretary for Environment, Safety and Health (EH) is reviewing the comments received and is developing a Notice of Proposed Rulemaking.

EM/EH Safety and Health Assistance Initiative

Under the EM/EH Safety and Health Assistance Initiative, DOE is developing

standardized curriculum and instructor guidelines. Additionally, they are developing evaluation criteria for course assessment. The purpose of the standardization is to provide consistent training complex wide, assure all courses are of the same high quality, assist grantees in meeting DOE expectations for course content and presentation, and assure that all workers are receiving the highest quality training. Throughout the development process, grantees are involved with EH and EM to provide their expertise and input for an effective and cost efficient program.

Program Evaluation

The Oak Ridge Institute for Science and Education (ORISE) have been contracted to evaluate the grantee program. When completed, the evaluation will be shared with the Congressional Armed Services Committee having expressed a desire to receive a program evaluation. In performing the evaluation, ORISE will utilize existing evaluation materials from grantees, NIEHS and the Operations Offices. Specifically, they have been charged to review training utilization decisions at DOE sites, determine if training is reaching target audiences and evaluate the effectiveness of program management. The evaluation results will be available in late 1995.

CONCLUSION

The HAZWOPER program was developed with the support of government, contractors and labor unions to ensure that the workforce is trained to conduct restoration activities in a manner which protects worker health and safety. The training program has, over the course of three years, progressed through the development phase to the implementation phase with over 7400 workers trained by seven grantees. In each phase of program development, members of the partnership assumed larger or smaller roles based upon program needs. For example, in the earliest phases, government and labor organizations had a more active role during the development of curriculum. In the implementation phase contractors and labor unions have a more active role as training needs are identified and met. As rules are developed for implementation of the program, all three partners will have active roles.

The partnership between government, contractors and labor organizations works to ensure that the training is timely, site specific, and meets minimum criteria. A goal of this partnership is to protect the safety and health of the workers through the fostering of close cooperation between management and workers, improving efficiency and quality of training, improving the ability to address worker concerns, and empowering all stakeholders to address site-specific safety and health needs.

55-5

THE DYNAMIC HASP: A TOOL FOR DAILY PROJECT MANAGEMENT

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ABSTRACT

This paper will present a methodology to incorporate the requirements of the Health and Safety Plan (HASP) into daily management activities at hazardous waste sites. It is the author's contention that the HASP can be a "living document" and not a document that is merely developed as a regulatory requirement and then forgotten. The requirements of the HASP closely mirror the day-to-day functions of project management and a comprehensive safety and health program. The two can be blended together to provide a safer and more productive environment; better than either can provide alone.

INTRODUCTION

In 1986, the Superfund Amendment Reauthorization Act (SARA) tasked the Occupational Safety and Health Administration (OSHA) to develop a standard to protect workers at hazardous waste clean-up sites. OSHA responded by promulgating 29 CFR 1910.120, the Hazardous Waste Operations and Emergency Response Standard, commonly referred to as HAZWOPER, in 1986. This standard, which was issued as a final rule in March, 1989, sets forth the requirements of the HASP (29 CFR 1910.120 (b)(4)).

As written, the HAZWOPER standard is a performance standard, as opposed to a proscriptive standard, such as a permissible exposure limit (PEL) standard. This distinction becomes critical: in that performance standards mandate that employers execute certain functions (including, but not limited to, site planning, monitoring, medical surveillance and training) to protect employees from safety and health

hazards. Therefore, following this logic, it is imperative that management take an active role in operations covered in the scope of the HAZWOPER standard.
When discussing management functions, four specific activities are considered classical; planning, organizing, leading (directing) and controlling. Many corporations in the private sector, DuPont, Alcoa, ITT, and RJ Reynolds, to name a few, have recognized the benefit of management involvement in worker safety and health. The HASP provides an ideal transition vehicle to integrate management functions with safety and health activities at hazardous waste sites.
Within the HASP components of the HAZWOPER standard, there is a requirement for a "safety and health risk or hazard analysis for each site task and operation found in the workplan." This component provides a bridge between safety and health activities and daily project management. In order to perform a hazard analysis, a task must be broken down into steps or phases and each phase analyzed for potential safety and health hazards. This process parallels the overall project planning management undertakes for a major construction or demolition project. Typically, a major project is broken down into phases, with each phase assigned a completion date and a projected cost. Based on these principles, the melding of these two "phase systems" into a single project management tool, focusing on safety and health, is a simple process.
While the hazard analysis component provides the most obvious link to managing daily activities, the HASP contains other components that can be used to blend safety and health and daily project management. Several other sections of the HASP, as required by the HAZWOPER standard, that can be incorporated into daily management activities are: employee and supervisory training; personal protective equipment (PPE) requirements; personal and environmental monitoring; and site control and decontamination procedures. The remaining HASP requirements, emergency response procedures, confined space entry procedures, medical surveillance and spill containment, are important considerations at hazardous waste sites, but typically do not change from day-to-day.
The concept that the HASP can be used as a daily project management tool does not exist solely as an intellectual one, but as a practical one, as well. The DOE facility located at Weldon Spring, Missouri, integrates their HASP with day-to-day project management activities. The Weldon Spring Site Remedial Action Project (WSSRAP), where the sole mission of the DOE is to protect human health and the environment, has developed a program that successfully blends day-to-day project management and the HASP. All activity at WSSRAP has been classified into one of three areas: quarry operations, chemical plant, and overall maintenance. Each of these areas function independently of the other in order to achieve the final goal of the site, which is the remediation of all hazardous substances in the facility. However, there is one common element in all areas of operation at Weldon Spring, the HASP.
When a remedial action scope of work is identified at any of the three WSSRAP functional areas, a "boilerplate" HASP is created by DOE and the managing contractor. This HASP is provided to the contractors submitting bids for the project. It then becomes the contractor's responsibility to address the specific hazards associated with the scope of work and incorporate them into the "boilerplate" HASP using specified HASP addendums. This final document is the site-specific HASP that meets the requirements of the HAZWOPER standard.
CASE STUDY
Daily Project Management
In order to facilitate the HASP as a tool for daily project management, WSSRAP project managers work with their safety support staff and communicate the day's activities and precautions to all workers. The project manager is careful to document the discussion and ensure that all workers attend.
Daily project management encompasses a number of activities. First, the day's events are planned and reviewed, as well as the potential hazards and exposures created by the day's operations. Next, the proper teams and equipment are assigned ensuring that the right people and equipment are available for the job. Then, the work is assigned, to include the day's final objective, the type of air monitoring that will be performed, and exposure limits that will be in force. Finally, since the safety officer attends all morning meetings, daily inspections are scheduled, and any results from the previous days monitoring (air, personal, or environmental) efforts are reviewed and discussed. Thus, each element of management, planning, organizing,

directing, and controlling, has been incorporated into the morning's daily safety and project management briefing.
To facilitate the briefing, the site uses either one of two HASP addendums: the Safe Work Plan (SWP) or the Task Specific Safety Assessment (TaSSA) form.
Safe Work Plan (SWP)
In order to ensure that the HASP is site and task specific, WSSRAP requires each contractor to submit SWP's for each of their routine assignments. A SWP typically includes:

Description of Work	Work Location
Est. Start Date/Time	Subcontractor
Equipment	Crew
S&H Hazard Evaluation	Preventive Measures
Hazardous Conditions	Dosimetry Requirements
Air Sampling	Doffing Sequence
H&S Technician Coverage	Badge Color Required For Entry
Approvals	Inspections

Minimum Personal Protective Clothing and Equipment
Special Instructions and Requirements

These headings are specific subsets of the HASP, and help the site comply with section (b)(4)(iii), pre-entry briefing requirements. The SWP also helps the site comply with section (b)(4)(ii), effectiveness of the safety and health plan, by incorporating inspections into daily activities.
The SWP is reviewed during each morning meeting, if the day's activities include that specific task. Any changes or revisions are documented at that time by the project manager. A separate "Briefing Record" form is completed for each meeting and for each SWP, further documents activities and changes to the HASP.
Task Specific Safety Assessments (TaSSA)
The TaSSA provides: safety awareness at the worker level of task performance; an awareness of the requirements for safe conduct of the activity at the inspector level; and a method of control of field activities. A TaSSA is required for all field activities, typically non-repetitive tasks that were not specifically addressed in the HASP or SWP (See Fig. 1). The TaSSA must be revised, dated, and signed daily by those personnel performing the task.
Fig. 1.
The TaSSA includes:
- Authorization names and signatures
- Summary description
- Risk assessment
- Comments/discussion
- Crew and field supervisor signatures
- Inspection records

Once written, a TaSSA must be reviewed and signed off by a competent person, as defined in the written procedures, although not necessarily by the site safety officer. As a measure of control, copies of TaSSA's are maintained at the site and with the Access Control Monitor. Therefore, safety inspectors will always have the TaSSA available during planned and random inspections (See Fig. 2.). The original TaSSA's are logged into a tracking system and filed in the Construction Management office.
Fig. 2.
One of the other aspects of the TaSSA is it's flexibility to deal with work stoppage when a new or previously unidentified hazard is discovered. As long as the form is properly completed and reviewed, a competent person can make a decision to proceed. This procedure minimizes work interruptions, which places an incentive on the part of the contractor to properly analyzes new or previously unidentified hazards. Thus, the TaSSA closes the gap between planned conditions and activities (HASP & SWP defined tasks) and non-routine and unplanned conditions and activities, while expediting the project.
Authority and Approvals
Original SWP's require the review and approval of the Responsibility Assignment Matrix (RAM) Team. The Team is comprised of multi-disciplined members with safety, management, environmental and engineering backgrounds. Since the Team members are more than one individual per discipline, SWPs can be reviewed quickly, if necessary,

as backup Team members have already been selected. Revisions to SWPs must follow the same procedure as original submittals.
TaSSAs receive reviews and approval by a defined "competent person." Depending upon the nature and degree of the hazards, the competent person will consult with the relevant safety personnel, when necessary. Since TaSSAs are written on a daily basis, there are no revision requirements.
Documentation
All SWPs are assigned a log number and tracked in a computer database, to include the revision number and status of pending approval or if approval has been denied subject to revising the SWP. A copy of the SWPs is also onsite (in the field) whenever these operations are occurring. Files are maintained on all SWPs, to include inspection records and any corrective actions recommended as part of these inspections. The daily SWP review is documented using the "Briefing Review Record." The TaSSA is also computer tracked, although log numbers are not yet a part of the tracking system. Final copies are collected daily and filed in the Contract Management office. Although competent person approval is required, these documents are reviewed periodically and at random at the direction of the DOE safety officer, to help verify that the forms are being completed appropriately.
BENEFITS
At WSSRAP, Safe Work Plans (SWPs) have been used for the past 5 years, with Task-Specific Safety Assessments (TaSAAs) being used for the last two years. Over the years, there has been an evolution of the plans, forms, and procedures when using these tools, and the site is convinced that the efforts provide worthwhile benefits and compliment the management planning responsibilities of DOE.
The mission of WSSRAP is the "protection of human health and the environment." Site management is convinced that the proper planning and implementation of safety procedures not only save lives, but significantly contribute to the minimization of unplanned events and interruption of timetables and milestones. Some examples of benefits received are presented in the Fig. 3 and Fig. 4.
Fig. 3.
Fig. 4.
In Fig. 3, you can see a significant increase in the number of manhours (e.g., remediation activities). As a result, you would expect to see a significant increase in the number and rate of injuries. WSSRAP has experienced only some linear increases, where the expected trend was for the increases to extrapolate upward. In Fig. 4, the severity has definitely come under control, although there is a slight increase in the 1993 trend due to a single lost work day case.
Prevention
In addition, daily project management and review of records provide WSSRAP opportunities to mitigate (prevent) circumstances before they become threatening or cause injuries or damage. For example, urine bioanalysis, routinely performed on site workers, indicated an upward trend in 50% of the work crew to a radiological exposure. Upon review of these records, the safety staff investigated work activity and discovered that the work crew was inadvertently modifying their protective cloth doffing procedures, due to some discomfort caused by the clothing ensemble. A training meeting was scheduled, where revised protection and decontamination measures were communicated to the workforce. As a result of a well developed HASP, one that planned for appropriate medical surveillance and a daily management strategy that reviews monitoring data, WSSRAP was able to implement mitigation procedures before exposures were above limits.
LESSONS LEARNED
By incorporating safety into daily project management and SWP and TaSSA procedures, the site has been able to evolve and improve their safety efforts and procedures, further benefiting the work product, schedules, and milestones. A number of specific "lessons learned" initiated changes to procedures and business as usual.
Field Authorization for Quick Turnaround
The TaSSA was developed so any new or unplanned hazards could receive appropriate review and approval prior to the work activity proceeding, but not significantly impeding the process. During the past few years, the planning documents (HASP & SWP) have been much improved, reducing the frequency of TaSSA's for routine activities.
Fixed Fee Contractor Performance
Fixed fee contractors will typically debate any procedure which will slow down work completion, once the contract is signed and notice to proceed is given. WSSRAP found

that appropriate change orders due to change in conditions were easier to document using the information in TaSSAs. In addition, change orders were negotiated fairly, realizing that sometimes there are legitimate changes to conditions or activities.

Subcontractor Participation in Hazard Analysis

Another benefit the site received when implementing SWPs and TaSSAs was subcontractor involvement in safety. Subcontractors were made responsible for completion of safety documents (SWPs and TaSSAs). Since accountability procedures were in place (review, approval, and inspections), the contractor is better motivated to plan their activities safely.

Time Out for Safety

It is never an easy decision to exercise stop work authority. Contractors generally get upset (especially fixed price contractors) when stopping work, and the purpose of stopping the work sometimes gets lost in the subsequent discussion. In response, the DOE Safety Office and its site contractors agreed to a strategy of "time out for safety." Instead of an official stop work, safety concerns observed during inspection receive a time-out. If a solution is not quickly resolved, then stop work authority is exercised.

Inspection Evaluation Criteria

At first, inspectors were evaluated on the number of sites/tasks they inspected per month. It was determined that, although a lot of inspections were being performed, unresolved unsafe practices were still contributing to accidents. The safety office revised their evaluation criteria, holding inspections accountable for identifying corrective actions.

EVOLUTION

Obviously, the program in place at Weldon Spring has produced some very positive results. The record of performance, indicated by the lost workday case rate, revision records and escalating manhours, shows that the integration of the HASP with daily project management provides tangible benefits. However, some areas of the plan are still in the evolution stages. For example, it was discovered that emergency drills were conducted, but not on a regular basis. The determining criteria for emergency drills was a change in conditions that mandated a change in the emergency preparedness plan.

When using a system on a daily basis, an unavoidable result is the generation of paperwork. That in itself is not a major problem, but maintaining, tracking and trending this paperwork is a formidable task. Fortunately, as these problems became apparent at Weldon Spring, worker safety was not compromised.

The majority of these recordkeeping obstacles center around the TaSSA form. While the field use of these forms increased at all three areas of operations, a consistent tracking and documentation system has not yet been developed. TaSSAs are consistently monitored by field supervisors and foreman. What is still needed is a closure action by "corporate" safety personnel. By providing this final link in the chain, "corporate" safety & health personnel will be fully integrated with the TaSSA system.

As is common with hazardous waste site operations, unknown hazards are often revealed as work progresses. At WSSRAP, these occurrences are addressed using the aforementioned TaSSA system. These unknown hazards may result in changes in procedures that may last ten minutes or ten days. This uncertainty, especially in longer intervals, creates a potential duplication problem in the TaSSA system. In other words, no criteria exist for determining when to submit repeat TaSSAs or if a revision to the Safe Work Plan is necessary. An established decision chain or flow chart to determine when to continue submitting TaSSA's or to amend the SWP could eliminate this and help contain the paper explosion.

When a TaSSA is issued, a copy of that form is required to be posted at the worksite and be available for review during inspections by management. Any management official is authorized to conduct an inspection at any time during the operation. When an inspection is conducted, the inspector is required to sign-off on the TaSSA form indicating employees are adhering to the changes specified in the TaSSA. If the employees are not working according to the procedures outlined in the TaSSA, the inspector is required to note this and modify work practices as necessary. These inspections occur randomly and appear to have little established protocol. The creation of an inspection protocol, focusing on safety & health, would assist individuals who lack formal safety & health training in the identification of hazards present at the job site. A standard protocol may also streamline the

inspection process, allowing for more frequent and thorough inspections.
The integration of the HASP with daily project management at WSSRAP appears to be 95% complete. The use of this system has shown positive results in that worker injuries have been reduced while manhours worked have increased, and activities are more hazardous. Injury trend information would be readily available and easily documented by an established trending system. While statistics are kept at WSSRAP, there exists little formal trending process to measure the success of the program. A system that tracks and trends inspection results, lost work day rates, manhours worked or any other industry norm for evaluating safety and health performance would be a benefit to the site.

SUMMARY

The HASP, required by OSHA (HAZWOPER), is designed to be a management planning document, as well as an instrument for daily project management. In order to fully implement the regulatory requirements of a site specific safety and health plan, 29 CFR 1910.120 requires:

- site health and safety plan (HASP);
- pre-entry briefings; and
- effectiveness inspections.

As a performance standard, OSHA allows many methods available to contractors for HASP compliance. WSSRAP has not only developed a model HASP, but requires the integration and participation of project management, safety staff and contractors in regard to hazard identification and control. By implementing safety on a daily basis, the HASP becomes dynamic; part of daily project management. It also improves the working atmosphere not only toward safety, but also toward improving product quality and timeliness. The site has experienced a documented savings in terms of worker injury and has mitigated potential exposure problems. WSSRAP expects to continue evolving and refining the HASP process in order to fully achieve its mission objectives; to protect human health and the environment.

ACKNOWLEDGMENTS

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55-6

AN APPROACH TO SAFETY ANALYSES AND DOCUMENTATION FOR DECONTAMINATION AND DECOMMISSIONING PROJECTS AT DEPARTMENT OF ENERGY NUCLEAR FACILITIES

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ABSTRACT

The requirements for documentation of safety analysis, safety impacts and hazards posed by the U.S. Department of Energy (DOE) facilities and operations are presented in a number of DOE Orders and related directives. In addition, the U.S. Nuclear Regulatory Commission (NRC) has been active in developing guidance for decontamination and decommissioning (D&D) activities for commercial nuclear facilities. However, specific guidance for application of DOE Orders, directives, and other guidance for preparation of safety documentation related to DOE D&D programs has not been developed. An approach to defining the requirements for safety documentation was needed to support D&D Program. This paper discusses the step-by-step approach developed for specifying safety documentation and for the integration of the safety analysis process into the D&D project sequence.

The step-by-step process is based upon the 'graded approach' as presented in DOE Order 5480.23. This approach utilizes the concept of addressing hazard identification and documentation of the safety analysis process. This concept is based upon an evaluation of hazards associated with the D&D activity to be implemented and the condition of the facility after deactivation, rather than the existing safety documentation prepared for production operations. Maximum use of existing project documentation, such as preliminary characterization information and preliminary hazards analyses, are considered in the safety analysis process. The

proposed approach integrates requirements of several DOE Orders including DOE Order 5480.11 5480.23, and 5481.1B, along with other relevant guidance (e.g., applicable NRC Regulatory Guides and NUREGS) into a logical sequence for defining safety requirements. This approach promotes development of safety documentation using the 'team concept'.

The result of this effort was the development of a decision-based process for defining the appropriate levels of safety evaluation and documentation for a facility, based on the existing [postulated] hazards and safety concerns at the time of initiation of D&D field activities. This process allows the safety analyst to make responsible decisions on how much and what type of analyses and documentation would be best suited for a project- or facility-specific situation.

There is no existing program within DOE for safety analysis that has been tailored specifically for D&D projects. As such, this process is of particular importance to the DOE Complex due to recent mission changes at many of the DOE field sites and the subsequent increase in growth of the national D&D program. Use of the proposed approach for development of safety documentation for D&D projects promotes early identification of issues and allows the use of existing standards, criteria, and project management system without creating unnecessary documentation and inhibiting the progress of these efforts.

INTRODUCTION

The requirements for documentation of safety analysis, safety impacts and hazards posed by the U.S. Department of Energy (DOE) facilities and operations are presented in a number of DOE Orders and related directives. In addition, the U.S. Nuclear Regulatory Commission (NRC) has been active in developing guidance for decontamination and decommissioning (D&D) activities for commercial nuclear facilities. However, specific guidance for application of DOE Orders, directives, and other guidance for preparation of safety documentation related to DOE D&D programs has not been developed. An approach to defining the requirements for safety documentation was needed to support the DOE D&D Program. This paper discusses a decision-based process developed for specifying safety documentation and for integration of the safety analysis process into the D&D project sequence.

CONCERNS WITH THE CURRENT SAFETY ANALYSIS APPROACH

The current safety analysis approach for DOE nuclear facilities was developed for facilities dedicated to producing nuclear materials for national defense. The mission of a majority of DOE sites has changed to environmental cleanup and D&D; as such, the requirements and content of safety analysis for production facilities may not be relevant and/or appropriate for D&D of facilities. The current approach for safety analysis for DOE nuclear facilities, along with concerns associated with this approach, are further discussed below.

Existing Requirements For Safety Documentation

A safety analysis must be performed for every DOE nuclear facility that develops and evaluates the adequacy of the safety basis for the specific facility. The safety basis parameters to be assessed include management, design, construction, operation, and engineering characteristics necessary to protect the public, workers, and the environment from safety and health hazards posed by the nuclear facility or nonfacility nuclear operations. The DOE has documented requirements for the safety analysis process in several Orders; however the Nuclear Safety Analysis Reports (DOE, 1992), DOE Order 5480.23 establishes safety analysis requirements for DOE contractors for the D&D of nuclear facilities.

The DOE Order 5480.23 defines the "level of concern" with respect to safety considerations within the framework of Hazard Classification, which drives the requirements for the preparation of safety analysis reports (SARs) for DOE nuclear facilities. All facilities classified as a Hazard Category 3 and above are required to comply with DOE Order 5480.23. Facilities that do not meet or exceed Hazard Category 3 threshold, but that still possess some amount of radioactive material, may be designated as Radiological Facilities. The Hazard Category is to be performed in accordance with the methodology specified in the Hazard Categorization and Accident Analysis Techniques For Compliance With DOE Order 5480.23, Nuclear Safety Analysis Reports (DOE, 1992a), DOE-STD-1027-92.

Radiological facilities are exempt from DOE Order 5480.23, but are not exempt from other DOE and DOE-related safety requirements. The DOE Order 5481.1B, Safety Analysis and Review System (DOE, 1986) identifies safety analysis requirements for facilities that possess significant nonradiological hazards that may not be

addressed by DOE Order 5480.23.
It is conceivable that many of the DOE facilities that have been deactivated and placed in a "D&D-ready" condition may be reclassified as Radiological Facilities and thus will not be subject to requirements of DOE Order 5480.23.
Hazard Categorization and Classification
The DOE Order 5480.23 requires an evaluation of the nuclear activities to be conducted and a classification of the processes, operations, or activities in accordance with the Hazard Classification criteria established in DOE Order 5480.23 and DOE-STD-1027-92. The hazards analysis should address (as a minimum):
Inventory of Hazardous Material - Including an inventory that envelops all radioactive and nonradioactive hazardous materials that are stored within the facility;
Evaluation of Potential Releases - Identification of energy sources that might contribute to the generation or uncontrolled release of hazardous materials;
Hazards Analysis Document - The Hazards Analysis Document (HAD) is the final document in the hazards evaluation process, and provides a more definitive hazards classification based on valuative data. The HAD consists of sections for: Hazardous Material Quantity, Form, and Location; Energy Sources and Potential Initiating Events; Preventative Features; and Mitigative Features;
Hazards Classification - The final facility Hazards Category is defined by the results of the HAD.
Traditionally, the majority of DOE production facilities have been designated as Hazard Category 3 and above. Per DOE Order 5480.23, the graded approach is to be used during the development of the Hazard Classification of a facility.
Graded Approach to Safety Documentation
To avoid the development of unnecessary and unwieldy safety documentation, DOE Order 5480.23 promotes a graded approach to the level of analysis required for a specific facility or operation. The graded approach was developed in an attempt to ensure that the level of information contained in safety documentation is consistent with the level of hazards posed by the facility or operation in question.
The objectives of the graded approach are to ensure that the requirements for analysis, evaluation, and documentation of safety requirements associated with a DOE facility are proportional to the potential hazards associated with the facility. The level of understanding and control of hazards to workers, the public, and the environment should be comparable for all facilities. For relatively simple, uncontaminated facilities, an acceptable level of understanding and control of hazards can be achieved with less sophisticated techniques and less detailed knowledge of facility characteristics than more complex operations. The key factors of the graded approach are:
Justification for the level of analysis and documentation for each hazard considered must be provided as part of the plan and schedule for SAR development and maintenance. The level of analysis and documentation for each facility must be commensurate with:
- The magnitude of the hazards being addressed;
- The complexity of the facility and/or systems being relied upon to maintain an acceptable level of risk; and
- The stage or stages of the facility life cycle for which DOE approval is sought.

The graded approach as addressed in DOE Order 5480.23 is specific for the development of SARs.
For D&D projects, as well as other nuclear operations, it is the evaluation of hazards that dictates the level and sophistication of safety analysis documentation. Although DOE Order 5480.23 generally requires that a new or upgraded SAR be developed for Hazard Category 3 facilities or activities, preparation and submittal of a SAR for environmental activities is required on a case-by-case basis. If a SAR is required, DOE Order 5480.23 allows the application of a graded approach according to the hazard category and the stage of the facility life-cycle. Also, facilities and activities may be segmented or partitioned to allow certain buildings, rooms within buildings, or individual activities to be managed under Hazard Category 3 while all other buildings and activities are managed as Radiological Facilities.
Safety Considerations During Deactivation
As defined by DOE Order 5480.23, the SAR is the focal document for definition of safety requirements for nuclear facilities, there are other documents and processes that must be considered when defining safety requirements for the deactivation of a

facility in preparation for D&D including: a Preliminary Characterization Report, Preliminary Hazards Analysis, and Technical Safety Requirements. Each of these documents are further discussed below.

Preliminary Characterization Report

The Preliminary Characterization of a facility involves the characterization of the hazardous wastes and materials that will be encountered during subsequent D&D operations. Information may be obtained by operating records, process knowledge, and operating personnel, and can be used to formulate the initial characterization of the radiological and nonradiological hazards associated with the facility. Information obtained from the Preliminary Characterization Report may be used to complete the Preliminary Hazards Analysis (PHA) for a facility.

Preliminary Hazards Analysis

The PHA should be performed upon turnover of a facility from deactivation to D&D. This analysis should be conducted based on assurance that the deactivation objectives have been met. The PHA is to compare the post-deactivation radiological material inventory to the hazard classification criteria of DOE-STD-1027-92. The resultant Hazard Classification should reflect the condition of the facility at the time of turnover to the D&D Program.

Technical Specification Requirements

Although preparation of a SAR may not be necessary to address the post-deactivation condition of the facility, a revised set of technical specifications and supporting safety analysis may be required. At a minimum, the existing Technical Specification Requirements (TSRs) or other equivalent safety limits which were used during operations should be revised or deleted to reflect the post-deactivation and pre-D&D condition of the facility. Revised surveillance and preventive maintenance requirements and procedures should be developed for the post-deactivation mode.

It should be noted that during the deactivation process, much of the radioactive substances, Special Nuclear Material (SNM), and hazardous chemical inventories are to be removed from the facility. This is performed to significantly reduce the total quantity of radioactive materials and hazardous chemicals that would be encountered during demolition and decontamination activities.

Safety Conditions for Deactivation

As defined in the Decontamination and Decommissioning Guidance Document - Draft (DOE, 1994) there are several safety conditions that should be addressed prior to transition of a facility from deactivation to D&D including the overall condition of the facility, radioactive and hazardous material inventory remaining, and existing (operational) documentation for the facility.

Ideally, facility deactivation will be planned and performed by the operating program as part of cessation of operations. This goal will be supported by an efficiently planned and executed deactivation program. Typically, the Office of Environmental Restoration (EM-40) will not accept a facility until completion of deactivation. Items that should be addressed during deactivation include (but not limited to):

Removal or stabilization of radioactive source terms;

Removal of SNM to meet less than level Category IV in accordance with DOE Order 5633.3;

Removal of hazardous and dangerous chemical inventories;

Ensure that confinement structures are structurally sound;

Deactivate, consolidate, or cascade the facility Heating, Ventilation, and Air Conditioning (HVAC) systems:

Control facility access to preclude personnel entry other than that required for quarterly radiation and other surveys;

Prevent personnel from utilizing the building and deactivate/clean all personnel support systems;

Remove, to the extent possible, all combustible materials;

Deactivate loss prevention systems and all other electrical systems; and

Install and monitor systems needed for surveillance until D&D activities commence.

The primary goal of deactivation is to meet the acceptance criteria of EM-40. A large part of the acceptance criteria is ensuring the facility is maintained in a safe condition following completion of deactivation activities.

PROPOSED APPROACH FOR D&D SAFETY DOCUMENTATION

The decision-based approach for documentation of safety analysis for D&D facilities focuses on maximum use of the graded approach and the condition of the facility

after completion of deactivation activities. For facilities approaching D&D, the safety documentation should be based on the activities remaining during D&D field activities. Documentation provided on facility operations that were terminated should be the minimum necessary to demonstrate the safety of the facility during D&D field activities. As shown in Fig. 1-1, a decision-based approach has been developed for safety documentation of facilities scheduled to undergo D&D. Elements of the decision-based approach, along with a comparison of the proposed approach to existing guidelines, is presented below.

Decision-Based Approach for Safety Analysis

The decision-based approach uses the graded approach in the development of safety documentation for D&D of facilities and focuses on hazards that exist after completion of deactivation activities. It is likely that most, if not all of the SNM will be removed during deactivation activities. Therefore; the proposed approach reflects the appropriate level of hazards associated with the facility scheduled to undergo D&D. In addition, the proposed approach allows for instances where EM-40 will accept a facility prior to completion of
Deactivation activities. The decision-based approach proposed in Fig. 1-1 is a seven-step process:

1. Review existing operational hazards/safety documentation;
2. Evaluate hazards of the deactivated facility;
3. Verify removal of hazardous/radioactive materials;
4. Determine whether the D&D facility exceeds Hazard Category 3 criteria;
5. Incorporation of the Safety Documentation into the Health and Safety Plan for D&D of the Facility;
6. Prepare a Safety Analysis Report per DOE Order 5480.23;
7. Integrate safety documentation into D&D project documentation.

Each of the key steps of the decision-based approach are further discussed below.

Step 1 - Review Existing Operational Hazards/Safety Documentation

The first step of the decision-based approach consists of reviewing the existing operational hazards/safety documentation. The review of existing safety documentation should focus on the factors used to formulate the existing Hazard Category to determine whether they are appropriate to support initiation of deactivation activities. Generally, the safety documentation for the operational facility will not have to be revised prior to initiation of deactivation activities; the operational envelope should be sufficient to address accidental conditions that may occur during deactivation.

Step 2 - Evaluate Hazards of the Deactivated Facility

The second step of the decision-based approach encompasses evaluating hazards of the deactivated facility. As discussed earlier, deactivation activities should address removal of nuclear and hazardous material from the facility. The types of hazards that would be expected to be encountered after completion of Deactivation would consist primarily of radiological exposure and occupational hazards (i.e. physical hazards). A thorough review of the remaining hazards is needed to ensure proper classification of the facility.

Step 3 - Verify Material Inventory

The third step of the decision-based approach includes verifying that the majority of radioactive and hazardous material has been removed from the facility. This step should be performed at the time of acceptance of the facility by EM-40. There may be instances where EM-40 accepts a facility prior to completion of deactivation activities. As such, the material inventories of the facility should be documented accordingly for identification of potential accidental conditions and proper classification of the facility.

Step 4 - Determine Whether Facility Exceeds Hazard Category 3 Criteria

The fourth step of the decision-based approach focuses on determination of the Hazard Category of the facility. The Hazard Category of the facility will determine the type and level of safety documentation to be completed for a facility scheduled to undergo D&D and should use the graded approach to the maximum extent possible. Because of the removal of nuclear and hazardous materials during deactivation, it is anticipated that the majority of facilities to undergo D&D will qualify as less than Hazard Category 3. The hazards of the facility are evaluated and subsequently documented in the remainder of the decision-based approach.

Step 5 - Incorporate Safety Documentation Into Health and Safety Plan

The fifth step of the decision-based approach encompasses documenting results of the

hazard classification process in the Health and Safety Plan for D&D of the facility. The Health and Safety Plan for D&D of a facility typically addresses the occupational hazards and radiological exposure issues that would be encountered during D&D activities. In addition, the Health and Safety Plan would address administrative requirements associated during implementation of field activities. If the facility does not exceed the Hazard Category 3 threshold, the Hazard Classification would be documented in the Health and Safety Plan and addressed as necessary to support D&D field activities. For facilities that still exceed the Hazard Category 3 criteria the safety analyst would perform the sixth step of the decision-based approach; preparing a SAR. The fifth step would not be applicable for those facilities that exceed the Hazard Category 3 threshold of the decision-based approach.

Step 6 - Prepare Safety Analysis Report Per DOE Order 5480.23

The sixth step of the decision-based approach entails documenting results of the safety analysis process in a SAR prepared in compliance with DOE Order 5480.23 using the graded approach. The review of safety conditions for the facility, radioactive and hazardous material inventories, Hazard Classification, and other elements of the safety analysis process, as required, are to integrated into formal documentation. The level of detail associated with the 21 elements of a SAR should be based upon the graded approach.

Step 7 - Incorporate Safety Documentation into Project Documentation

The seventh, and final step of the decision-based approach consists of incorporating the safety documentation into the D&D project documentation. Both the Health and Safety Plan and the SAR should be incorporated into the Decommissioning Plan for the facility. It is anticipated that for a majority of the facilities to undergo D&D only a Health and Safety Plan would be required; a SAR would not be required to be completed due to the level of hazards to be encountered during D&D field activities.

Evaluation of the Initial State Safety Conditions

A major element of the proposed decision-based approach is evaluation of the safety conditions that remain at the time of D&D of the facility. The evaluation of safety conditions should focus on the radioactive and hazardous material inventories that remain after completion of deactivation activities. Ideally, all of the nuclear and hazardous material should be removed during deactivation; however, experience has shown that many of the facilities scheduled to undergo D&D still have radioactive and hazardous material remaining in the facilities and process lines. A thorough and proper review of material inventories should be conducted and incorporated into the safety documentation. In addition, although Deactivation focuses on removal of energy sources from the facility, occupational hazards may still exist that would influence the level of detail required for the safety documentation. Both material inventories and occupational hazards should be addressed during development of the safety documentation.

Evaluation of D&D Project Safety Requirements

The Decontamination and Decommissioning Guidance Document - Draft (DOE, 1994) provides guidance on the overall DOE D&D process and delineates key D&D decision points for large projects. There are seven phases to a typical D&D project. Of those seven phases, safety documentation is developed primarily in the first and third phases: during transition of a facility to EM-40 (Phase I) and during the environmental review phase of a project (Phase III). Fig. 1-2 shows the interrelationship between the overall D&D process and the safety analysis process. Phase I of the D&D process starts with termination of operations, including establishment of a surveillance and maintenance program and ends with achievement of safe shutdown and transfer of a facility to EM-40. Project documentation associated with Phase I includes a preliminary characterization and hazards analysis, establishment of an effective S&M program, and initiation of the EM-40 budget cycle. Work performed in support of the proposed activity or project during Phase I is primarily preparatory in nature and provides the framework for development of the project plan and preliminary engineering work. Safety documentation associated with D&D of a facility may be prepared at during Phase I of the D&D process; however, it is anticipated that the majority of safety documentation would be completed during Phase III.

Phase III of the D&D process addresses environmental and safety documentation required to complete D&D of the facility. Engineering work is performed to define

and estimate the D&D alternatives and other engineering studies are performed to support the preparation of the appropriate National Environmental Policy Act (NEPA) documentation. Phase III of the D&D process ends at D&D key decision 2, Approval of the D&D Alternative and Approval to Start Detailed Engineering. During Phase III the safety documentation for the proposed activity or facility is completed and included as a part of the Decommissioning Plan. For those facilities that are below the Hazard Category 3 threshold the Health and Safety Plan may suffice as the final safety documentation.

Comparison of the Proposed Approach to the Existing Guidelines

The existing guidelines for safety analysis documentation were developed to support a production mission for national defense. As such, the requirements and guidelines address issues that would be associated with operational activities. The proposed approach was developed for facilities that have terminated operations and are to undergo D&D; therefore, issues associated with deactivation and D&D of a facility are addressed. These issues focus on declassification of a facility from Hazard Category 1 to below the Hazard Category 3 threshold and activities conducted during deactivation (i.e., removal of energy sources). The proposed approach fulfills the intent of DOE Orders 5480.23 and 5481.1B to establish and evaluate the adequacy of the safety bases of facilities and integrates safety documentation into the D&D process as defined in the Decontamination and Decommissioning Guidance Document - Draft (DOE, 1994).

CONCLUSIONS

The decision-based approach provides a process for defining requirements and developing safety analysis documentation for facilities scheduled to undergo D&D. The approach assists the project team in defining the appropriate level of documentation required to support D&D and maximizes use of the graded approach in completing safety analysis documentation. The seven-step process is a tool to be used by the safety analyst to focus the level of effort required to complete safety analysis documentation and assists integration of the safety analysis documentation into the overall D&D process.

Findings and Observations

There is no existing program within the DOE for safety analysis that has been tailored specifically for D&D projects. As such, this process is of particular importance to the DOE Complex due to recent mission changes at many of the DOE field sites and the subsequent increase in growth of the national D&D program. Use of the proposed approach for development of safety documentation for D&D projects promotes early identification of issues and allows the use of existing standards, criteria, and project management system without creating unnecessary documentation and inhibiting the progress of these efforts.

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55-7

DEVELOPMENT OF AN AUDITABLE SAFETY ANALYSIS IN SUPPORT OF A RADIOLOGICAL FACILITY CLASSIFICATION

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ABSTRACT

In recent years, U.S. Department of Energy (DOE) facilities commonly have been

classified as reactor, non-reactor nuclear, or nuclear facilities. Safety analysis documentation was prepared for these facilities, with few exceptions, using the requirements in either DOE Order 5481.1B, Safety Analysis and Review System; or DOE Order 5480.23, Nuclear Safety Analysis Reports. Traditionally, this has been accomplished by development of an extensive Safety Analysis Report (SAR), which identifies hazards, assesses risks of facility operation, describes and analyzes adequacy of measures taken to control hazards, and evaluates potential accidents and their associated risks. This process is complicated by analysis of secondary hazards and adequacy of backup (redundant) systems.

The traditional SAR process is advantageous for DOE facilities with appreciable hazards or operational risks. SAR preparation for a low-risk facility or process can be cost-prohibitive and quite challenging because conventional safety analysis protocols may not readily be applied to a low-risk facility.

The DOE Office of Environmental Restoration and Waste Management (EM) recognized this potential disadvantage and issued an EM limited technical standard, No. 5502-94, Hazard Baseline Documentation. This standard can be used for developing documentation for a facility classified as radiological, including preparation of an auditable (defensible) safety analysis.

In support of the radiological facility classification process, the Uranium Mill Tailings Remedial Action (UMTRA) Project has developed an auditable safety analysis document based upon the postulation criteria and hazards analysis techniques defined in DOE Order 5480.23, Nuclear Safety Analysis Reports. The auditable safety analysis focuses on hazards and potential accidents associated with relocation of mill tailings at UMTRA sites. The majority of hazards related to UMTRA activities are construction hazards. These hazards are addressed in the auditable safety analysis. The auditable safety analysis also demonstrates the limited need for backup (redundant) systems at UMTRA sites. This, in turn, expedites preparation of the auditable safety analysis by greatly reducing the need for discussions regarding adequacy and sufficiency of redundant systems.

INTRODUCTION

Within the last 10 to 12 years, views have changed regarding safety analysis documentation required for U.S. Department of Energy (DOE) facilities, including what must be addressed, what is adequate, and what approvals are required. This new perspective is due, in part, to the publics' demands for assurance that DOE facilities are operated in a safe manner and not endangering the workers or the public. At the same time, assurance must also be given that the environment is protected.

Partially in response to public demands, the DOE has issued two principal Orders pertaining to safety documentation: DOE Order 5480.23, Nuclear Safety Analysis Reports (Change 1, dated 10 March 1994)(1), and DOE Order 5481.1B, Safety and Analysis Review System (dated 23 September 1986)(2). DOE Order 5481.1B was the initial Order regarding development of Safety Analysis Reports (SAR). In the spring of 1992, DOE Order 5480.23 was issued, defining additional elements to be included in the SAR and superseding DOE Order 5481.1B, SAR Requirements for Nuclear Facilities. By the end of 1992, all DOE nuclear facilities were required to prepare SARs in accordance with DOE Order 5480.23.

Developing SARs at most DOE nuclear facilities was very time-consuming and costly. Different evolutions of the SAR document were required (Preliminary Safety Analysis Report [PSAR], Final Safety Analysis Report [FASR], and possibly a Safety Evaluation Report [SER]); backup (redundant) systems had to be identified so that critical safety systems could continue to function; and adequacy and sufficiency analyses were required for relevant structures, systems, and components (SSC) that were part of the facility operations or processes.

DETERMINING ADEQUATE SAFETY DOCUMENTATION

"Adequate" safety documentation continues to be a challenge at DOE facilities that have low hazards, have processes with the potential for only localized on-site consequences, and/or have a limited mission life. For example, the Uranium Mill Tailings Remedial Action (UMTRA) Project, based in Albuquerque, NM, is responsible for successful remediation of mill tailings (residual radioactive materials contained in sand or loose soil) generated during the processing of uranium ore. This DOE project includes 24 processing sites and over 5000 vicinity properties located in 10 states and on 2 Indian reservations. Remediation of the mill tailings is necessary because of potential adverse health effects from prolonged exposure to

residual radioactive materials.
The remediation process is divided into two distinct tasks: 1) either encapsulation on-site or relocation of the mill tailings, and 2) verification of effective containment of the mill tailings through a ground water monitoring program.
Task 1 ensures encapsulation of the mill tailings at a location that is considered to be geologically safe, either on-site or off-site. This geologically safe repository is referred to as a "disposal cell." The relocation process commonly consists of loading the mill tailings into large dump trucks, transporting them to and depositing them at the disposal site, shaping the deposited mill tailings into their final configuration using bulldozers and scrapers, and then placing a final cover of clay and rip rap rock over them. The majority of site activities relating to task 1 are those commonly encountered on a construction site using heavy equipment.
This construction orientation is also reflected in the lack of redundant systems for UMTRA Project site activities. The SSCs at UMTRA sites are not required to provide continuous operational support, as would be expected at a moderate- to high-risk facility. For example, at a to moderate- to high-risk facility, HEPA filtration systems might be required to operate continuously. This would commonly include a standby diesel emergency power system to ensure that HEPA filtration can continue to function. SSCs used at UMTRA Project sites are not required to be operational to maintain system integrity. For example, if a haul truck or conveyor system has a failure, the equipment is replaced, operations continue without the component, or operations cease until repairs can be made. In addition, no system failure can result in an unacceptable worker exposure or environmental release.
Task 2 site activities include regularly scheduled sampling of ground water from wells drilled adjacent to the disposal cell. While task 2 activities will continue well into the next century (2010), the personnel performing ground water sampling will not be exposed to any hazards other than those encountered in the general work place.
The majority of UMTRA sites are located in the southwestern portion of the United States where mills have been closed. Because of the mill closures, most of the towns adjacent to UMTRA sites are quite small or nonexistent. It takes approximately 2 to 3 years to accomplish task 1 at each UMTRA Project site. Actual periods of operation during each year of task 1 may only be 9 to 10 months long because of inclement weather.
When all of these factors are taken into consideration, it is readily evident that following the traditional PSAR/FSAR process for each UMTRA site would be cost prohibitive. In addition, the UMTRA site task 1 activities could conceivably be finished prior to receiving approval on the FSAR. Clearly, the UMTRA Project needed to have a more timely and cost-effective method of developing its safety analysis documentation.
As the UMTRA Project reviewed its options regarding developing acceptable safety analysis documentation without placing unacceptable constraints on project dollars, DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports (3), was issued by the DOE in December of 1992. This standard provides a consistent method for determining the hazard category of a DOE facility, based on an "unmitigated release" of radioactive material. After reviewing UMTRA Project sites against the criteria specified in DOE-STD-1027-92, Project personnel determined that UMTRA sites could be classified as "radiological" verses "nuclear."
While this radiological facility determination relieved the UMTRA Project of having to develop a traditional SAR, the question of what would considered "acceptable" safety analysis documentation still remained. DOE Order 5481.1B does not apply to construction activities and it had insufficient depth and breadth to use for guidance in developing the safety documentation.
AUDITABLE SAFETY ANALYSIS (ASA) STRATEGY
In August 1994, while the UMTRA Project was evaluating guides to use for the safety analysis development process, the Office of Environmental Restoration and Waste Management (EM) issued a limited technical standard, DOE-EM-STD-5502-94, Hazard Baseline Documentation. In this standard, an auditable safety analysis (ASA) report is defined as the type of safety documentation required for a radiological facility. The ASA report must describe, in an auditable (defensible) manner, safety processes and programs used at the radiological facility.

To meet this requirement, the UMTRA Project used the principal safety criteria and SAR elements defined in DOE Order 5480.23. DOE Order 5480.23 was chosen for several reasons: its volume of definitive safety analysis content and structure, its recognition by the DOE community as the definitive written article on SAR development; and its guidance regarding what site activities should be addressed, how the SAR report should be structured, and how to focus SAR report development. Because the UMTRA Project chose to use DOE Order 5480.23 and it was not mandated by the DOE Headquarters, sections of the Order that were not considered applicable to the UMTRA Project did not have to have substantial objective evidence generated to justify their nonapplicability.

For example, Topic 8, "Inadvertent Criticality Protection," is clearly not applicable to the UMTRA Project. In a traditional SAR, this type of determination would require fairly extensive documentation to support it. In the ASA report prepared by the UMTRA Project, this section is simply not included. However, memorandums were generated for UMTRA DOE Project Office concurrence, documenting which elements of the Order would not be included in the ASA report. This approach allowed the ASA report to be focused on issues directly relevant to the UMTRA Project while still documenting all decisions made governing the report development process.

Because most of the UMTRA sites have the same basic types of SSCs, the UMTRA Project Office elected to develop a programmatic ASA report that addressed remedial action activities used at all of the sites. This allowed development of a single ASA report rather than 24 site-specific ASA reports.

The ASA report addresses site activities including those related to operation of heavy equipment, conveyor systems, and decontamination facilities. Site programs addressed in the ASA report include operations, training, maintenance, fire protection, quality assurance, decontamination, and emergency response. Project issues addressed in the ASA report include hazard analysis and facility classification, organization and administration, and derivation of technical safety requirements.

The ASA report includes an executive summary, a description of the process for development of the ASA report, a discussion of applicable elements of DOE Order 5480.23, a listing of applicable regulatory documents, and references. In addition, an integral element of the ASA report is a description of site hazards in sufficient detail to determine if existing controls used to limit target exposures are adequate.

In addition to the guidance provided in DOE Order 5480.23, the UMTRA Project also used the safety system analysis techniques contained in the Management and Oversight Risk Tree (MORT) fault tree process. The combination of these two types of review approaches allowed identification of appropriate hazards, review of existing barriers to control target exposure to the hazards, and a determination of the adequacy and sufficiency of the barriers. This same logic is used throughout the ASA report to provide a consistent technical baseline for the review process. It also allowed hazards relating to SSCs and programmatic elements to be discussed.

For example, SSCs of a area used for storage and dispensing of fuels (referred to on the UMTRA Project as a "fuel farm") are discussed as follows:

> FIRE PROTECTION: The quantities and types of fire protection equipment required to support UMTRA activities are based upon the fire hazard analysis techniques as defined in DOE Order 5480.7 and National Fire Protection Association (NFPA) 101. While each site hazard characteristics may vary slightly, all sites have potential fire hazards and use fire protection equipment as described below:
>
> Fuel Farm: Most of the UMTRA sites have a fuel farm area used for storage and dispensing of gasoline, diesel fuel, motor oils, and lubrication products. The fuel farm area is also used for storage of used motor oils for subsequent removal by an off site vendor.
>
> Hazards: Explosion and combustible fire hazards due to large volumes

of
flammable and combustible liquids stored in close proximity to each other.

Controls: Administrative controls include placing no-smoking placards at all fuel farms, developing site procedures, and training site personnel regarding the prohibition of parking construction equipment within 50 feet of fuel farm areas. Fuel farm storage tanks containing gasoline and/or diesel fuel are grounded in accordance with the requirements of NFPA 30, and dispensing containers are bonded in accordance with 29 Code of Federal Regulations (CFR) 1926.152.

Barriers: Earthen berms around perimeter of fuel farm; metal stanchions at fuel farm dispensing areas; and placement of fuel farms at remote locations.

Equipment: All fuel farms are provided with portable fire extinguishers (hand-held and/or wheeled). Sizes and quantity of the fire extinguishers are based upon volumes of flammable and combustible liquids stored. In addition, portable fire extinguishers (hand-held and/or wheeled) are placed in accordance with the spacing requirements defined in NFPA 10 and NFPA 30.

Governing Documents: Construction Environmental Safety and Health (ES&H) Management Plan, Training Plan, Site-specific Health and Safety and Hazard Communication Plans, Fuel Farm Operations Procedure, and Site Worker Course.

While this technique allows potential operational and/or process hazards to be readily identified and evaluated, programmatic elements (e.g., maintenance, operations, quality assurance, and training) need a slightly different approach. Programmatic elements commonly do not present hazards. Ideally, the level of detail developed for a programmatic element should be balanced with the type of work activity involved. For example, an overtly constrictive maintenance program for a low-hazard-level activity (e.g., changing light bulbs in a laboratory) is not desirable nor is it cost effective.
To determine what programmatic elements should be included in the ASA review process, the UMTRA Project used the functional area approach, as defined in the Environment, Safety and Health Configuration Guide (5), issued by the DOE in July 1993. The ES&H Configuration Guide provides uniform architecture for identifying and defining major ES&H topics, or "functional areas," based upon required activity. The UMTRA Project personnel determined that the following functional areas effectively control site activities or processes: configuration management; engineering, design, and construction; emergency preparedness; operational readiness review; environmental protection; fire protection; maintenance; management systems; occupational safety and health; operations; packing and transportation; quality assurance; radiological protection; safeguards and security; training; environmental restoration and waste management; and personnel.
Each of these functional areas was examined for potential applicability to the ASA report, and appropriate programmatic functions were included. For example, the maintenance management program is discussed as follows:

INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

The UMTRA Project has developed a maintenance management program using the graded approach criteria defined in DOE Order 4330.4B, Maintenance Management Program. The program has been implemented at the site level to ensure that equipment and components can operate dependably without excessive down time. While each site's maintenance management program can vary, depending on actual SSCs used, most of the

maintenance programs have the following elements in place:

Organization: UMTRA subcontractors are responsible for the day-to-day maintenance activities of all subcontractor equipment, including preventive, predictive, and corrective duties. The site maintenance group commonly consists of a supervisor and two to six support personnel.

Training: UMTRA maintenance personnel are trained by having new personnel work with experienced site staff. All training is performance-based to ensure that site personnel have been effectively trained prior to preforming tasks independent of direct supervision. Training records for maintenance personnel are retained at the site until completion of the contract.

Testing: All equipment, systems, and processes are tested in accordance with manufacturers' specifications and with site procedures.

Facility condition assessment surveys: Site management and subcontractor personnel perform daily and weekly walk-around inspections to verify condition of equipment and components. Selected pieces of equipment (haul trucks, scrapers, conveyors, rock separators, etc.) are subjected to a walk-around inspection prior to each use or daily start-up.

Work order system: Each site subcontractor is responsible for developing a work control system suitable for the equipment being serviced. As a minimum, all work order system(s) meet the applicable requirements of DOE Order 4330.4B and 49 CFR 396.

Maintenance procedures: Each site subcontractor is responsible for determining the depth and breadth of procedures required to support maintenance activities. All maintenance procedures are reviewed for adequacy and sufficiency by DOE Project Office personnel or a designee.

Preventive maintenance: Each subcontractor establishes a preventive maintenance program to ensure that all pieces of construction equipment, including haul trucks and supporting components, are effectively maintained.

Seasonal facility preservation: The majority of UMTRA sites have a winter shut-down program in place due to inclement weather. All heavy equipment is either removed from the site or winterized to prevent damage due to freezing temperatures.

Equipment repair history: All subcontractor maintenance programs are required to maintain relevant records of repairs to equipment. As a minimum, site programs meet the applicable requirements of DOE Orders 1324.2A and 4330.4B and 49 CFR 396.

Governing Documents: ES&H Management Plan, Standard Operating Practices (SOP) Manual, Maintenance Management Program Plan, and site- specific procedures.

All elements of the ASA report analyzing hazards and/or programs follow this same logic and all determinations of acceptability are supported by reference to adherence or implementing plans and/or procedures.

APPROACH TO THE DEFENSIBLE REQUIREMENT

As discussed previously in this paper, one of the requirements established by DOE-STD-5502-94 is that the ASA report be defensible. To meet this requirement, the UMTRA Project included in the ASA report a reference to the actual plan or procedure prepared to control site hazards or address programmatic issues. In addition, all plans and procedures referred to in the ASA report were reviewed for adequacy and

sufficiency in controlling the site hazard or governing the site program. This feature of the UMTRA ASA report also provides a convenient reference when procedures need to be revised or updated.

ASA REPORT RESULTS

The completed ASA report is approximately 75 pages long and is currently at the DOE UMTRA Project Office for final review and approval. Costs to the UMTRA Project have been minimal. Personnel responsible for development of the ASA report had previous experience in generation of SARs ,and the ASA report was generated over a period of months while other Project tasks were performed. This is significant because development and approval of a traditional SAR can take years.

The completed ASA report is already being used to orient new personnel or subcontractors to the unique aspects of UMTRA Project activities. It also allows site-specific activities or issues to be identified.

The ASA report will be reviewed on a yearly basis to ensure that all relevant safety processes are still identified. This yearly review also allows equipment or processes no longer used for UMTRA activities to be deleted from the report. This review process ensures that the ASA report reflects only current site activities.

CONCLUSIONS OF THE ASA REPORT PROCESS

The ASA report process has allowed the UMTRA Project to demonstrate that site processes are being conducted in a safe and defensible manner without placing undue constraints on budgets or site schedules.

By generating a Project-wide ASA report, individual sites with like processes can be addressed in a single document.

The ASA report can be used to identify potential equipment and procedure requirements for new UMTRA sites being started.

Because the ASA report is reviewed on a yearly basis, new processes or activities can be readily identified. In addition, processes no longer used can be readily deleted.

Due to the structure established for the ASA report, revisions can be accomplished in a timely and cost-effective manner.

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Session 56 -- Who's Listening: How to Communicate Better

Co-chairs: Margery Olson, SAIC;
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56-1

THE PRESS: FRIEND, FIEND OR FOE?
HOW ABOUT: TEACHER, CRITIC, OMBUDSMAN

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ABSTRACT

There appears to be strong anti-press phobia among scientists and engineers. More often than not, the press is viewed as playing an adversarial role always looking for that "gotchya" sensational story attached with a tabloid headline. The press has

a role as a useful teacher, critic and ombudsman. The press actually can be a great tool for getting to the public. Journalists can be proponents or adversaries. And there are many venues other than daily newspapers for doing so: trades, education and scientific journals, university publications and local periodicals.

The scientific and engineering community, in many cases, does not have the universal acceptance by the public. How can this community increase its credibility and obtain a broad range of coverage on such pressing issues as environmental risk, implementing innovative cleanup technologies, economic and job development from the weapons complex's scientific resources?

A recent General Accounting Office analysis found new, innovative cleanup technologies are not being used at DOE sites. Once reason cited for this was how local stakeholders were often not familiar with these newer technologies and associated them with an unacceptable level of risk. Although numerous conferences have been held and thousands of papers have been presented on the technological capabilities for addressing U.S. environmental problems, the public is still not convinced the scientific community is credible.

This paper intends to address the need for scientists/engineers to move beyond just talking to each other in a vacuum, and to discuss options for developing an educational as well as political forum. Points and counterpoints will be made by a former DOE Assistant Secretary, a newsletter publisher and a journalist from the trade press who are experienced in both sides of the issues.

INTRODUCTION

Almost 50 years have passed since the end of World War II. The final events that ended the war were the bombing on Hiroshima and Nagasaki. The two Japanese cities were decimated with the birth of the nuclear weapons era. Although over 40 million people were killed during the period of the war from 1939 to 1945, the horrible legacy of the mushroom clouds emanating from the deserts of New Mexico and Nevada are indelibly scorched in the minds of each succeeding generation born after the war.

At the time President Truman made the decision to use the Atomic Bomb tens of thousands of engineers, scientists and craftsmen had contributed to what each thought was their contribution to ending the war. Nuclear energy was harnessed but the birth of a nuclear weapons industry emerged and was entwined with all future developments. Nuclear technology evolved into weapons, energy in the peaceful harnessing of the atom and medical isotopes for diagnostic and treatment applications never received supremacy of independence. The Manhattan Project was the acme of the technical application of a multidisciplinary capability-democracy and free enterprise at its finest. As the applications of nuclear technology developed, the weapons sites of the wartime production became multifaceted industrial incubators. Technology of chemical separation, major construction, scientific laboratories with the most complete grouping of the nations scientific excellence was assembled to maintain the peace through application of the weapons of mass destruction, nuclear energy for naval propulsion, nuclear energy for electrical generation and nuclear medicine.

In 1948, a select team was assigned the task of reviewing the operations of the wartime facilities in their transition to peace time application. The report, by the AEC Safeguard Industrial Health Advisory Board (1), conclusion was:

a. Methods used during war time emergency may not be acceptable for continued peace time applications;

b. Radioactive and hazardous materials are being discharged into the environment and will have long term consequences; and,

c. When these facilities are returned to the public, the public will be a hard taskmaster.

Now that the communist threat of massive force appears to be dissipated and the United States appears to at least be at a lower risk of a massive outside force threat, DOE, the agency that assumed the legacy of the Atomic Energy Agency and its successor, the Energy Research and Development Agency, is under attack from the four estates of the realm and at a crossroads for survival.

The benefit of nuclear deterrence is now castigated from all sides, the end products of its legacy, Uranium, Plutonium and radioactive by-products, are addressed as the most toxic substances known to man by a technically weak press.

President Clinton has targeted a major budget reduction for the Department of Energy (DOE), $26 billion over 5 years. Congress, in its effort to reduce the size of

government, has indicated a potential for eliminating the Energy Cabinet post and the agency. It appears that the hero of the cold war, the too-cheap-to-meter potential for nuclear energy, the hope for fusion energy , solar, wind, clean coal, human genome and the electric car may suffer the ignominious distinction of being eliminated in the first knee jerk reaction of the "Reinventing the Government" revolution.

The question raised by the authors is how could this happen when major efforts were being initiated to open research facilities of one of the largest scientific resources in the world, an agency that saved the world for democracy, now focusing 100,000 employees, 40,000 scientists and billion of dollars to environmental accountability, restoration and development of technology of which the Department of Commerce touts as a $300 billion year industry.

It may be serendipitous, coincidence or just bad timing, but there appears to be inadvertent or possibly a recognition by the Fourth Estate that the Department of Energy in the past two years had reached the end of its useful purpose.

The authors conducted an extensive review of two years of press coverage on the Department of Energy by the national press, the trade and scientific journals and media in general-not a scientific study but at least a sufficient cross-section-to attempt to identify any possible cause for the almost universal conclusion that something is wrong with the Department of Energy.

The authors reviewed two years of DOE-related articles in the three major national newspapers: The New York Times, The Washington Post and The Los Angeles Times. In addition, two years of trade periodicals, i.e., Engineering News, Science, Physics Today, MIT's Technology Review and the Oil and Gas Journal. Also researched were magazines such as OMNI, Popular Science, Popular Mechanics, Business Week, Fortune, Time, Newsweek and U.S. News and World Report. All were reviewed over the two year period to see if there was any obvious trend in national coverage of DOE while the nation was regrouping for "Reinvention", "Contract for America" or any other potential government evaluation.

It was interesting to see a developing pattern of self-inflicted bleeding and purging by the Department of Energy during this two year period resulting in a overwhelming impression of long-term mismanagement and public irresponsibility. The Washington Post and The New York Times, as expected, carried the most features on the Department of Energy in the two year period; twenty-two (22) major stories. Table I lists the various publications that were reviewed and the results of specific areas tabulated as positive or negative in presenting a view of the Department as would be perceived by the general public are tabulated in Table II. As can be seen from Table I the dominant inference of the articles is negative. The predominant negatives in Table II articles were on mismanagement, inefficiency and inept managing of programs. The second major area of national negative press was related to radiation experiments in the 40's, 50's, 60's and 70's. In the majority of reports no credit was given to DOE for internal findings, corrective actions that were implemented without GAO audits or Congressional Committee hearings. However, due to the wire services of the three major newspapers read inside the Beltway-The Washington Post, The New York Times, and the Los Angeles Times-the Local Constituency Daily receives the end product of these three major national reports. If their perception is negative, then the whole country gets a negative perception of DOE.

If one considers that The New York Times, The Washington Post, and the Los Angeles Times have the major wire services to the regional local newspapers and the dominant theme is negative then Congress and its constituents see a total negative impression of the Department of Energy and as a result the current "Knee Jerk" reaction to significantly reduce or eliminate the Department.

The authors then reviewed what are considered to be the prestigious trade journals that cover Department of Energy related items in science, technology and management. The periodicals reviewed were Science, Physics Today, Science News, Engineering News Record, Chemical and Engineering News, Oil and Gas Journal, and Scientific American. As would be expected of the 69 articles related to Department of Energy 60 were positive and 5 were negative, 4 were either neutral or non-committal.

One of the unusual aspects of the research identified a major deficiency in an area where the Department is committing $250 million to technology related to the electric car. Three publications, Science, Atlantic Monthly, and Business Week ran cover stories on the future of electric vehicles and battery research. None of the

articles reference DOE or the US Advanced Battery Consortium or National Electric Vehicle Program funded by DOE. GAO in GAO/PEMD-95-7 (2), Electric Vehicles stated that the US program is fragmented and that the US success may be overcome by foreign competition. If after spending $250 million with the major auto manufacturers and battery manufacturers DOE is not mentioned in any of the major stories investigative press is some what deficient to say the least. The Oil and Gas Journal Dec. 19, 1994 (3), praised former Deputy Secretary Henson Moore, current Deputy Secretary Bill White and wrote a five page laudatory article on Deputy Assistant Secretary Reginal Spiller, but wrote an editorial in the same issue saying it's time to get rid of DOE.

The Feb. 20, 1993 Science News (4) contained a cover story on technology and water purification and never mentioned one DOE facility even though DOE is spending a half-billion dollars per year in groundwater research. Bioscience Magazine Sept. 1994 (5), wrote a ten page article "Renewable Energy: Economic and Environmental Issues." The article referenced five DOE published papers out of 60 referenced documents, but failed to mention a single DOE energy lab even though DOE spends $2 billion on renewable energy and related areas per year. Hazwaste Magazine (6) wrote a cover story on "Nuclear Decontamination and Decommissioning" and failed to mention DOE experience although DOE has decommissioned over 50 reactors, West Valley Reprocessing Plant and innumerable hot cells and provided the major research and funding for the TMI-2 Decon. Talk about having your light under a bushel.

That now gets to the fundamental issue is the press friend, foe or ombudsman? The mass media in most cases appears technically inept. This, combined with the transition from the secrecy mentality, the subliminal technologist's fear of being used or misquoted and the big brother mentality of the Department of Energy's press office, the engineer/scientist has limited ability to provide technical articles to the press/media which can be understood by the average taxpayer. The New York Times runs a technology section once per week, The Washington Post runs a two column technology quarter page daily and a periodic technology supplement and DOE stringers are always looking for a one-up on their competitors. Of the three major publications that printed positive articles about DOE-Discovery, OMNI and Popular Mechanics-they published layman articles provided by federal laboratories to the magazine technical writers. DOE labs and lab contractors wrote over 3,000 technical papers last year; Waste Management '94 produced 900+; Spectrum, 94-600; American Nuclear Society, 150; American Chemical Engineers, 300; Society of Mechanical Engineers, 150; and another 800 papers were produced throughout the other major disciplines. Most were written for technical understanding and not for educational purpose.

It may be too late to save key elements of DOE from bureaucratic extinction, but it is imperative that the work being done within the various facilities of the Department become visible to the media, Congress and the taxpayer. The expenditure of $6-8 billion in research each year must compete with medicare, education, crime, AIDs and state and municipal infrastructure.

The authors found one publication issued quarterly by Lawrence Berkeley Laboratory, Research Review (7), which met the quality and educational level that could be used for publication in magazines of the class of OMNI, Popular Science and Sky (United Air Lines) that would provide a more balanced perspective for the expenditure of $6 billion of the taxpayers money. The LBL Review was able to publish in almost layman's terms articles on quarks, clouds and climate, cell aging, human genome and DNA mapping which could be published directly in any public technical or semi-technical magazine and could be a model for other DOE laboratories.

The press is neither friend nor ombudsman without input. From the limited study conducted by the authors, the FOE is the scientific community within DOE who publish only for other scientists, ergo providing hymnals to the choir. In today's diminishing funding availability in a reinventing knee jerk budget, the oft heard academic slogan "publish or perish" is the need of the future. It may be too late for the current DOE system but the future requires scientific literacy for public support and only the scientist and engineers can help the press/media become technically literate.

This could be accomplished by planning to have at least ten articles published each year in laymen or layperson journals from each laboratory or facility. That would be only 300 topics per year. To keep America competitive the utilization of 40,000 focused scientists supported by taxpayers requires more effort than hymnals for the

choir. The press to today's hyper environment thrives on "bad news is good news." To make the press's technical job easier and the scientific career more secure and, more importantly, recognized as a national resource of considerable economic return, requires directed effort layperson technical journalism and an understanding that the public trough supports hogs and chickens. The chickens are dedicated to continuous support and the hogs are committed to a one-time banquet. It appears that the banquet may be over and the chickens need to learn how to crow to continue to be fed and feed a technically deficient press.

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HYPERMEDIA: A TOOL FOR EFFECTIVE STAKEHOLDER INVOLVEMENT

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ABSTRACT

Computer-based multimedia communication has recently emerged as an effective tool for many kinds of presentations, and its special features -- accessibility, consistency, user appeal, and clear communication -- make it especially effective for augmenting public participation activities. Jason Associates Corp., in cooperation with S.M. Stoller Corp. and Coleman Research, developed a multimedia product for the U.S. Department of Energy to summarize for general audiences the key information contained in the 3,400 page DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement. This product has been and continues to be used to inform the public about spent nuclear fuel management. Experience deploying this application in venues throughout the country has resulted in some valuable lessons learned: 1) Location defines the amount of use, 2) Increased use of sound, video, and graphic enhancements does not always improve communication, 3) Use of kiosks can have significant advantages and disadvantages, 4) Youth can be a conduit to adults.

INTRODUCTION

Meaningful dialogue is critical to the success of decision-making processes, especially those involving public agencies, but dialogue requires that all involved parties first have a clear understanding of the issues at hand. While agencies are recognizing that effective communication produces better, more easily implemented decisions, the task can be particularly challenging when they must present audiences with large volumes of technical information.

Computer-based multimedia communication has emerged over the past five years as an accessible and effective tool for many kinds of presentations. While there are a variety of multimedia applications, they all share common characteristics. They typically bring together, in a common digital format, several media -- including text, illustrations, photos, sound, animation, and video -- so that a presentation becomes multi-sensory and user-driven, rather than simply a passive, linear experience.

Multimedia can be operated on a variety of platforms, including personal computers, kiosks, projection devices, interactive games, and Internet. Well-designed multimedia presentations -- whatever the platform -- get the attention of audiences and improve retention of ideas at a recall rate two or three times that of static presentations, making them ideal for educational or training purposes (1). By

employing a variety of media, they help users to understand concepts that are difficult to convey through text alone. More recently, the increasing popularity of laptop computers has made portable presentations possible for business.
The U.S. Department of Energy faced a particularly challenging communication task in 1994 as part of its preparation of a complicated environmental impact statement. The Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement (SNF & INEL EIS), a 3,400-page multi-volume document issued in June 1994, covered spent fuel management decisions for the entire DOE complex as well as site-specific decisions about environmental restoration/waste management activities at INEL. Because the National Environmental Policy Act requires Federal agencies to obtain public comment on environmental impact statements and to involve stakeholders in making decisions about them, DOE sought innovative ways of communicating information to the public about this large, technically complex document and its subject.
As one tool for improving communication and encouraging public involvement in spent nuclear fuel management decision making, DOE elected to create a computer-based multimedia application. "Choosing a Course," the interactive multimedia (or hypermedia) product designed and developed for DOE by Jason Associates Corporation and S.M. Stoller Corporation, Coleman Research, summarized for general audiences the key information contained in the draft of the SNF & INEL EIS. Included were background information about the DOE weapons complex and the spent nuclear fuel issue, opportunities for citizen involvement, descriptions of potentially affected sites, alternatives to spent nuclear fuel management and their implications, and site-specific implications for the various alternatives (2).
The goal was to produce a multimedia product that was strictly consistent with the content of Volume 1 of the draft SNF & INEL EIS (the portion that dealt with DOE programmatic spent nuclear fuel management), but that was more user-friendly than the draft document itself. While the text and graphics for "Choosing a Course" were drawn directly from the SNF & INEL EIS, the multimedia format made it possible to select a level of information (and supporting photographs and graphics) appropriate for general audiences.
DOE elected to use different platforms for the EIS hypermedia for different purposes. Since DOE and the U.S. Navy planned to conduct public meetings about the SNF & EIS in potentially affected communities across the U.S., they chose a kiosk version of the product for those occasions. Kiosks are floor-mounted cabinet installations designed to make information accessible to many people and are commonly found in shopping malls, airline terminals, and other public places. Users typically operate multimedia products in kiosk form by making selections on a touch-sensitive monitor screen.
For individual use, hypermedia was made available on disk for home computers, where it can be operated by making selections with a mouse. Interested citizens may request copies of the product for either Macintosh or PC. In addition, the application "Choosing a Course" available on Internet via the INEL Home Page (http://INEL.GOV/).
Kenneth A. Chacey, deputy director of DOE's Office of Spent Fuel Management, noted that a multimedia product was particularly well suited to the task of providing information to stakeholders. " The power in hypermedia is the ability to make full use of a multi-faceted medium to convey complex activities to the general public through meetings, interactive workstations, and information superhighways, like Internet," he said.
TECHNICAL DESCRIPTION
Before any communication tool can be created, one must first identify the audience and messages to be communicated. DOE's goal was to provide information to the public so that they could make informed decisions about the complex issue of spent nuclear fuel management. They wanted to make available to the public a product that would enable average citizens to understand the problems that DOE faced in managing and storing spent nuclear fuel, as well as the various alternatives that DOE was considering as a solution.
The requirements for this particular product were that it be
  User friendly -- with a transparent user interface
  Platform independent (IBM/Macintosh) -- a run time version that did not require the user to purchase additional software

Lowest common denominator hardware requirements so that it could be run on a typical home computer.
In order to reach as broad an audience as possible and to make the product platform independent, the developers identified existing software that allowed creation of a run-time version that did not require the user to purchase special software to run the program and did not require special royalty fees from the client or the user. For use on garden-variety home or office computers the application was developed with certain constraints in mind -- including disk size, rapid access memory (RAM), and (primarily with IBM) the possible absence of a sound card. As a result of these constraints, the developers chose to limit this product to three media: text, photographs, and drawings.
The characteristics of multimedia that make it effective for communication in business, training, and education also make it ideal for public participation activities and help to overcome these barriers.
Accessibility: Navigating a series of technical documents in search of a particular item of information can be difficult for both agency representatives and stakeholders involved in decision-making processes. In these circumstances, well designed multimedia products can offer fast, user-controlled access to specific information.
The developers of "Choosing a Course," for example, approached information design from the user's point of view, using the items that DOE felt stakeholders would most want to know about as primary topics. The computer-based, menu-driven format allows users to locate and select topics quickly via the kiosk touchscreen or Macintosh/PC mouse. The advantage for DOE staff and representatives was that they could use hypermedia as a training tool to prepare themselves for public meetings, while stakeholders could get quick answers to many questions.
Opening with an initial animation screen, the body of information is contained under five key topics represented by icons (or buttons). This table of contents represents the first decision point in the program and gives the user the option of selecting one of five topics about spent nuclear fuel (Fig. 1).
Fig. 1. Flow chart schematic of first level architecture for "Choosing a Course."
Overview summarizes important information about the spent nuclear fuel program and the SNF and INEL EIS and provides other relevant background material.
Public Participation outlines the requirements of the National Environmental Policy Act relating to public involvement in decision making, as well as the relationship of the SNF and INEL EIS to various other NEPA documents.
By selecting Current Site Information , the user can get information about any site at which spent fuel management activities are, or may at some point be, taking place, including DOE sites, U.S. Navy facilities, university and research reactors, and potential ports of entry for foreign research reactor fuel. This section provides detailed information about the five DOE sites being considered for major roles in spent nuclear fuel management
Management Alternatives allows the user to select any of the alternatives to spent nuclear fuel management, to compare the alternatives, and to see the national implications of each. This selection includes maps illustrating the spent fuel transportation activities that would be required under each alternative
Selecting Site-Specific Information , allows the user to select a site, then see the implications for that site specifically for any of the spent fuel management alternatives. Included is information on the follow topics: transportation, environment, facilities, socioeconomics, health and safety, and risk.
Under each of the five topics in the table of contents more information is layered. Each topic expands into a tree of options with further options tiering off of successive sublevels. This layering of information allows users to browse deeper and deeper into the subject matter or allows them to stay at very summary level information. Multi-level design is one of the defining features of this system's architecture. A user can truly "get into" the information and spend hours perusing the system or just read the summaries and be in and out in 10 minutes. The advantage of the technology and architecture is that it allows different users to use the system to meet their individual informational needs in a non-linear way.
DISTRIBUTION AND DEPLOYMENT OF THE SYSTEM
An obvious target audience is the interested public, meaning those who are likely to attend public meetings, visit reading rooms, or ask to be placed on the mailing list. Providing this group with access to the kiosk and/or desktop version provides

them with the opportunity to quickly get to information of interest and as a result provide more meaningful and focused comments. By having the kiosk available prior to public meetings DOE hoped to increase the quality (i.e. on topic, and related to the decision being discussed) of oral comments received. The deployment of kiosks was targeted to both of these distinct audiences.

One goal of public information is to make information more accessible and digestible. Accessibility can be defined in terms of convenience and usability. Printed information is convenient because it can be delivered by mail to one's house, however, receiving a 3,400 page document by mail does not make the information accessible if you are only looking for answers to specific questions of interest. Summaries and brochures can be helpful, but often lack detail and are inherently linear in the way information is organized. Interactive multimedia can make detailed technical information more usable and convenient. As a result of improved communication to stakeholders, DOE would improve its chances of receiving better information from stakeholders.

Because the interactive multimedia system requires a computer system to run, distribution of the desktop version is limited to those who own or have access to computers. Kiosks, however, can be appropriately placed in nearly any venue from public meeting sites, public libraries, and reading rooms to shopping malls, public schools, public buildings, and museums. Kiosks, therefore, offer a significant advantage over other forms of information in that they can place the information where it is convenient for the public rather than forcing the public to find the information. This provides opportunities for outreach to an audience that is often referred to as the "silent majority."

Reaching the "silent majority" is not likely to move them to activism and may not result in additional public participation, but it does offer an expanded opportunity for individuals to become informed about the subject and allow them to make the decision as to whether or not they wish to become involved.

To reach these two distinct audiences, a kiosk deployment action plan that identified appropriate venues throughout the country specifically targeting communities being considered as alternatives for spent fuel management. Upon completion of the program, kiosks loaded with the system were sent to public meeting locations.

At publication, kiosks or systems were located at appropriate sites in 5 locations in Idaho including the Shoshone-Bannock Reservation, 2 locations in Georgia , 2 locations in Washington D.C., 1 location in Tennessee, and 1 enroute to Nevada. Kiosks have been placed in shopping malls in Idaho Falls and Pocatello, Idaho; the Eastern Idaho State Fair; Pocatello Idaho City Hall; the INEL Pocatello outreach office; the Shoshone-Bannock Public Library; the Idaho Falls public library, the Idaho Falls Airport; lobbies of SRS and INEL buildings; University of South Carolina , Ruth Patrick Science Center; and the American Museum of Science and Technology in Oak Ridge, TN. Additionally, kiosks or demonstrations have been provided at conferences and workshops including meetings of DOE's Transportation External Coordinating Group, the Whitehouse conference on Technology for a Sustainable Future; a DOE conference on diversity, several national meetings of the American Nuclear Society, the DOE NEPA Compliance Officers Meeting, the International Association of Public Participation Practitioners annual meeting, and the Waste Management 95 Symposium.

LESSONS LEARNED

Experience deploying this application in venues throughout the country has resulted in some valuable lessons learned: 1) Location defines the amount of use, 2) Increased use of sound, video, and graphic enhancements does not always improve communication, 3) Use of kiosks can have significant advantages and disadvantages, 4) Youth can be a conduit to adults.

Location. At public meetings and at conferences and workshops, the amount of use the system got was proportional to the convenience of its location. When the kiosk or system was place in traffic patterns or in open areas, it was used more frequently. When the system was in a separate room, away from a traffic pattern or meeting room, use decreased.

When possible, the location of the system should be optimized to ensure maximum exposure. Often, room design or space constraints limit control of location, but this variable should always be considered and optimized.

Use of audio, video, and graphics. While the ability to use the above media are

what sets multimedia apart from printed information and posters, they must be thought through with the communications objectives in mind. A similar system that had a more linear design but relied more heavily on sound and video was not as popular with a large group of students because it did not contain much depth of information in comparison to the subject system. The two systems, however, where designed with different communication objectives in mind. The key driver of choices to use or rely on audio, video, and graphics should always be communication goals and target audience.

Another lesson relating to use of sound is that it can be off-putting. Ambient noise levels vary in both location and time. If sound is an integral part of the multimedia design, volume levels must be adjusted at the deployment location to ensure it is intelligible under all or most conditions and is not too blaring when ambient noise levels are low. If sound is an augmentation not an integral part of the design, the system should allow sound levels to be adjustable and allow the option to turn sound off when it is not desirable or should allow users to "short-circuit" sound when it is not needed.

Advantages and disadvantages of kiosks. Use of kiosks can have significant benefits, but those benefits come with a price. Having the system in a kiosk allows placement in locations such as lobbies of buildings, shopping malls, libraries, etc., without staffing or security. This allows greater access and minimal labor effort. The system can also be operated on a standard computer but in that configuration requires staffing to deter theft. While the size of a kiosk is an inherent disincentive to theft, it also complicates shipping and handling logistics. To ensure maximum utility and realize greatest benefit, one should consider whether or not the staffing will be available and then determine which system configuration is most desirable.

Reaching adults through youth. Depending on where and how the system is deployed, it often attracts youth, in part because it may resemble a video game, and in part because youth are not as intimidated by technology as adults. While this may not seem like the intended use of the system, it does have benefits and provides some value added. First, youth are often accompanied by adults who often look over the shoulders of their children. Additionally, adults are frequently lured to the system by their children who have higher curiosity. Other parents use the system to teach there children about how the technology works. Finally, youth can carry messages back home to parents resulting in achieving some additional level of awareness and communication. While use of the system by youth does not always bring these benefits, they cannot be discounted.

CONCLUSION

While accurate public information is only one part of community decision making, it is a first step in building consensus (3). "Choosing a Course" enabled users to visualize and compare implications of all proposed alternatives side by side on a site-specific basis, thus allowing them to formulate their own ideas of a successful solution and encourage their participation in the decision-making process.

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ADVANCED TECHNICAL TRAINING FOR AN INTERNATIONAL AUDIENCE OF PROFESSIONALS, SCIENTISTS AND ENGINEERS: VIDEOCONFERENCE/VIDEOTAPE FROM THE UNIVERSITY OF NEW MEXICO

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ABSTRACT

Budget cuts and increasing demands for a technically competent work force are problems facing industry and government today. Training requirements must be met in a cost-effective manner. The videoconference format is a proven method that saves

money on training and travel costs. The videotapes produced from training programs extend the productivity in any facility.
However, designing a training series that meets the needs of a diverse audience composed of scientists, engineers, and management from government, industry and universities is extremely challenging.
National Environmental Technology Network (NETN), a department of The University of New Mexico's College of Engineering, has a proven track record in developing and producing effective videoconference training programs for industry, government, national laboratories, and universities. To date, seven successful series have been completed: Total Quality Management (two series, one for service organizations and one for manufacturing); Hazardous Waste Management; Waste Minimization and Pollution Prevention; Environmental Risk Management; Radioactive Waste Management, and Mixed Waste Management. Two series are in production: Decommissioning (6 parts) and Pollution Prevention (5 parts). National awards from the United States Distance Learning and the United States Environmental Protection Agency attest to NETN's leadership in providing advanced technical training using distance learning techniques.
This presentation describes how to develop, produce, and market an environmental television series and reveals the keys to successful videoconferencing by describing the past successes of NETN's videoconference training. The presentation characterizes how videoconferences can be used as university-developed advanced technical training for shifting the work force from defense projects to environmental remediation and protection. It also illustrates how the effectiveness of this training is enhanced by partnerships that evolve between the originating university and the receiving sites. Additionally, the presentation addresses the technical aspects of the production of past series.
INTRODUCTION
Professionals must continue their education throughout their careers to stay abreast of the latest technologies. For example, to keep up with developments in their field, engineers require retraining on an average of every four years. The need varies, depending on the precise discipline. Additionally, many technical professionals currently face significant changes in their careers, as their positions shift from defense-related activities to environmental fields.
With the budget cuts and increasing demands for a technically competent work force confronting industry and government today, the emerging training requirements must be met cost-effectively. The videoconference format represents a proven method for communicating the most up-to-date information concerning technical, environmental and management topics. According to Daniel Barron, distance education is a means of ". . .taking quality education to the people who need and want it. Distance education is to instruction what bookmobiles and branch libraries are to reading" (1).
Through this method, thousands of professionals can receive the most current information and participate in training programs simultaneously, via satellite, without incurring the cost of travel and perdiem expenses. The series are structured to stimulate creative interaction and promote a significant exchange of information among technical professionals and stakeholders confronting the increasingly complex environmental challenges of today's world.
Broadcast on Cband, Kuband, and through NTU, these videoconference series can be received anywhere in the United States. Most organizations already have the equipment necessary for receiving these instructional television broadcasts. Those not having this capacity can purchase a satellite dish for a minimum investment, a one-time cost for equipment. Videotapes produced from training programs and the accompanying instructional workbooks extend the capacity for training in any facility. Those who participated in the live training sessions have an opportunity to review material and clarify important points. Those who could not attend the live broadcasts can watch the videotapes and complete the training as their schedules permit.
During the live broadcasts, the training experience extends far beyond that of being a viewer. The programs afford every participant the opportunity to interact by means of question/answer sessions throughout each program, via a toll-free telephone number, e-mail, or the use of a fax machine. The interaction is amplified by accepting questions before the programs air, throughout the programs, and after the programs are completed.

Designing a training series that meets the needs of a diverse audience of scientists, engineers, and managers from government, industry and universities is extremely challenging. As Reza Azarmsa explains, "Much like any other area of mediated communication, teleconferencing is both a science and an art" (2). National Environmental Technology Network (NETN), a department of the University of New Mexico's College of Engineering, has an exceptional record for developing and producing effective videoconference training programs for industry, government, national laboratories, and universities.

NETN is part of an environmental consortium funded by the Department of Energy (DOE). The Waste-management Education and Research Consortium (WERC) is composed of New Mexico State University, University of New Mexico, New Mexico Institute of Mining and Technology, Sandia National Laboratories, Los Alamos National Laboratory, and Navajo Community College. Members have teamed together to help solve the environmental problems facing our nation, collaborating in three areas: research, education, and training.

In addition to a variety of videoconference series, six to eight full-credit environmental courses are broadcast from the three member universities on KU-band satellite each semester. WERC also sponsors a summer research lecture series and offers a certification program.

PAST SUCCESSES OF VIDEOCONFERENCE TRAINING AND THE TWO UPCOMING SERIES

To date, NETN has produced seven successful videoconference series: Total Quality Management (two series, one for service organizations and one for manufacturing); a 14-part Hazardous Waste Management series, an 8-part Waste Minimization and Pollution Prevention series; a 7-part Environmental Risk Management series; an 8-part Radioactive Waste Management series, and a 6-part series on Mixed Waste Management. During 1995, NETN will be producing and broadcasting two new series. The first, Decommissioning, is a six-part series addressing decommissioning within the Department of Energy's Environmental Management program. The second series, Pollution Prevention, examines technologies for applications from government installations to cottage industries.

These live, interactive videoconference series have been broadcast to over 167 sites with diverse audiences. The training has spanned the United States, reaching 8,000 participants in 48 states-100 industry sites, 64 military/government installations, 32 universities, and 46 hospitals.

National awards from the United States Distance Learning Association (USDLA) and the U. S. Environmental Protection Agency attest to NETN's leadership in providing advanced technical training using distance learning techniques. These series have received awards for the past three years, and most recently, NETN earned an award for "Most Outstanding Distance Education Network" from the USDLA.

HOW TO PRODUCE AN ENVIRONMENTAL TELEVISION SERIES

To select a topic for a videoconference series, NETN surveys DOE sites, requesting information about their current training needs. When sufficient interest is expressed in one particular area, and the funding for production procured, the topic becomes the focus of the next videoconference series.

In the early stages of developing a videoconference, NETN conducts an "Environmental Experts" workshop, where environmental professionals meet to listen to speakers and participate in panel discussions and question/answer periods. To simplify information transfer, workshop participants complete surveys for NETN. These surveys ask respondents to describe the issues they consider critical for professionals in these fields to address; list the concepts that need to be communicated to stakeholders; identify recognized national experts in the fields; and provide names of periodicals, national organizations, and associations important for people involved in this area of expertise.

The next step consists of advisory board meetings, which lay the foundation for the videoconference. National experts involved in previous programs, scientists from the institutions that represent WERC, and stakeholders are all invited to attend. To provide an overview of a series' production, speakers with past experience in televised training describe the roles and responsibilities of those involved in the process. For example, technical advisors identify program leaders and make sure they are willing to serve in that capacity. The technical advisors must see the overall program content and make sure all aspects of appropriate topics are covered. They also work with the NETN staff and provide the last line of review for participants' instructional manuals.

The individual program leaders recruit nationally recognized experts in the field for their program. They maintain a balance in the program content and may also serve as presenters.
The program presenters need good communication skills in order to deliver their material and to interact with the participants during question/answer periods. Presenters are encouraged to incorporate graphics, video clips, humor, and real-life examples to keep their presentations interesting. Each presenter submits charts, suggested readings, test questions, glossary terms, and a brief biography to NETN, which prepares the material in a standard format for the participants' instructional manuals.
During the next portion of the advisory board meeting, members begin the planning process by developing a proposed series outline. The outline consists of specific program topics, any particular emphasis to be considered for each topic, and possible case studies.
As the topics develop, the technical advisors and the advisory board suggest possible program leaders -- experts in the field with backgrounds in industry, business, and government. NETN contacts these individuals, and those who agree to lead join with the advisory board and NETN staff to form the nucleus of the evolving series. The program leaders then select five or six more people with expertise in their program topics to serve as presenters.
Because NETN seeks presenters from all over the United States, much of the program design occurs during a series of conference calls. Once a plan is established for the individual programs, the program leaders meet with the NETN staff in Albuquerque to fine tune the content of each program and assure continuity for the series as a whole. This not only reduces redundancy, it also confirms that the participants will receive an adequate amount of technical background information. Then, as the scheduled air dates approach, conference calls among those involved in individual programs continue as needed.
Meanwhile, behind the scenes, a team of graphics artist, writers, and editors prepare the presenters' material for the instructional workbooks and return the final draft to the presenters and the technical advisors for review. The workbooks for each program contain the following information: intended audience, program description, presenters' charts, suggested readings, test questions, presenters' biographies and addresses, various questionnaires, an information request form, and a participant's evaluation form.
The finalized manuals are sent to each site prior to the program broadcast. Two copies of the workbooks are sent -- one unbound, for duplication and distribution to all students planning to attend the training, and one bound, to serve as reference material and augment the site's videotapes of the programs.
Simultaneously, the video producer pulls together all the elements necessary for live broadcasts. Scripts are written and case studies are chosen. Arrangements for on-location productions at the sites are made, the case studies are shot and edited for the broadcasts. All elements for live broadcast are orchestrated with the studio, down to the selection of soundtrack music.
Prior to the first program's broadcast, NETN conducts a two-hour facilitator training program. Experience has shown that site facilitators contribute to the overall success of this televised training when they function as proactive members of the team. The facilitator training program covers a wide range of instructional material. Facilitators learn technical details, such as how to establish a comfortable environment for the viewers and how to check the operation of the equipment. They learn how to find the appropriate audience and publicize the upcoming training at their site. They receive tips about effective ways to interact with series advisors and are encouraged to seek experts within their organization who can enhance the learning experience.
In addition to the televised training, facilitators receive a training packet to serve as a reference during the course of the series. Given this overview of what to expect from the distance education experience and how to make it meaningful for the participants, the facilitator often serves as the foundation for the partnership that develops between the university and the receive sites.
One week after the facilitator training program, the first program is broadcast. Every program within a series contains a minimum of five or six presenters from industry, government, and universities with varied scientific backgrounds and expertise. Each program devotes one segment to regulations that pertain to the topic

of discussion. To increase effectiveness, live presentations are mixed with video footage that demonstrates the technology involved in the program topic, as well as relevant case studies that illustrate practical uses of the material. Customized workshops and on-site training are also available.

Publicizing this training event occurs at two levels. NETN can assist the sites with publicity by developing a customized brochure, news releases, and advertisements. A brief promotional tape is also available to the sites. On the broader spectrum, NETN notifies former participants of the upcoming series, writes press releases, and develops detailed brochures for distribution. To help ensure a wide array of participants, the NETN Director publicizes the videoconference training at trade shows and at national meetings on distance education and environmental issues.

THE KEYS TO SUCCESSFUL VIDEOCONFERENCING

The keys to producing effective videoconference series extend into every facet of production. In the preliminary stages, well-defined topics and carefully selected presenters with excellent credibility set the stage. Comprehensive instructional workbooks enable the audience to follow along easily and record their notes. Question/answer periods with ample time for discussion encourage interaction that facilitates technology transfer and lasting partnerships among the sites.

Regular feedback from the participants following each program enables the production staff to constantly improve on the quality of the series. When a program concludes, participants complete two forms: a Technology Questionnaire and a Participant Evaluation. The facilitator returns these forms to NETN, where a thorough review of the evaluations is completed. This direct input from the receiving sites allows the technical advisors, program leaders, presenters, and production staff to improve the quality of the series by capitalizing on what went well, and modifying the programs to make improvements where indicated.

Timing is also an important consideration. Experience shows that short programs, regularly offered, are the most effective. For example, NETN broadcasts these live videoconferences at regular intervals, on the second and fourth Wednesdays of each month. And finally, the role of the on-site facilitator cannot be overemphasized. Well-prepared facilitators can play a proactive role in distance education, as they bridge the gap between the television studio and the audience and promote an open exchange of ideas and technology transfer.

HOW VIDEOCONFERENCES CAN PROVIDE ADVANCED TECHNICAL TRAINING FOR SHIFTING THE WORK FORCE FROM DEFENSE PROJECTS TO ENVIRONMENTAL REMEDIATION AND PROTECTION

As the work force shifts from defense to environmental careers, working professionals need environmental information for current and future jobs. They require immediate, high-level training in specific environmental fields. NETN's goal is to assist the Department of Energy's Risk Management Program in complying with the Congressional mandate of a 30-year remediation initiative.

Advanced technical training and technology transfer represent a proven method of disseminating valuable information to engineers, scientists, researchers, hospitals, universities, and laboratories around the world. Videoconferences provide unique networking opportunities, allowing participants to exchange ideas with recognized environmental experts. During the interactive sections, participants are encouraged to ask specific questions that pertain to their particular site. Throughout the series, audience members develop the tools to change directions in their careers, with a strong emphasis on environmental clean-up and future use of a site.

Transmissions via satellite offer economic advantages, such as savings on travel, per diem, and workshop registrations. The price per site allows unlimited attendance during the live broadcasts. To prolong the life of the information contained in these series, the sites have access to videotapes of the programs and a bound copy of the instructional workbooks for their library.

After each series concludes, participants receive certificates of completion and earn Continuing Education Units (CEUs) through the University of New Mexico.

The Department of Energy currently receives NETN's environmental series in all DOE Operations offices across the United States and at DOE Headquarters. This is part of the DOE effort to retrain their work force from defense to environmental careers and to support environmental technology transfer.

Recognition and endorsement have been extended to NETN by the American Society of Mechanical Engineers (ASME) and the Hazardous Waste Action Coalition (HWAC).

HOW EVOLVING PARTNERSHIPS BETWEEN THE ORIGINATING UNIVERSITY AND THE RECEIVING SITES ENHANCE THE TRAINING

NETN achieves balance and diversity by seeking presenters from industry, government, and universities. Working closely with experts from such varied backgrounds has enabled NETN to develop strong partnerships among the presenters as well as the receiving sites.

The ongoing interactive programs promote mutual awareness and understanding. Unique partnerships can evolve through a distance education experience. For example, Sandia National Laboratories and Los Alamos National Laboratory subscribed to NETN's first series on Total Quality Management in 1990. Both organizations have participated in every subsequent series, and a mutually beneficial understanding has developed among the University of New Mexico and both national laboratories. Because individuals feel free to ask site-specific questions, participants throughout the nation come to understand each other better, discovering common ground and seeking common solutions.

TECHNICAL ASPECTS OF PAST SERIES AND THE UPCOMING DECOMMISSIONING AND POLLUTION PREVENTION SERIES

As mentioned earlier, the technical aspects of a videoconference series -- program content and specific emphasis -- are determined through the input from viewers of previous series, advisory board meetings, and an Environmental Experts workshop. NETN's first two distance education initiatives pertained to Total Quality Management (TQM). A 14-part series offered an engineering/ management approach to TQM, and a 6-part series focused on TQM as it applies to service organizations. Hazardous Waste Management, the next series, consisted of the following 11 programs: Program 1) Introduction: What Is Waste?; 2) Risks Associated with Hazardous & Radioactive Wastes; 3) Transport Processes Related to Wastes; 4) Waste Form Modification; 5) Site Characterization; 6) Sampling and Analysis; 7) Soil and Groundwater Remediation I: Physical/Chemical Processes; 8) Soil and Groundwater Remediation II: Biological Processes; 9) Radiation and Radioactive Materials; 10) Radioactive and Mixed Wastes Management; 11) Waste Minimization and Series Close.

Waste Minimization and Pollution Prevention - adhering to the logical progression of the content, NETN followed this series with an 8-part videoconference. The series began with an overview which defined waste minimization and described the need. Then the training progressed through the following topics: 1) Overview -- Why Minimize Waste?; 2) Where Do We Start Waste Minimization?; 3) How Does Recycling/Reuse/Reclamation Make Economic Sense?; 4) Are the Right Product/Process Designs Being Addressed?; 5) Solvents and Organic Chemicals; 6) How to Implement Minimization in Metals, Plating, and Electrical Interconnects; 7) How to Minimize "End of Life" Problems; and 8) Planning and Preparing for the Future.

Environmental Risk Management - a 7-part series sought to bridge the gap between technical issues and social concerns. Program titles included 1) Risk: Terminology, Concepts, Methods, Applications and Why Risk Communication Is Difficult; 2) How Do We Decide What Is Risky?; 3) Identifying the Gap: Divergent Technical and Social Methods for Risk Characterization; 4) Quantification of Models for Risk Assessment; 5) Risk Characterization: Synthesis and Communication; 6) Risk Assessment: Communicating Results and Public Perception; and 7) Risk-Based Decision Making: A Final Act?

Radioactive Waste Management was produced concurrently with the Environmental Risk Management series as an 8-part series (broadcast on alternating Wednesdays). This series began with 1) An Introduction to Radioactive Waste Management, and the following programs introduced 2) Interactions Between Radiation and Matter; 3) Decommission and Decontamination; 4) Transportation Issues in Radioactive Waste Management and UMTRA; 5) Low-Level Radioactive Waste Management; 6) High-Level Radioactive Waste; 7) Transuranic Radioactive Waste Management; and 8) New and Other Technologies for Radioactive Waste Management.

Mixed Waste Management, a 5-part series began broadcast on April 6, 1994, and concluded on June 1, 1994. Generated by government facilities, research laboratories, hospitals and universities, mixed waste is a major concern in remediation projects, including the clean up of Department of Energy weapons complexes and old radioactive waste disposal sites. The mixed waste dilemma became apparent as the course addressed the most basic questions: "What are the characteristics of mixed waste?" "What are our treatment options?" "How do we store mixed waste?" "Can we dispose of it?" and "What is the citizen's role in the process?"

The series consisted of 1) Generation and Characterization of Mixed Waste; 2)

Storage of Mixed Waste; 3) Minimization and Treatment of mixed Waste; 4) Technical Development of Mixed Waste Issues; and 5) Remediation Activities.
Decommissioning, a six-part series, began on February 8, 1995, with an introduction to the series with a goal and vision statement from U.S. Department of Energy. An inventory and history of the major decommissioning sites with overviews and the various phases of the decommissioning process were included.
Each program consists of technical presentations by experts focusing on Decommissioning terminology, problems involving contamination and radiation along with successful approaches. This series will stress the importance of good planning, pre-job briefings, and dress rehearsals with mock-ups to uncover problems and same time. Participants learn the techniques for dismantling and segmenting equipment and structural members and demolishing structure.
Pollution Prevention will be broadcast on alternating Wednesdays beginning on May 3 as a 5-part series. Pollution Prevention is far more powerful, more painless, less costly than continually degrading the environment by just "managing waste." Only through massive educational and awareness programs can the attitudes and values of the people be made sensitive to the need for waste minimization/pollution prevention and the seriousness of everyone's responsibility and moral obligation to protect the environment for future generations.
The end-of the pipe approach to waste management is a losing strategy. Striving to be in environmental compliance is not the answer; and managing waste is not only expensive, but poisonous to the environment and to the health of the public. Pollution prevention -- the front-end approach--makes business sense and is less destructive to the environment.
Communication and participation are critical elements in our national efforts to identify and explore environmental challenges. NETN develops distance education programs that highlight technologies and options for waste management and environmental remediation.
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56-5
OHIO'S STATEWIDE LOW-LEVEL RADIOACTIVE WASTE EDUCATION PROGRAM: PHASE II - OUTREACH
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ABSTRACT
At The Ohio State University, a team of faculty members from the Nuclear Engineering Program and OSU Extension are conducting a multi-phase state wide low-level radioactive waste education program. The program's objective is to provide information that Ohio's citizens and their elected officials can use to participate in discussions and decisions related to low-level waste. In Phase I, the team developed a set of 27 fact sheets and four exhibits addressing a variety of topics related to low-level waste. The team's goal was to make these educational materials accurate, easy to understand, and unbiased. An extensive review process was developed to help meet these goals. A set of overhead transparencies was developed from the fact sheet materials to be used in presentations.
In Phase II, Outreach, the materials are being distributed across Ohio using a variety of channels with emphasis on the County Extension Offices, which are located in each of Ohio's 88 counties. Over 7,000 sets of fact sheets have been distributed to State Legislators and their aides, County Commissioners, hospitals, local health departments, civic organizations, high school teachers, college faculty, the media, and other interested people. Exhibits have been displayed at professional meetings, county fairs, mall shows, and the Ohio State Fair. Presentations have been made to a variety of groups, and a 20-minute slide/tape program has been prepared for use at short meetings. A day-long media workshop has been held, and a second one is planned. Summaries of the fact sheets have been prepared for use in the print media

and distributed to all newspapers.
Effectiveness of the entire program is difficult to determine at this early date. However, questions asked at the presentations and responses to pre- and post-presentation questionnaires indicate that participants understand the material discussed and use it when thinking about low-level waste.

BACKGROUND

In 1991 a team of faculty members from the Ohio State University (OSU) Nuclear Engineering Program and OSU Extension* recognized that over the next few years Ohio's citizens and their elected officials would be making decisions related to low-level radioactive waste disposal. The faculty members thought some Ohioans might wish to have a source of accurate, easy-to-understand, unbiased information as they began to discuss and make those decisions.
Accordingly, the faculty team designed a program to develop and distribute materials on low-level waste. Funding was sought, and the Midwest Interstate Low-Level Radioactive Waste Compact Commission agreed to support the program in phases. To date, two phases have been funded. In the first phase materials were developed, and in the second they are being distributed. This paper outlines the materials development phase and provides details of Phase II, Outreach, which is currently underway. (Phase I, Materials Development, was described in a paper presented at Waste Management '94.)

PHASE I - MATERIALS DEVELOPMENT

The primary product of the Materials Development phase was a set of 27 fact sheets. Each fact sheet addressed a single topic and was printed on one or two 8 1/2" x 11" pages. The topics were grouped into four categories:
1. Radiation science
2. Characteristics of low-level radioactive waste
3. Technology
4. Historical and legal aspects.

Radiation science fact sheets addressed sources and effects of ionizing radiation as well as key concepts such as radioactive decay, half-life, and shielding. Characteristics of low-level waste discussed in the fact sheets included sources, volumes, and typical contents. Storage, treatment, disposal, and transportation technologies were among those considered. Fact sheets in the "historical and legal" category presented information on the Low-Level Radioactive Waste Policy Acts, the Midwest Compact, and Ohio's activities to date with respect to low-level waste.
A very thorough review process was used for the fact sheets. Each one was reviewed by all members of the faculty team, three or four experts on the topic addressed, and a five-person panel consisting of a nuclear safety expert, a representative of a national environmental group, a science educator, a physician working in nuclear medicine, and a retired judge. Finally, all low-level waste generators in Ohio, all Ohio environmental groups the Project Team could identify, and anyone else who expressed an interest in the fact sheets' contents were invited to review the draft documents. About 20 groups and individuals accepted the invitation and received a copy of each fact sheet when the final draft was available. Each comment from every reviewer was cataloged and considered. Comments from such a diverse group of reviewers helped the authors provide more complete information, make the fact sheets easier to understand, and remove previously undetected bias.
Professional graphics were used to illustrate the fact sheets. The documents were printed in four colors on recycled paper using soy ink.
Using the material developed for the fact sheets, the Project Team prepared four table-top exhibits, one addressing each of the four major categories of fact sheet topics. Each exhibit is arranged on a 4' x 8' velcro board and can be folded to fit in a carrying case one person can easily handle.
In addition, overhead transparencies were prepared for presentations to groups requesting information on low-level waste. Materials were developed for presentations ranging from a 3-hour seminar for those interested in details to a 30-minute overview that might be appropriate for a civic organization's luncheon meeting.

PHASE II - OUTREACH

Ohio State University's Statewide Low-Level Radioactive Waste Education Program is unique in that it is a cooperative effort between faculty from the Nuclear Engineering Program with expertise in radioactive waste management and faculty from OSU Extension with expertise in environmental science and education. The outreach

phase of the program is heavily dependent on the County Extension Offices which are found in each of Ohio's 88 counties. County Extension Agents are well-trained and highly respected teachers who are recognized within their communities as sources of sound information on a wide variety of topics.

Three target audiences were identified for the outreach phase: local and state officials, the general public, and the media. Programs to reach each of those target audiences have been established at both the county and state level. County programs are conducted by the County Extension Agents while state level programs are carried out by Project Team members. Each of the 88 County Extension Offices has participated in a basic information program, while in some counties where interest in low-level waste issues is greater the Extension Office has undertaken additional activities. Activities at the state and county level are described in the following paragraphs.

County Level Activities

All County Extension Offices received 30 copies of each fact sheet. These fact sheets are kept in a file in each Extension Office and are distributed to people who request them. Extension Agents order additional fact sheets from the Project Team as needed. Several counties have requested additional fact sheets for use by community organizations.

Each County Extension Office distributed sets of fact sheets to all County Commissioners and local health departments. Bound sets were distributed to local libraries. In addition, a set of five summaries of the fact sheets written by a journalist working with the Project Team were distributed to the local newspapers and broadcast media in each county. Newspapers or stations with environment or science reporters received sets of the fact sheets to be used as reference documents.

A hotline was established by the Project Team in Columbus, Ohio, and County Extension Agents can use the line to get answers to questions people in their county may have about low-level waste.

Several County Extension Agents have developed additional programs to provide information to the public and officials in their counties. Using funds provided by the Low-Level Waste Educational Program, the Extension Agents in some counties have arranged half or whole day meetings at which Project Team members make detailed presentations on low-level waste to local officials. Other Extension Agents have set up the exhibits at mall shows and county fairs. One agent displayed the exhibits at an Earth Day Celebration.

State Level Activities

The Project Team in Columbus is responsible for the state level activities. Fact sheet distribution has been an important part of those activities. Sets of fact sheets have been sent to many groups including:

- State legislators
- Ohio's delegation to the U.S. Congress
- Ohio Hospital Association
- League of Women Voters
- University faculty
- Civic organizations
- High school science and social studies teachers
- Professional organizations.

Table-top exhibits have been displayed at a number of statewide meetings. In addition, a display on low-level waste was prepared for the Ohio Sate Fair which attracts about one million people annually. The display was attached to the sides of a wooden frame whose enclosed volume equalled the volume of low-level waste from Ohio in 1993. That exhibit was dismantled and reassembled at the Farm Science Review where the attendance is usually 30,000 to 50,000. One set of fact sheets was displayed, and order forms were provided so that people could request the sheets of interest to them. Two hundred requests were received.

A day-long media workshop was held in June 1994. Presentations were made by Project Team members and other university personnel on radiation science, low-level waste characteristics, waste management technologies, historical and legal aspects of low-level waste issues, methods of presenting technical concepts to a non-technical audience, and sources of information on low-level waste. Ohio's legislature had not yet begun to consider legislation on low-level waste, and only a dozen people attended. The workshop will be repeated March 9, 1995, and a larger turnout is

expected since low-level waste legislation was introduced in Ohio's Senate in January 1995. A media hotline has also been established and publicized. Questions on low-level waste may be faxed to the Project Team, and answers will be provided quickly.

Throughout Phase II, members of the Project Team have been traveling across Ohio to meetings at which they make two- or three-hour presentations to local decision makers. The purpose of the talks is to provide attendees with information they need to participate in discussions and decisions related to low-level waste management in Ohio.

Presentations to all audiences begin with a review of the basic science related to radiation. Key vocabulary is introduced, and the audience learns about topics such as half-lives, ionizing radiation, background radiation, and shielding. This segment of the presentation is followed by discussions of laws and regulations, characteristics of low-level waste, and related technologies such as treatment, storage, disposal, and transportation.

Questions from the audience are encouraged, both during and after the presentation. Covering the science at the beginning of the talk provides the audience with the terms and concepts needed to ask specific questions and establishes a framework within which the speaker can structure answers that all members of the audience will understand. All questions asked at the public meetings are noted, and a file is being kept of the types of questions people ask. It will be used to identify the need for new fact sheets or additional material in the presentations.

Reaction to presentations is quite positive. Project team members believe that some reasons for the positive reaction are:

The project team's goal is to teach fundamental concepts, not to persuade people to take a particular action or position

Great effort was expended to ensure that the teaching materials were accurate, easy to understand, and unbiased

Presentations focus on dealing with the waste that exists, not discussing whether it should exist

Fundamental scientific principles and key vocabulary are presented first - and in a manner that all members of the audience can understand

Questions are encouraged throughout the presentation, and all are answered respectfully and as thoroughly as possible

Audiences are small (10 - 50 people), affording an opportunity for personal interaction with all attendees.

As legislation on low-level waste is being considered in the Ohio Legislature, it is expected that more civic and professional groups will want short presentations on low-level waste at business or luncheon meetings. To date, Project Team members have followed the principles listed above in making these short presentations. However, if the demand for such presentations increases, it may not be possible for Project Team members to accept every invitation to speak.

To meet the anticipated demand for short, general presentations on low-level waste, a 20-minute slide/tape program has been prepared. It covers some of the main points made in the longer presentations. The slide/tape program will be available through the County Extension Offices. The first several times the program is used, a Project Team member will introduce it and answer questions after it has been shown. Questions asked most frequently and their answers will be provided to County Extension Agents so that they will be able to field some questions when a Project Team member is not present.

SUMMARY

A unique team of faculty from the Ohio State University Nuclear Engineering Program and OSU Extension has prepared a variety of educational materials on low-level radioactive waste. These include fact sheets, exhibits, and visual aids for presentations. A very thorough review process was established to help ensure that the materials were accurate, easy to understand, and unbiased.

The materials have been distributed throughout Ohio using the well-established and highly respected OSU Extension network and a variety of other vehicles. Seven thousand sets of fact sheets have been distributed; the exhibits have been used over 30 times, and an similar number of presentations have been made, primarily to local and state officials. Requests for materials and presentations continue to flow into the project office.

Response to the materials and presentations has been quite positive. The public

generally seems to find them to be accurate, easy to understand, and unbiased. The most direct evidence of the audience's reaction has come at public meetings where participants seem to quickly grasp the fundamental concepts and vocabulary and demonstrate their understanding by posing provocative questions or offering well-formed opinions and suggestions.

56-6
COMMUNITY ATTITUDES TOWARD EPA'S HANDLING OF THE MONROE COUNTY, INDIANA PCB SUPERFUND CLEANUP
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ABSTRACT
For the past several years, the United States Environmental Protection Agency Region 5 (EPA) has been conducting cleanup efforts in Monroe County, Indiana, a community with widespread PCB contamination. The EPA, Potentially Responsible Party (PRP), City, County and State entered into a consent decree with the Federal District Court in May, 1985 requiring the PRP to destroy the PCB-contaminated materials in a municipal solid waste (MSW)-fueled incinerator.
Some citizens have vigorously opposed the incinerator, leading EPA to create some innovative community relations activities (Smith, et al, 1992). Most recently, the EPA and contractor staff teamed with the Indiana University Center for Survey Research to conduct a telephone survey of a random sample of Monroe County residents. The study assessed residents to measure their attitudes regarding communication between the EPA and the public and about selected local environmental issues.
The survey team completed 527 telephone interviews asking Monroe County residents a variety of questions regarding the PCB cleanup. In particular, questions targeted public knowledge of the technology suggested for the clean-up. Of those interviewed, 76.8% were aware of the proposed incinerator. Of these, almost 90% had concerns about incineration fueled by solid waste or some other fuel source.
EPA is perceived as a believable source of information; 68.7% affirmed EPA as a believable source. Before establishing a local office and maintaining a presence in the community, mistrust of the EPA was common. Survey results thus indicate that despite general opposition to incineration, public attitude toward the EPA is not similarly negative.
As a result of the survey, and some state legislation, EPA initiated new directions regarding the Bloomington area cleanup efforts to incorporate some of the citizens' concerns. EPA persuaded the other consent decree parties to explore alternatives to incineration and in February 1994, the parties officially launched a joint effort to assess alternative remedies.
BACKGROUND
Westinghouse Electric Corporation designed, manufactured and sold electrical capacitors and transformers containing polychlorinated biphenyls (PCBs) in Monroe County, Indiana from 1959 until the late 1970's. Their off specification PCB-laden capacitors and transformers were dumped in local landfills and quarries throughout the sixties and the early seventies (Lillich, 1981).
In the early eighties, the Indiana State Department of Health (ISDH), at the county health department's request, conducted a preliminary assessment/site inspection. Their sampling results indicated PCB concentrations to 330,000 parts per million (Osborn, 1991). The six Monroe County sites were then ranked using the Hazard Ranking System to determine the sites' potential hazards to the environment and public health. Four scored above the 28.5 required to be added to the EPA's National Priorities List sites (EPA, Sept. 1990). ISDH then requested that the EPA assist them in determining the extent of PCB contamination in Monroe County. After some of the EPA's preliminary studies were completed, both the city of Bloomington and the EPA filed suit against Westinghouse (Schmall, 1983).
The next step in the Superfund Enforcement Process requires that a remedial investigation be conducted to assess the extent and nature of the contamination (EPA, 1988). A feasibility study is then prepared to examine and evaluate various remedial alternatives. Following a public comment period on the EPA's preferred

alternative and the draft FS report, the EPA chooses a remedy and outlines it in the Record of Decision. No formal remedial investigation-feasibility study was ever done for the Monroe County PCB-contaminated sites by the EPA. The National Contingency Plan that the EPA was working under at the time, did not require a remedial investigation/feasibility study. Also, because the EPA was preparing a lawsuit against Westinghouse, many of the decisions made regarding alternatives assessment were not documented (EPA, July 1990). Consequently, the EPA did not issue a record of decision for the sites.

Early in the Superfund process, the EPA's method of developing case information for responsible party lead sites was done by conducting an Enforcement Decision Document so a record of decision was not performed for the sites (EPA, 1990). In addition, because the EPA was in litigation with Westinghouse, they kept many documents confidential. The citizens were not allowed to see the Administrative Record until the EPA had settled the case and then it was only made available through a Freedom of Information Act (FOIA) request.

These events led to public mistrust of the EPA. Most of the early public meetings regarding these sites were contentious and the EPA personnel were called names like "baby killer" and citizens, armed with pies, aimed them at the EPA personnel (EPA Public Meeting, 1988).

In May, 1985 a consent decree was signed among the EPA, the state of Indiana, Monroe County, the City of Bloomington, and Westinghouse. The consent decree identified six PCB-contaminated sites and outlined the terms between the above-listed parties in removing the contamination. One of the requirements of the consent decree is that Westinghouse design and construct a municipal solid waste fueled incinerator to destroy the estimated 650,000 cubic yards of PCB-contaminated soil and materials. Westinghouse is also responsible for disposal of the resulting incinerator ash. Once the consent decree was signed, initial removal actions were implemented in 1986 to stabilize the sites until the final remedy was permitted and executed.

After the initial removal actions were completed, Westinghouse submitted the incinerator and ash landfill permit applications (August, 1991). It seemed as though the cleanup would proceed, but the local and state officials began putting up roadblocks. The City of Bloomington and Monroe County passed ordinances and the State of Indiana passed a bill mandating that a study of alternatives to incineration be conducted before permit applications for a hazardous waste incinerator could be considered. All three of these parties are legally bound to the consent decree yet they have put into law statutes impeding the cleanup process.

During the early removal actions, the citizens accused the EPA of not keeping them informed of the ongoing cleanup actions. The main objection was that the EPA personnel were located in Chicago, therefore it was difficult for citizens in rural Indiana to communicate with them. In a bold administrative decision, the EPA began an aggressive Community Relations program in Monroe County, Indiana (See HMC '92 proceedings).

In January 1989, the EPA opened a Public Information Center/telephone hotline managed by an EPA contractor and staffed with a local resident. A Citizens Information Committee was established which required the EPA remedial project manager and the work assignment manager to meet monthly with local citizens in Monroe County. The meetings were videotaped and aired on the local public access channel. EPA contractor staff wrote fact sheets and distributed them via the local Sunday paper. Citizens were taken on site tours as the cleanup progressed and residents were visited before a cleanup action was taken in the neighborhood. Finally, the EPA set up and maintained four information repositories in the county so that a FOIA request was no longer needed to obtain site cleanup information. Residents could go to the local office or one of the four information repositories.

As a way to evaluate the public's perception of the cleanup remedies and the EPA's performance during the cleanup, a telephone survey of a random sample of Monroe County residents was conducted. Survey results indicate that the EPA's commitment to Monroe County has helped the agency to develop trust with the citizens and has helped them to get information to the public in an efficient manner.

METHODS

The EPA and contractor staff teamed with Indiana University Center for Survey Research to conduct a telephone survey of Monroe County, Indiana residents to measure their attitudes regarding communication between the EPA and the public and about selected local environmental issues. Some specific areas addressed in the

survey included awareness of the proposed incinerator to burn PCB contaminated materials in Monroe County and concerns about the incinerator.
Two independent samples of telephone numbers were randomly generated using a random digit dialing procedure. One sample contained 1377 possible residential telephone numbers from the exchanges that serve Monroe County. The second sample contained 190 telephone numbers in the 857 exchange (Indiana University residence halls and family housing). At each residential telephone number, a respondent was randomly selected when there was more than one adult in the household. There was no random selection of respondents when students in residence halls were reached. Because it is so difficult to randomly select respondents in fraternities/sororities, persons living in that type of household were not interviewed.
Data were collected by telephone using the University of California Assisted Survey Methods software. The questionnaire was pretested three evenings and the questions were modified based on the comments collected from the pretest sample group. The survey began on April 19th, 1993 and ended on May 9th, 1993. Interviews were collected seven days per week at various times throughout the day and night.
There were 527 interviews completed of randomly selected persons within Monroe County. There are approximately 48,000 residences with phones in Monroe County. About 12,000 of the telephones are in Indiana University residence halls and family housing. A fully proportional sample would overrepresent students living in residence halls, therefore the sample was stratified to produce 89 completed interviews with students living in residence halls and family housing and 438 with the remaining residents of Monroe County. The average length of the completed interviews was 15 minutes.
RESULTS
Of the 527 respondents interviewed, 84.2% claimed to be aware of PCB contamination in Monroe County. Of these, 87.8% voiced specific concerns about the presence of PCBs in Monroe County. The two primary concerns were possible health related problems and potential contamination of groundwater.
While 76.8% of respondents were aware that an incinerator to burn PCB-contaminated materials is supposed to be built in Monroe County, only about half of these were aware that the incinerator is required to be built by a federal consent decree.
In a number of public meetings, opponents of the incinerator have raised objections to the use of municipal solid waste as a fuel source for incineration. They suggest that solid waste used as a fuel source is an untested technology (Herald Times, Jan. 13, 1993). It has also been suggested that other fuel sources such as natural gas would be more feasible alternatives.
Survey questions were designed to assess public reaction to solid waste as a fuel source as well as to assess possible responses to alternative fuel sources. Also of interest was whether public opposition was limited to the incinerator's fuel source or whether there was a more basic objection to incineration as a technology.
Of those respondents who were aware of the proposed incinerator, 79.3% were either somewhat or strongly concerned about using an incinerator fueled by solid waste to destroy PCB-contaminated materials. It does not appear, however, that fuel source is the fundamental area of contention. Over 80% expressed specific concerns about the use of solid waste to fuel an incinerator. Similarly, over 80% expressed specific concerns about alternative fuel sources such as natural gas. Regardless of fuel source, respondents expressed concern about the use of an incinerator to destroy PCB-contaminated materials in Monroe County:
"It just concerns me that they would burn it and that dioxin (one of the byproducts from what I understand) [would be released]...the way the winds blow, it would blow right into Bloomington."
"It could disperse over a wide area...solving one problem [of] pollution with another form that would contribute to air pollution."
"I'm concerned about incineration of any [sort] because what is incinerated is going to go into our air and the air we must breathe and live in...[I am concerned about] any type of incineration fueled in any manner...I can't see whether it would make a difference if it was solid waste or gas. The residue will be in the air."
It is evident, then, that Monroe County residents generally oppose the use of incineration to destroy PCB-contaminated materials. It is likely that this public sentiment was influential in the decision of local and state government officials to halt the incinerator permitting process until alternatives to incineration could be assessed.

Despite public aversion with the consent decree remedy, dissatisfaction does not extend more generally to the EPA. Of Monroe County residents interviewed, 68.7% agree that the EPA is a believable source of information regarding PCBs in Monroe County. Since the local EPA Public Information Center in Bloomington was established, over half of respondents interviewed feel that communication between the EPA and the community regarding the PCB situation in Monroe County has improved. Of those who have ever contacted the local PIC about any environmental issue, 75.0% felt that the staff was helpful.
Shortly after EPA conducted this survey, the Indiana state legislature enacted a law which prevented the state from issuing a permit for the construction or operation of a hazardous waste facility used for incineration of PCBs unless the technology used at an equivalent hazardous waste facility has demonstrated a destruction efficiency of 99.9999%. The public's sentiment as demonstrated by their answers to the survey along with their pressure on the legislature to enact this law, effectively halted the process. EPA responded by persuading the other government parties and Westinghouse to examine alternatives to incineration. In February 1994, the consent decree parties announced that they would work together in a good faith effort for an alternative to incineration.
CONCLUSIONS
Despite an on-going public information campaign by the EPA to educate Monroe County residents about incineration as a technology, public perception of incineration remains overwhelmingly negative. Regardless of fuel source, residents oppose the use of incineration to destroy PCB-contaminated materials. Attitudes toward the EPA, however, have been positively influenced by local interaction between the EPA and the public.

Session 57 -- Innovative Approaches for Environmental Restoration
Co-chairs: John Lehr, USDOE;
Veronica P. Cornell, US Energy Corp.
57-1
USING BENCHMARKING TO ENHANCE LOS ALAMOS NATIONAL LABORATORY'S ENVIRONMENTAL RESTORATION PROJECT
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ABSTRACT
Los Alamos National Laboratory (LANL) is actively exploring more efficient and cost-effective methods of implementing the Environmental Restoration (ER) Project. Under this premise, the LANL ER benchmarking study was undertaken to compare characteristics of LANL's ER Project to similar ER and remediation projects at other Department of Energy (DOE), Department of Defense (DoD), and private sector facilities. A team of process experts from the LANL Environmental Management Program Office, worked closely with DOE and benchmarking consultants to identify ER-related elements, including performance measures and metrics, to be studied and compared. From this effort, a benchmarking protocol was subsequently developed with significant input and support from DOE. Data was gathered on each program element which enabled LANL to gauge the performance of its ER Project against the performance of similar environmental remediation efforts at other institutions, and to identify "best-in-class" characteristics of other environmental remediation programs. LANL is currently using the benchmarking results to assist in enhancing the performance of its ER Project.
BACKGROUND - INFORMATION GATHERING
The objective of the study was to identify opportunities to enhance performance of the LANL ER Project. To accomplish this objective, LANL's Environmental Management (EM) Program Office contracted with Environmental Resources Management (ERM) to jointly conduct a benchmarking study of the EM ER Project. The benchmarking protocol was developed with input from the Department of Energy (DOE). The protocol covered the following programmatic-level elements of the ER Project:

Status of Overall ER Effort	Procurement
Deliverables/Associated Penalties	Contractor Support
Analytical Chemistry Services	Staffing Costs
Waste Volumes/Management	ER Project Status
Technical/Regulatory Strategy	Site Closure
National Environmental Policy Act	Future Land Use
Corrective Action Management Units	Management Structure
Public Involvement Strategy	Temporary Units
Programmatic Technical Support	Support Functions
Site Prioritization	Pilot Studies

Participants in the ER benchmarking study included:
DOE Facilities: Idaho National Engineering Laboratory (INEL); Oak Ridge National Laboratory (ORNL) in Tennessee; and Nevada Test Site (NTS).
DoD Facilities: Tooele Army Depot in Utah; Pease Air Force Base (AFB) in New Hampshire; and Marine Corps Camp Lejeune in North Carolina. McClellan AFB in California and Tinker AFB in Oklahoma have also been asked to participate, but a response has not been received.
Industrial Facilities: FMC; Allied Signal; and General Electric (GE).
To date, completed protocols have been received from the three DoD facilities, two DOE facilities, and one industrial facility. One industrial facility submitted only a partially completed protocol, and the LANL protocol is also complete.

The Benchmarking Process

To facilitate identification of best-in-class performance among different ER programs, the four step process described below was followed.
Step 1: Develop Metrics - First, metrics were designed to facilitate comparison of performance among different ER projects. For example, in the present study, metrics were developed that allow comparison of the performance of ER programs with different start dates, lifecycles, and budgetary plans. These normalizing metrics facilitated "apples-to-apples" comparison of performance.
Step 2: Apply Metrics to Performance Data and Identify Performance Gaps - Metrics were applied to identify project element performance gaps. Metrics were developed for comparison of progress on investigations, designs, and cleanups. The metrics were also designed to normalize for differences in project budgets, timelines, and rates of budget expenditure.
Step 3: Interpret Gaps - If a performance gap was identified, differences in program characteristics (e.g., ER project strategy, regulatory framework, etc.) were examined to explain the observed performance gaps, and to identify and prioritize performance improvement opportunities.
Step 4: Review Additional Information - Best practices employed by participating DOE, DoD, and industrial facilities were reviewed, and follow-up interviews were conducted with participating facilities to explain the observed performance gaps, and to identify and prioritize performance improvement opportunities. Additionally, literature searches and discussions with environmental professionals were used as follow-up to further validate the explanations of observed performance gaps.

INTERPRETING THE FINDINGS - IDENTIFYING CAUSES OF PERFORMANCE GAPS

Regulatory Framework

Differences in the ER Project strategy employed by each facility and the resulting regulatory framework negotiated by each facility were reviewed. For example, ER programs at Camp Lejeune, Tooele Army Depot, and INEL are regulated by Federal Facility Agreements (FFAs), while Pease AFB operates under a Base Realignment and Closure Plan (BRAC). As part of their strategy, each of these facilities aggressively negotiated a flexible schedule with concurrent regulatory review of deliverables, involving the regulators as partners throughout the ER process.
In contrast to these flexible regulatory frameworks, the Hazardous and Solid Waste Amendments (HSWA) module driving the Los Alamos ER Project restricts LANL to following the traditional Resource Conservation and Recovery Act (RCRA) Facility Investigation/Corrective Measures Study (RFI/CMS) approach to characterization and remediation. Noncompliance results in a Notice of Deficiency, which, if not corrected, results in a Notice of Violation from EPA Region VI. To date, the Los Alamos ER Project has not successfully engaged regulators to obtain a more flexible regulatory framework.

PRELIMINARY RESULTS

The following discussion is limited to programmatic-level elements of the ER Project that are significant and for which sufficient information is available to adequately discuss and interpret the findings.

Progress on RCRA/CERCLA Activities

Findings: Fig. 1 compares facility performance on completed RCRA/Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) activities. Table I provides the background information for Fig. 1. The metric is designed to normalize facility performance for differences in planned schedules. The DoD facilities and DOE facility A show more progress on expedited site cleanup and no further actions (NFA) than LANL. DoD facilities and DOE facility A have already completed all planned expedited cleanups. The remaining DoD facility has completed approximately 75% of its planned expedited cleanups. Los Alamos has completed approximately 10% of its planned expedited cleanups. Progress on receiving approval on NFA recommendations is greater for all the DoD facilities surveyed than LANL. Similarly, DOE facility A has received approval for close to 45% of their planned NFA sites. Los Alamos has received approval for approximately 28% of its planned NFA sites. However, LANL has a greater total number of completed expedited sites and NFAs.

Interpreting the Findings: DoD's strategic approach places a greater emphasis on accelerated cleanups and obtaining approval for NFAs. Opportunities to accelerate cleanups are evaluated on an ongoing basis as a means of reducing risk and expediting the overall cleanup process. This tactic varies somewhat from the traditional emphasis on investigation, design, and cleanup activities.

DOE facility A's performance can be linked in part to the ER program strategy negotiated with regulators. As part of its FFA, DOE facility A negotiated (and successfully implemented) a formal regulatory process that emphasizes facilitating site investigations. The process also accelerates cleanups and NFA recommendations. In addition, at DOE facility A, sites are aggregated into common categories for NFA, Phase II investigation, or accelerated cleanup. This process of aggregation accelerates cleanups by incorporating concepts that are functionally equivalent to presumptive remedies.

LANL has developed strategies to accelerate site investigations, cleanup, and NFA determinations. These strategies were submitted to EPA Region VI in the LANL ER Project Installation Work Plan (IWP). However, these strategies have not been accepted by EPA, primarily because Region VI mandates compliance with traditional HSWA module RFI/CMS process requirements. Similar strategies were proposed in the recent ER Project Action Plan, but have yet to be successfully negotiated or implemented.

It is important to note that the Los Alamos ER Project consists of more individual sites than the other ER programs in the study. Since the majority of these sites are potential candidates for NFA or accelerated cleanup, a negotiated strategy within the regulatory framework on accelerated cleanup and NFA recommendations will enhance performance.

Analytical Chemistry Services

Findings: Analytical chemistry services are an example of an ER Project element where LANL compared unfavorably and immediate performance gaps (costs and turnaround times) were identified.

Interpreting the Findings: Los Alamos ER Project analytical chemistry services have resulted in excessive costs and delays in receiving analytical chemistry results. Excessive analytical chemistry data turnaround times and internal data validation turnaround times cause delays in decision making contributing indirectly to some of the observed performance gaps (e.g., delaying NFA determinations).

Deliverables and Associated Penalties

Findings: In contrast to Los Alamos, none of the other participating facilities have received Notices of Deficiency (NODs) for their deliverables (e.g., RFI Work Plans, RFI Reports, etc.), nor have they received Notices of Violation (NOVs). To date, Los Alamos has received NODs for most major deliverables submitted to EPA Region VI.

Interpreting the Findings: The performance gap between LANL and the study participants appears to be the result of the regulatory framework negotiated by the other facilities, which includes flexible program schedules with concurrent regulatory review of deliverables. This involves the regulators early and throughout each facility's ER process. In contrast, Los Alamos is regulated by EPA Region VI under the relatively inflexible HSWA module.

ER Project Waste Management

Findings: The ER programs of facilities participating in the study vary significantly in terms of their current level of integration with waste management (WM) activities. DOE facility A, like LANL, does not have low-level mixed waste disposal capacity, while DOE facility B has onsite low-level mixed waste disposal capacity. DOE facility A and B representatives indicated that while the ER and WM activities are separate, waste management strategies are thoroughly integrated. In contrast to the DOE facilities, the industrial and DoD facilities do not separate waste management from ER functions.
Interpreting the Findings: Currently, LANL WM waste treatment, storage, and disposal (TSD) capacity is constraining the LANL ER Project. Additionally, the LANL WM Program focuses primarily on TSD for wastes generated by Laboratory operations, not ER Project-generated waste. Improved integration of waste management planning and funding for TSD of ER Project-generated waste at LANL is essential to enhancing ER Project performance.
NEPA
Findings: The policy of the three participating DoD facilities is that the CERCLA process is functionally equivalent to the National Environmental Policy Act (NEPA) and, therefore, the requirements of NEPA are satisfied by completing CERCLA processes. DOE facility A indicated that NEPA requirements are integrated in a manner similar to the DoD facilities because the CERCLA process is deemed to be functionally equivalent to NEPA by DOE.
Interpreting the Findings: NEPA requirements are explicitly excluded from the LANL HSWA module relative to EPA decision making; however, NEPA is still a DOE requirement for RCRA-regulated ER projects. As a result, approximately 2 to 5 percent of the Los Alamos ER Project budget is directed at meeting NEPA requirements. The updated Los Alamos ER Project FY95 baseline projects significant expenditures for NEPA compliance.
Table I
IMPROVEMENT OPPORTUNITIES - WHERE DO WE GO FROM HERE?
Strategic Planning
A primary root cause of performance gaps identified during the study is the lack of a comprehensive ER project strategy that is properly formulated, negotiated and implemented. Figure 2 is a cause-effect diagram outlining some of the key elements that must be addressed. The study indicated that regulatory strategy (and tactics for achieving strategic objectives) is a key element of a comprehensive ER project strategy. The study pointed to the need for DOE and LANL to reach consensus on a regulatory strategy for aggressively and continuously negotiating with regulators on specific issues. Figure 3 summarizes several of the regulatory issues that should be considered in formulating a regulatory strategy. The strategy must be developed with stakeholder input.
Additionally, institutional requirements and waste TSD options (see Fig. 4), should also be considered during formulation of a comprehensive strategy. Policy decisions, such as acceptable cleanup goals, acceptable levels of risk, cost and liability for offsite waste TSD must be negotiated as part of the ER Project's regulatory strategy.
Negotiations with Regulators
Several negotiating positions and tactics to address performance gaps (i.e., improvement opportunities) in LANL's current regulatory strategy are presented in Table II. In addition to the areas presented in the table, negotiating positions are being developed in areas such as analytical suites and detection limits for chemical analyses, Corrective Action Management Units and Temporary Units (CAMU/TU), and land disposal restrictions (LDRs).
Analytical Chemistry Services
The Los Alamos ER Project recently formed a Sample Management Process Problem-Solving Committee to address analytical chemistry service issues. Additionally, improved turnaround times and reduced analytical chemistry costs are being negotiated aggressively with contract laboratories, and included as contract clauses, with penalties for late submittal of data.
ER Project Waste Management
The ER Project Manager has identified a Waste Management Coordinator within the ER Project to deal with waste management planning and funding for TSD of ER Project-generated wastes. The Coordinator is tasked with integrating waste management and pollution prevention strategies within EM.

NEPA

Addressing duplicative NEPA requirements for major remedial/corrective action decisions in the future will have a significant impact on the Los Alamos ER Project budget (5 to 10 percent) and schedule. If DOE was to remove this requirement, between $60 million and $130 million could be saved over the life of the ER Project. A determination must be made on the NEPA functional equivalency issue and use of HSWA decision documents and activities (e.g., proposed plans, corrective measures studies, etc.) to meet public participation requirements.

SUMMARY

As a result of the benchmarking study, performance gaps were identified in several LANL ER Project areas, the most significant being: analytical chemistry services, strategic planning, regulatory strategy, waste management, and NEPA. Steps are underway at LANL to address each area to enhance the cleanup schedule and cost performance of the ER Project.

57-2

IMPLEMENTING THE HANFORD ER BASELINE DURING A TIME OF FUNDAMENTAL CHANGE

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ABSTRACT

Establishing a sound project control system during dynamic change is critical to the success of the Richland Environmental Restoration Project (RERP). Once a baseline is established, it is equally important to implement it effectively. The RERP baseline is complex, involving hundreds of activities at the 560-square-mile site.

The major challenge to implementing the Hanford ER baseline is the dynamic environment that continually impacts the program. This environment includes evolving regulatory priorities, extensive stakeholder involvement, budget restraints, workforce restructuring, and contract reform. This paper describes how the RERP baseline was developed and how DOE and Bechtel are using a disciplined process to implement the baseline in this changing environment.

INTRODUCTION

Environmental restoration at the Hanford Site is a major technical challenge involving large quantities of contaminated soil, numerous burial areas with radioactive and mixed wastes, extensive areas of groundwater contamination, and more than 100 buildings to be decontaminated and decommissioned. An equally challenging task is implementing a baseline to manage the scope, schedule, and cost of the program.

The "cleanup" work at Hanford can be divided into two major mission categories -- waste management and environmental restoration. Waste management includes the treatment, storage, and disposal of wastes, such as the high-level liquid wastes stored in 177 underground tanks, and management of spent fuel. Westinghouse Hanford Company, the Hanford M&O contractor, is responsible for waste management, along with the associated maintenance and operations necessary to manage the 560-square-mile Hanford reservation.

Environmental restoration includes restoration of areas contaminated during past production and disposal activities. It includes remediation of buried waste sites, contaminated soil columns, contaminated groundwater plumes, as well as decontamination and decommissioning of surplus defense production facilities and associated infrastructure support facilities.

Prior to 1994, Westinghouse Hanford Company was responsible for both waste management and environmental restoration at Hanford. In July 1994, Bechtel Hanford, Inc. (BHI), became the Environmental Restoration Contractor, assuming responsibility for a portion of the work that was being performed by Westinghouse.

The Environmental Restoration Contract is a demonstration of a new type of contract at DOE sites. The concept of having a separate contractor to manage environmental restoration originated in 1990 with the Department of Energy's Alternate Contracting Task Force, which looked at ways to lower costs and improve efficiency in DOE's nationwide environmental restoration program. One concept the task force developed was a separate contractor with the single mission of managing the remediation of a contaminated site to the final end-state condition.

Prior to BHI's start of work as the ERC, a detailed baseline for the RERP had been

developed using a strategic planning effort termed the Optimized Baseline Project. This baseline was viewed as a pre-conceptual level baseline; it was adopted by BHI for purposes of configuration management and contract performance measurement. As the Hanford ER project moves forward, the challenge will be to successfully implement this baseline during a time of rapid and dramatic change.

CURRENT CHALLENGES

As the Hanford ER project moves from the "assessment" stage to "cleanup," there is no significant experience base to guide the way. Work of this type has not been done on a large scale before, so there is little experience to indicate what to do and how to do it (scope management), how much it will cost (cost management), and how long it will take (schedule management).

In addition to the lack of an experience base, conditions surrounding the project are changing. These changes stem from a combination of technical, financial, and political issues. Some of the major factors being experienced include:

Restructuring of the Tri-Party Agreement (TPA). The TPA is an enforceable agreement signed in 1989 between DOE, the Environmental Protection Agency, and the Washington Department of Ecology, specifying milestones for Hanford cleanup. Major changes in the ER portion of the TPA were negotiated between the three parties in the summer of 1994.

Changing funding scenarios. The ER scope of work for FY 1995 was budgeted at about $233 million. DOE productivity challenges reduced funding for this work to about $201 million. Further reductions due to internal DOE funding decisions and actions by the Office of Management and Budget have reduced funding to about $150 million. All of these changes have taken place since October 1, 1994.

Increasing budget restrictions. DOE's nationwide environmental management program enjoyed rapidly increased funding in the early 1990s, reaching an annual budget of about $6 billion. By 1994, projections were that this growth would slow, roughly matching the rate of inflation for the rest of the decade. Following the election of November 1994 and the resulting emphasis on reducing the Federal budget major reductions in future funding are anticipated.

Changing priorities. The focus of Hanford ER activity has shifted over the past two years. A program that once was heavily oriented toward assessment work on many parts of the site has become a program focused on doing actual cleanup at the part of Hanford close to the Columbia River. D&D work, which had been projected as a future activity, has been accelerated to reduce the cost of "mortgages," the annual expenditures needed just for surveillance and maintenance of aging buildings.

Recognizing limitations of technology. A significant part of the Hanford ER effort was directed at groundwater remediation. Experience was showing that there was minimum payback from many of these expenditures. This has resulted in a reevaluating the actual risk posed by groundwater contamination and deciding whether it would be better to work on improving technologies before tackling the groundwater issue.

DEALING WITH CHANGE

Each of the environmental conditions listed above presents a formidable challenge to successful implementation of a project management system. Taken together, they pose enormous challenges for managing the Richland ER project.

To deal with this changing environment requires a disciplined project control process. There are many factors that can trigger a change in the baseline. New data or regulatory actions can cause a change in scope. Budget changes due to actions by DOE, OMB, or Congress can cause change in funding and schedules. Actual field experience can cause a change in scope, cost and schedule simultaneously.

ESTABLISHING A PROJECT CONTROLS SYSTEM

Validated Baseline

When BHI assumed responsibility for the Hanford ER project in July 1994, the existing baseline had been developed under an M&O contracting system that evolved from a production paradigm. BHI is changing the project controls system toward a project management approach, developed by Bechtel through experience on other DOE programs, on Department of Defense environmental projects, and on commercial projects.

This "projectized" project controls system was developed in three months through a cooperative effort between DOE and BHI. The BHI staff included Bechtel employees who transferred to Hanford and Westinghouse project controls staff who moved to BHI during the contract transition.

To avoid the cost of software development, the system uses off-the-shelf software

(Primavera Parade). To ensure that all recipients of data (DOE-Richland, DOE-HQ, regulators, public) get accurate information, the former system and the new system will operate in parallel until March 1995.

Change Control Process

As these changes develop, they are analyzed and are incorporated into the baseline through a change control process. The process begins with the project teams. The BHI organization is structured around matrixed project teams. Each project team is composed of the engineering, construction, environmental compliance, health and safety, quality assurance, project controls, and support personnel needed for a specific task. The DOE-Richland Environmental Restoration staff, which oversees BHI, is also organized along project lines. This "mirror image" between DOE and BHI results in an efficient working relationship.

Trending and Variance Analysis

When a project team recognizes a potential change that is likely to affect the baseline, they identify it as a trend and begin to collect data to support the trend. Each project team has a DOE representative, who can attest to the correctness of the potential trend. The trend is then taken to a Change Order Review Board, made up of BHI and DOE staff, which reviews the change and recommends action. The change is then formally submitted to the appropriate DOE official for approval. DOE and BHI have developed a close working relationship which allows for timely identification and action.

CONCLUSION

Hanford environmental restoration is a complex program that has undergone major changes in the past two years. This climate of change is likely to continue and accelerate. To continue making cost-effective progress during this time of change requires a proven baseline, a project-oriented controls system, and a disciplined change control process.

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VALIDATING THE CONCEPTUAL SITE MODEL:
A FOCUSED ALTERNATIVE FOR REMEDIAL INVESTIGATIONS INVOLVING RADIOACTIVE, MIXED AND HAZARDOUS WASTE SITES

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ABSTRACT

Remedial action decisions for Superfund sites are compounded by uncertainties regarding future land use and, particularly for sites involving radioactive and mixed wastes, a shortage of cost-effective clean-up options. Remedial investigations (RIs) for such sites typically are resource-intensive field investigation efforts that attempt to fully characterize all aspects of the site. This type of RI often fails to meet its intended purpose--to provide information to assess risks to human health and the environment as needed to support the development, evaluation and selection of appropriate remedial response actions. An alternative approach, based on the conceptual site model, allows the RI to focus on the most important components of the site: contaminant sources, release mechanisms, migration pathways, and receptors. The conceptual model is "validated" by quantifying the relationships between these components and integrating the results of the baseline risk assessment, resulting in a quantitative description of the source term, release rates, migration rates, flux and exposure (risk).

INTRODUCTION

For sites listed by the U.S. Environmental Protection Agency (EPA) on the National Priorities List (NPL) and thus consigned to the Superfund cleanup program, remedial investigations (RIs) typically constitute the most costly phase of the program prior to the actual remediation or cleanup of the site. The basis for the RI is provided in regulations under the National Oil and Hazardous Substances Contingency Plan (NCP), which are based on the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA). EPA published guidance on the conduct of RIs that details the scope and objectives of the RIs as well as the suggested format for preparing RI reports (1). As stated in the regulations and guidance manual, the purpose of the RI is to collect sufficient data to support the assessment of risks (to both human health and the environment) as well as the development, evaluation and selection of

appropriate response (cleanup) actions.
The traditional RI, as established by EPA guidance and precedent at hundreds of NPL sites nationwide, involves 1) a "scoping" (planning) phase, the primary output of which is a Work Plan that establishes the objectives and rationale for the RI; 2) a field investigation, where data are collected through such activities as sampling and analysis of a site's soil and groundwater; 3) data evaluation, in which the data collected for the field are reduced and evaluated; and 4) preparation of the RI Report, which documents the results of the field investigation and data evaluation efforts. Most RI Reports are developed with the following elements:

an introduction with background discussion
a description of the investigation focusing on field sampling, analysis and measurement activities
an analysis of the physical characteristics of the site
a description of the nature and extent of contamination (by media)
a discussion of contaminant fate and transport
the baseline risk assessment (human health and environmental)
summary and conclusions, including remedial action objectives

A recurring criticism with the traditional approach to conducting RIs is the disconnect between the characterization and risk assessment: project teams responsible for deciphering the physical characteristics of the site or delineating the nature and extent of contamination typically have only limited interaction with the risk assessment teams responsible for the human health or ecological risk assessments. Consequently, the characterization and risk data are not well integrated, leading to RI Reports where all contamination is defined and discussed, even where trivial, and all risks are calculated, even where such exposures are highly unlikely.
At most Department of Energy (DOE) facilities on the NPL, implementation of the Superfund program is complicated by the presence of radioactive wastes, uncertainties regarding future land use and a shortage of cost-effective cleanup options. Most DOE facilities on the Superfund list contain a large number of sites that need to be evaluated to determine the need for and scope of appropriate response actions. Unfortunately, it is not always clear just how clean the sites have to be or what methods will be used to do the actual cleanup. Developing the scope of work and objectives for an RI at such sites inevitably requires the project team to anticipate all the data needs for the full range of possible land use or remediation scenarios, resulting in a poorly focused RI that strives to characterize and document, in sometimes excruciating detail, all aspects of the site--physical, biological, chemical and radiological.
A poorly focused RI produces vast amounts of data that can not be effectively evaluated and integrated into the decision-making process for the site. Almost anyone who has been involved with RI activities at DOE sites has seen the products of this labor--typically an immense, multi-volume treatise laden with hundreds of data tables, charts and figures that fill entire bookshelves. The trend toward gigantism in the RI trend challenges even the most assiduous technical reviewers and is especially alarming to the stakeholders (e.g., surrounding communities), who in most cases are clearly intimidated by the sheer volume of information.
Generating data is an expensive activity: the more data that are generated, the more the project will cost, and not just in terms of field sampling and analysis, but also in the validation or verification of the data, managing the data in databases, evaluating the data for the RI report, and developing interpretations and summaries of the data to be included within the report. The disproportionate resource requirements required for overly ambitious characterization efforts are a significant drawback to conducting traditional RIs, particularly at large or otherwise complex sites. Depending on the conditions at the site, an RI may take several years to complete, from planning to issuance of the final document, and run up costs in the tens of millions of dollars. The upward pressure on costs continues as subcontracting expenses, health and safety burdens and layers of oversight and management increase.
Many RIs fail to meet their intended purposes, which is to efficiently provide information needed to assess risks and support the development, evaluation and selection of remedial alternatives. Even with the trend toward smaller, more manageable projects and broader use of removal-type response actions, RIs (or analogous site characterization efforts) will continue to be a key step in the

cleanup process and a principal foundation for remedial action decisions. An alternative approach that better integrates characterization and risk data, resulting in a more focused, user-friendly RI document, is clearly warranted. Development of this method is timely, given the efforts in DOE (and elsewhere in the Federal government) to streamline the environmental restoration process, accomplish more with the same or smaller budgets and more effectively communicate site conditions and risks to the public.

DESCRIPTION OF CONCEPTUAL SITE MODEL

Conceptual site models are widely used to illustrate the relationships between contaminant sources, migration pathways and potential receptors at a site. These qualitative models are often used in the scoping process for site investigations to guide sampling activities or sometimes to merely demonstrate a cursory understanding of the site. In its latest guidance on the Data Quality Objectives (DQO) process, EPA identifies developing a conceptual site model as a key step in the planning process for data collection activities (2). Risk assessments may also use conceptual site models but generally adhere to standard EPA guidance that complicates efforts to integrate risk-related findings on exposure pathways and receptors into the characterization-based understanding of the site.

The limitations associated with the conceptual model's uses usually stem from the non-specific manner in which they are often constructed. Typically, qualitative models proposed for sites do not maximize the use of existing, site-specific knowledge and as such may depict dimensionless source areas, and every (or nearly every) conceivable contaminant release mechanism, secondary source, migration pathway, and exposure point. This type of all-inclusive model sets the stage for an overly ambitious sampling plan and, once the data are collected and the interpretations begin, the models are usually discarded in favor of the standard approach that says that any data worth collecting is worth discussing.

Proper construction to ensure maximal usefulness of the initial site conceptual models during the scoping or planning phase of RI work requires a thorough integration of historical data into the characterization objectives and rationale for the proposed RI. The search for historical data should include searches for site records regarding source characteristics, environmental setting (e.g. geology, hydrology), site operational history, previous investigations or other environmental studies and aerial photographs. It is important to recognize the usefulness of historical data for preliminary modeling purposes even in instances where the quality of data is less than ideal or unknown. All too often investigators dismiss historical data sets based on a desire to ensure rote compliance with current analytical methods or data quality expectations. Instead, emphasis should be placed on the ultimate quality goal of maximizing data usability. Though historical data may have to be qualified it is often quite useful in the planning phase of the RI for constructing a qualitative conceptual model that focuses on the likely contaminant sources, probable release mechanisms, expected pathways, and realistic receptor locations.

Conceptual models, whether applied to an entire site or an individual source of contamination within a site, generally include the following components:

Contaminant Sources are wastes or areas of contamination that have released contamination to the environment or have the potential to do so; in some cases, the source areas also are associated with risks to human or environmental health (e.g., an area of highly contaminated surface soil). Primary sources are typically the wastes themselves; secondary sources are highly contaminated media (soils, sediment, etc.). For preliminary conceptual models, details on the nature and extent of source areas will vary based on the amount of historical (background) information found.

Release Mechanisms are defined as any process that results in the transfer or migration of a contaminant from the source into environmental media such as air, soil, sediment, surface water or groundwater. Typically there is enough information available for a site and its environs to generate a list of known, probable, or potentially active release mechanisms.

Contaminant Migration Pathways are environmental media (e.g. air, surface water, groundwater) that have become contaminated and are capable of carrying the contamination away from the source to the point of exposure. Migration pathways can serve directly as exposure pathways in risk assessments.

Exposure Points are the locations where human or environmental receptors may come into contact with or otherwise be affected by the contaminants from the site.

Examples include contaminated soils in a source area and a creek or stream that receives contaminated run-off or groundwater discharges.
An example of an initial conceptual site model for a site in a relatively humid climate that has buried waste, shallow groundwater circulation and perennial streams and rivers nearby is shown in Fig. 1. On-site receptors are limited to site workers (assuming an industrial-type land use) and environmental receptors (wildlife, vegetation); downstream receptors include residential and recreational users of surface water and environmental resources.
Fig. 1.
VALIDATING THE CONCEPTUAL SITE MODEL
Refining, verifying and quantifying the relationships and processes identified in the conceptual site model is analogous to the validation process used to confirm analytical data and numeric-based models. Validating the initial conceptual site model involves quantifying the relationships between the model components, transforming the qualitative model developed during the scoping phase into a quantitative description of the source term, release rates, migration rates, flux and exposure. The process begins when the first data are coming in from the field and ends with the production of the RI Report. The specific activities associated with the validation process will vary depending on the physical setting, contaminant dynamics, characterization objectives and follow-on decisions to be supported by the RI data. An overview of the validation process is presented in the example site description.
EXAMPLE SITE
An RI based on a conceptual site model was recently completed for a site used for the disposal of radioactive and chemical wastes during the late 1950s and 1960s. The site is a burial ground where low level radioactive waste, transuranic wastes, fissile wastes and hazardous chemical wastes (e.g., spent solvents) were buried in hillside trenches and auger holes in an area of approximately 30 hectares. In addition to the trenches and auger holes, the site included several small landfills, surface impoundments with contaminated sediments and sludges, and underground waste storage tanks. In addition to on-site soils and groundwater, contamination from the wastes have impacted adjacent streams, which are the principal conduits for the off-site migration of (and potential exposure to) site-related contamination.
RI Planning
The initial conceptual site model developed during the scoping phase identified buried wastes as the major source, percolation of precipitation liquids through the wastes as the predominant release mechanism, transport in shallow groundwater as the primary migration pathway, and discharge into a perimeter stream and downstream river as the primary exposure points (for off-site human receptors; exposure points for environmental receptors occurred both on- and off-site). Additional sources (impoundments, landfills and underground tanks) and release mechanisms (interflow during storms and surface runoff) were also identified, but their relative significance was not known.
The initial model was qualitative in nature, consisting of little more than a series of cross-sections depicting the spatial relationships of the model components with directional flow indicators (e.g., downward arrows indicating percolation). It was used to establish the characterization objectives for the RI field investigation, specifically to focus the data collection efforts on information needed to confirm the various hypotheses associated with the conceptual model. These objectives included:
  Source term definition focusing on an inventory of waste types and contaminants of concern as well as waste packaging, placement methods, dates, and locations.
  Identifying principal and secondary release mechanisms, including the partial inundation of trenches by a rising water table, formation of transient, perched zones of saturation in trenches and auger holes, and direct leakage from impoundments and tanks.
  Delineation of surface and subsurface contaminant migration pathways and use of a calibrated hydrologic model to define the relative contributions of run-off, interflow and groundwater discharge to surface water flow from the site.
  Quantifying current and predicted, pathway- and contaminant-specific fluxes from each source area.
  Defining contaminant concentrations at primary exposure points both on- and off-site (as needed to determine concentration terms for use in risk evaluations).

The RI field investigation was designed to address these objectives and conducted using the observational approach to ensure maximum flexibility in the field. Flexibility was an essential component due to the uncertainty associated with most components of the initial conceptual site model. For example, flow monitoring and sampling after storm events early in the field program indicated that these events were major contributors to the overall flux of contamination from the site. The storm event monitoring and sampling program was expanded significantly to facilitate a better understanding of this mechanism. Conversely, it was shown that surface water sampling in the perennial streams that border the site was a more useful measure of the contaminated groundwater discharge than sampling of seeps, and therefore the seep sampling program was significantly reduced.

Data Evaluation

Data evaluation efforts were the principal activity of the RI. The framework for data evaluation was provided by the need to refine the site conceptual model to reflect, as accurately and completely as possible, the principal contaminant types and sources, release mechanisms, transport media and routes, and intermedia transfers, and then to link this model with risk-based considerations of present and potential exposure pathways, exposure point locations, and receptors. This approach entailed a much more rigorous analysis of the site conditions and contamination data than is typically performed for an RI.

The site was divided into two separate drainage basins and further subdivided into smaller study areas based on surface water and groundwater drainage divides because most of the release mechanisms and transport media (as described in the initial conceptual model) were water-based. Divisions within drainages were influenced by previously established solid waste management unit (SWMU) designations and knowledge of source characteristics and/or containment techniques. Each study area contained at least one contaminant source area.

A hydrologic model was constructed for each study area. The model was needed to define and quantify (when possible) the hydraulic processes associated with existing or potential releases of contamination and the subsequent movement of that contamination within the site. These models were depicted using cross sections through contaminant source areas and parallel to the predominant or most likely contaminant migration path(s). Each model extends to the nearest perennial stream or on-site drainage discharge point. In addition to presenting the source's position relative to migration routes (potential and observed), the models identify release mechanisms and intermedia transfers, and present the annual water budgets for individual flow pathways.

A focused sampling and analysis program had generated a large database of analytical data that was then pared down to a smaller subset of contaminants based on input from the human health and ecological risk assessment activities as well as an analysis of background concentrations of chemicals and radionuclides. A listing of analytes of concern for each study was developed for each study area and subsequent data evaluation efforts focused only on the specific occurrences of those analytes determined to represent actual contamination and/or a potential risk.

Analytical results and hydrologic models were combined into a study-area-specific conceptual model that identified the principal sources and contaminants, primary release mechanisms, and the distribution of contaminants within active migration pathways (Fig. 2). The model also presented the results of flux calculations performed for primary analytes (based on contribution to risk). Flux calculations were pathway specific (i.e., surface runoff, interflow, and groundwater) and based on the average annual flow rate and the average concentration detected in that pathway (if available). Flux calculations for surface runoff were based on average soil concentrations, erosion potential, and sediment/soil transport in surface runoff.

Fig. 2.

A separate conceptual site model package was prepared for each study area, which were then used to construct conceptual models for each of the 2 main drainage areas. In addition, fluxes from each study area were combined with the data from downstream receiving waters to generate overall surface water release contributions, which were presented on an area-specific, drainage-basin-specific, and site wide basis for the major contaminants of concern.

RI Report

A major goal of the RI was to generate a comprehensive, yet user-friendly document

that summarized the current knowledge of the site's contaminant dynamics, exposures and risks. To this end, a site-wide conceptual model, based on a compilation of the area- and drainage basin-specific models, was developed and presented in a Technical Summary Volume (<100 pages) that also summarized the environmental setting, included a detailed remediation and operable unit strategy as well as remedial action objectives for the site.
Important features of the model were its ability to identify, illustrate, and rank the sources of contamination (i.e., wastes), the applicable secondary sources/migration pathways, and the release mechanisms that link the two, as well as the pathway-specific on-site risks for both residential and industrial exposure scenarios. The model reflected the hydrologic processes (primarily shallow groundwater) that govern the spread of contamination from the sources to the on-site and off-site groundwater, surface water, sediments, and soils, creating the potential for risks. Release mechanisms were primarily bathtubbing in trenches and inundation of buried wastes with groundwater, particularly during wet season (high base) conditions and, to a lesser extent, surface runoff, storm flow, and percolation of rainwater through the wastes. To facilitate the assessment of the relationships between contaminant sources, release mechanisms, secondary source media, and area-specific contributions to off-site risks, the source areas were ranked by contribution to the total off-site risk.
CONCLUSION
Use of the conceptual site model offers many advantages to the traditional approach for characterization activities, namely the effective integration of risk with the contaminant sources and physical-based processes governing contaminant release, transport and exposure. Although successful application of the method requires a significant level of data evaluation and understanding, the end product will more clearly address project objectives and better support the follow-on decision making activities for remedial action.
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57-4
MEETING THE CHALLENGE OF MOVING HANFORD ENVIRONMENTAL RESTORATION TO A PROJECT MANAGEMENT APPROACH
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ABSTRACT
Implementing the Environmental Restoration Contract (ERC) at the Department of Energy (DOE) Hanford Site is a major step in the process of moving the Hanford Environmental Restoration Project from operations to environmental restoration. This paper describes how the DOE Richland Operations Office and Bechtel Hanford, Inc. (BHI), the new ERC, are working together to implement this change.
BHI assumed responsibility as the Hanford ERC on July 1, 1994, after a four-month transition period. Since the contract began, DOE and BHI have taken a series of steps to set up an organization and establish management systems tailored to the environmental restoration mission. These include: 1) organizing by projects to establish clear lines of accountability and responsibility; 2) establishing project teams and providing them with the resources needed to complete their assignments; 3) implementing productivity improvements and cost savings; 4) forging a partnership between DOE and BHI; and 5) establishing relationships with regulators and stakeholders.
BACKGROUND
The Environmental Restoration Contract (ERC) at Hanford is a demonstration of a new type of contract at DOE sites. The concept of having a separate contractor to manage

environmental restoration originated in 1990 with the Department of Energy's Alternate Contracting Task Force, which looked at ways to lower costs and improve efficiency in DOE's nationwide environmental restoration program. One concept the task force developed was a separate contractor with the single mission of moving a contaminated site to full and complete restoration, rather than having environmental restoration managed by the management and operations (M&O) contractor.

Goals of the new contracting strategy included improving management control of the environmental restoration program, reducing cleanup costs, and facilitating more timely restoration of the sites. Two demonstrations of this concept were planned, one at Fernald and one at Hanford.

The Fernald Environmental Restoration Management Corporation (FERMCO), a subsidiary of Fluor Daniel Corporation, became the Fernald ERC in September 1992. A team headed by Bechtel Hanford, Inc. (BHI) was selected for the Hanford assignment in January 1993. Start of work was delayed by a protest from unsuccessful bidders. Selection of the Bechtel team was reaffirmed by DOE in February 1994; BHI started a transition in March; and the full contract took effect July 1. The Bechtel team includes three preselected subcontractors: CH2M Hill Hanford, Inc.; IT Hanford, Inc.; and TMA Hanford, Inc.

In contrast to Fernald, where FERMCO has responsibility for the entire site, BHI is responsible for only one part of the Hanford mission. There are two major categories of "cleanup" work at Hanford -- waste management and environmental restoration. Waste management includes the treatment, storage, and disposal of wastes, such as the high-level liquid wastes stored in 177 underground tanks. Westinghouse Hanford Company, the Hanford M&O contractor, is responsible for waste management, along with the associated maintenance and operations necessary to manage the 560-square-mile Hanford reservation. These activities account for more than 80% of the Hanford budget.

Environmental restoration, BHI's area of responsibility, is the cleanup and restoration of areas contaminated during past production and disposal activities. It includes cleanup of burial areas, liquid waste disposal sites, and contaminated groundwater, as well as decontamination and decommissioning of reactors and other facilities no longer in use. These activities account for approximately 20% of the Hanford budget.

TRANSITION

Environmental restoration at Hanford is a complex task, and making the transition into the new contract was also complex. Previous contractor transitions at other DOE sites normally involved a complete changeover from one company to another, with the new contractor bringing in a management team, assuming full responsibility for the site, and hiring the full existing workforce. For the Hanford ERC, Bechtel was assuming only a portion of the work being performed by Westinghouse.

This set of circumstances posed many challenges in the areas of management systems, health and safety, milestone compliance, and human resources. Westinghouse and Bechtel worked together to identify employees directly employed by Westinghouse in the ER program and determine their role in the new organization. An equally complex task was identifying the Westinghouse indirect and part-time support staff and determining which would move to the new organization. Effecting this transfer of several hundred employees was made even more difficult because they were transferring from an operations organization into a project management organization.

Personnel from DOE-Richland and Bechtel met early in the transition with their counterparts at Fernald to learn from their experience in making the transition there. The information provided by Fernald was extremely helpful to Hanford in making a successful transition.

A steering committee made up of managers from DOE, Westinghouse, and Bechtel was formed to provide guidance during the transition. Managers from Bechtel met with their counterparts in the Westinghouse organization to learn about activities in their areas of responsibility, review staffing, and identify critical tasks to be completed during transition.

The sheer volume of work in the human resource area was challenging. Westinghouse provided Bechtel with lists of employees who worked full-time in the ER program, and additional lists of people who provided indirect support to the program. BHI invited the Westinghouse employees to orientation sessions of about 50 people each, held in the evenings, to explain the new organization and describe the process that would be used to screen and hire employees. At these meetings, employees were given a phone

number to schedule individual interviews. The BHI team interviewed almost 600 Westinghouse employees, hired about 500, and determined where they would be employed in the new organization. Throughout the process, the two companies worked together to provide information through the site employee newspaper and other channels. During this period Bechtel also set up program controls, safety and health, environmental compliance, procurement, and accounting systems. All of these functions had to be fully operational by July 1.

A key to success during this period was a detailed transition plan, developed by BHI and approved by DOE. This plan listed all actions that had to be completed for BHI to be ready for full contract operations, and designated points of contact within DOE, Westinghouse, and Bechtel for each item. A series of formal readiness reviews were held in June. In these reviews, BHI described actions that had been taken to achieve each requirement in the transition plan. DOE management determined if the area was ready for operations and identified action items to be completed after transition. DOE determined in the readiness reviews that the BHI team was ready to assume full responsibility for the Hanford ER program effective July 1.

ACTIVITIES TO DATE

Since the contract began, DOE and BHI have taken a series of steps to set up an organization and establish management systems tailored to the environmental restoration mission. These include: building a partnership between DOE and the BHI team; organizing by projects to establish clear lines of accountability and responsibility, giving project teams the resources needed to complete their assignments; implementing productivity improvements and cost savings; negotiating agreements with organized labor; and establishing relationships with regulators and stakeholders.

The organization of the Hanford ER project illustrates a fundamental difference between an operations approach and a project management approach. An operations approach is organized by operating functions, using an in-house workforce focused on repetitive tasks. It makes minimal use of subcontractors and has numerous management layers. A project management organization has few management layers, is built around project teams, and uses specialty subcontractors to perform much of the work.

The BHI organization at Hanford is built around project teams. Each project team is composed of the engineering, construction, environmental compliance, health and safety, quality assurance, and support personnel needed for a specific task. The Environmental Restoration staff of DOE-Richland, which oversees BHI, also organized along project lines. This "mirror image" between DOE and BHI results in an efficient working relationship.

The working relationship was further enhanced by a project management retreat held in June. This facilitated retreat was a very worthwhile exercises. Purposes of the retreat were to:

- Build a solid foundation of mutual respect and partnership between BHI and DOE
- Identify the primary expectations each organization had of the other, and use these expectations to define the distinct roles of DOE and BHI
- Identify the issues and opportunities facing the project and put the organization interface in place to resolve these issues
- Create and then commit to an overall "Commitment Statement" for the project

This retreat introduced key players in DOE-RL and BHI to each other and helped motivate both organizations to focus on common goals and to better define roles and responsibilities. The working relationship developed during this exercise has been continued through a "Results Management Team" made up of managers from the DOE and BHI organizations. The team meets weekly to:

- Enhance the velocity and effectiveness of the transition
- Instill unified, integrated project leadership
- Focus on accountability for results and managerial control
- Challenge the team to reach for and begin producing breakthroughs in results
- Instill a "project" culture and build momentum in the team
- Identify and resolve issues openly and effectively
- Create a win-win context among the DOE and the contractors

After this teaming arrangement was in place and working at the upper management level, it was then instituted at the project level in December. Representatives from each project team met to build a foundation of partnership, identify priorities and expectations, identify issues and opportunities, and create a commitment statement for each project.

The first measurement of the effectiveness of this "win-win" relationship between DOE and BHI came in December with DOE's fee determination report for the first three months of the contract (July through September). The BHI contract is a Performance Based Fee contract, in which a fee pool is available based on performance only. DOE-RL's Performance Evaluation Committee defines specific performance objectives and criteria prior to each rating period and then monitors BHI's performance to each of these criteria. Based on the ratings, BHI can earn anywhere from zero to 100 percent of the available fee. For the July-September period, BHI earned 88% of the available fee. This high rating, despite the many difficulties posed by the transition, indicates that the teaming arrangement between DOE and the contractor is working.

During the last three months of 1994 and the early part of 1995, the Hanford ER program has focused on funding issues. Reductions in the FY 1995 funding and projections of further reductions in FY 1996 and beyond have resulted in a series of budget studies. The ability to perform these studies has been enhanced by having a defined, project-oriented baseline in place.

CONCLUSION

DOE and BHI have made the transition into this new type of contract and are showing progress in expediting remedial action and improving cost-effectiveness. There are, however, still many challenges remaining. These include:

Further refining the contract to remove provisions that were appropriate for an M&O contract but not for environmental restoration.

Closing on a labor agreement.

Developing an environmental restoration strategy for the site that focuses on "doing the right things" and not merely "doing things right."

The biggest challenge for the Hanford ER project will be to continue showing progress during a time of reduced funding and continued regulatory and public pressure to show results.

57-5

ISSUES RELATED TO UNCERTAINTY IN PROJECTIONS OF HAZARDOUS AND MIXED WASTE VOLUMES IN THE U.S. DEPARTMENT OF ENERGY'S ENVIRONMENTAL RESTORATION PROGRAM*

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ABSTRACT

Projected volumes of contaminated media and debris at U.S. Department of Energy (DOE) environmental restoration sites that are potentially subject to the hazardous waste provisions of the Resource Conservation and Recovery Act are needed to support programmatic planning. Such projections have been gathered in various surveys conducted under DOE's environmental restoration and waste management programs. It is expected that reducing uncertainty in the projections through review of existing site data and process knowledge and through further site characterization will result in substantially lowered projections. If promulgated, the U.S. Environmental Protection Agency's Hazardous Waste Identification Rule would result in potentially even greater reductions in the projections when site conditions are reviewed under the provisions of the new rule. Reducing uncertainty in projections under current and future waste identification rules may be necessary to support effective remediation planning. Further characterization efforts that may be conducted should be designed to limit uncertainty in identifying volumes of wastes to the extent needed to support alternative selection and to minimize costs of remediation.

INTRODUCTION

Identifying quantities of contaminated environmental media that are subject to hazardous waste regulations under the Resource Conservation and Recovery Act (RCRA) is an issue confronted at most hazardous waste sites being addressed under either RCRA corrective actions or under the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA). At U.S. Department of Energy (DOE) sites, the further presence of radioactive contaminants in such media adds the prospect that media may be considered both hazardous and radioactive waste, or mixed waste. The administration of the hazardous waste regulations for a given site, including DOE sites, is usually the province of the environmental protection agency of the state in which the site resides. Authority over radioactive wastes, on the other

hand, has traditionally been the responsibility of federal agencies, principally DOE and the U.S. Nuclear Regulatory Commission (NRC). With the 1992 passage of the Federal Facility Compliance Act (FFCA) amending RCRA to bring federal facilities into compliance with applicable federal and state laws, the administration of mixed waste is conducted jointly under both federal and state authorities.

Identifying and quantifying hazardous and mixed wastes is an important activity within DOE's waste management (WM) program and within the environmental restoration (ER) program, in particular. Such activities have major implications regarding the cost of remedial activities and regarding waste management and waste treatment decisions within the program.

This paper discusses the general nature and sources of uncertainties in projections of volumes of environmental media and debris (i.e., contaminated equipment and structures) potentially requiring management as hazardous or mixed wastes within the ER program and suggests means for reducing them. Hazardous and mixed wastes have been singled out for evaluation because of the inherent difficulty in estimating quantities of such wastes in environmental media and debris, because of the large number of technologies available for the required treatment of such media, and because of the high costs associated with treatment and management. In addition to evaluating the nature, sources, and general level of uncertainty in the volume estimates, the nature of impacts of uncertainty on various waste management options are suggested. Lastly, recommendations are made as to what relative levels of uncertainty may be acceptable for supporting various action alternatives and how such levels may be attained through reevaluating current information and through conducting further characterization.

MANAGEMENT OPTIONS FOR HAZARDOUS AND MIXED WASTES

Application of Waste Regulations to Contaminated Environmental Media

Under RCRA, states with authority can classify contaminated environmental media as hazardous waste under two basic provisions of the rules (40 CFR 261): 1) the medium is a characteristic hazardous waste because it has failed a specific characteristic test; or 2) the medium is a hazardous waste under the mixture rule. The mixture rule provides that any product of the mixing of a listed hazardous waste with another solid waste is itself a hazardous waste (not to be confused with mixed waste, i.e., waste that is both hazardous and radioactive). Listed hazardous wastes are derived from specific industrial processes and activities and include many of the substances that often contaminate environmental media, including, for example, spent solvents. Consequently, under the mixture rule, regulations for listed wastes can, in certain circumstances, be extended to environmental media. Given these two provisions, states have had, and continue to have, broad authority to classify contaminated media as hazardous waste. Similarly, by extension, such authority also applies to the classification of media contaminated with mixed waste.

With respect to characteristic wastes, definitive classification requires testing of hazardous characteristics, principally the toxicity characteristic, which is tested using the toxicity characteristic leaching procedure (TCLP; 40 CFR 261, Appendix II). (The characteristics of ignitability, corrosivity, and reactivity can be ruled out for most environmental media and debris.) Preliminary or presumptive classification, on the other hand, can be done on the basis of general information (including process knowledge, spill logs, and material inventories) or even by applying standard assumptions regarding releases from vessels and the like. But definitively establishing that a particular medium is not a characteristic hazardous waste, once brought into question, requires specific characteristic testing.

To establish that a particular medium is hazardous waste under the mixture rule is a less definitive process. The mere presence of a listed hazardous waste constituent may cause a medium to be classified as hazardous (or mixed) waste. No de minimis levels exist under the mixture rule. Once a waste (medium) is classified as hazardous waste under this rule, it can only be declassified by petitioning the U.S. Environmental Protection Agency (EPA) to delist the waste, a time-consuming process. Changes to the hazardous waste identification rules are being considered that will offer relief from the mixture rule, as discussed in a later section of this paper.

Regulations under the mixture rule are not explicitly directed to environmental media, e.g., soils or groundwater, but are applied to such media to the extent they can be considered solid wastes. The mixture rule does, however, specifically address contaminated debris, under the so-called debris rule (40 CFR 268.2[h]). Debris is a significant waste element in the ER program because many restoration projects

involve the demolition of buildings and other structures. It is assumed that the mixture rule applies equally to debris and environmental media in the ER program. In the identification of mixed waste, the above hazardous waste criteria, along with prevailing radioactive waste criteria, are applied independently. Mixed wastes are usually initially established to be radioactive wastes on the basis of NRC regulations and DOE Orders and later determined to be also hazardous wastes under RCRA.

With respect to radioactivity, much of contaminated media at ER sites falls into the category of low-level waste, which is defined as radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in Section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste) (10 CFR 61). Two other classes of radioactive wastes of importance at ER sites are 1) transuranic (TRU) wastes, defined as containing more than 100 nCi of alpha-emitting transuranic isotopes with half-lives greater than 20 years per gram of waste (DOE Order 5820.2A), and 2) 11e.(2) byproduct material (primarily uranium mill tailings), referring to the portion of metal-bearing ore remaining after extraction of uranium (DOE Order 5820.2A). As the identity of such wastes is already well established at most ER sites, details of the application of radioactive waste classification criteria will not be presented here.

Management and Treatment of Generated Waste

Contaminated environmental media are not technically wastes until "generated" through, for example, excavation as in the case of soils. Once generated, that which is classified as either hazardous or mixed waste is subject to the treatment, storage, and disposal regulations under RCRA (40 CFR 262, 40 CFR 263, and 40 CFR 264). Such wastes that are being considered for disposal are subject to land disposal restrictions (LDR) under RCRA. The LDR rules (40 CFR 268) prohibit the disposal in or on land of wastes that exhibit toxicity or the potential to leach contaminants into groundwater above specific thresholds. Characteristic hazardous wastes are explicitly prohibited from such disposal unless treated to the point that they no longer exhibit the characteristic. Wastes with listed hazardous constituents are subject to treatment standards specific to the constituents or to treatment by a specified technology. Verifying that such treatment standards are met for either waste again requires that the waste be characterized. Often it can be established that wastes, including environmental media, identified through the mixture rule already meet treatment standards in the absence of any treatment.

Alternative Management Strategies

One avenue that has been opened to reduce the administrative burden of LDR rules is the institution under RCRA of the corrective action management unit (CAMU) concept. Under this concept, placing remediation wastes generated at a facility as part of a RCRA corrective action into or within a CAMU is not considered land disposal and such wastes therefore are not subject to LDR (40 CFR 264, Subpart S). Remediation wastes (media) may be excavated within a CAMU, treated in a second CAMU, and redeposited in the excavated area of the original CAMU without triggering LDR. Also, waste disposal units within a CAMU are not subject to minimum technological requirements under RCRA. The responsibility and authority for protecting health and the environment under this approach is assumed under the prevailing RCRA corrective action program. The use of CAMUs is expected to reduce delays in cleanup efforts.

A second development currently under way that addresses the issue of identification and management of hazardous waste is EPA rulemaking aimed at reform. The proposed Hazardous Waste Identification Rule (HWIR) (57 FR 21450, May 1992) addresses, among other things, wastes classified as hazardous under the mixture rule, and should benefit environmental restoration efforts. The proposed HWIR would establish an entry and exit system for prospective wastes. The rule would employ health-based, contaminant-specific concentration limits, termed "bright line" numbers, as criteria for entry or exit of the system. The limits could be used to identify hot spots in environmental media that would be subject to RCRA hazardous waste rules, while removing those requirements for media falling below the criteria. The latter media would still, however, be subject to site-specific cleanup standards.

The rule was proposed in 1993, then withdrawn after comments were reviewed. It is expected to be reproposed by September 1995. Large quantities of contaminated media at ER sites currently subject to RCRA regulations, including the mixture rule, could be affected by the new HWIR. Similarly, characterization efforts related to identifying hazardous and mixed wastes in environmental media would have to conform

to the new rule, if promulgated, possibly requiring reevaluation of existing data and/or recharacterizing media in light of the "bright line" numbers.

NATURE AND SOURCES OF UNCERTAINTY IN ESTIMATES OF MEDIA VOLUMES SUBJECT TO RCRA

Several surveys of ER sites have been conducted within DOE's environmental management (EM) program, gathering information on estimates of contaminated environmental media and debris that are potentially subject to subject to hazardous waste regulations under RCRA. Surveys include the Contaminated Media/Waste Data Call, the estimates prepared to support the Programmatic Environmental Impact Statement (PEIS), and estimates gathered to support the Baseline Environmental Management Report (BEMR). The surveys relied on site characterization data and process knowledge available at ER sites. The current analysis does not examine any particular estimate but describes the nature and sources of uncertainty in estimates of this type in general. The volume estimates generated are assumed to be inherently conservative, both with respect to the volumes of media that might ultimately be removed for remediation, and thus generate solid waste under RCRA, and with respect to the volumes of solid waste that may ultimately be determined to be hazardous or mixed waste under RCRA. This conservatism arises in general from a desire on the part of waste managers to include all media that will have to be addressed in the future, whether or not it is technically hazardous or mixed waste, and as a result of the generally small amount of characterization data available with respect to current or future hazardous waste criteria.

Many surveys to estimate volumes of contaminated media are conservative by design, often forcing the assignment of suspect media into a particular waste class. Volume estimates, particularly of hazardous and mixed wastes, therefore may start from a conservative base. Moreover, the estimated volumes of contaminated media and debris are often just a first approximation based on limited data or on simple modeling. Such volume estimates are often done in a conservative manner so as to avoid misidentifying any contaminated areas.

With respect to identifying regions within contaminated media that may be potentially hazardous or mixed waste, authoritative data needed to make such identifications are often sparse and, where available, may not have been collected for the purpose of characterizing media according to hazardous waste criteria. Further data needed to define the boundaries of such regions are particularly lacking or may be completely absent. Often, simply the known or suspected presence of a characteristic or listed hazardous constituent is all the information available as a basis for classifying media and developing volume estimates.

In cases where the mixture rule may apply, a particular medium can be classified as hazardous or mixed waste on the basis of very low levels of hazardous constituents. At many contaminated sites, the mere presence of a listed constituent has been used to preliminarily identify a medium as potentially subject to RCRA without regard to the origin of the constituent. However, if the source of the hazardous constituent was not a listed waste, such as spent solvents, for example, in the case of certain volatile organic constituents, then the medium is not subject to the stringent criteria of the mixture rule. Instead, the quantitative criteria for characteristic hazardous wastes apply. In that regard, data for determining the status of environmental media with respect to hazardous waste characteristic criteria using the TCLP are particularly limited at ER sites.

Because a final determination of the sources of hazardous constituents may not have been fully established at many ER sites, large quantities of contaminated media may be assumed to be subject to RCRA under the mixture rule that in fact are not. The low levels of hazardous constituents in much of this media indicate that the media may be identified as nonhazardous after hazardous characteristic testing using the TCLP.

Current estimates of waste volumes potentially subject to regulation as hazardous or mixed (hazardous and radioactive) waste under RCRA may therefore be considered to represent a reasonable upper bound on hazardous and mixed wastes that may be generated in the ER program. The level of uncertainty varies from site to site and from medium to medium according to the level of information available. The greatest volumes as well as the greatest relative uncertainties are probably associated with groundwater and soil.

The volumes of media that will ultimately have to be addressed in the program, however, are probably substantially less than these upper bound projections, even if changes in waste identification rules are not forthcoming. Efforts to refine the

boundaries of contaminated areas will surely reduce volumes. Given the economic incentives, characterization efforts to support such refinements will be easily justified. Much greater reductions in volumes are possible, however, under scenarios that provide relief from the mixture rule, such as under the proposed HWIR described above. Some forms of rule changes are to be expected over the multiyear course of the ER program. With respect to some mixed waste, volume reductions would in fact reflect reclassification from mixed waste to the prevailing radiological waste class, although refinement of the volumetric extent of the radiological component through further characterization or monitoring during excavation is certainly also possible.

MANAGING UNCERTAINTY IN VOLUME ESTIMATES

Impacts of Declining Volume Estimates on Action Decisions

If existing volume estimates of hazardous and mixed wastes represent an upper bound, as asserted above, then reductions in uncertainty in these estimates will correspond to declining estimates of volumes. As volume estimates decrease, management decisions addressing the wastes may change.

There are three major factors influencing any scenario under which hazardous and mixed waste cleanup actions might proceed under the ER program: 1) the extent to which the current action plan under current RCRA waste rules is carried out, 2) the extent to which the CAMU concept can be applied to additional ER sites, and 3) and the extent to which proposed revised hazardous waste identification rules affect the ER program. Declining volume estimates will have impacts on cost estimates and associated action decisions under the various possible scenarios.

Under any scenario, significant reductions in hazardous and mixed waste volumes from current estimates will impact the budgeting of remedial actions, the selection of action alternatives, and the selection of treatment technologies and process options. Table I presents the nature of impacts of declining volume estimates on these activities in terms of the three scenario factors. The impacts increase according to the degree that volume reductions are realized through improved characterization and to the extent that the CAMU concept and revised hazardous waste rules apply to a given scenario.

Preliminary selection of both the general alternative actions and specific treatment processes at various ER sites may be expected to change in the directions suggested in Table I as cleanups progress and requirements change.

Reducing Uncertainty in Volume Estimates

Refinements to volume estimates will be required on a continual basis as the ER program is carried out. For initial planning purposes, current volume estimates may have to be evaluated and revised before major planning decisions are made. That is, the uncertainty in current estimates may be too high to support a comprehensive action plan relative to an environmental medium that is potentially hazardous or mixed waste. In general, reducing uncertainty to support planning will require 1) reexamining existing data available at the sites with the expressed purpose of improving volume estimates and 2) conducting further characterization to improve estimates where existing data are insufficient. The extent to which these activities need to be carried out varies from site to site.

Several measures can be taken to reevaluate existing site data. First, if characterization data are available, those data should be evaluated for representativeness. The inferences drawn from the data can then be applied to the volume of medium represented by a given sample. Second, it is very important that in cases where RCRA characteristic hazardous waste criteria are being applied, rather than the mixture rule, this fact be firmly established and characterization data applied according to the applicable concentration criteria. In such cases, a particular medium can be classified on the basis of available TCLP data or, lacking that, to a great extent on the basis of direct analysis of the medium. In the latter case, a solid medium can be determined to be nonhazardous if contaminant concentrations in the medium do not exceed 20 times the TCLP criteria. (This criterion ensures that TCLP leachate could not possibly exceed the criteria.) For instance, if all soil samples representing a given volume of medium have no concentrations exceeding 20 times TCLP criteria, it may be inferred that none of that volume is hazardous or mixed waste. In cases where wastes are identified on the basis of the mixture rule, improvements in estimates may be accomplished through careful review of sample locations.

Beyond these steps, where data are sparse, the application of process knowledge may

improve volume estimates. Such knowledge might include spill logs, area use patterns, chemical inventories, or a mass balance analysis. In some cases, site-specific contaminant transport modeling, in which existing data and process knowledge are applied, may be appropriate to refine volume estimates of contaminated media.

In cases where additional characterization is required, either to support planning or implementing an action, a focused and efficient sampling and analysis plan should be developed with well conceived data quality objectives. In such development, the sampling problem and action decisions should be well defined and measurement uncertainty goals set accordingly. The first step in this process is to identify the leading action alternative for the affected media. Next, consideration of costs per volume of media and total volumes of media affected should be used to indicate the levels of uncertainty in volume estimates that may be acceptable and thus the level of characterization required. Table II presents a qualitative ranking of uncertainty as driven by overall costs that might be acceptable under various action alternatives for a given volume of waste.

The general observations shown in Table II would further be a function of the total volumes of waste ultimately identified. For example, percentage uncertainties in media volumes acceptable at a 1,000 m3 level may be unacceptable at a 100,000 m3 level. An additional factor to be considered relative to uncertainty in volume estimates of mixed waste is whether further characterization will result in reducing volumes of the media that are either potentially hazardous or radioactive, or both. Another factor that will affect characterization requirements is the extent to which new hazardous waste identification rules are adopted in the future. Given concentration-based criteria, increased levels of characterization will be required to define areas of media above and below the criteria. The levels of uncertainty attained in these efforts will depend on a balance of the costs of characterization versus those of treatment and/or disposal of excess media.

SUMMARY AND CONCLUSIONS

Contaminated environmental media at DOE ER sites are subject to the provisions of RCRA and may be classified as hazardous or mixed waste on the basis of characteristic testing or the mixture rule. Estimated volumes of such media that are potentially hazardous or mixed waste are large and are assumed to be conservative because 1) surveys are generally based on first-order approximations of the boundaries of contaminated regions, 2) current levels of characterization do not support identification of subregions, or hot spots, of media exceeding RCRA criteria, and 3) some media may be classified as hazardous waste on the basis of a misapplication of the mixture rule. It is expected that improving media characterization and reviewing RCRA criteria will result in declining volume estimates. Reduced waste volumes, in turn, will have impacts on cost estimates and on the selection of remedial action alternatives. Improved volume estimates, therefore, may be necessary to support remedial action planning. Such improvements may be effected through review of existing characterization data and process knowledge, or through further characterization of contaminated media. The levels of uncertainty acceptable in further characterization efforts will depend on the nature of the contamination, the volume of contaminated media involved, the type of remedial alternative selected, and the overall costs of the remedial actions.

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CHARACTERIZATION OF HAZARDOUS WASTE RESIDUALS FROM ENVIRONMENTAL RESTORATION PROGRAM ACTIVITIES AT DOE INSTALLATIONS:
WASTE MANAGEMENT IMPLICATIONS*

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ABSTRACT

Investigators at Argonne National Laboratory (ANL), with support from associates at the Pacific Northwest Laboratory (PNL), have assembled an inventory of the types and volumes of radioactive, toxic or hazardous, and mixed waste likely to be generated over the next 30 years as the U.S. Department of Energy (DOE) implements its

nationwide Environmental Restoration (ER) Program. The inventory and related analyses are being considered for integration into DOE's Programmatic Environmental Impact Statement (PEIS) covering the potential environmental impacts and risks associated with alternative management practices and programs for wastes generated from routine operations. If this happens, the ER-generated waste could be managed under a set of alternatives considered under the PEIS and selected at the end of the current National Environmental Policy Act process.

As the ER Program progresses over the next 20 to 30 years, it is estimated that areas identified for restoration within certain DOE installations will generate about 2.5 106 t (1.7 106 m3) of nonradioactive solid waste through site restoration activities. The ER residuals will typically include process residues and sludges; contaminated debris, soil and concrete; bulk metals; and stabilized monolithic solids. A significant fraction of this waste will require solid landfill space for disposal, and a portion will need to be treated and/or disposed of at specially permitted facilities. Because DOE has little or no on-site capability for treating and disposing of the types of nonradioactive or hazardous wastes that the ER Program will generate, the current practice is to transport wastes of this type off-site for treatment and disposal. Conceivably, under the currently structured PEIS alternatives, nearly all of the nonradioactive ER waste inventory could be targeted for off-site commercial facilities for treatment and/or ultimate disposal. This practice could result in unnecessary, costly, and potentially risky site restoration actions, including the transportation of approximately 120,000 truckloads of nonradioactive ER wastes over the next 30 years.

INTRODUCTION

The U.S. Department of Energy (DOE) has supported the compilation and analysis of an environmental restoration waste inventory for consideration in the preparation of the Programmatic Environmental Impact Statement (PEIS) for its Environmental Restoration (ER) and Waste Management (WM) Programs. The information presented in this paper is taken, in part, from this effort. Under the PEIS, the DOE is evaluating and comparing the risks associated with current practices and programs for managing its wastes with those associated with alternative waste management scenarios. The specific risks and environmental impacts being evaluated are human health risks and costs associated with transportation and/or treatment, ecological risks, land-use impacts, and other impacts.

Routine operations at DOE facilities generate radioactive, nonradioactive, and mixed waste. These wastes are managed under DOE's WM Program. Wastes that are nonradioactive such as those that exhibit the hazardous characteristics of corrosivity, ignitability, reactivity, and/or chemical toxicity are managed in accordance with regulations implementing the Resource Conservation and Recovery Act (RCRA), the Toxic Substance Control Act (TSCA), and hazardous and toxic waste legislation at the state level. The DOE commonly refers to wastes covered by RCRA or TSCA regulations and by state regulations as hazardous waste (HW). Radioactive wastes generated at DOE installations are grouped by the level and type of radioactivity present and are managed separately in accordance with special regulatory programs set forth in the Atomic Energy Act and by the U.S. Nuclear Regulatory Commission. The PEIS is assessing the risks and costs associated with the transportation, treatment, and disposal of HW, high-level radioactive waste (HLW), transuranic waste (TRUW), low-level radioactive waste (LLW), and low-level mixed waste (LLMW). Waste that has both radioactive and hazardous components is known as mixed waste. Mixed waste potentially includes all radioactive waste types, but typically LLMW and TRUW, and must be managed in accordance with both sets of rules. This paper focuses on the characteristics of the nonradioactive HW that will be generated under the ER Program, and examines the impact such waste will have on DOE's current practice of sending virtually all nonaqueous HW off-site for treatment and disposal.

Wastes that will be generated under DOE's ER Program (ER wastes) are nonroutine, meaning that the wastes will be generated once in the course of remediating DOE facilities. These wastes will include HW, LLMW, LLW, and TRUW, although in proportions substantially different from those generated by routine operations. The quantitative and qualitative relationships between wastes generated under the WM Program (WM wastes) and ER wastes are shown in Fig. 1. These values represent the best estimates currently available on the amounts of hazardous, mixed, and radioactive wastes that DOE is likely to generate each year over the next few

decades (1-4). Figure 1a shows the projected annual quantities of HW, LLMW, LLW, TRUW, HLW, and other wastes that will continue to be generated yearly by routine, ongoing DOE operations and managed under DOE's existing WM Program. Figure 1b shows the quantities of HW, LLMW, LLW, TRUW, and other wastes that can be expected each year, on average, over the next 30 years under the ER Program. The values for the WM wastes have a higher level of confidence because they are based on studies of actual waste generation records (e.g., waste manifests) and therefore are rooted in historical experience. The values for the ER wastes are less certain because they are based on incomplete waste characterization data and computer-generated projections. No allowance has been made in either group for future reductions due to waste minimization initiatives or regulatory changes, such as the expected 1995 redefinition of RCRA hazardous wastes, which might exclude many of the contaminated media wastes from regulation in later years.

The focus of this paper is on the group of HW that will be generated under the ER Program and the impact this waste could have on DOE's current practices for managing HW under the WM Program.

WASTE MANAGEMENT PLANNING ALTERNATIVES

The DOE is considering alternative waste management scenarios for each of the five waste types (HW, LLMW, LLW, TRUW, and HLW). These scenarios differ mainly as to how the 45 DOE sites could manage their chemical wastes, including transportation, and treatment, storage, and disposal (TSD). Portions of the waste might be handled on-site and the rest handled off-site (by outside commercial firms). Determining how to distribute the waste between on-site treatment and off-site TSD facilities leads to a comparison of multiple alternatives requiring the evaluation of human health risks, costs of treatment and transportation, and ecological, land-use, and socioeconomic impacts. The alternatives currently under consideration by DOE include:

Baseline (No Action) Case The baseline or no-action alternative refers to the continuation of the current distribution of chemical hazardous wastes between on-site treatment and off-site commercial TSD vendors and the continuation of current (as of 1992) routing of hazardous waste trucks from the DOE generation site to the final TSD location.

Decentralized Case Under decentralization, treatment activities at DOE sites would increase between 5 and 10%. As a result of this increase, use of commercial TSD vendors would decrease, likely reducing the risks from transportation.

Five-Region Case (Regionalized No. 1) The five-region case builds upon the decentralization alternative by treating, at DOE installations, approximately 50% of the waste that is currently treated, stored, and disposed of off-site. Instead of being transported to commercial facilities, this waste would be transported (if necessary) to one of five designated DOE-owned-and-operated hazardous waste treatment facilities. The five DOE installations selected for this treatment (for evaluation purposes) are the Hanford Site (HS), Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Oak Ridge Reservation (ORR), and Savannah River Site (SRS). All treated waste would be disposed of at permitted commercial landfills.

The Two-Region Case (Regionalized No. 2) Under the two-region case, 90% of all chemical hazardous waste (nonwastewater [nonaqueous] hazardous waste) currently treated by commercial TSD vendors would be transported to one of two DOE-owned-and-operated hazardous waste treatment facilities. The two DOE installations selected for this treatment (for evaluation purposes) are INEL and ORR. All treated waste would be disposed of at permitted commercial landfills.

The appropriateness of these alternative strategies for managing the ER wastes, especially the ER HW, is currently being evaluated. The key issue is the fact that none of the current alternatives include a provision for on-site disposal capability at strategic DOE installations. Even though varying degrees of treatment capacity changes are explored, none of the alternatives considers whether the resulting treatment residues should be disposed of on-site and, if so, what impact this would have on existing landfill capacity. The question of whether more on-site landfill space should be built into future planning for DOE waste management is important, given the large volume of wastes expected to be generated under the ER Program. The systems now in place to manage HW from routine operations and the potential application of these systems to the management of HW that will be generated under the ER Program are examined below.

HAZARDOUS WASTES FROM FACILITY OPERATIONS

Hazardous wastes that arise out of routine ongoing operations at DOE installations are generated in more or less the same quantities year after year. These wastes, also called "as generated" HW, are managed under DOE's WM Program. Recent updates to the HW inventories, compiled in the hazardous waste risk assessment modeling database (5), show that about 1.26 106 t of HW is being generated each year in the course of weapons manufacture and retirement, nuclear fuel processing, research, electroplating, photographic work, building and equipment maintenance, and other normal activities at DOE facilities nationwide.

Nearly all of this waste, over 98% (1.24 106 t) of the WM HW total, is hazardous wastewater derived from production or laboratory operations. Wastewater produced in large volumes is readily treated on-site, and DOE has several large on-site wastewater treatment plants to process virtually all of the hazardous wastewater it generates. These plants are located at the ORR, Kansas City Plant (KCP), Sandia National Laboratory-New Mexico (SNL-NM), SRS, Rocky Flats Plant (RFP), and INEL.

The remainder of the WM "as generated" HW produced each year is relatively small in volume compared with the hazardous wastewaters. Data from 1991 to 1992 indicate that DOE produces between 0.02 and 0.09 106 t of other HW each year, including undiluted or concentrated waste chemicals, chemical mixtures and solutions, spent solvents, oils, greases, ordnance chemicals, acids, caustics, photographic chemicals, plating wastes, and other types like petroleum- or mineral-based waste materials (5). Such wastes including solids, liquids, and compressed gases that are corrosive, toxic, reactive, explosive, and/or ignitable are generated in relatively small quantities within the DOE complex and are contained in drums, tanks, cylinders, and lab packs. Occasionally, this group of HW includes a few truckload quantities of polychlorinated biphenyls (PCBs) or asbestos-contaminated equipment, soil, and debris from cleanup operations at various DOE facilities. Although a small part of these wastes is being incinerated or deactivated at existing on-site facilities, most of the HW requiring organic destruction or organic removal and recovery is transported off-site to commercial RCRA and TSCA facilities for treatment and disposal. As a result of this relatively low demand, DOE has few active on-site RCRA-permitted hazardous waste treatment facilities and no RCRA landfills.

HAZARDOUS WASTES FROM ENVIRONMENTAL RESTORATION

The nonradioactive ER wastes are nonroutine wastes that will be generated when site-specific environmental restoration plans are implemented. These plans call for the long-term cleanup of previously buried wastes and contaminated soil, groundwater, surface water, buildings, structures, and equipment at over a dozen DOE facilities within the next 20 to 30 years. These ER wastes may be hazardous under RCRA or state regulations, toxic under TSCA, or solid waste with low contamination levels that can be disposed of directly in landfills. The data currently available suggest that a significant fraction of these wastes may not be hazardous or toxic under RCRA, TSCA, or state regulations. Thus, until more definitive information is available, the authors of this paper have chosen to discuss all nonradioactive ER wastes together and to conservatively refer to them collectively as ER HW.

Environmental site survey data (e.g., borehole characterization data) were used as input to the automated remedial assessment restoration (ARAM) computer model (6) to create a database on the types and volumes of ER HW that will be generated throughout the DOE complex under the ER Program (7). Similar ER databases were created for the mixed waste and radioactive waste types. The data identify the DOE installation where each hazardous waste stream will be generated, the activity responsible for generating the waste, the physical type of each waste stream, the projected total volume, and, where known, the chemical constituents likely to be present and the estimated average concentration of each chemical constituent expected in each waste stream.

The ER Program is expected to generate over 80 separate waste streams producing a total of nearly 2.5 106 t (or 1.7 106 m3) of HW over the next 20 to 30 years. These wastes are expected to come from remediation of buildings, structures, equipment, pits, lagoons, disposal sites, land, and groundwater at 14 DOE installations. A rank order of those installations by the projected amounts of HW expected is given in Table I. The restoration activities projected for INEL are estimated to produce the largest volume of ER HW, over 1.5 106 t (1.0 106 m3), in the DOE complex.

Unlike the HW in the WM Program, most of the waste streams listed in the ER HW

database are not wastewaters. The ER HW is listed either as solid residuals from on-site treatment of buried wastes and contaminated media or as untreated contaminated debris from facility demolition and other decontamination operations. Analyses of these data indicate that five basic types of projected ER wastes will be generated, as follows:

Monolithic solids (900,000 m3) These solid wastes consist primarily of asbestos and metal-contaminated soils or treatment residues (such as contaminated clay and silt soil fractions from soil-washing operations) that have been solidified and stabilized by grout/lime/flyash fixation or by vitrification. More than half (54%) of the ER wastes from all sources are expected to be in the form of monolithic solids.

Contaminated debris (773,600 m3) The debris is primarily composed of previously buried asbestos abatement waste. It also includes chemically tainted concrete dust from decontamination or surface scraping of concrete structures, as well as contaminated metal parts from dismantled equipment, linings, and frames. About 46% of all ER wastes will be debris.

Solid process residues and sludges (3,140 m3) These wastes are typically wastewater treatment sludge solids or spent activated carbon residues from groundwater and soil vapor extraction treatment systems. Solids or sludges from solvent recycling operations may also be included.

Contaminated soils (85 m3) A small amount of lightly contaminated soil will be produced that will not be stabilized due to its low level of contamination.

Aqueous wastes (22 m3) The aqueous wastes originate from incinerator scrubber waters, ash quench waters, or aqueous streams containing dilute amounts of solvents from solvent recovery operations.

Essentially all of the ER HW will be secondary or by-product waste (i.e., relatively dilute) because it will be derived from the cleanup or treatment of previously buried HW and remnants of those wastes that were generated, leaked, spilled, released, stored, dumped, removed, or disposed of sometime in the past. The ER HW differs physically from the WM HW that is associated with normal or routine operations. The ER HW will be mostly solid in form (monoliths and debris), whereas the WM HW will be mostly aqueous in form (wastewater). In terms of their intrinsic hazards and risk management requirements, the ER wastes will be comparable to the hazardous wastewaters that are generated every day by normal operations because, like the wastewaters, the hazardous constituent concentrations will be relatively dilute. Much of the waste included in the ER HW group may, in fact, be appropriately classified and managed as nonhazardous solid waste.

PRELIMINARY TREATMENT AND DISPOSAL FINDINGS FOR ER HAZARDOUS WASTES

A preliminary treatment and disposal allocation plan was developed to identify the most appropriate way to manage the projected wastes generated by the ER Program. It must be recognized that this plan is based upon the limited data available for proper waste characterization, which necessitated some basic, but reasonable, assumptions to be made about the data available on waste chemical profiles and the hazardous or toxic characteristics likely to be exhibited by each ER hazardous waste stream. The assumptions used in developing the allocation plan are described below:

Assumption 1 None of the ER HW will be listed RCRA wastes. None is known to have been generated by listed RCRA waste processes (described in 40 CFR Part 261), and none is known to be mixed with, contain, or be derived from listed RCRA wastes. Therefore, treatment and management of the ER HW has been assumed to be guided strictly by the hazardous waste characteristics exhibited by each waste. This assumption may not apply if any of the ER wastes are derived from treating listed RCRA wastes or mixed with listed wastes, or are found to contain listed RCRA hazardous wastes. Examples to be checked include incinerator ash solids, scrubber water residues, and solvent recovery sludges.

Assumption 2 The aqueous ER HW was assumed to be not characteristically corrosive (i.e., will not have a pH of <2 or >12). This is very likely to be true given the nature of the processes generating the aqueous wastes (oil/water separator or scrubber water).

Assumption 3 The ER HW was assumed to be nonexplosive, nonreactive, and nonignitable. This assumption is reasonable on the basis of the chemical profiles observed so far and the available knowledge about the typical characteristics of remedial wastes involving soil, water, debris, and process solids.

Assumption 4 All stabilized wastes (monoliths) and unstabilized soils were

assumed to pass the toxicity characteristic leaching procedure (TCLP) test and to be ready for landfill without further treatment or handling. This assumption may be questionable for some of the wastes that contain relatively high levels of toxic heavy metals. Examples are monolithic solids of soils and other stabilized wastes, such as those at the RFP and SRS installations. These wastes are shipped for disposal as bulk monoliths at RCRA-permitted landfills after TCLP testing.

Assumption 5 Wastes containing <50 mg/kg PCBs were assumed to be exempt from TSCA regulations and may be disposed of as solid wastes, unless other hazardous characteristics or contaminants are present. Waste sludges and liquids containing >50 mg/kg PCBs will be incinerated, and PCB-containing soils and monoliths will be landfilled at off-site commercial facilities with TSCA permits. The treatment of PCB-contaminated waste at one of DOE's on-site incineration facilities is possible if a TSCA permit to allow thermal treatment of PCB-containing waste can be obtained. An exception to this set of assumptions was made for ER waste sludge/solid process residue at the Portsmouth Uranium Enrichment Center (PUEC). This waste is very unusual in that it will probably have high PCB levels (7,500 mg/kg) in combination with very high levels of mercury (10,000 mg/kg) and other toxic heavy metals. It was assumed that it could be treated on-site with solvent extraction to remove PCBs as a special handling procedure and that the PCBs could be extracted from the mercury and metals, allowing each to be treated or recovered separately. This waste was not assigned directly to incineration because incinerators that are permitted to burn PCBs may not be permitted to accept the waste due to its high mercury level. It was also not assigned to metal recovery because mercury recovery furnaces (which could be used to recover the mercury from this waste) may not be able to take the waste contaminated with PCBs.

Assumption 6 All surface-contaminated bulk metal scrap was assumed to be sent to a metal recycler for metal recovery. This assumption is probably valid, although it will depend on the type and amount of contamination present. The RCRA rules governing the management of contaminated debris should be consulted when dealing with this type of waste.

Assumption 7 All contaminated concrete debris was assumed to require stabilization prior to disposal. In fact, depending on the type and degree of contamination, some might require other forms of treatment and some might need no treatment at all before disposal. The RCRA rules governing the management of hazardous debris should be consulted when handling this waste.

With these assumptions and the information available in the database, ER wastes with apparent hazardous RCRA profiles were assigned to one or more of the following treatment and disposal groups:

Stabilization, incineration, or aqueous treatment;
Organics removal and recovery for fuel-blending or recycling;
Metals removal and recovery for recycling; and/or
Direct disposal at a RCRA-permitted landfill.

Wastes with clearly nonhazardous RCRA profiles (such as asbestos debris or soil with an elevated aluminum content and no other contaminants) were allocated to direct disposal at solid waste landfills.

On the basis of the assumptions provided and the preliminary data available, the assigned treatment and disposal groups for each ER hazardous waste stream appear to be reasonable choices for the large majority of the waste streams evaluated. As additional information becomes available, some redirection of wastes to other forms of treatment prior to disposal should be expected. This might occur through further site characterization, where necessary, before site restoration was initiated. Further refinement of the most appropriate treatment and disposal options could occur at the time of generation of the ER HW. This might be done with testing to confirm identification and characterization of the waste streams. With this testing, a more clear determination could be made of the appropriateness of the assigned treatment group(s) to ensure compliance with federal and state regulations.

The quantities of ER HW that are expected to require treatment and land disposal, or direct landfill, are summarized in Table II (the values shown for direct landfill do not include treatment residues that would also require landfill, although these residues would ultimately have to be considered). These ER waste management requirements are compared to recent estimates of HW routinely generated and managed (in the WM Program) by DOE in the course of normal facility operations. The data show that DOE is currently treating about 1.2 106 t/yr (1.2 106 m3/yr) of

hazardous wastewater in on-site aqueous treatment systems. Virtually no aqueous hazardous wastes are sent off-site. The DOE also provides a few on-site facilities for deactivation of explosive and other ordnance wastes it generates, as well as some incineration capacity. For the most part, however, DOE sends its nonaqueous hazardous wastes off-site for treatment and disposal. Currently, there are no permitted on-site RCRA or TSCA landfills, or landfills with the required capacity, available within the DOE complex for disposal of toxic or hazardous wastes.
Analysis of the data in Table II reveals that the distribution of ER HW among the various treatment or disposal options is quite different from the current HW pattern. For example, only very small amounts (about 1 t/yr) of HW from the ER Program will need to be incinerated, deactivated, or sent to aqueous treatment systems. However, about 4,000 t/yr will be appropriate for metal or organic recovery operations. These treatment systems, including some removal and recovery operations, are currently operational at certain DOE installations and/or are being considered for expansion at designated DOE treatment centers under the evaluation of WM HW alternatives for the PEIS. By far, the bulk of the ER HW (almost 80,000 t/yr) will require stabilization and/or direct landfill. Stabilization and land disposal options at DOE installations are not currently being considered in the PEIS for the HW alternatives. Without this type of on-site treatment and disposal capacity, nearly all of the ER wastes would need to be directed to commercial facilities for ultimate disposal. This practice would result in over 113,000 truckloads of waste requiring transportation to off-site commercially permitted facilities over the next 30 years. Most of this waste would originate at INEL and ORR.
The data in Table II show little or no evident impact from the ER-generated waste in 1991 and 1992 on the off-site treatment and disposal requirements under the WM Program. If the ER Program had been in full operation in 1992 (at the generation rate indicated in Fig. 1), the values shown for off-site WM treatment and disposal would be much larger, approaching the projected off-site values shown for ER waste treatment and disposal. The amount of ER wastes generated in 1992 is likely to have been between 2,000 and 3,000 t, certainly no more than 6,000 t, which is less than 10% of the potential ER waste loads projected for future treatment and disposal. As the ER Program is more fully implemented, the average annual quantity of ER wastes projected to be generated and require treatment or direct disposal will increase dramatically, approaching 84,000 t/yr (56,000 m3/yr or 3,772 truckloads per year) at steady state.

THE ER PROGRAM: IMPLICATIONS FOR HAZARDOUS WASTE MANAGEMENT

Three key points can be made to distinguish the differences between the HW generated by the ER Program compared with that generated by the WM Program, as it relates to management requirements for waste treatment and disposal. First, on a relative basis and considering the total waste generated, nearly all of the WM-generated HW is wastewater. On the other hand, ER-generated HW is predominantly in solid form. Second, due in part to the waste form, the management of ER wastes would involve much more direct land disposal than the management of WM wastes. Finally, much of the ER wastes might not be hazardous or toxic and, if necessary, could be stabilized and disposed of on-site. These distinctions that is, WM HW being dominated by treatment requirements (e.g., wastewaters) and ER HW being dominated by disposal requirements lead to the conclusion that alternatives for waste management of one group of waste may not be appropriate for the other. Because the current HW alternatives are treatment-oriented, they apply more to the management of WM HW than to the management of ER HW.
The basis for these findings can be clarified by examining the annual waste treatment and disposal projections for all waste types under both WM and ER Programs. This will provide an overall view of the magnitude of the need for treatment systems compared with disposal systems. Figure 1 helps provide this perspective. Focusing on the WM waste treatment and disposal requirements (Fig. 1a), two important observations can be made. First, treatment technology is the dominant practice in the management of the WM wastes. At least 95% of the WM wastes require treatment. Second, LLW and hazardous wastewaters are the predominant waste types that must be managed, accounting for approximately 94% of all wastes in the WM Program. It is important to note that the residues (sludges) from wastewater treatment requiring land disposal would amount to less than 0.1% of the treated wastewater volume. These data clearly indicate that the current structure of the WM Program is not appropriate or easily adaptable to the disposal (landfill) of large

volumes of waste typical of the ER Program. Switching our focus to the overall management of all ER wastes (Fig. 1b), it is clear that treatment and disposal requirements are split 70/30 between ER wastes needing the application of treatment technology and ER wastes that can be disposed of directly in landfills. Nearly 900,000 m3 of ER wastes will require land disposal, somewhere, every year for the next 30 years. This waste volume is approximately 56% LLW, 37% LLMW, and 6.9% HW. These landfill waste estimates probably include most of the residues from treatment of the ER wastes, which will also need to be disposed of in landfills. Finally, the available data for the ER HW indicate that because of the very low contamination levels present and the stable form of the waste (e.g., monolithic solids), much of the ER HW may in fact not be hazardous or toxic as defined under RCRA and TSCA and therefore may require little or no treatment prior to disposal.

Even if these compelling data are not considered in decisions involving how the ER Program will be fully implemented and how the ER waste loads will be folded into the WM Program, the ramifications of such decisions should be carefully considered. For example, looking at HW management alone, if DOE continues its current practice of shipping most of its nonaqueous HW off-site for treatment and disposal, the result would be a sharp rise in the combined annual off-site shipments of ER HW and WM HW from current levels of about 10,000 t/yr (6,700 m3/yr) to about 90,000 t/yr (60,000 m3/yr). When this ER HW generation rate reaches steady state, DOE will be shipping wastes off-site at nearly nine times the current shipping rate for a period of about 30 years. Nine of every ten truckloads leaving DOE sites will be transported to commercial landfills where the wastes will be directly landfilled or stabilized first and then landfilled. Only one in ten will be directed to a commercial treatment facility.

Looking from another angle at the ramification of a waste management practice that relies almost entirely on off-site disposal and incorporating the projected LLMW disposal loads with the HW disposal loads, over the next 30 years the amount of WM and ER HW trucked off-site to commercial landfills would fill 240 football fields one yard deep from "goalpost to goalpost." It would take 4,000 truckloads per year traveling millions of highway miles over the next 30 years just to move all the HW off-site for stabilization and disposal. It would take another 68,000 truckloads per year to transport the other ER wastes off-site for disposal. That is 72,000 truckloads per year filling over 4,000 football fields one yard deep from "goalpost to goalpost" with HW and LLMW.

In view of this prospect, on-site options to stabilize and dispose of these wastes at DOE facilities over the next 30 years needs to be carefully examined. The great majority of the projected ER HW (and LLMW) that is designated for stabilization and disposal is relatively low-hazard waste, much of which might actually be nonhazardous. Two DOE facilities in particular, INEL and ORR, are expected to generate almost 80% of the ER HW that will require stabilization and landfilling. Managing these wastes on-site at INEL and ORR might be more sensible than sending them off-site. Certainly, on-site stabilization and disposal could eliminate the hazards and risks associated with highway or railway transportation and would avoid generating most of the air pollution associated with such transportation.

Alternative management scenarios are needed to provide options for developing on-site stabilization and disposal facilities at key DOE installations. Such scenarios are especially important in planning for the ER Program wastes. In particular, the feasibility, costs, risks, and public acceptability of creating on-site stabilization and landfill facilities for the ER HW at ORR and HS and for the LLMW/LLW at ORR and INEL should be carefully examined and weighed against the alternatives currently under consideration.

CONCLUDING REMARKS

On the basis of these considerations, careful attention should be given to the option for on-site treatment and disposal over the next 30 years of the bulk of DOE's nonradioactive ER HW. The preliminary projections of the mostly high-volume, relatively low-hazard ER wastes suggest that the feasibility of installing on-site stabilization and landfill facilities with sufficient capacity for managing the ER wastes and other hazardous wastes should be carefully considered and planned for prior to the initiation of full-scale restoration actions. Such planning is especially important for installations like INEL, HS, ORR, which within the next two to three decades are projected to generate a large portion of the ER wastes that would be targeted for direct landfill. Certainly, in-situ stabilization and

landfilling could eliminate the potential hazard and risk associated with waste excavation, handling, loading and unloading, and over-the-road transportation. The feasibility, costs, risks, and public acceptability of creating on-site stabilization and landfill facilities for managing nonradioactive ER wastes at ORR, HS, and INEL should be carefully examined and weighed against the costs, risks, and public acceptability associated with the off-site transportation, treatment, and disposal alternatives. Depending on how the ER Program is fully developed and implemented, considerable potential exists for significant reductions in costs, safety hazards, and health and environmental risks.

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PROTOCOLS FOR IN SITU FIELD SCREENING AT THE COMPLETION OF DECONTAMINATION OR REMEDIATION

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ABSTRACT

Current practice for assessing the effectiveness of surface decontamination or soil remediation is to obtain an in situ beta/gamma or alpha measurement for residual activity assumed to be on the surface, or to obtain a sample for radiochemical analysis for residual assumed to be distributed in the medium of interest. However, as decommissioning regulatory standards become increasingly more restrictive, current practice may not provide sufficient assurances, during the conduct of the work, that the decommissioning standards have been met and the decontamination and decommissioning (D&D) workforce can be demobilized. The alternative, laboratory sample analysis with longer count times, is not sufficient or desirable for operational surveys for the following reasons:

1. Laboratory sample analysis becomes prohibitively expensive, if expanded to provide "close" support to D&D activities.
2. Laboratory sample analysis is not appropriate for verifying the decontamination effectiveness on surface intended for unrestricted release for recycle or reuse.

This work is intended to provide an in situ technique that can be used for screening surveys of residual surface activity and to supplement limited sampling for residual volumetric activity. This protocol is based on measuring all pertinent particle emission rates including alpha, beta, and photon. By observing surface flux of the various particle emission at different distances from the surfaces, applying basic principles of beta and photon shielding/attenuation, and performing isotopic analysis to identify the isotopes present, the following information could be obtained:

1. Extent of residual activity (size of source)
2. Uniformity of source distribution (hot spots)
3. Source geometry (surface vs. volume distribution)
4. Isotopic ratio (limited to 2 or 3 different isotopes)

5. Estimate of surface or volumetric residual activity level.
This protocol, when coupled to proper instrument selection and use, should provide a cost effective alternative to extensive laboratory sample analysis for assessing decontamination or remediation efforts while work is still in progress.

INTRODUCTION

Radiological measurement requirements of this protocol for decontamination and decommissioning (D&D) or remediation in general, are very much the same as the requirements of the operational activities handling similar isotopes. The differences between operational radiation measurements and D&D or remediation radiation measurements, that the health physics professional must be sensitive to, are the degree to which a particular requirement must be addressed. This sensitivity becomes progressively more acute during the course of the D&D or remediation activities with greatest emphasis on the requirements for completing and documenting the final survey accurately and cost effectively. This paper is directed more toward the issues involving in situ measurement rather and correlation of different types of measurements to one another to find the extent of residual activity (size of source), uniformity of source distribution and determination of hot spots, source geometry (surface vs. volume distribution), isotopic ratio (limited to few isotopes), and surface or volumetric residual activity level. Sample acquisition and analysis and correlation to these measurements will not be discussed in this paper. Sample analysis is an area deserving a separate presentation.

PERFORMANCE CHARACTERISTICS

During the course of decommissioning, a number of factors impact the interpretation of measurement results from a source of residual radioactivity present in contaminated surfaces or volumes. Factors that are significant to proper understanding of the measurement results include:

1. Source-detector geometry
2. Type of radiation and its energy range
3. Survey protocols
4. Relationship between calibration or source check and in situ measurement results

Of the four factors listed above, the first three are of the main concern of this paper. The last factor is considered to be of a universal consideration and concern under any protocol and should follow the proper procedures according to the accepted standards.

The first factor includes the geometry and self-attenuation of the in situ source and attenuation characteristics of intervening materials between the source and detector. Source geometry and self attenuation are significant for gamma, beta, and alpha emissions. In most in situ measurement situations, intervening materials (e.g., dust, air, surface coatings, etc.) are more significant for beta and alpha measurement interpretation than for gamma measurements. The effects of source geometry, source composition, and proximity to a point of interest have been developed and applied to field measurements from the beginning. In the 1950's, relationships for source geometry and attenuation for gamma emissions were developed initially for reactor design (1), and relationships for beta emissions were developed for medical applications (2).

The second factor is significant because most calibrations and/or source checks data are obtained from a single source calibration or check source, hopefully, somewhat representative of the conditions anticipated in situ. However, many decommissioning and remediation projects involve more than a single radionuclide, requiring the health physics professional to pay detailed attention to the results of in situ and/or calibration measurements as a function of site specific isotope energy ranges.

The third factor relates to the manner in which a particular survey is performed to detect and evaluate the intensity and geometry of a particular source. Specifically, the factors of significance implied here are source-detector distances, expected source geometry, etc.

GAMMA SOURCES

The relationships for gamma attenuation for unshielded (air only) sources, or sources external to interspersed shielding, are generally described by one of the following conditions:

  point source - attenuation 1/r2
  line source - attenuation 1/r
  planar source - attenuation = 1.0 for r < ~5x

attenuation 1/r for ~5x < r < ~10x
attenuation 1/r2 for ~10x << r
where:
r - distance from source
x - primary dimension of source

When shielding or self attenuation is involved the following formula could be used:

$$Ji(r,Ei) = J(0,Ei).B.\exp\{-ml(Ei).r\}$$

Eq. (1)

Ji(r,Ei) = Attenuated photons of Ei at a distance r
J(0,Ei) = Photons of Ei at a distance 0 (source)
ml(Ei) = Photon linear attenuation coefficient
r = Distance from source to detector
Ei = Photon energy
B = Material build up factor

From these relationships, it would appear feasible to obtain an indication of an unknown source geometry by taking measurements at various distances from the source. This approach would be most useful where a spill or leakage has occurred and the extent of the activity can be estimated, where accessibility may be a problem (e.g., subsurface soil or behind barriers). The other factor that could impact such an approach is the effects of photon energy on the relationship. To study these effects, a series of shielding models were devised and executed using a computer shielding code (MICROSHIELD 4.0, Ms 92) to compute the photon flux and exposure rates at various distances from the source. Three source geometries were selected for the purpose of these models: point, disk, and cylindrical. For each geometry, three different exposure locations (1, 33,66,100 cm) were chosen from source surface. For the point source geometry, the source was shielded by three different shield thicknesses (5, 15, 30 cm) for each exposure location. The shield material was concrete with a reduced density of 1.7 g/cm3 to simulate soil. For the disk source geometry, the seven different radiuses (5, 20, 50, 100, 200, 500, and 1000 cm) were used for each exposure location. For the cylindrical geometry, both source thickness and source radius were varied for all three exposure locations. Three different source thicknesses (5,15, 30 cm with same density as used for point source geometry shield) and seven different radiuses (5, 20, 50, 100, 200, 500, and 1000 cm) were selected. The source in each geometry case consisted of six photons with energies ranging from .1 to 3 MeV (.1, .2, .5, 1, 2, 3 MeV) and each with a strength of 1 photon/s, 1 photon/cm2, and 1 photon/cm3 for the point, disk, and cylindrical geometry, respectively.

Table I shows the effects of source size, distance between source and detector, and shield on particle exposure attenuation for the above described source geometries for a 1 MeV photon. It was noted from results of the photon fluxes in air that photon energy is a small effect (<1%) over an energy range from ~100 keV to ~3 MeV for planar source (disk). For the point source, photon energy has a more pronounced effect on the photon flux at the exposure locations as the distance from source increases. For volumetric source, this effect depends on the source radius and thickness, and varies from 10-50%. From these observations, extension to a field situation for measurements at ~0.6 m and ~1.0 m from the surface of the source would enable an estimate to be made of the source geometry. The only unknown remaining is whether the source is planar or a volume. This can be determined by coupling the photon measurements to a series of beta or alpha measurements, particles which are normally present in addition to the photons (gamma). This approach will be expanded after discussion of beta source attenuation effects.

TABLE I

BETA SOURCES

The following discussion is based upon the Loevingers equations for beta dosimetry (4). The basis for the beta dose distributions from a particular source is the point source function:

$$J(x) = ke\text{-}vx/(vx)2$$

Eq. (2)

J(x) is the dose in rads per disintegration at a distance x from a point source of beta particles, and v is the beta absorption coefficient in the medium surrounding the source, and k is the constant of proportionality that is a function of the beta particle energy. The beta dose function is extended to provide solutions to dose

distribution relationships for area and volume sources.
The solution to the beta dose distribution function is dependent upon resolution of parameters that vary with the beta particle energy. Two parameters that must be determined prior to the beta dose solution are the average (effective) beta energy for a beta spectrum, and the beta attenuation coefficient.
Average Beta Particle Energy and Beta Attenuation Coefficient
The radioactive decay by beta emission is characterized by a continuous energy spectrum of beta particles with a discrete endpoint energy for each specific isotope. Since the absorbed dose in a medium is directly proportional to the energy of the beta particle undergoing the energy loss transformation, the absorbed dose distribution for a spectrum would involve integration of the distribution function over the energy range of the spectrum. It is, however, practical to characterize the spectrum by a single average, or effective, energy that results in the same absorbed dose distribution. In general, the average beta spectrum energy (Eb) is approximately one-third of the spectrum endpoint energy (Eo). More specifically, the average beta energy is a function of both the atomic number (Z) of the initial nucleus, and the spectrum endpoint energy. This relationship is explained in detail in Hi 58. For an isotope of interest, the beta endpoint energy (5,6) along with this information could be used to determine the average spectrum energy.
Empirical relationships have been derived to express the beta energy response for the attenuation coefficient (v) in various media. Loevinger (2) gives the formula
Eq. (3)
where E/E* is a measure of distortion of the beta energy spectrum (beta particle emission tangential to nucleus) and the "allowed" spectrum (beta particle emission radially to nucleus). Values for the constants a1 and a2 in various absorbing media are given in Table II.
TABLE II
Beta Dose Rates in Air
The point source beta rate function given by equation (2) is expanded and solved for attenuation in air for a point source; a zero thickness; infinite planar source; and an infinitely thick, infinite area source.
Point Source
The point source beta dose rate function for a distance x in air is given by the expression.
Eq. (4)
where:
J(x) - Beta dose rate at distance x (Rad/disintegration)
x - Distance from source to dose point (gm/cm2)
v - Attenuation coefficient in air (cm2/gm)
k - (1.28 x 10-9) p2 v3 Eb a
p - 0.001293 gm/cm3
Equation (4) may be rearranged with unit conversions to produce an expression of the form:
Eq. (5)
where:
Db (Eo, d) - Beta dose rate at distance (d) for beta energy (Eo) - Rad/Hr/Curie
d - distance from source to dose point - cm (=x/p)
The generalized expression for beta dose rate as a function of beta endpoint energy is plotted and presented in Ref. 7.
Infinite planar source of zero thickness
The generalized beta dose rate function at a distance x in air for a zero thickness source of infinite planar area is given by the expression:
The function D(Eo,d)/Ebs is in units of Rad/hr/mCi/cm2/MeV
where:
s - Surface activity in mCi/cm2.
The generalized infinite plans source function is plotted and presented in Ref. 7.
Infinite planar source of infinite equivalent thickness
The generalized beta dose rate function at a distance x in air from an infinite area, infinitely thick source is
This generalized function is in units of Rad/hr/mCi/cm3/MeV
where
t - Volume Source in mCi/cm3
This beta dose function is descriptive, for example, of the dose distribution above

water pools more than several centimeters thick. The volume source relationship as a function of beta endpoint energy is presented in Fig. 1.
Fig. 1.
FIELD APPLICATION PROTOCOL: CONCLUSION
The following protocol is offered as an example of the application of combining the gamma and the beta source geometry relationships to estimate an in situ source geometry. The sample to follow is for a situation involving a source of contaminants in soil with the strength of 1 gamma (E = 1 MeV) and 1 beta (E0 = 1 Mev) emission per second.
1. Scan areas, using a NaI detector and ratemeter to find location(s) of elevated readings.
2. At highest reading point, take the following measurements:
  a) Beta- contact, 0-1 cm [Go(b)]
        - 15 cm from surface [G6]
        - 30 cm from surface [G12]
b) Gamma- contact, 0-1 cm [Go(g)]
        - 50 cm from surface [G50]
        - 100 cm from surface [G100]
3. Form the ratio G50/G100 from the gamma readings and use Fig. 2 to estimate an approximate source radius. This is useful for comparing the measurements to the hot spot criteria.
4. Form the ratio G15/G30 for the beta readings and, using Fig. 1 and gamma flux/dose information in Table I (also Fig. 2) to estimate the contaminant depth in the soil. This is achieved by comparing flux/dose behavior of the beta and gamma readings and model prediction at various locations above the contaminated surface and correlating these results to the ratio of the beta reading to gamma readings and corresponding model predictions at the same location.
5. Determine the response function (conversion factor) for the gamma detector as a function of effective gamma energy. This becomes important for interpretation of measurement results and correlation of these results to shielding model results.
6. Combine these functions with measurement results in Step 2 and Step 4 to estimate the total dose rate for the measurement and corresponding source distributions. Note that this approach is based on the assumption that the activity is uniformly distributed.
Fig. 2.
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57-8
A REVIEW OF SUPERFUND RECORDS OF DECISION FOR RADIOLOGICALLY CONTAMINATED SOIL SITES
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ABSTRACT
Thirty-nine Records of Decision written on radiologically contaminated Superfund sites were reviewed in the process of developing a presumptive remedy for soils contaminated with radionuclides. Twenty-five of the RODs were for actions taken to

remediate contaminated soil and/or sediments.

INTRODUCTION

The purpose of this study was to review technologies used at other radiological contaminated sites in the United States as described in Superfund Records of Decision. Records of decision signed for sites contaminated with radionuclides during the period of 1985 to 1993 were analyzed in-depth. This review in turn was intended to support the development of a presumptive remedy for soils contaminated with radionuclides. The review also examined the effectiveness of decisions made for radiologically contaminated soils.

Objectives of the study were:

to determine the use of and importance placed on risk assessment in deciding the remedies for radiologically contaminated sites,

to determine the drivers used in past decisions, and

to determine the technologies which have been used successfully to remediate radiologically contaminated sites.

A total of fifty one radioactively contaminated Superfund sites were identified. Thirty-nine CERCLA RODs published for those contaminated sites between 1983 to 1993 were reviewed for this report. Two additional RODs were written and approved during that time period, but were unavailable for review (Denver Radium Superfund Site Streets and Weldon Springs Chemical and Raffinate Plant). The thirty-nine RODs and one declaration reviewed, were tabulated for media type, the primary action selected, and for any additional actions listed. Of these RODs

25 addressed radiologically contaminated soil and/or sediments (such as pond),

7 primarily addressed surface and/or ground water issues only,

1 of which included other contaminated materials other than soil or water, such as buildings, construction material or debris, sludge, bulk packaged waste, etc.;

4 addressed other contaminated materials only, such as buildings, construction material or debris, sludge, bulk packaged waste, etc., and

3 RODs addressed non-radiogenic contamination only, such as heavy metal, or VOC contamination.

The majority of radioactively contaminated sites for which RODs were available had either radium (and daughter products) or thorium (and daughter products) as the primary contaminants of concern. The most common radionuclides listed as contaminants of concern were radium and members of its decay series, including radon gas, then thorium-228/230/232, uranium-234/235/238, cesium-137, plutonium-238/239/240, americium-241, strontium-90, cobalt-60, actinium-227, protactinium-231, and tritium (listed by high to low number of occurrences)(Fig. 1). The primary and additional remedial actions selected, and the media contaminated for the 25 RODs that primarily addressed radiologically contaminated soil and/or sediments are summarized in Table I. These RODs also addressed other contaminated media or contamination as follows:

3 included surface and/or ground water issues,

14 included other contaminated materials other than water, such as buildings, construction material or debris, sludge, bulk packaged waste, etc., and

11 were also contaminated with heavy metals and/or VOCs.

The primary actions selected for the radioactive contamination sites are as follows:

75% of the RODS (18) selected excavation and disposal for the primary treatment.

one of these RODs addressed contamination at multiple sites and listed two primary actions dependent upon the level of contamination, some sites were to be excavated with permanent off-site disposal, and some sites determined to have low level contamination with risks within EPA's "acceptable range of risk" were listed as no action;

20% of the sites (5) selected containment and/or restricted access.

contamination at two sites was left in place with improved administrative controls and restricted access, and

the selected action for three sites was containment (leaving contamination in place with capping and/or institutional controls).

the remaining 5% (2 sites) selected physical separation and/or chemical extraction for the preferred treatment.

Both sites are located on the Idaho National Engineering Laboratory; one is in the process of implementation, and at the other site, Post-ROD treatability studies did not show that the treatment option would meet disposal criteria and a consolidation and containment option was substituted.

Twenty-one of the RODs provided for a total of 23 additional actions to be taken to either mitigate immediate risks or to remediate other contaminated media (Table I). The alternatives considered and the non-compliance criterion/criteria were tabulated. Five of the documents also provided for continued monitoring and maintenance of site contamination. Of the eighteen sites with planned excavation and/or removal of contamination, temporary onsite storage was provided for at seven sites. Additional treatment actions were selected at 4 sites; the treatments were described in two of the documents as providing for shielding of residents and reduction of radon emissions, and the other two documents provided for stabilization of the contaminated soil prior to disposal. Other secondary actions were relocation of residents, reconstruction of residences, pump and treat water, and decontamination of other media such as contaminated roofs and equipment.
In the detailed analysis of alternatives, nine criteria are used to rate or compare the efficacy of each alternative. The number of times each class of criteria (threshold, primary evaluation, or post-public comment) were the basis for rejecting an action are tabulated in Table II. Of the 82 alternatives described, 43 were rejected because of non-compliance with the threshold criteria (protection of human health and environment or ARARs), 29 non-compliance with the primary evaluation criteria, and 7 because of post-public comment criteria. The 29 instances that alternatives were eliminated for non-compliance with the primary evaluation criteria were distributed mostly among the first three of the five (5) criteria (long-term effectiveness & permanence, reduction of toxicity, mobility or volume, and short-tern effectiveness). Three alternatives were rejected because of unproved reliability or because off-site disposal sites were not available for either the type or volume of waste.
In two of the RODs, the selected action did not comply with the ARARs, but waivers were granted. Two other RODs also documented non-compliance with primary evaluation criteria (in both cases it was the short-term effectiveness criteria), and additional actions were taken to circumvent any problems or mitigate the risks the selected action posed. The availability of off-site disposal was recognized to be a problem at seven (7) sites, and the additional action was the provision of temporary on-site disposal.
In the 25 RODs addressing radiologically contaminated soils, risk assessments were not presented in detail but results from risk evaluations or assessments from other documents (such as the remedial investigation) were summarized. In 4 of the documents, no risk values were given, and it was concluded that there was no risk or that the risks were insignificant without any evidence offered. Baseline cancer risk levels (current risk no-action conditions) were identified for either current occupational or current residential cancer risks for the other 21 RODs reviewed. Additional future cancer risk values were reported in 12 of these RODs.
The risks reported in the RODs represent individual excess cancer risks associated with lifetime chronic exposure to carcinogenic contaminants present at the sites. Theoretically, a 1E-6 (10-6) risk indicates that if an individual were exposed every day throughout life to the concentrations associated with the risk level, his or her individual probability of developing cancer resulting from that exposure is unlikely to exceed one in one million.
Exposure point concentrations were generally estimated conservatively for all pathways assessed. Only two (Maxey Flats, and Monticello) sites reported that population studies had been conducted. Children were the only sensitive sub-population evaluated in any of the risk assessments.
Few of these sites involved actual current exposure of workers or residents that were working or living on-site. Due to the time frame (prior to the publication of applicable guidance) when many of these RODs were written there was little consistency in what risks were reported or how risks were calculated. Potential current exposure scenarios used were the same or similar to the EPA standard default assumptions for both occupational and residential scenarios in most of the RODs. Potential current exposure to an intruder was assumed at one site (BOMARC). Assumptions made were in some cases not adequately described and the determination could not be made as to the appropriateness of the assumptions. Current risk was often based on exposure pathways, or scenarios, which may or may not exist.
Current cancer risks for 15 of the 21 (71%) reporting sites were greater than 1E-04 (Fig. 2). Six of the 21 (29%) of the RODs reported current risk values within the NCR acceptable range of risk or less (1E-06 to 1E-04). Current risks for two of the

sites were less than 1E-06. Additionally, for five of the RODs there were hazard indices with values greater than 1, indicating possible non-carcinogenic health hazards to current occupational receptors at four of the sites, to current residential receptors at two sites, and via other scenarios at two sites.
The difference between current and future risks involved a future change in site use that would involve more frequent exposure, additional or new receptors or additional pathways of exposure. Residential development was the most often cited future-use scenario. Future risks were based on current concentrations of radionuclides in surface soils allowing for 30 to 100 years of decay for the majority of sites.
Future residential risks were discussed quantitatively in only 12 of the RODS (Fig. 3). Nine (75%) indicated future potential cancer risk greater than 1E-04 and three (25%) were within the NCRs acceptable range of risk (1E-06 to 1E-04). Future risk values for two sites were within the acceptable range. Hazard indices of greater than 1 also were calculated for 4 of the 12 sites. All of the future risks were based on hypothetical exposure scenarios.
Of the six RODs reporting current risk values within the NCR acceptable range of risk, two also reported future risk values within the acceptable range. This indicated that risk was not one of the criteria used in deciding the remedial actions described in these RODs.
There was a lack of assessment of the degree of risk reduction associated with remedial alternatives. This constitutes a major weakness in the decision-making process. Of the RODs reviewed, risk reduction was evaluated quantitatively for only two sites and was mentioned qualitatively in five other RODs. Risk reduction played a limited role in the evaluation of the remedial alternatives. The degree of risk reduction associated with the alternatives evaluated were not well-defined in the majority of decisions reviewed, and the cost-effectiveness of the remedies selected could not be determined. However, most RODs concluded in some way that the selected alternative was cost effective.
From the detailed analysis of alternatives, what appeared to be the most significant factors in the selection of remedial alternatives was the compliance with the Threshold Criteria (protection of human health and environment and ARARs compliance) and technical feasibility. However, all RODs planned for remedial actions to taken even for the sites with risks lower than 1E-06.
Twenty seven (27%) percent of the RODs selected the least costly alternative other than no-action, and thirty-two (32%) percent selected the most costly remedy. The geometric mean for cost of remediation per ROD was 7.4 million dollars. The cost of the selected actions for most (14 of 21) of the sites was less than 7 million dollar each as shown on Fig. 4. In 73 percent (16 of 22) of the RODs, the average additional cost of the more costly alternative over the selection action was $31 million additional.
Ecological risk was not well developed in any of the RODs reviewed. Again in most cases there were simple summaries of other documents stating no risk, or risk not unacceptable. In five cases, there was no presentation of ecological or environmental risk. In one ROD, no ecological risks were evaluated because the assumption was made that if human health was protected, there would be no environmental or ecological risk. Cleanup levels that assure protection of public health may not always assure protection of surrounding ecosystems, and the lack of extensive assessment of ecological risks, which is sometimes warranted, is another major weakness in determining appropriate remedial alternatives.
Of the 25 documents reviewed, only one discussed in detail the uncertainties of the risk assessment and the assumptions made. Most of the documents mentioned that the evaluation of risk was uncertain or did not mention uncertainty. The one document that developed the discussion of uncertainty well, did not draw any conclusions. Therefore risk and uncertainty could not have played a determining role in the decision process.
Most (16) selected excavation with off-site disposal as their preferred alternative. With the problems in finding a facility for permanent disposal, many sites have had to store excavated materials on-site. Permanence of the selected alternative but actual is far from permanent.

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GAS CYLINDER DISPOSAL PIT REMEDIATION WASTE MINIMIZATION AND MANAGEMENT
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ABSTRACT
A remediation of a gas cylinder disposal pit at Sandia National Laboratories, New Mexico has recently been completed. The cleanup prevented possible spontaneous releases of hazardous gases from corroded cylinders that may have affected nearby active test areas at Sandia's Technical Area III.
Special waste management, safety, and quality plans were developed and strictly implemented for this project. The project was conceived from a waste management perspective, and waste minimization and management were built into the planning and implementation phases. The site layout was planned to accommodate light and heavy equipment, storage of large quantities of suspect soil, and special areas to stage and treat gases and reactive chemicals removed from the pit, as well as radiation protection areas.
Excavation was a tightly controlled activity using experienced gas cylinder and reactive chemical specialists. Hazardous operations were conducted at night under lights, to allow nearby daytime operations to function unhindered. The quality assurance plan provided specific control of, and documentation for, critical decisions, as well as the record of daily operations. Both hand and heavy equipment excavation techniques were utilized. Hand excavation techniques were utilized. Hand excavation techniques allowed sealed glass containers to be exhumed unharmed.
In the end, several dozen thermal batteries; 5 pounds (2.3 kg) of lithium metal; 6.6 pounds (3.0 kg) of rubidium metal; several kilograms of unknown chemicals; 140 cubic yards (107 cubic meters) of thorium-contaminated soil; 270 cubic yards (205 cubic meters) of chromium-contaminated soil; and 450 gas cylinders, including 97 intact cylinders containing inert, flammable, toxic, corrosive, or oxidizing gases were removed and effectively managed to minimize waste.
LIST OF ACRONYMS
INTRODUCTION
Beginning in 1984, several investigations and assessments, including a Resource Conservation and Recovery Act (RCRA) Facility Assessment (RFA), were conducted at Sandia National Laboratories, New Mexico (SNL/NM) as part of the Department of Energy (DOE) Environmental Restoration (ER) Project. These investigations revealed 172 potential release sites which required investigation and possible corrective action.
One site, known as the Gas Cylinder Disposal Pit (GCDP), was located in the southeast corner of the environmental test range known as Technical Area (TA) III (Fig. 1). The dimensions of the excavated pit area were approximately 80 by 180 feet (24 by 55 meters) by 10 feet (3 meters) deep. The pit-bottom configuration and depth suggested that fill material was likely to be less than 6 feet (1.8 meter) thick, and this indicated the volume of the affected soil was no more than 3,200 cubic yards (2,443 cubic meters).
Fig. 1.
The pit was reportedly dug in 1963, but prior to its excavation, the area in which the site was located may also have been used to dispose of high-explosive (HE) residues by detonation. Following its excavation, the pit was used for the disposal of lithium hydride and other reactive chemicals from various SNL/NM laboratories, and from 1980 to 1984, the pit was used for the disposal of 400 lecture-size [2 by 12 inches (5 by 38 centimeters)] gas cylinders by detonation with shaped-explosive

charges. The cylinders reportedly contained a variety of gases, including some toxic materials and unknown materials which could not be shipped off site. This site also had been used occasionally for temporary storage of potentially explosive chemicals, including picric acid. Visual surveys revealed glass shards, bottle remains, and gas cylinder remains. Table I lists chemicals thought to have been disposed in the pit. Based on the survey of historic activities, no radioactivity was suspected.

TABLE I

The GCDP site was originally scheduled to be investigated during the course of a RCRA Facility Investigation (RFI) by drilling shallow boreholes into areas likely to have aggregations of gas bottles or chemicals. Because of health and safety concerns during drilling, this type of intrusive characterization was not considered to be the optimal course of action. In addition, hazardous wastes were thought to be present in unstable storage containers that could pose a threat of spontaneous release, resulting in possible explosion and/or potential for exposure to hazardous waste as a result of a container failure.

Two non-intrusive investigations, a walk-over health and safety survey and a surface geophysics survey of the pit, were undertaken to provide more information for the appropriate course of action. The initial walk-over survey was conducted in Level B personal protective equipment (PPE) and served to identify partial and intact lecture bottles, amber glass containers containing unknown solid and liquid materials, and previously unidentified radioactive material. Drilling in the pit was determined to be unacceptable; therefore, complete exhumation and removal of the contents of the pit in association with an accelerated and voluntary corrective action for site cleanup was planned.

WASTE MANAGEMENT PLAN

SNL/NM is required to practice pollution prevention as part of activities associated with investigation and remediation of sites. Consideration is given to the costs and liabilities associated with management of wastes produced as a result of these activities at the beginning of each planning phase. The remediation of the GCDP was planned from the perspective of waste minimization and waste management, as well as careful attention to aspects of safety, construction, and remediation. The management plan for investigation-derived waste (IDW) was developed with the cooperation of the SNL/NM Generator Interface and Waste Operations Departments, and was formulated to be consistent with existing guidelines of the U. S. Department of Energy (DOE), U. S. Environmental Protection Agency (EPA), and New Mexico Environment Department (NMED).

The excavation operations conformed to specific objectives outlined in the quality assurance project plan prepared for this project. Various field tests and inspections were required during excavation and material segregation. The field inspections typically required a pass/fail decision. These decisions included: 1) identification of pit materials; 2) removal and segregation of clean and suspect soils; 3) removal and segregation of breached and unbreached material containers; and 4) evaluation of cylinder contents. The decision tree for these operations is shown in Fig. 2. Following a surface clearance, excavation proceeded from north to south with iterative magnetic surveys.

Fig. 2.

The majority of waste to be created during pit excavation at the GCDP was believed to be soil and debris, decontamination water, personal protective equipment (PPE), ordinary trash, and gaseous and solid chemicals. Waste to be produced during operations was expected to be contaminated with metals, chemicals, pressurized gases, high explosives (HE) residues, and radioactive soils. Waste was categorized as nonregulated, hazardous, radioactive, and mixed (radioactive with hazardous). The majority of waste by volume was expected to be nonregulated.

Field screening was performed during all soil excavation activities to facilitate segregation. Both real-time instrumentation and field test kits were used to screen for various constituents during the course of the project. Metals in soils were evaluated using X-ray fluorescence (XRF); volatile organic compounds (VOCs) were evaluated using Drager tubes, photoionization (PID) and flame ionization (FID) detectors; HE was evaluated using EXSPRAYTM; and radiological constituents were evaluated using a pancake Geiger-Muller (GM), an alpha scintillometer, and a sodium-iodide (Nal) gamma spectrometer (Table II). Screening for various gamma-emitting radionuclides was also performed using gamma spectroscopy. Action levels for field screening included mean background plus 2 standard deviations for

XRF and radioactivity, and any positive reading for volatile organics and HE. Detection of constituents above the action levels triggered material segregation and resulted in possible additional sampling. Soils and debris were staged in specific engineered locations (Fig. 3) until results of laboratory analyses were available and the waste types were determined.
TABLE II
Fig. 3.
Prior to soil placement in the staging area, a grab sample of up to five aliquots of material was collected randomly from the bucket of the front-end loader. These grab samples were combined to form a single composite sample representing the soil contained in the staging area. Excavated soil was stockpiled onsite in prepared staging areas east of the disposal pit. These areas were bordered by reinforced concrete barriers ("Jersey" barriers) on three sides, an earthen berm on the fourth side, and a lining of very low density polyethylene sheeting. The staging areas were built to accommodate 60 cubic yards of soil each, but could be modified easily to accept more. All soils were covered with fiber-reinforced poly sheeting secured by sandbags for protection against wind and rain. Stained or discolored soils were segregated in separate areas, and sampled to determine possible hazardous or radioactive characteristics. Analysis of the suspect soils included HE (EPA Method 8330), VOCs (EPA Method 8240), SVOCs (EPA Method 8270), TAL metals (EPA Methods 6010 and 7000 series), and isotopic uranium, thorium, and plutonium. Table II also provides a list of the laboratory analytical methods used during this project.
Debris (wood, metal scrap including breached cylinders, thermal batteries, glass, and bulk chemicals) from the pit was staged in small engineered staging areas. Pressurized gas cylinders and intact chemical containers recovered from the excavation were staged separately, individually, and away from all other materials. Before packaging, all debris and scrap removed from the pit was screened for radioactive contamination using a Nal radiation detector, and swiped and counted using a gas proportional counting system.
The section of the pit containing radioactive soil was sectioned into four quadrants approximately 30 by 40 feet (10 by 12 meters). Each quadrant was investigated and excavated separately, in the interest of waste minimization. Every bucket of soil and debris was screened in the field using a Nal radiation detector. Soil having readings greater than background plus 2 standard deviations was considered to be contaminated and was removed. Upon excavation, radioactive soil was segregated and drummed in the pit itself.
All containers and staging areas were marked with appropriate labels. All unknown materials, pending analytical results, were labeled "Investigation-Derived Waste." Materials whose characteristics were known were labeled as nonregulated, hazardous, radioactive, or mixed, as appropriate. Container labels were marked in permanent ink and included waste source, suspected contaminants, results of field screening, contents, the dates accumulation began and ended, associated sample numbers, and the project manager's name and telephone number. Labels were numbered sequentially. All solid waste drums were stored on wooden pallets.
Decontamination water was produced during heavy equipment decontamination. This decontamination was performed at a temporary decontamination pad constructed for the project. Decontamination water was collected in open-topped, lined 55-gallon drums for ease of handling and sampling. Once full, drums were sampled and analyzed for TCLP metals, VOCs, and SVOCs. Personnel decontamination was largely conducted on a dry basis, but contingency for wet decontamination (3-gallon sprayers with soap and water, and 3-gallon sprayers with water only) was available. In addition, a decontamination trailer was installed at the site at the boundary of the controlled area; this facility housed two showers which could be used as a contingency for personnel decontamination. Personnel decontamination water also was collected in open-topped, lined drums, and analyzed for TCLP metals, VOCs, and SVOCs. Liquid waste drums were stored on secondary containment pallets to contain spills or leaks.
All disposable PPE was field screened for radioactive contamination, HE, and organic contamination by PID and FID, and stored as IDW in open-topped, lined, 55-gallon drums upon completion of each day's work. Uncontaminated PPE was segregated from all other wastes. The drums were labeled according to their contents (e.g., "suspect clean as indicated by field screening"). PPE suspected to be radioactive was double-bagged in yellow plastic at least 3 millimeters thick and placed in drums visibly printed with the word "radioactive."

All ordinary trash was stored in plastic household or lawn trash bags upon completion of each day's work. Ordinary trash was considered to be nonregulated and was placed in a commercial dumpster labeled "Nonregulated."

WASTE MINIMIZATION

Federal and state laws, as well as DOE Orders, require that SNL/NM plan, implement, and document the minimization of all types of waste, including such things as water discharges and air emissions. Therefore, waste minimization practices were incorporated into field activities at the GCDP site to reduce the quantity of waste generated.

The following list includes specific methods of waste minimization implemented during the project.

Waste management and decontamination procedures were written in order to document activities, provide preplanning and quality control, reduce time in the field, avoid the use of unnecessary equipment, and avoid the generation of excess waste.

Nonintrusive investigations were completed before intrusive work; no hazardous or radioactive waste was produced during these investigations. These investigations also focused attention and subsequent remediation activities on problem areas which included only 56% of the total pit by surface area.

Daily activity/health and safety "tailgate" meetings included waste minimization and disposal topics, and described specific tasks which site workers were to perform during that work period. Site workers were encouraged to become a part of the planning team and to identify additional waste minimization opportunities.

Dry decontamination procedures were used to avoid creating decontamination fluids for personnel decontamination.

All excavation equipment was kept in the pit during excavation activities, and only when excavation was completed was the equipment removed and decontaminated. Only materials that could easily be surface surveyed were used for PPE.

All PPE and decontamination rags were surveyed for radioactive contamination as part of personnel decontamination. Wherever possible, PPE was reused. Typically Level "A" and "B" suits were sanitized and reused 15 times.

All debris, scrap, and soil were screened for radioactive contamination, and an aggressive program of field screening was employed for HE, organic and inorganic gases, and metals. Continuous air monitoring was performed for gaseous releases in the controlled-atmosphere of the special operations area to which cylinders of gas removed from the pit were taken. If material screened "positive," it was segregated and containerized with like materials.

An onsite laboratory for gas analyses, as well as onsite treatment capability for these gases, provided immediate identification and decision making regarding the management of gases, and eliminated the need to ship unknown samples offsite. Only nonregulated treatment-process wastes were ultimately produced.

Site personnel did not mix waste types or mix soil and decontamination fluids. Daily efforts to prevent cross contamination were practiced. The key to this capability was site setup and preparatory activities which provided a large area for segregated storage.

Waste was stored in the smallest appropriate containers, often 5-gallon plastic pails for scrap metal and bulk chemicals.

Aliquots of soil were taken from every loader bucket and composited for waste classification analyses, and the bulk of the material was stored onsite prior to disposal; much of this material was segregated as a result of field screening. This process was time-consuming, but resulted in the minimum amount of regulated waste being sent offsite for disposal. Uncontaminated soils were returned to the pit.

Investigative geoprobing was used exclusively for subsurface verification sampling. This method removed only the amount of soil necessary to advance the core barrel and eliminated waste cuttings and drilling fluids, and produced little or no dust.

RESULTS

Excavation was a tightly controlled activity using experienced construction crews and gas cylinder specialists. All field crews had additional specialized training in waste management and waste minimization, as well as Radiological Worker training. Remediation proceeded from north to south in the pit, following the slope contour. A trackhoe excavator was used for coarse excavation during nonhazardous phases of removal. The trackhoe was positioned in the pit itself. Soils were placed 1) in the bucket of a front-end loader positioned out of the pit; 2) directly in drums brought

into the pit via the loader; or 3) directly into 40-yard rolloff bins. All soils were passed through a 2-inch mesh screen before removal or containerization. Archaeological-type hand excavation of gas cylinders, reactive chemicals, and thermal batteries was performed by experienced crews using non-sparking shovels after the trackhoe could not continue. Excavation activities were performed in either Level "A" or Level "B" PPE, including work in the trackhoe or backhoe, and during archaeological-type hand excavation. A total of 48 person-hours were spent in Level "A" PPE, 1,728 person-hours in Level "B" PPE, and 612 person-hours in Level "C/D" PPE for support purposes during the excavation phase of the project.
A total of 140 cubic yards (107 cubic meters)of thorium-contaminated soil was drummed directly in the pit and subsequently staged onsite. A total of 270 cubic yards (205 cubic meters) of chromium-contaminated soil (associated with a group of ruptured and intact thermal batteries) was removed from the pit. A total of 1,200 cubic yards (1,096 cubic meters) of suspect soil was removed and staged until waste characterization was made (this soil was to be placed back into the pit).
Several dozen thermal batteries; 5 Ib (2.3 kg) of lithium metal; 6.5 Ib (3.0 kg) of rubidium metal; several kilograms of unknown chemicals; and 450 gas cylinders, including 97 intact cylinders containing inert, flammable, toxic, corrosive, or oxidizing gases were removed and effectively managed to minimize waste. Thermal batteries were disposed of as hazardous waste. Lithium metal was repackaged and disposed. Rubidium metal was given to the SNL/NM "Metals Reapplication Program." Based on evaluation, no intact cylinder was in condition to be shipped offsite, because of corrosion, nonfunctional valving, dents, or other miscellaneous damage. These cylinders were deemed unstable and unsafe, and the contents had to be rendered safe.
Cylinder evaluation was a multi-step process (Fig. 4). Initially, when cylinders were excavated, each underwent an in-field visual inspection; cylinders which were breached were discarded as scrap metal, and those which appeared to be intact were subject to closer inspection in the pit itself. This inspection included a survey using common field-portable direct reading instruments (including a GOW-MAC gas leak detector, and Gastech Model 1314 in the ppm combustibles mode), as well as a soap film solution to determine if leaks were present. Cylinders were stored in the special operations area and visually checked and monitored daily using the PID and FID.
Fig. 4.
The next phase of cylinder evaluation was similar to the initial evaluation, but was performed under controlled conditions, and cylinder cleaning was included as an initial step. Each container was given a unique identification number. All leak testing was repeated. Each container was examined in detail, and physical measurements including specific defects, wall thickness, color codes, special construction or special metal alloys, and type and construction of the valve were recorded on preprinted forms. All cylinders were photographed using a Polaroid instant camera. Color copies of the pictures together with a copy of all field data were mailed to the ETSC-Schaumburg, Illinois, office for review and consultation on possible courses of action.
Cylinder content analysis was performed in an onsite mobile laboratory containing an Infrared Spectrophotometer and a Mass Spectrometer. Cylinders were connected to the instruments by a vacuum manifold, using either their existing valving, or a new valve installed on the side of the cylinder wall using proprietary "hot tapping" called high pressure container access (HPCA). The HPCA installation is somewhat similar to installing an in-line saddle on an existing conventional plumbing system. Small quantities of gases were allowed to bleed through the system if still pressurized; alternatively, helium gas was used to purge the cylinder if the pressure in the cylinder was at zero pounds per square in. gauge (psig).
The fourth step was management, or on-site stabilization, of gases from cylinders. On-site stabilization of reactive gases from unstable cylinders was performed after registration with the City of Albuquerque Air Quality Division. Either registration or permitting with the City of Albuquerque was required because of the contents of certain cylinders. A permit would be required if the City determined that a significant amount of hazardous air pollutants were being emitted (case-by-case basis, no emission limit set) or the precontrolled emission rate exceeds 10 pounds per hour or 25 tons per year; otherwise registration was required. Registration was the method of choice because of the small quantities of materials found. This course

of action was also consistent with SNL/NM's existing agreements regarding permitting of potentially explosive and unstable chemicals. In that agreement, NMED determined that the treatment of potentially explosive and unstable chemicals to render them non-hazardous was exempt from permitting and substantive requirements as outlined in the New Mexico Hazardous Waste Management Regulations. All treatment process residue was disposed through the Permitted Hazardous Waste Management Facility at SNL/NM. Management of any specific gas from a cylinder depended on the gas identified. A multiple decision pathway on the management and disposition of contents of each group of cylinders was initially compiled. A decision on the "best" option for the management of gas from a particular cylinder was then determined, although contingencies were also retained. In addition, the possibility of making an immediate on site decision to protect human health and the environment if a toxic release, explosion, or fire occurred during handling was retained as an option for all cylinders.

The preferred option for flammable, toxic, corrosive, or oxidizing gases was transfer of the gas to a DOT-shippable container, if the original cylinder contained sufficient quantity of gas (Table III). The next options for some reactive or toxic gases included neutralization, oxidation, or hydrolysis. For non-toxic or halogenated flammable gases, another option was flaring. Certain very toxic materials required special handling or transfer of contents to a new package without treatment. For inert gases, the preferred option was release to the atmosphere. Descriptions of management methods for pressurized gases are detailed in Table IV.

TABLE III

TABLE IV

CONCLUSION

The voluntary corrective action remediation of a Gas Cylinder Disposal Pit at Sandia National Laboratories, New Mexico was recently completed. This facility operated from 1963 to about 1984. The project was planned from a waste management perspective, waste management and waste minimization were built into the planning phases, and were aggressively implemented throughout the course of the project. The original scope of the project was for the removal of 20 intact gas cylinders, and the removal of 3,200 cubic yards (2,443 cubic meters) of contaminated and uncontaminated soil.

A site-specific, written waste management plan was produced at the start of activities. The site layout was planned to accommodate heavy equipment, storage of large quantities of suspect soil, and special areas to treat and dispose of gases in cylinders recovered from the pit during operations. Continuous geophysical investigations and radiological surveys focused the investigation on "hot spots" only and immediately reduced the investigation to 56% of the original estimate. Excavation was tightly controlled and managed, employed both hand digging and heavy equipment to maintain safe operations, and, as a result, further reduced the quantities of waste generated. All debris and soils were field screened before comprehensive laboratory analyses were performed to confirm and quantify contamination. Waste segregation was strictly enforced for all waste streams. Contents of cylinders were determined using an onsite mobile laboratory, and the management of gases, including treatment of hazardous gases, was performed onsite. Steps were also taken to minimize the generation of decontamination fluids and used personal protective equipment.

Several dozen thermal batteries; 5 pounds (2.3 kilograms) of lithium metal; 3.7 pounds
(1.7 kilograms) of rubidium metal; several kilograms of unknown chemicals; 140 cubic yards (107 cubic meters) of thorium-contaminated soil; 270 cubic yards (205 cubic meters) of chromium-contaminated soil; and 450 gas cylinders, including 97 intact cylinders containing inert, flammable, toxic, corrosive, or oxidizing gases were removed and effectively managed to minimize waste. These figures amounted to a four-fold increase in the amount of intact cylinders removed from the pit, yet only 12% of the original estimate of 3,200 cubic yards (2,443 cubic meters) of soil waste generated during the course of the removal. A summary of waste streams and waste quantities is provided in Table V.

TABLE V

Pre-planning was the key ingredient in a successful recipe for remediation. Efficient site layout allowed ease of operations, generous contingency for the unexpected, adequate temporary storage space, and the facilities to allow treatment

of hazardous and toxic gases. Effective and usable waste management documentation provided for smooth and efficient operations, a high level of quality assurance and quality control, and the flexibility to modify operations when needed. Finally, application of the principles of waste minimization showed that remediation could be focused and selective, minimize cost and schedule, and guaranteed that waste volumes were as small as possible. These approaches, or parts of them, could be applied immediately, even to sites which are currently in the process of being remediated, throughout the DOE complex or elsewhere. The cost savings would be significant, quality control would be improved, and disposal volumes would be decreased.

Session 58 -- High Level Waste: Spent Fuel and HLW Storage/DWPF
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58-1
IN-SITU CHARACTERIZATION OF METAL FUEL STORED IN THE HANFORD K BASINS
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ABSTRACT
The Hanford K Basins contain roughly 2,000 tons of uranium metal fuel previously irradiated in N Reactor. Current plans call for the fuel to be moved to a new storage location in the next few years. This paper describes near term activities to gather data on the mechanical condition and chemical state of the fuel and associated sludge. These on-going in-situ, nondestructive, examinations are intended to complement hot cell examinations which are planned for the near future. Characterization data will feed decisions on interim storage, long term storage and the environmental documentation process as the fuel is relocated.
Several techniques have been, or are planned to be, employed during the conduct of examinations in the K Basins.
1. A video survey using underwater cameras has been made for the open top canisters which hold fuel in the K East Basin. These data have given early assessments of cladding degradation, canister corrosion, fuel swelling, and canister/fuel interactions.
2. Ultrasonic measurements of the water level in gas traps connected to sealed canisters have been used to provide indications of gas generation in or water leakage into the canisters.
3. Sampling of gas and water from closed canisters of fuel in the K West Basin will furnish indications of fuel corrosion, fission product release, and canister integrity and will guide choices of fuel to be retrieved for detailed hot cell examinations.
4. Depth measurements and qualitative properties assessments for the sludge on the floor of the basins and in the bottom of canisters have been made using underwater video techniques.
INTRODUCTION
Two water-filled Hanford K Basins are currently utilized to store 2,000 tons of zirconium alloy clad uranium metal fuel assemblies irradiated in N Reactor prior to 1987. Each of the two basins is 38 m long by 20 m wide and is filled with water to a depth of 4.9 m. Half of these fuel assemblies (i.e., those in the K East Basin) are stored in open top canisters made of aluminum or stainless steel and half (in the K West Basin) are in sealed vented water-filled canisters made from the same materials. The design of a canister is shown in Fig. 1.
Fig. 1.
The individual N Reactor fuel assemblies each consist of two concentric circular fuel elements separated by an intervening coolant channel. Assemblies are up to 66 cm long and 6.14 cm diameter. Optimally 14 inner and 14 outer elements are stored in one canister. A canister consists of two identical barrels welded together with seven fuel assemblies per barrel. In practice fewer than 14 fuel assemblies are often found in one canister.

Efforts are under way (1) to gather information on the mechanical condition and chemical state of the metal fuel and associated sludge stored in the Hanford K Basins. These characterization data are intended to support the eventual transportation, processing, and interim storage of this fuel. Most of the current data gathering activities are being performed in the basins without fuel movement. These in-situ nondestructive examinations are intended to complement hot cell examinations which are ongoing or planned for the future. The techniques employed include a video survey of open storage canisters, determination of water/gas interface levels in gas traps for sealed canisters, sampling of gas and water from sealed canisters, (for chemical analysis) and measurement of sludge depth and sludge volume.

NEAR TERM CHARACTERIZATION DATA

Video Survey

Using an under-water camera a survey has been made of the open-top canisters which hold fuel in the K East Basin. These data have given early assessments of the degree of cladding degradation, endcap damage, canister corrosion, fuel swelling, and canister/fuel interactions. An example of fuel corrosion associated with a broken or detached endcaps is presented in Fig. 2.

Fig. 2.

A quantitative assessment of the percentage of deteriorated fuel has been made and a significant fraction are damaged. However the majority of the fuel elements appear to be intact and uncorroded (as far as can be determined from an overhead view). Those elements which show evidence of deterioration probably are those which sustained endcap damage during reactor discharge. Oxidation of uranium fuel by water has caused deformation and detachment of endcaps due to fuel expansion caused by formation of low density oxides relative to the original metal. A small minority of fuel elements have split cladding with major segments of missing fuel. Table I summarizes the amount of fuel damage observed during the video survey of the K East Basin.

Corrosive and mechanical interactions between fuel and canisters are not seen. Stainless steel canisters appear to be essentially uncorroded in the K Basin water environment. The older aluminum canisters show large numbers of corrosion nodules which predate the institution of strict water chemistry control measures in the basins.

TABLE I

Water Level Measurements for Gas Traps Associated with Sealed Canisters

The canisters in K West Basin have sealed lids as shown in Figs. 1 and 3. These closed canisters contain a venting device which allows excess gases (such as hydrogen liberated during fuel oxidation) to escape. Ultrasonic measurements of the level of the gas/water interface in these devices have been used to provide indications of gas buildup due to fuel corrosion or of water intrusion due to a failure of the canister seal. Figure 4 is a plot of the data obtained which indicates that the majority of the gas traps contain largely gas while a smaller population contain predominantly water (perhaps indicative of leaking canister lids).

Fig. 3.

Fig. 4.

Gas and Water Sampling from Sealed Canisters

Strategies and equipment have been developed for sampling of gas and water from the closed canisters of fuel in the K West Basin. Entry into the canisters will be made through valves (Fig. 1) originally designed to load canisters with water and covergas. The samples obtained in this manner will furnish indications of fuel oxidation, fission product release and canister seal integrity. These data will also guide choices of fuel to be retrieved from the K Basins and examined in detail at the Hanford hotcells.

Sludge Depth Measurements

The sludge on the floor of the basins (mainly in K East) consists of a mixture of canister/rack corrosion product, oxidized fuel, sand, and other debris of external origin. The range of characteristics ascertained to date is summarized in Table II. Analysis of sludge inside of the canisters has not yet been done but is planned for the near future. Depth measurements and qualitative properties assessments of K East Basin sludge have been made using under water video techniques. These measurements are summarized in Table III. Large expanses of the K East Basin are covered by a

depth of only a few cm of sludge with isolated accumulations of up to 18 cm deep. Small pit areas appended to the basin contain sludge up to 100 cm deep in isolated spots. Integration of the depth measurements yields the sludge volumes listed in Table III, for the basin proper and the pits.

TABLE II

TABLE III

Hotcell Examinations

Examinations in hotcells of fuel with a history of K Basin storage are expected to commence in the very near future (2). Initial examinations will focus on the fuel stored in the sealed K West canisters. Damaged and pristine elements will undergo optical and dimensional measurements. Follow-on destructive examinations for fuel will include metallography, thermogravimetric measurements, and radiochemical analysis. It is also expected that determinations of the physical, chemical, and processing characteristics of basin and canister sludge will be performed in parallel with the fuel examinations.

CONCLUSIONS

Accelerated in-situ characterization of K Basin spent metal fuel will influence near term facility decisions (3) and allow informed choices of fuel and sludge samples targeted for more extensive hotcell examination. The condition of the elements in the K East Basin ranges from pristine to corroded fuel with ruptured cladding. The magnitude of the sludge mitigation campaign has been bounded through determination of sludge volume. Finally data pertinent to the choice of wet and dry storage options are being obtained from sealed canisters in the K West Basin.

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58-3

HANFORD TANK FARMS VADOSE ZONE MONITORING PROJECT

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ABSTRACT

The Hanford Tank Farms, and in particular the single-shell tanks (SSTs), are one of the U.S. Department of Energy's (DOE) most important facilities in terms of environmental management and remediation. Until the high-level nuclear waste is removed from the SSTs, this waste requires proper management to minimize releases to the environment. Contamination present in the vadose (unsaturated) zone also is of concern. Accurate and defensible information is needed for current tank management decisions and for longer term tank waste disposal and closure. Tasked with conducting a monitoring program to measure the subsurface contamination, the DOE Grand Junction Projects Office (GJPO) intends to acquire spectral gamma-ray data from nearly 800 boreholes at the Tank Farms. GJPO personnel will collect these data

with specially designed logging trucks equipped with high-resolution Spectral Gamma Logging System units to measure radionuclide concentrations. This nonintrusive measurement method will provide data for Hanford's cleanup mission, monitoring program, and baseline characterization.

OVERVIEW

Large quantities of high-level nuclear waste were produced during the processing of plutonium for national defense at the Hanford site in southeastern Washington State. From approximately 1943 to 1987, large underground storage tanks were used to store millions of liters of high-level radioactive liquid wastes. The term "Tank Farm" refers to the areas on the Hanford site where groups of these subsurface tanks are located. U.S. Department of Energy Richland Operations Office (DOE-RL) officials estimate that 67 of the 149 single-shell tanks (SSTs) have leaked in the past, releasing approximately 1 million liters of high-level mixed waste into the vadose zone sediments under and surrounding the tanks.

DOE-RL started the Hanford Tank Farms Vadose Zone Monitoring Project in 1994. The project is a scientific investigation to determine the types and extent of radioactive contamination in the subsurface as a result of SST leaks. Subsequently, DOE-RL tasked the DOE Grand Junction Projects Office (GJPO) and its contractor, Rust Geotech, with this investigation.

GJPO's background in subsurface measurements and calibration is based on 50 years of experience that includes the National Logging Program conducted in the late 1970s and early 1980s. During this program, millions of borehole-feet were logged in support of the National

Uranium Resource Evaluation (NURE) program. GJPO constructed and maintains a complete suite of subsurface calibration facilities that are recognized as the national standards for calibration of subsurface logging equipment. Scientists at the GJPO used these calibration facilities to develop innovations in calibration and data analysis methods that will also benefit this project.

TANK FARMS BACKGROUND

Constructed between 1943 and 1964, the tanks are known as SSTs because their walls are a single shell of steel-lined concrete (1). These SSTs, located in Tank Farms in the 200 East and West Areas at the Hanford site (see Fig. 1), are as much as 2.5 meters below ground surface and more than 46 meters above the unconfined groundwater table beneath the Hanford site. Sixteen of these tanks have individual capacities of about 52,250 liters, and 133 tanks have individual capacities that range from 503,500 to 950,000 liters.

Fig. 1.

The 149 SSTs reside within seven waste management areas, with each area containing between 10 and 40 SSTs. Waste management areas A-AX, B-BX-BY, and C are located in the 200 East Area. Waste management areas S-SX, T, TX-TY, and U are located in the 200 West Area.

An estimated 67 of the 149 SSTs leaked in the past, releasing approximately 1 million liters of high-level waste into the vadose zone sediments surrounding and under the tanks. The addition of waste to the SSTs was halted in 1980, but the tanks continued to serve as waste storage vessels. Double-shell tanks, first put into service about 1968, are the only tanks currently receiving waste. A double-shell tank has two steel-lined concrete shells separated by an annulus. None of the 28 double-shell tanks are known to have leaked, and they are not included in this monitoring program.

DOE-RL installed nearly 800 steel-cased boreholes in the sediments around the SSTs. For many years, gross gamma-ray logging was performed in these boreholes to detect leaks from the tanks. Because detecting leaks in the sediment really is detecting a problem after it has occurred, DOE-RL initiated new programs to provide in-tank monitoring data for leak detection. The role of vadose zone monitoring for leak detection diminished during succeeding years as the tanks were pumped of free liquid and as better in-tank monitoring methods were established.

Current data needs focus on determining the types and distribution of contaminants under and surrounding the SSTs. This information is necessary to determine which tanks have leaked in the past and to use as a basis in making decisions about tank operations. Subsurface radionuclide concentration data represent part of the characterization information required to prepare an Environmental Impact Statement (EIS) and an SST closure plan required by the Resource Conservation and Recovery Act (RCRA). Data from this project will serve DOE's Hanford cleanup mission and will

increase the ability to meet scheduled milestones in the Hanford Federal Facility Agreement and Consent Order (2), also known as the Tri-Party Agreement because it was co-signed by DOE, the Washington State Department of Ecology, and the U.S. Environmental Protection Agency (EPA), Region 10, in May 1989.

PROJECT OBJECTIVES

The three primary technical goals for the project are

To obtain baseline monitoring data for the SSTs with the new high-resolution Spectral Gamma Logging System (SGLS) units.

To develop recommended improvements to the existing gross-gamma monitoring to provide defensible, scientific methods to enhance routine leak detection.

To evaluate promising new technologies for determining the nonradiological hazardous constituents in the vadose zone and for studying geologic formation properties as an essential component of transport modeling.

Federal and State agencies and Hanford stakeholders approved these project objectives. The Washington State Department of Ecology, the 1990 DOE Tiger Team, EPA, and the U.S. General Accounting Office reviewed and prepared reports on the viability of vadose zone monitoring and the effectiveness of the existing gross-gamma leak-detection program. These reports, as well as reviews by DOE-RL, determined that borehole geophysical logging is a technically viable and cost-effective method for monitoring and characterization purposes.

Spectral Gamma Logging Systems Baseline Monitoring

Greenspan Inc., Houston, Texas, built two new SGLS units for the Westinghouse Hanford Company specifically for this project. GJPO personnel will operate the units at the Hanford site to collect data for characterization and monitoring needs. Characterization data needs include

Determining the types, concentrations, and distribution of radionuclides under and surrounding the SSTs.

Identifying contamination sources (which tanks have leaked).

Establishing a baseline for comparison of data for short- and long-term monitoring objectives.

Knowing the types and concentrations of radionuclides will help in understanding the scope of the current environmental remediation problem and in evaluating the economics of remediation alternatives required by RCRA. Identifying sources of contamination is important for current operations because special controls or operations may be performed on those tanks that are known or suspected to have leaked. At present, the number of tanks and the areal extent of the leakage are not well known. An established baseline is necessary to use as a basis for evaluation of future short- and long-term monitoring data.

Monitoring data needs include

Performing leak detection.

Providing tank surveillance.

Assessing long-term performance and risk.

Monitoring changes in the distribution of contamination in the vadose zone is a primary objective. SGLS data collected in the boreholes can detect new, or previously undetected, areas of contamination. Surveillance aspects involve determining any near-term changes of contaminant concentrations in the subsurface and the cause of those changes. Long-term monitoring will provide data for input into performance and risk assessments that are needed for integration into the EIS and RCRA investigation and closure plans.

If adequate long-term monitoring data can be obtained to show that radionuclides in the vadose zone are stable during an established institutional control period, a leave-in-place remediation option may be justified. Such an option would result in a savings of billions of dollars compared to any type of removal action.

GJPO personnel will collect SGLS baseline monitoring data from all boreholes surrounding the SSTs. These data will be reduced to radionuclide concentration profiles for correlation between boreholes. Computer software programs will model the cross-borehole correlations to show the three-dimensional (3-D) contaminant distribution. These 3-D diagrams (see Fig. 2) provide a powerful interpretation tool to locate contamination sources. Determining the sources is an important key in identifying which tanks have leaked and, as a consequence, which tanks may require special management considerations.

Fig. 2.

The final product of the SGLS baseline monitoring project is a series of reports to

provide radionuclide concentration data and 3-D interpretation diagrams and to identify, as well as possible, which tanks have leaked. Establishment and maintenance of an extensive database will permit comparisons with future data to address the long-term migration monitoring aspects of the project.
Data, reports, and conclusions from the SGLS baseline monitoring project will be available for integration into various site-wide monitoring programs, systems engineering studies, risk assessments, the EIS, and the SST Closure Plan. SGLS project data will increase the understanding of the magnitude and scope of any cleanup operations and will designate which tanks may deserve special treatment during waste retrieval operations to prevent further degradation of the vadose zone environment.

Review of Previous Monitoring Program

As part of the Tank Farms Vadose Zone Monitoring Project, GJPO will review the previous gross-gamma leak-detection program. Review components include data acquisition, reduction, interpretation, and instrumentation calibration and maintenance to determine the utility of the program as a leak-detection tool for the Tank Farms.
On the basis of the evaluation of previous logging practices, GJPO will prepare a scientific-based test plan for selected procedures, instruments, and methods. The goal of the test plan is to select ways to optimize previously used measurement systems, procedures, and data-reduction methodology to support the current in-tank leak-detection program.

Technology Evaluation and Demonstration

Several new technologies currently being developed may benefit the Tank Farms monitoring program. GJPO will evaluate new technologies for capabilities to identify hazardous constituents and to determine soil formation properties and will select technologies that could be immediately beneficial to the project. Four promising areas of investigation are

- Multispectral neutron-gamma logging
- Neutron porosity measurements
- Prompt-fission neutron logging
- Gamma-gamma density measurements

Following the demonstration of the selected technologies at Hanford, GJPO personnel will calibrate the methods and the hardware.

FEATURES AND CAPABILITIES OF THE SPECTRAL GAMMA LOGGING SYSTEMS

Westinghouse Hanford Company, a DOE-RL contractor, designed two logging units specifically for this project, and Greenspan Inc. of Houston, Texas, built the units to those specifications. Greenspan and GJPO personnel performed testing and calibration of the units at the GJPO calibration facility.
Future calibrations of the SGLS units will be conducted at the Hanford site. Data will be collected from borehole models moved to Hanford from a DOE calibration facility at Spokane, Washington. The calibration factors determined from the GJPO measurements will be applied to the Hanford data to establish a correlation between the Hanford models and the GJPO standards, thereby certifying the Hanford models for future calibrations of the SGLS units.
Each logging unit is installed on a four-wheel drive, diesel-powered truck (see Fig. 3). The truck engine powers an electrical generator that supplies all of the required electrical power, and the hydraulic system drives a cable winch and logging boom. The winch and boom handle the cable that suspends the gamma-ray sensor package, or sonde, in the borehole. This cable also contains electrical lines that carry signals from the gamma-ray detector to the recording apparatus in the truck.
Fig. 3.
In a typical operation, the sonde is moved to the bottom of the hole, then gamma-ray data are collected as the sonde returns to the surface. Typically, the sonde moves to a particular depth, stops and gathers data for approximately 100 seconds, then ascends to the next depth, which is usually 15 centimeters higher. Because a typical borehole is about 36 meters deep, a borehole log will consist of about 240 individual readings. About 6 to 8 hours will be required to log one borehole.
After placing the sonde in the borehole, the logging system computer completely controls the logging process. The computer is programmed to control sonde movements and data gathering processes and to continuously monitor essential mechanical and electrical logging components. The computer responds to a malfunction by sounding an alarm. If the logging engineer fails to correct the problem, the computer dials a

cellular telephone and sends a message to an operations control center, then terminates the logging operation.

Gamma-ray detectors within the sondes are crystals of high-purity germanium. The gamma-ray energy resolution of these semiconducting detectors is far superior to that of scintillation detectors, such as sodium iodide or cesium iodide, customarily used in petroleum and minerals industry gamma-ray logging. This excellent energy resolution allows the data analyst to identify the gamma-ray radionuclides, in most cases, and to calculate the concentrations of the radionuclides precisely.

Because germanium detectors must be maintained at the temperature of liquid nitrogen (-200 C) during operations, each sonde contains a metal flask, or dewar, that holds enough liquid nitrogen to cool the detector for about 12 hours. Each logging truck carries a 94.6-liter dewar of liquid nitrogen that permits replenishment of the sonde dewars in the field.

PROJECT MANAGEMENT

The Hanford Tank Farms Vadose Zone Monitoring Project is planned and will be managed as a project with established scope, cost and schedule baselines, and performance measurement elements. GJPO-prepared documentation requisite to the performance of this project include, but are not limited to, calibration and operating procedures, monitoring plans, and management and operating plans to ensure the integrity and defensibility of the collected data. The following specific documents represent the types of documents GJPO personnel will use to conduct the Hanford Tank Farms Vadose Zone Monitoring Project:

The SGLS Calibration Plan describes data collected at the GJPO calibration facility that were used to determine the appropriate calibration and environmental correction factors for the SGLS. The properties of the standards are compiled in Field Calibration Facilities for Environmental Measurement of Radium, Thorium, and Potassium (3).

The Calibration Report contains (1) descriptions of measurement methods; (2) calibration constants for potassium, uranium, and thorium; (3) calibration functions for analysis of artificial nuclides; (4) casing correction functions; (5) corrections for variations in diameter of a water-filled hole; and (6) other measurement factors, such as spatial deconvolution parameters. Standard propagation-of-uncertainties methods will be used to estimate experimental uncertainties associated with all measured and calculated quantities. This report will be a "chart book" for the logging systems.

The Interim Operating Procedures designates the procedures to be used during extensive field testing of the SGLS units following calibration and before initiation of the Tank Farms baseline monitoring. Field testing will be conducted with the GJPO calibration and maintenance facilities. This testing will consist of logging the calibration boreholes and testing the electronics, draw works, hydraulic systems, software programs, and data management systems. Operators will use this document during training to operate the systems, verify the quality of data, and process data. GJPO personnel will modify the interim operating procedures during the field testing to prepare the Operating Procedures for Logging. These final operating procedures will be reviewed and approved before baseline monitoring is initiated at the Tank Farms.

The Baseline Monitoring Plan presents the scope and objectives of the technical program, including data quality objectives, methodologies, procedures, and instrumentation used for data collection with the new SGLS units in the Tank Farms. Other elements of this plan are analytical requirements, quality controls, required reports, personnel qualifications, data handling, data reduction, and data quality assessments. This plan will be implemented following completion of the field testing.

The Data Processing and Storage Plan describes management of borehole data acquired at the Tank Farms with the SGLS units. GJPO personnel will reduce and interpret these data and prepare results for presentation in reports. These postacquisition tasks will be performed with specialized data processing hardware and software systems other than the onboard computers in the SGLS units. The Database Interface Plan will define the Hanford databases that should receive raw log data and reduced data and the databases that should support data processing.

CONCLUSIONS

The SSTs at Hanford are one of DOE's most important environmental concerns. Until the waste is removed from the SSTs, it must be properly managed to minimize releases

to the environment. Information is needed for current tank management decisions and for long-term tank-waste disposal and closure. A vadose zone monitoring program at the Tank Farms is necessary to provide that information for both monitoring and characterization data needs.
DOE-RL will benefit from GJPO expertise in borehole logging. The Hanford Tank Farms Vadose Zone Monitoring Project is based on more than 50 years of experience in subsurface measurements combined with many years of experience in successful project management.
Development and implementation of this vadose zone characterization and monitoring project will enhance the success of DOE-RL's efforts to meet scheduled milestones in the Hanford Federal Facility Agreement and Consent Order (known as the Tri-Party Agreement because it was co-signed by DOE, the Washington State Department of Ecology, and EPA, Region 10, in May 1989). By implementing a vadose zone monitoring program for the SSTs, along with evaluating and upgrading the existing leak-detection monitoring program, DOE-RL will demonstrate a commitment to meeting the Tri-Party Agreement by providing data useful for Tank Farms operations, environmental remediation planning, and postclosure monitoring.
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58-4
NEW METHODS FOR DETERMINATION OF INTERSTITIAL LIQUID LEVELS IN HANFORD WASTE TANKS
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ABSTRACT
The key to the leak detection program for many tanks at Hanford is the method used to evaluate the apparent interstitial liquid interface (ILL) within the pore space of the solid waste medium (either crystalline or sludge). Three new approaches were introduced in the summer of 1993 (count rate, derivative, and sigmoid), all of which significantly improved the accuracy and repeatability of interstitial liquid level values from neutron survey data. This paper summarizes the three new methods and details a case study in which, as a direct result of this improved analysis, a tank that had been declared an "assumed leaker" was reclassified as "sound" for the first time in Hanford's 50 year history.
INTRODUCTION
Many of the 177 underground nuclear waste storage tanks at Hanford do not have liquid waste surfaces, and potential leakage must be evaluated by determining the Interstitial Liquid Level (ILL). The in-tank photograph shown in Fig. 1 indicates why a surface level measurement alone cannot be used as a valid leak detection indicator in many tanks. Since 1985, a total of 61 liquid observation wells (LOWs) have been installed for the purpose of leak detection, with additional installations continuing. A LOW is a hollow tube made of fiberglass, steel, or "TEFZEL" (Registered trademark of DuPont), which is capped at the bottom, and inserted into the waste to a point near the bottom of the tank. The interior of this tube can be opened to the atmosphere via surface risers, but is isolated from the waste, thus providing a surveying environment which is free from direct contamination (as shown in Fig. 2). These LOWs are surveyed weekly, using wireline logging techniques common

to the geophysical and petroleum industries. Both thermal neutron and gross gamma ray probes are used, but the neutron probe normally provides higher quality data. The resulting surveys, (plots of depth vs. count rate), are then evaluated using interpretation software to determine the exact depth of the liquid interface. Plotting the derived liquid interface against time can document trends and changes in liquid levels. These interface changes, when analyzed with statistical methods, provide the basis for an effective leak detection program.
Fig. 1.
Fig. 2.
As mentioned above, the two probes most commonly used for this analysis are the thermal neutron probe and the gross gamma probe. The gross gamma probe at Hanford consists of a single Geiger-Mueller detector and its associated electronics, which acts very much like a point-source detector. Since the waste in Hanford's single-shell tanks was generated from 25 to 50 years ago, any short-lived isotopes have long since decayed, and are present only at insignificant levels. The primary gamma-producing isotope that remains is Cs137, which is in a highly water-soluble form in most cases. Although this solubility normally results in a clearly identifiable change in gamma activity at the liquid interface, the change in count rate is not normally as definitive as that provided by the neutron probe. In most cases, the gamma profiles are less sensitive to changes in fluid saturation, and are normally used to support and verify the primary neutron data. The remainder of this paper will focus on a new and more detailed analysis of the neutron count profiles.
The underlying physical principle behind a thermal neutron probe is its response to moisture, more specifically, to the hydrogen associated with water. The neutron source, located near the bottom of the probe, consists of 1.5 Curies of americium/beryllium (AmBe), which emits "fast" (high-energy) neutrons at approximately 4.5 MeV. Two of the most common neutron detectors are made from either helium 3 (He3) or boron trifluoride (BF3), both of which are gasses and emit an alpha particle when a thermal neutron is absorbed. The initial energy level of the source neutrons, however, is much too high to allow capture by either detector. The neutrons must achieve a significantly lower energy level before they can be detected. It is the rate of this neutron "thermalization" process that is of primary interest.
One of the most effective fast neutron moderators known is hydrogen, primarily because its mass is virtually equal to that of a neutron. In a high-hydrogen environment a fast neutron will slow to thermal energies nearer the source, and will thus be captured by a detector (yielding high count rates). In virtually all other environments, the neutrons will maintain their initial energy levels far longer (i.e., yielding lower count rates at the detector). Since high hydrogen densities are normally associated with liquid water (or organics in some cases), this makes the thermal neutron probe a very sensitive moisture detector. In practice, the BF3 detector is used at Hanford instead of the He3 detector because of its superior level of discrimination in high-gamma environments. A typical neutron probe response across a liquid/solid or liquid/vapor interface goes from high counts in the liquid phase to low counts in the solid or vapor phase, and normally exhibits an "S", or sigmoid-shaped response curve (unless modified by nearby porosity and saturation changes).
BACKGROUND
The initial LOW survey program was formally established in 1985, along with basic software to determine the liquid interface from neutron and gamma scans. This interpretation software was designed primarily for ease of use, and was expected to yield only rough approximations of liquid levels. The major elements of this original analysis are summarized below followed by the resulting limitation.
  The data was collected only at intervals of 0.1 foot (1.2 inches), and the results were also recorded in these increments. No interpolation between data points was attempted. Limitation: interpolation could have improved resolution.
  The computed liquid interface was obtained by visually setting crosshairs on a monitor screen. Limitation: graphical resolution was poor and the depth scale was not expanded adequately, leading to potentially large errors in depth.
  Results often could not be repeated, even by the same technician. Limitation: increased data scatter, particularly when different technicians were used.
  If there was no significant visible difference between the established baseline plot (taken several years earlier) and the current scan, the difference was recorded

as "zero", and no quantitative analysis was attempted. Limitation: the difference was often significant, although barely visible on the compressed graphic scale then in use.

A flaw in the software allowed the interface to be established at a data point which was not on the recorded data curve. Limitation: data could be offset by as much as 1 to 2 inches.

Analysis of the "top of tank" marker was not attempted. Limitation: Without a fixed internal depth reference all systematic errors in the van depth system translated directly into ILL calculation errors.

Baseline values used for leak detection (both surface level and ILL data) were fixed single values. Most of these values had been set several years earlier. Limitation: If natural evaporation, condensation or settling is occurring the baseline should change with time at a relatively constant slope. In the past, this inflexible approach led to some naturally evaporating tanks being declared "assumed leakers" when fixed and outdated limits were exceeded.

Since tolerances were large, however, this method was considered acceptable for many years. Over time it simply became the "standard", and the limitations of this approach were not questioned or re-evaluated. Most tanks were stable enough so that this method was adequate for gross leak determinations.

The primary focus of this paper is waste tank SX-102. This is a single-shell tank constructed in 1954 which began receiving waste as soon as it was completed. The tank exhibits a relatively thin solid crust (approximately 1.0"-1.5" thick, most likely floating on liquid), and the liquid observation well (LOW) data is the official means of leak detection. Although a surface level conductivity device is operable, it has been in the "intrusion mode" for several years, with only occasional surface level measurements taken (Fig. 3).

Fig. 3.

In the summer of 1993 the ILL on SX-102, as determined by LOW neutron surveys, exhibited a sharp decrease (based on the established evaluation technique at the time), ultimately exceeding the decrease criteria set for leak detection. As shown in Fig. 4, the data indicates a relatively constant level for several years (with a few notable variations) followed by a sharp decrease. The apparent sudden decrease follows a classic leak profile, and as a result of this profile and the official leak detection criteria being exceeded, the tank was declared an "assumed leaker". Preparations for emergency pumping of this tank were initiated, which is a very expensive and manpower-intensive operation. Tank SX-102 was on the "Hydrogen Watch List", a safety classification which requires a special safety analysis and associated precautions prior to allowing work in or around the tank.

Fig. 4.

CURRENT METHODOLOGY

The apparent leakage of SX-102 prompted a fresh look at the ILL interpretation techniques in use at the time. It soon became apparent that a significant amount of error could be generated (as outlined in the "Background" section), and that a much more accurate analysis was technically feasible. One of the most damaging past practices was that of calling the change in level "zero" if no separation was clearly visible between the baseline plot and the current survey. (Some of these apparent "zero changes" resulted from the poor graphic presentation available.) It was this practice that generated most of the flat portion of the ILL plot prior to the sharp decline. Once the decline became large enough to be identified, the rate of decrease was magnified as a result of the other systematic errors in the interpretation technique, as previously discussed.

As a direct result of identifying the limitations of the existing system, three new analysis techniques were developed. The methods are summarized below, followed by a corrected interpretation of the SX-102 ILL data.

Determination of the exact depth of the liquid interface currently involves three distinct analysis techniques. All three approaches (count rate, derivative, and sigmoid) have been applied to each tank, and the technique indicating the most consistent, stable and repeatable results has been chosen as the "official" method for that tank. Fortunately, the historical database of raw survey data extends back to 1985, and a thorough evaluation is possible to determine the most applicable method.

An important part of all three new approaches is analyzing the "top marker". This feature is identified by the sharp increase in counts at the interface between the

tank vapor space and the cement structure of the dome. Figure 2 illustrates this feature for SX-102. Since this feature cannot move it is used as an internal depth calibration. Where this feature is sharp and well-defined (typically in 4 inch risers) the software shifts all of the raw data by the amount required to place the top marker exactly on depth. The resulting ILL values exhibit far less data scatter, since most depth problems associated with the LOW survey van have been removed. Where the LOW has been installed in a riser larger than 4 inches (up to 12 inches) the sharpness of this feature degrades, and cannot be used reliably in about one-third of the tanks.

Most neutron survey data generates acceptable results with all three approaches, but difficult analytical situations occasionally occur. In some tanks, one or two of the available methods fail for various reasons. The profile of every tank, however, has been successfully analyzed using at least one of the three approaches outlined below.

The Count Rate Method

The first method is known as the "count rate method". This simplistic but accurate approach relies on taking an average count rate immediately below the expected interface (100% liquid saturated) and an average count rate immediately above the expected interface (<100% saturated) across a pre-determined depth "window". The two values are then averaged, and the depth corresponding to this "average" count rate is designated as the ILL depth. In practice, this is an iterative process, with the initial "estimate" of the liquid interface being the extrapolated current baseline for the tank, which represents the best guess at the expected depth on the survey date. (Remember that a sloped baseline is used, and the "expected" ILL varies with time.) An "offset" is then assigned above and below that depth to establish a "count rate window" on either side of the expected interface. The resulting windows are intended to represent the stable counts above and below the expected interface. Total count rates are taken in both of these windows, to provide a liquid-filled count rate below the interface and a partially-saturated count rate above the interface. A midpoint count rate is determined by averaging the two and comparing the result with the initial depth estimate. If the two depth values do not match within the convergence criteria, then the newly-determined value is used as a starting estimate and the process is repeated until the values converge. Deviation from the expected baseline is then computed and presented as part of the quality control and data reporting process. (A typical count rate interpretation is shown in Fig. 5.)

Fig. 5.

The Derivative Method

In the derivative method, a "depth window" is established which symmetrically covers both sides of the feature to be evaluated. This is the only method which is not iterative, and does not tie the initial guess to the baseline extrapolation. Within this window, the raw survey data (taken at 0.1 foot increments) is smoothed using a cubic spline function; this function is then broken into equal 0.01 foot increments to improve depth resolution. The resulting output function has a smooth and continuous first and second derivative. (This is one of the inherent mathematical characteristics of a cubic spline curve fit.) The derivative of this function is then calculated every 0.01 feet, and the liquid interface is defined as either the minimum or maximum value of the derivative (depending on slope). This method defines the liquid interface as the exact point (within 0.01 feet) of maximum slope (maximum rate of change). In most cases, the derivative method yields a slightly different absolute value than the count rate technique, but both methods are comparable, and the long-term data trends do not differ significantly. The derivative method is illustrated in Fig. 6.

Fig. 6.

The Sigmoid Method

Any time raw data follows a predictable pattern, it can be evaluated by curve fitting to a mathematical function of similar shape. In most cases, this mathematical function will behave more consistently and will exhibit less scatter than the original raw data. This approach was applied to the classic "S" shaped response of a neutron probe measuring a liquid interface. Equation 1 is the exponential function used to simulate this response.

Eq. (1)

Virtually any sigmoid shape can be fit to this function by appropriate adjustment of

the variables a, b, c, and d. In practice, a non-linear optimization routine is used to adjust the values of these four variables in order to maximize the regression coefficient (R2) between the raw data and the resulting curve-fit function. Once the best possible fit is achieved, the midpoint b (point of symmetry) is defined as the liquid interface, (as illustrated in Fig. 7). As with the count rate method, the resulting answer is fed back into the equation as a starting value, and the routine iterates to convergence. In approximately 50% of the tanks, the sigmoid method proved noticeably superior to the other two methods.
Fig. 7.
Determining the Best Method
To determine the best method, each tank is analyzed by all three methods, with all historical data being re-evaluated. The resulting outputs are plots of the liquid interface (level) vs. time. From these plots, the analyst can determine increasing and decreasing linear trends, as well as the standard deviation of the data points about the trend line. The most appropriate method is the one which produces the tightest data cluster about the linear trend line (minimum standard deviation), while also providing the fewest number of data analysis failures. (The most common problem is a "failure to converge".)
Variations from baseline greater than 2 standard deviations are considered "suspect", and the surveys are repeated. Repeatable variations greater than +/- 3 standard deviations are considered legitimate anomalies, and are reported as potential leaks or intrusions. Once the optimum method and analysis parameters are chosen for a tank, they are stored in the central computer control file, ensuring that the analyses for all past and future surveys will be made on a comparable basis.
CASE STUDY, TANK SX-102 LEAK DETERMINATION
In May of 1993, waste tank SX-102 appeared to experience a sudden downturn in the interstitial liquid level (ILL), primarily due to the poor resolution of the existing analysis technique (as previously discussed). Because of the apparent sudden decrease in ILL, the tank was declared an "assumed leaker", and preparations were being made to perform an "emergency pumping" of the tank. During these preparations, the data was re-analyzed using the techniques just described, and a steady, long-term decreasing trend became apparent, with no major changes in slope. Although the new analysis techniques were in their infancy at the time, they proved conclusively that the decrease in liquid level was due to evaporation, which had been taking place for at least nine years. This was the first time a sloping baseline (taking into account natural evaporation and/or condensation trends) had been applied to raw tank data. A psychrometric study of the apparent evaporation trend was commissioned, resulting in a calculated evaporation rate which very closely matched the new ILL interpretation shown in Fig. 8. The study included physical, thermal, and hydraulic modeling, and yielded a best estimate of evaporation rate equal to 0.68 inches/year. The linear correlation of ILL data provided a slope of approximately 0.63 inches/year. These numbers match within the expected errors of the two methods used. As a result of this new interpretation, the tank was re-categorized as "sound", and a multi-million dollar emergency pumping operation was avoided.
Fig. 8.

58-6
VITRIFICATION OF ACTINIDE SOLUTIONS IN SRS SEPARATIONS FACILITIES
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ABSTRACT
The Savannah River Site (SRS) Separations Canyons will eventually undergo Decontamination and Decommissioning (D&D). In order to D&D these facilities, material (such as Americium and Curium) currently in tanks must be dispositioned. Actinide solutions in these tanks contain extremely valuable nuclear isotopes for which viable applications exist. The National Heavy Element and Advanced Neutron Source (ANS) program at Oak Ridge can use this material; however, a process to convert this solution to a solid form for safe transport does not currently exist. Actinide glass compositions were developed through Office of Technology Development (OTD) research. This research formed the cornerstone of a process to vitrify the

Americium and Curium (Am/Cm). This process has been recommended by Westinghouse and Savannah River Company as the most favorable option for Am/Cm disposition. Work continues towards developing a full scale prototype system to demonstrate the process and train operators.

The system would be installed in the Multi-purpose Processing Facility (MPPF) of F-Canyon. This is an ideal location since it provides viewing windows, good shielding and access to the process with manipulators. The system will consist of a melter capable of at least 1-2 kg/hr glass output, feed systems, and an off-gas system. The glass will be poured into small canisters so the material can be safely stored or shipped to its destination. Two feed streams will ne directed to the melter; glass marbles and a nitric acid solution which contains the material to be vitrified. The two streams are designed so that they will mix in the melter and form the appropriate composition to produce as a result of wet feeding which dictates the requirement to handle condensate in the off gas. The condensate will be collected, sampled for fissile material and discharged to the canyon waste tank. The melter proposed for this program is a platinum/rhodium alloy bushing melter commonly used in the glass fiber industry; however, this is a new and unique application for this melter technology. The melter is an alloy box with tabs on each end for an electrical connection. Current is passed through the bushing and heats it similar to a resistance heater. The unique features of the bushing melter for this application include the following: It will be enclosed so the off-gas can be contained and processed. The melter feed will be wet, that is the melter will be required to supply adequate power to vaporize the liquid portion of the feed and then melt the solids along with the glass beads entering the melter in a separate feed stream.

There are many specialized waste forms in the DOE complex which can not be easily processed in facilities like the Defense Waste Processing Facility (DWPF) at SRS or have a potential future use. The process being developed will provide the capability to convert specialized or unique forms of nuclear material into a stable, solid and safe form which can be stored indefinitely, shipped or reprocessed as the need arises. The successful demonstration of this technology to vitrify isolated actinide material could have future application in numerous locations across the DOE complex.

INTRODUCTION

The actinide vitrification system is being developed for several reasons. A long term goal for the Savannah River Site (SRS) is to Decontaminate and Decommission (D&D) facilities. At least one tank in the F-Area Separations Canyon contains valuable isotopes which the National Heavy Element and Advanced Neutron Source (ANS) program at Oak Ridge can utilize. A minimum of three options were investigated for dispositioning the Americium/Curium (Am/Cm) solution currently stored in an F-Area Separations facility tank. The evaluation determined that vitrification was the favored option since the end product is a solid glass log which can be safely stored or transported.

In order to D&D F-Canyon, stored and inprocess material must be processed or removed from tanks in the facility. This program was initiated to process a specific tank which contains Am/Cm in a nitric acid solution. Some tanks in the facility may be emptied to waste tanks. However, the solution in question could impose adverse effects if this were done. The replacement value for the Americium and Curium isotopes was estimated to be between one and two billion dollars. The quantity of material in this tank will supply all of the ANS program's needs for these isotopes for the foreseeable future. This program is serving two goals of dispositioning the tank of solution and supplying Oak Ridge needed isotopes. By stabilizing the isotopes in glass, they can be safely stored indefinitely for future use.

ACTINIDE VITRIFICATION SYSTEM DISCUSSION

The vitrification system must be operated remotely due to the radioactivity of the Am/Cm solution. The design of the equipment used in this system must ensure high reliability to minimize the amount of maintenance required. The vitrification system will be installed in the Multi-Purpose Processing Facility (MPPF) in F-Canyon. The MPPF is a unique facility in section eighteen if the canyon. It is the only section which has shielded oil filled windows for viewing the operation. Master Slave Manipulators are used for operating equipment in the shielded cells. There are eight window / manipulator stations in the facility, however, only one or two will be required for the vitrification system.

Each station has a removable rack that process equipment is mounted on. Two of the current racks which are no longer in service will be removed to accommodate the vitrification system. New racks will be fabricated to support the vitrification system. The new racks will be fully assembled and tested in a non-radioactive facility to ensure all the integrated systems function properly prior to installing them in the MPPF. Current estimates indicate that it will take approximately 300 hours of operation to process the Am/Cm solution. The expected process parameters are shown in Table I. Table II and III show anticipated glass compositions for this process to vitrify Am/Cm. A schematic of the system is shown in Fig. 1 and each major system is described in detail below.

TABLE I

TABLE II

TABLE III

Fig. 1.

Feed System

The feed consists of two components, a nitric acid solution which contains dissolved Am/Cm and a glass forming frit. The two feed streams must be fed in the correct proportion to make an acceptable glass product. Frit is a dry material similar to sand. Separate systems will be used to feed each stream into the melter lid. The two feed materials will be mixed in the melter and be poured out the bottom as a glass product.

The nitric acid solution will be pumped from a storage tank to a smaller feed tank in the MPPF. The solution is fed using a constant head arrangement as shown in Fig. 1. The solution is airlifted from the bottom into a small vessel above the feed tank at a rate slightly higher than will be fed to the melter. An overflow from this small vessel is returned to the feed tank and a outlet below the liquid level feeds the melter. The fluid head pressure delivers the liquid feed to the melter through an orifice to regulate flow. The liquid feed system has a fixed flow as determined by the equipment setup. Orifice changes can be made as required to alter the process parameters or feed a completely different solution. The liquid feed tube at the melter top will likely require cooling to prevent the solution from boiling in the feed line.

The dry frit feeder will have an adjustable feed rate which must be coordinated with the fixed solution feed rate so the ratio is in the target range. The frit will be fed using a weight loss feeder which determines the feed rate by monitoring the weight loss in the frit hopper over time. This stream will be delivered to a conveying system that will transport the material to the top of the melter where an air lock transfer device will ultimately discharge the dry feed into the melter. The air lock is necessary to maintain a seal on the melter and minimize the quantity of air infiltration so the melter off-gas system is not overloaded. It will also prevent the vapors in the melter head space from contaminating or possibly plugging the dry feed system.

Melter

The melter chosen for this vitrification system is called a bushing melter, which is basically a platinum/rhodium alloy box with tabs on each end. Electrical bus bars are connected to the tabs on each side of the alloy box which acts like a resistance heater when current is passed through it. Up to 15,000 amps at 2 volts are used in the glass fiber industry for drawing glass fibers. The vitrification bushing will have one drain hole as opposed ti several hundred fiber drawing holes in a typical glass fiber bushing.

The novel attributes of the bushing melter that will be used for vitrification include wet feed and an enclosure above the melt. The feed for this melter includes dry frit and a solution of material to be stabilized as opposed to standard bushings which are fed glass beads or molten glass. The vitrification bushing must provide the heat of vaporization for the transport fluid as well as the heat necessary to melt the glass frit and oxides delivered in the feed solution. Using wet feed creates off-gas which must be contained and treated. A top will be installed on the melter which will seal the system so the off-gas can be drawn into the off-gas system. The top will also have two feed nozzles, one each for the wet and dry feed to enter the melter. In addition, a large port in the melter top directs the off-gas stream into the treatment system. The glass product is poured into stainless steel canisters through a single nozzle in the bottom of the melter. The nozzle is sized to ensure the desired pour rate at the expected glass viscosity.

Pouring is controlled by an air jet that freezes the glass in the pour nozzle and/or a platinum alloy flow stopper which acts as a heat sink.

Off-Gas System

This system will function to contain and treat the off-gas originating from the melter. It is desirable to maintain control over the stream coming from the melter since it may contain some small amount of radioactive particulate. The system will operate to cool and scrub this process byproduct and any air introduced into the system for pressure control purposes. The volume of off-gas is dictated by the vaporization of the nitric acid solution and the plant air introduced to control pressure in the bushing melter. A slight negative pressure is maintained inside the system to ensure that if a leak occurs the flow is into the system not out. This feature minimizes the possibility of further contamination of the shielded cells facility.

The off-gas system originates with the melter. A melter top is used to contain any volatiles emanating from the melter. This is a unique feature which has not been used on bushing melters before. The melter top has a port where the off-gas exits and enters the treatment system. There is a control air connection in the off-gas line immediately above the melter. An automatic valve that supplies control air is directed by the signal from a pressure transmitter connected to the melter vapor space. As dynamic changes effecting the melter pressure take place, a closed loop control system reacts to supply more or less air to the off-gas system which maintains a pressure of negative five inches of water vacuum inside the melter. The stream then travels through the pipe to a quencher where the off-gas is quenched with a water spray. The quencher acts to cool, remove particulate and dilute the condensate. The two phase flow from the quencher flows into an off-gas condensate tank (OGCT) where particulate settles out and condensate is collected. The condensate level is monitored and pumped to a waste tank as needed. The vapor fraction of the off-gas stream exits the condensate tank through a port in the top and is directed into a high efficiency mist eliminator (HEME) where particulate is entrapped. The HEME condensate is drained back to the OGCT.

Once the off-gas exits the HEME it is directed to a heater which raises its temperature about ten degrees Celsius. This raises the temperature above the stream's due point to prevent condensation in the High Efficiency effectiveness and increases the pressure drop across the filter. The HEPA filter is the last component the off-gas flows through prior to entering the canyon process vessel vent system (PVVS).

Glass Storage Canister System

The glass product will be poured into stainless steel canisters which are approximately three inches in diameter and fifteen inches high. An automated system for handling the canisters remotely is required due to the radioactivity of the vitrified material and the fact that it will be performed in a shielded cells facility. A turntable capable of holding approximately eight canisters will be used to position empty canisters under the melter drain nozzle. Once a canister is filled, glass flow from the melter is terminated and the turntable rotates to position a new canister under the melter pour spout. Glass pouring can then be resumed so the next canister is filled.

Several of the turntable positions will hold full canisters and allow them to cool. Heat transfer from these canisters will be evaluated to determine the cooling rate. Insulated canister slots on the turntable may be required if the cooling rate needs to be adjusted to ensure the glass anneals properly. One station on the canister turntable will be setup to weld a cap on each canister. A mechanism to inventory and position canister caps so they can be welded is necessary in this remote application. The turntable location after cap installation is used to remove the full canister and insert an empty one.

A material handling system for the canisters must be developed. It will include a pick and place type device to automatically remove the full canisters and insert a new one. The full canisters must not only be removed from the turntable but transported to a location where they can be removed from the shielded cell facility and transported to their destination. Empty canisters will be delivered to the turntable to replace the full ones that are removed. The outer surface of the sealed canisters will be decontaminated to facilitate transportation of the sealed vitrified material.

CONCLUSION

The actinide vitrification system being developed at SRS provides the capability to convert specialized or unique forms of nuclear material into a stable solid glass product that can be safely shipped, stored or reprocessed according to the DOE complex mission. This project is an application of technology developed through funds from the Office of Technology Development (OTD). This technology is ideally suited for vitrifying relatively small quantities of fissile or special nuclear material since it is designed to be critically safe. Successful demonstration of this system to safely vitrify radioactive material could open up numerous opportunities for transferring this technology to applications throughout the DOE complex.

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58-7
INITIAL RESULTS FROM THE CANISTERED WASTE FORMS PRODUCED DURING THE FIRST CAMPAIGN OF THE DWPF STARTUP TEST PROGRAM

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ABSTRACT
As part of the Defense Waste Processing Facility (DWPF) Startup Test Program, approximately 90 canisters will be filled with glass containing simulated radioactive waste during five separate campaigns. The first campaign is a facility acceptance test to demonstrate the operability of the facility and to collect initial data on the glass and the canistered waste forms. During the next four campaigns (the waste qualification campaigns) data will be obtained which will be used to demonstrate that the DWPF product meets DOE's Waste Acceptance Product Specifications (WAPS).
Currently 12 of the 16 canisters have been filled with glass during the first campaign (FA-13). This paper describes the tests that have been carried out on these 12 glass-filled canisters and presents the data with reference to the acceptance criteria of the WAPS. These tests include measurement of canister dimensions prior to and after glass filling, dew point, composition, and pressure of the gas within the free volume of the canister, fill height, free volume, weight, leak rates of welds and temporary seals, and weld parameters.

INTRODUCTION
The Defense Waste Processing Facility (DWPF) will vitrify high-level nuclear waste currently stored in underground tanks at Savannah River Site. After processing, the nuclear waste and glass frit will be mixed and then fed into the DWPF melter. The molten radioactive waste glass will be captured by stainless steel canisters as it pours from the melter pour spout. After glass filling, the canisters will be sealed by insertion of a temporary plug in the nozzle. The temperature of the nozzle will be at least 167oC higher than the temperature of the plug at time of insertion. The nozzle (with inserted sleeve) will then contract during cooling to form a

water-tight seal with the plug. The leak rate of the temporary seal of the canistered waste form will then be measured in the Melt Cell. When the leak rate is acceptable, the canistered waste form will be moved to the Canister Decontamination Cell where it is decontaminated using an aqueous slurry of glass frit. Smears will then be taken to ensure that the amount of smearable radioactive materials present on the outside surface of the canistered waste form is within specification. After this is confirmed, the canistered waste form will be transferred to the Weld Cell where the plug and sleeve are pushed down into the nozzle. A final weld plug will be inserted into the mouth of the nozzle, and a final weld will be made by simultaneous application of force and electric current for a short time duration (upset resistance welding). Smear tests will again be taken at this time. The canistered waste form will then be transferred to the Glass Waste Storage Building.
This report summarizes the results of testing of the canistered waste forms produced during the first campaign of the DWPF Startup Test Program. Durability results of the waste glass from this campaign will be reported separately.

CANISTER DIMENSIONS BEFORE AND AFTER GLASS FILLING

The vendor procurement specifications for length and diameter are:

Height: 117.94 to 118.06 inches or 299.57 to 299.87 cm
Diameter: 23.88 to 24.12 inches or 60.66 to 61.265 cm

The diameter (measured at 36 different locations) and the length of canister S00004 were measured prior to glass filling and the results demonstrated that the length was within the specification, but several diameter values were lower than the specification by a maximum of 0.02 cm. Perpendicularities for this canister were also measured

The ranges and the maximum changes in dimensions for canister after glass filling (canistered waste form) were:

Canistered waste form length:	117.994 to 118.001 inches or 299.705 to 299.723 cm
Maximum change over empty canister:	0.007 inch or 0.018 cm
Canistered waste form diameter:	23.896 to 23.957 inches or 60.696 to 60.851 cm
Maximum change over empty canister:	0.039 inch or 0.099 cm
Canistered waste form perpendicularity:	-0.114 to +0.093 inch or -0.290 to +0.236 cm
Maximum change over empty canister:	0.039 inch or 0.099 cm

It is interesting to note that after glass filling, all of the measured diameters were within specification. This implies that the canister became more round after filling, which is consistent with previous findings by SRTC. The canister became slightly less perpendicular with filling.

The Waste Acceptance Product Specifications (WAPS)(1) requires that the unfilled canister shall have an outer diameter of 61.0 cm (+1.5 cm, - 1.0 cm) and an overall length, after accounting for the closure method, of 3.000 m (+ 0.005 m, - 0.020 m). The maximum plug height using upset resistance welding for final closure is 0.092 inch or 0.234 cm. The unfilled canister S00004 readily met these requirements.

The WAPS also requires that the canister after glass filling and final closure shall be such that it will stand upright without support on a flat horizontal surface and will fit completely without forcing when lowered vertically into a right-circular, cylindrical cavity, 64.0 cm in diameter and 3.01 m in length. This specification was also met. In fact, the changes in these canister dimensions after being filled with glass are insignificant and provide assurance that canisters procured to the current DWPF specifications will readily meet the WAPS.

ICC LEAK RATES BEFORE AND AFTER DECONTAMINATION

After filling a canister with glass, a temporary heat shrink seal is made between an

Inner Canister Closure (ICC) plug and the sleeve in the nozzle. This temporary seal is intended to prevent water from entering the canister during the decontamination process. This ICC seal must be leak tight to less than 2 x 10-4 atmcc/sec for helium, a limit demonstrated to preclude inleakage of water into the canister. Water exclusion from the canister is important in order to prevent internal corrosion of the stainless steel canister. Decontamination is then carried out by frit blasting the canister surface with an aqueous slurry of glass frit.
The helium leak rate of the normal ICC seal on canister S00145 was measured three times before and after decontamination. The results were:

Before:	8 x 10-6 atmcc/sec	After:	9.3 x 10-5 atmcc/sec
	6 x 10-7 atmcc/sec		9.0 x 10-5 atmcc/sec
	-3 x 10-7 atmcc/sec		8.6 x 10-5 atmcc/sec

The before decontamination leak rates were essentially at the limit of sensitivity of the equipment.
The measured leak rate of the ICC plug of canister S00114 was 7 x 10-5 atmcc/sec. As planned, the nozzle was then heated to drop the ICC plug and sleeve into the canister and a new seal was made using a repair plug. The leak rate of this repair ICC seal was measured both before and after decontamination.

Before:	1.3 x 10-4 atmcc/sec	After:	1.2 x 10-4 atmcc/sec

The results for canisters S00114 and S00145 demonstrated that seals with leakrates less than 2 x 10-4 atmcc/sec for helium can be made. In fact all 12 canisters filled during this campaign met this requirement (Table I). Furthermore, it was demonstrated that aqueous frit blasting did not significantly change the leak rate of the seal.
TABLE I
CANISTER WELD PARAMETERS
The final weld is made by forcing the weld cap into the nozzle opening under a simultaneous application of DC current. The parameters of force, current, and time of application of current used to make the final upset resistance weld of nine of the canisters produced during FA-13 are presented in Table II
TABLE II
The nominal values to be used for making final welds are a force of 80,000 pounds, a current of 248,000 amps, and a time of 1.58 seconds.
The parametric study2 for weld performance determined acceptable windows for force and current of 90,000 15,000 lbs and 248,000 22,000 amps, respectively at 1.58 seconds. Hence all welds were made within the parametric window for acceptable welds.
WEIGHTS OF CANISTERED WASTE FORMS
The weights of the empty canisters were measured in DWPF prior to glass filling. After being filled with glass, the canister weight as measured on the pour turntable (PTT) with the bellows raised was also recorded. After arrival at TNX (a Savannah River Technology Center facility), all canistered waste forms were weighed using a calibrated overhead crane scale. The weights of the canisters were measured at TNX to determine if it would be appropriate to use the DWPF weight system to approximate the fill height. The net glass weight for each canister sent to TNX is given in Table III, along with the corresponding weight measured at DWPF.
TABLE III
At DWPF the weight of the glass was obtained by subtracting the weight of the empty canister from the weight of the glass filled canister as measured on the pour turntable with the bellows raised. At TNX, the weight was obtained using the crane scale measured weight minus the combined weight of the empty canister plus the weights of the canister closure plugs. The weights of the closure plugs are as follows:

Weld plug	4 pounds
ICC plug	10 pounds
ICC repair plug (if used)	12 pounds

In all cases the weights measured at DWPF were greater than those obtained at TNX.

These differences, listed as deltas in Table III, range from 32 to 96 pounds. These results indicate that the pour turntable provided a fairly accurate estimate of the canistered waste form weight with the bellows raised.
The differences between the weights of the filled canisters with the bellows lowered and raised are provided in Table IV. The delta increased with filling such that the average additional weight of the bellows was 278 pounds after filling, an increase of 60 pounds over the average difference of the empty canisters. Hence, the force and weight observed were not constant as the canister was filled with glass. The difference is manageable since 60 pounds corresponds to only 1.5 inches of glass fill height. However, significant scatter existed in the delta from canister to canister. The maximum differential was 120 pounds which corresponded to 3 inches of glass fill height. These results suggest that weight alone as an indicator of fill height has an uncertainty on the order of 2 inches.

TABLE IV

FREE VOLUME AND FILL HEIGHTS

The WAPS requires that the canister must be filled with glass to a height equivalent to at least 80% by volume of the empty canister. For a DWPF canister, 80% corresponds to a fill height of 86 inches.
The glass fill height was measured on S00146 and determined to be 90.2 inches. This corresponds to a canister that has been filled to approximately 84%.
The free volume of canister S00146 was determined to be 119.6 liters as described previously(3). The total volume of an empty canister has been estimated to be ~735 liters. Using this value, the glass occupies 83.7% by volume of this canister. Both free volume and fill height measurements indicate that the acceptance requirement was met.
For a Westinghouse built canister which uses stainless steel plate for the cylinder, the typical wall thickness is 0.43 inches and the diameter averages ~ 23.95 inches. Canister S00146 was representative of these dimensions. Using these dimensions, the volume occupied by a 1 inch thick section of the canister was calculated to be 6.86 liters. The measured density of the glass from canister S00146 was found to be 2.695 g/cc.(4) Therefore, each one inch section of the canister, with no voids, should contain 40.7 pounds of glass.
The DWPF canister has a reverse dished bottom which reduces the volume of glass a canister can hold. In addition, the bottom head has a thickness of 0.5 inches. Since the fill height is measured from the bottom of the canister, the amount of volume excluded by the dished bottom and wall thickness can be estimated to accurately determine the glass volume. The excluded volume for the bottom skim cut was calculated to be approximately 4.1 liters, while the half inch wall thickness causes an additional 3.43 liters to be excluded. Therefore, the total volume excluded is 7.53 liters. This corresponds to a 1.1 inch slice of the canister.
For canister S00146, the fill height was measured at 90.2 inches. Therefore, the amount of glass present should be:

(90.2 - 1.1) 40.7 pounds/inch = 3626 pounds.

The measured weight of the glass at TNX was 3606 pounds. This corresponds well with the calculated value, but it does not take into account two factors. First, a void (shrinkage cavity) was detected in this canister several inches below the top surface of the glass. This void was not spherical, but was approximated to be a sphere of 7 inches in diameter. Therefore, the approximate calculated volume would be 3 liters. This corresponds to ~17 pounds of glass which must be subtracted from TNX measured weight (3606 pounds) and brings the calculated and measured values even closer together. The other factor was the fill height itself, since the top surface was concave. As the canister wall was approached from the center axis, the glass fill height increased slightly. It may have decreased again at the wall, but it was difficult to determine since numerous pieces of glass had broken away from the top surface. This latter effect, although small, has not been accounted for in this analysis.
The fill heights of a total of five canisters were measured at TNX. Measurements were performed using standard measuring devices after the nozzles had been removed from the canisters. The distance to the glass surface was measured four times at 90 degree intervals around the base of the canister nozzle, therefore all measurements were towards the center of the canister (in the concave region). These four

measurements were then averaged and compared to the fill heights calculated from the canister weights using the 40.7 pounds/inch factor, as above. The measured and calculated fill heights are shown in Table V.

TABLE V

The maximum difference between the measured and the calculated fill heights was 1.1 inches which along with the data on free volume, indicates that no significant glass porosity or voids existed in the canisters.

DEW POINT OF THE GAS WITHIN THE FREE VOLUME

The dew point of the gas within the free volume space of canister S00146 was measured as previously described(3). The dew points obtained with the two hygrometers (calibrated the hygrometers within + 2C) were:

Hygrometer #1: 5.1C

Hygrometer #2: 2.6C

Average dew point: 3.9C

The dew point was also monitored as a function of internal gas pressure. This was accomplished by slowly pumping the system and monitoring the dew point as a function of gas pressure. The dew point (hygrometer # 1) fell from 4.9C at 700 Torr to 3.0C at 600 Torr to 0.9C at 500 Torr. The partial pressure (vapor pressure) of water in gas is a linear function of the overall gas pressure. Hence the dew point should fall as the pressure is reduced. The measured dew points as a function of overall pressure followed this expected dependence.

The observed dew point within the canister is evidently dependent on the dew point of the air within the melt cell. The temperature and relative humidity for the melt cell when each canister was sealed was recorded and is shown in Table VI.

TABLE VI

Canister S00146 was sealed at a dew point of 13C in the melt cell. If the dew point in the melt cell at the time of sealing was higher, then the dew point in the canister free volume space is expected to be higher. The dew point in the melt cell at the time of sealing was higher than 13C for all but one of the other canisters. This implies that the other canisters have dew points within the free volume greater than 0C. It is important to control the dew point of the gas within the canister free volume. If the temperature of the canistered waste forms within the Glass Waste Storage Facility at SRS reaches 0C, then dew will form on the inside walls of the canister for those canisters having dew points greater than 0C. The concern here is that liquid water can lead to corrosion of the canisters.

The leak rate from the ICC seal of S00146 was 2 x 10-5 atmcc/sec for helium, a rate which is an order of magnitude less than required for this seal. (See section on ICC Leak Rates before and after Decontamination). Therefore, it has also been demonstrated that no significant inleakage of water occurred during the decontamination process.

INTERNAL PRESSURE OF THE GAS WITHIN THE FREE VOLUME

The internal gas pressure of S00146 was measured as 741 Torr at 31C. Since the waste acceptance criterion limits the internal gas pressure at a temperature of 25C, the measured gas pressure of 741 Torr was converted to a pressure at 25C. This pressure became 726 Torr (0.96 atm or 14.0 psi) at 25C, which readily meets the acceptance requirement of a pressure less than 1.5 atm or 22 psi at 25C immediately after filling and sealing.

COMPOSITION OF THE GAS WITHIN THE FREE VOLUME

Mass spectrometric data obtained on the air within the free volume of canister S00146 did not reveal any compounds, other than air, within the sensitivity of the equipment. The acceptance criterion of the WAPS is: No detectable foreign materials present in the free gas. Therefore this acceptance criterion was met. Scans of the canister free volume gas were obtained using a mass spectrometer from 1 to 200, 1 to 50, 40 to 95, and 90 to 200 mass units. These data were compared to equivalent scans taken from room air, which was considered the standard since the gas within the free volume space of the canistered waste form should contain only air from the melt cell. Consequently, the mass spectrometer was looking for changes between the composition of the gas within the free volume and within the room air. The only difference noted was in the amount of carbon dioxide present. Relative to Argon 40,

approximately 25% less carbon dioxide was present in the canister air than in room air.
Scans using a multiplier to increase the sensitivity were also performed from 90 to 200 and 45 to 90 mass units. These regions were where detection of foreign materials would be expected. These data were again compared to the scans obtained with air, and no differences were noted.
Sensitivity of the Extrel Questor II process mass spectrometer was estimated in two ways. The first method used isotopes of argon as a sensitivity indicator with Ar-38 present at ~ 6 ppm. For organics, benzene concentrations of 1, 10, and 100 ppm in air were used. Both techniques demonstrated sensitivities less than 10 ppm, with an estimate of 5 ppm. Hence, detection of foreign materials was bounded by this sensitivity.
CONCLUSIONS
The results of the testing on canistered waste forms produced during the first campaign of Startup Testing at DWPF have demonstrated that the DWPF is ready to continue on with the next four Waste Qualification campaigns.
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TECHNOLOGY DEVELOPMENT FOR THE REMEDIATION OF THE DEPARTMENT OF ENERGY'S RADIOACTIVE

WASTE TANK PROBLEMS
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ABSTRACT
The Department of Energy's Radioactive Tank Waste Remediation Focus Area is integrating the management of technology development for the remediation of the radioactive waste tanks across the DOE complex. The Focus Area is organized to address the key technology development needs in the areas of Structural Integrity and Waste Analysis; Waste Dislodging and Conveyance; and Waste Pretreatment and Processing. These efforts are directed towards demonstrating innovative technologies which will reduce costs, enhance worker safety, and/or enable the safe remediation where technological solutions are currently unavailable or inadequate. The Tank Waste Focus Area is developing sensors that will minimize the handling of samples for laboratory analysis, alleviating the safety concerns associated with handling. To assist in planning for retrieval of the waste from the tanks, technologies to map the interior of the tanks are being developed. Remotely controlled waste retrieval systems, that are capable of dislodging the sludge and saltcake fractions of the wastes and conveying the dislodged waste out of the tanks are also being demonstrated. The separation technologies under development will separate tank wastes into low-level, TRU and high-level fractions, thereby significantly reducing the volumes of high-level wastes requiring costly treatment and disposal. Low-level waste (LLW) treatment technologies are also being demonstrated to reduce waste

volumes and produce waste forms which are chemically and physically durable. Examples of the technologies in each of these areas will be discussed.

INTRODUCTION

Radioactive and hazardous wastes, generated from the production and processing of nuclear weapon materials, have been stored in more than 330 underground storage tanks (USTs) nationwide across the Department of Energy (DOE) complex beginning in the 1940's. The majority of the waste, over 400,000 m3 (100 million gallons), is stored at five DOE sites: the Hanford site in Richland, Washington; the Fernald site in Ohio; the Idaho National Engineering Laboratory near Idaho Falls, Idaho; the Oak Ridge Reservation in Tennessee, and the Savannah River Site in South Carolina. The remediation of these tanks is specified in compliance agreements for each of the sites. Specific remediation milestones and requirements have been established for each site. At Hanford, for example, recent renegotiations of the Tri-Party Agreement (TPA) (between DOE, the EPA, and the Washington Department of Ecology) have revised the cleanup strategy and timetable.

Two waste storage tank design types are prevalent across the DOE complex: single-shell wall and double-shell wall designs. The tanks are made of stainless steel, concrete, and concrete with carbon steel liners, and their capacities vary from 5,000 gallons (19 m3) to 106 gallons (3,785 m3) (see Fig. 1). The tanks have an overburden layer of soil ranging from a few feet to tens of feet. The physical and chemical compositions of the wastes vary from tank to tank. Tank waste consists of several physical forms: sludge, supernate, and salt cake. Most of the waste is alkaline, and contains the following major chemical constituents: nitrate and nitrite salts (approximately half of the total waste), hydrated metal oxides, phosphate precipitates, and ferrocyanides. The 640 MCi of radionuclides are distributed primarily among the transuranic (TRU) elements and fission products, primarily 90Sr and 137Cs. In-tank atmospheric conditions vary in severity from near ambient to temperatures over 93oC, and tank void-space radiation fields can be as high as 10,000 rad/h.

Fig. 1.

In response to the need for remediating UST waste, the DOE Office of Technology Development created the Underground Storage Tank Integrated Demonstration in 1991. Planning for the remediation of DOE's radioactive waste tanks has also been the responsibility of the DOE Offices of Waste Management and Environmental Restoration. As part of its "New Approach to Technology Development", DOE created the Radioactive Tank Waste Remediation Focus Area in 1994 to integrate and coordinate the tank waste remediation technology development efforts throughout these Offices. The mission of the Focus Area is to develop and apply technology systems, using an integrated approach, to safely and efficiently accomplish tank waste remediation across the DOE complex. The near term goal of the Tank Waste Focus Area is to demonstrate the ability to retrieve and treat liquid tank waste by June, 1997. The ultimate goal is to demonstrate and deliver safe and cost-effective solutions that meet regulatory requirements and are acceptable to the public and other stakeholders. The Tank Waste Focus Area also strives to develop industrial partnerships and to encourage the commercialization of the technologies developed. To meet these goals, DOE has established a Focus Area management team that brings together technology developers (from the Office of Technology Development) and representatives of the organizations that will manage the remediation (the Offices of Waste Management and Environmental Remediation). DOE has also selected an "Implementation Team Lead Organization" to lead the technology development in the Field. For the Tank Waste Focus Area, DOE's Richland Operations Office, in collaboration with other Operations Offices, has been designated as the Lead Organization. Technical leadership is provided by the Pacific Northwest Laboratory, teamed with other National Labs.

The Tank Waste Focus Area has focused its technology development on those fields where currently available technology does not meet the technical and safety requirements for tank waste remediation. Efforts are also being conducted to enhance baseline technologies and develop technologies that may be needed should existing technologies prove inadequate. The Focus Area seeks to integrate the development activities across DOE's EM Offices so that site specific problems are addressed from a national perspective. The fundamental problem areas that are shared by most of the participant sites and have been designated as sub-program elements are: Structural Integrity and Waste Analysis; Waste Dislodging and Conveyance; and Waste Pretreatment and Processing. Characterization of tank wastes has historically been

very expensive, and has failed to obtain representative data for many tanks. The Tank Waste Focus Area is developing sensors that will enable in-situ waste analysis and minimize the handling of samples during laboratory analysis, alleviating the safety concerns associated with handling. To assist in planning for retrieval of the waste from the tanks, technologies to map the interior of the tanks are being developed. In collaboration with the Robotics Crosscutting Program of OTD, the Tank Waste Focus Area is developing remotely controlled waste retrieval systems, that are capable of dislodging the sludge and saltcake fractions of the wastes and conveying the dislodged waste out of the tanks. These devices are designed to fit through access ports as small as 12 inches in diameter, and work in highly radioactive environments. Waste separation techniques will separate tank wastes into low-level, TRU and high-level fractions, thereby significantly reducing the volumes of high-level wastes requiring costly treatment and disposal. Low-level waste (LLW) treatment technologies will reduce waste volumes and produce waste forms which are chemically and physically durable. The following sections will discuss each area in some detail.

STRUCTURAL INTEGRITY AND WASTE ANALYSIS

Safe remediation of high-level waste from tanks is impossible without knowledge of waste's chemical and radioactive composition, and basic physical parameters. A goal is to obtain the majority of this information on an in-situ basis. In addition, information is needed on the physical location of the waste surface, the tank walls, and any obstructions (e.g., risers) within the tank. This requires the development of innovative methods for placement of sensors into the tanks and their subsequent positioning within the tank.

Chemical Speciation

The waste analysis projects are focused on developing technologies which identify the chemical speciation, moisture content, and rheological properties, such as density and viscosity, of the waste. Tank waste characterization currently involves removing core samples from each tank and subsequently evaluating these samples in a hot cell. Retrieval and analysis of a single core sample can take up to six months and cost more than a million dollars. In-situ characterization technologies will significantly decrease the cost and time required for analysis. In addition, in-situ methods will reduce worker exposure and be much safer than any methods which require direct handling and transportation of high-level waste. In-situ analysis is also expected to better measure the characteristics of the waste as it exists in the tanks.

Chemical speciation analysis methods currently being developed include spectrographic methods, such as Laser Raman and infrared. One method analyzes the Raman shifts in scattered light to identify and quantify chemical compounds, while the other utilizes vibrational spectroscopy in the near-infrared (NIR) region to detect moisture levels. While Raman spectroscopy will not replace regulatory approved analytical methods, its development as a screening technique for both ex-situ and in-situ characterization offers a significant reduction in time, cost, and secondary waste generation, minimizes radiation exposure to personnel, and provides a valuable guide to the sampling and analysis of core waste material. Remote measurements in harsh chemical and radioactive environments are made possible with the use of rugged fiber optic cables and probes, which transmit light to the waste and the signal to the spectrograph.

In-situ characterization of core sections will be achieved with the use of a Raman fiber optic probe fitted inside a cone penetrometer. Such a probe is being developed at the Lawrence Livermore National Laboratory. Another potential in-situ use is for chemically profiling the waste surface inside a tank. The feasibility of applying laser Raman spectroscopy to tank waste characterization is being demonstrated with both non-radioactive (cold) and radioactive (hot) tests. Analyses of pure materials and surrogates show that Raman scattering can be used to identify major waste components, including ferrocyanide, ferricyanide, nitrates, and nitrites. In addition, a remote fiber optic probe was installed in a radioactive-test cell (hot cell) at Hanford and is being used to record Raman spectra of real tank waste material. Two hot cell campaigns provided operational experience and spectral data for tank waste materials, some of which have significant fluorescence. Stable, high powered laser diodes (785 nm emission) were ordered as a means to mitigate the fluorescence. A 0.4 watt, high power doubled NdYAG laser was successfully integrated into the cold Raman test system. Probe testing was completed with four fiber optic

probe designs including two commercially available probes. The result indicated that in-line filters were effective in reducing silica Raman noise and that probe field-of-view is a strong sensitivity issue with slurry materials.
A non-contact, imaging Raman spectrometry system for the rapid chemical screening of underground storage tank wastes is also being developed by the Lawrence Livermore National Laboratory. The basis of this system is an optical field-of-view Raman surface scanner coupled to an imaging spectrograph with a two-dimensional array detector that provides spatially resolved chemical information about the tank waste material. The spatial data will be used to provide a chemical map of the waste surface. Fiber optics will be used to provide the remote interface between the tank waste environment (i.e., either hot cell for core samples, or in tank for in-situ waste analysis) and external instrumentation. The non-contact imaging scanner optical head is being designed to be compatible with both existing Raman hot cell systems at Westinghouse Hanford Company and commercially available units. The potential for using the non-contact imaging system with other spectroscopic techniques is being explored.
A prototype, laboratory bench-top model of the non-contact imaging Raman core scanner has been constructed and its utility in the chemical characterization of mock core samples has been demonstrated. The ability of the system to characterize the two-dimensional surface of a sample has been shown. A comparative study between the single point, contact fiber optic Raman probes and the laser scanning, non-contact imaging Raman spectroscopy has been performed. Factors considered in the comparative study were: available laser power at the sample, the effect of distance from the sample surface to the collection, the effect of background lighting, the interference from background Raman scattering, and other method-specific interference. The detection limit of ferrocyanide was established for each technique. The non-contact method proved to be more sensitive by an order of magnitude (resolution of 0.01 weight percent ferrocyanide).
The Westinghouse Savannah River Company at the Savannah River Technology Center has developed a Near-Infrared (NIR) spectroscopy system with a fiber optic probe, to measure the moisture content of the tank waste, an important safety parameter when ferrocyanide is present. The moisture system is being developed for both hot cell and direct in-situ waste applications. NIR spectroscopy is a well established technology, with many commercially available components and modules. Water normally interferes with IR analysis due to its strong optical absorption bands located around key vibrational absorption bands. With the NIR system, the water concentration of the tank waste is determined from the optical spectra back-scattered from the waste surface. The moisture-sensing system uses a calibration model developed with partial least squares fit analysis from standard materials having known moisture content. The spectra from unknown samples are then processed using this mathematical model to produce a water content value.
The initial testing of this moisture sensing method with tank simulant materials has been completed. The calibration model was built and tested with four waste tank simulants that are chemically and physically representative of the tank salt cake and slurry mixtures found in Hanford's underground storage tanks. The NIR system correctly predicted moisture within a 5 wt% error for all four tank materials. The errors were reduced to 2 to 3 wt% when the composition of the samples closely matched those used to build the model. These moisture measurement errors are well within the requirements to establish the safety status of a waste tank containing ferrocyanide materials.
Surface Topography
Knowledge of the topography of the waste in the tanks is needed prior to the deployment of in-situ sensors and during the retrieval operations. Two methods are being developed to address that need. The laser range finder (LRF) will provide rapid but coarse mapping of the tank interior. A structured light technique will be used for detailed mapping of the tank interior. Structured Light and Laser Range Finder technologies are well established and are based on relatively simple concepts. Essentially, the hardware consists of a laser, a receiver (such as a camera), and data processing equipment.
For the structured light system, the position and direction of propagation of the laser beam is known and controlled. The camera shows the two-dimensional projected position of the beam on the surface to be mapped. Simple trigonometric calculations are performed to determine the position of the surface. In comparison to the Laser

Range Finder, the structured light approach is slower, but gives a much finer resolution in all three dimensions. The Structured Light System has been demonstrated to perform with an accuracy and repeatability of better than 0.5 inch in field testing. It can be used to map the interior of the tank, including the surface contour, the condition of the walls and dome, and the location of in-tank hardware. The Structured Light Surface Mapping System will be deployed through either single or multiple risers. In the case of multiple riser deployment, the laser and the camera are inserted in separate risers, giving a very accurate and efficient "pitch and catch" set-up. For those cases where it may be necessary to deploy the entire system through a single penetration, the system will be capable of being deployed through a single 4" riser, albeit with some loss of accuracy and detail. In 1992, a surface mapping system based on the structured light approach was deployed at Fernald to map waste surfaces in the K-65 silos. The system contributed significantly to the successful completion of a CERCLA Removal Action Milestone in December 1993. The successful field deployment of structured light technology at Fernald made it highly desirable to both commercialize the technology and transfer it to the Hanford site.

The Laser Range Finder has a lower resolution than the structured light system, but is capable of providing near real time images. These capabilities qualify it for use in supervisory control and/or periodic surveillance. The laser range finder uses a laser to measure the distance to an object. Given the distance to this object in a known (controlled) direction, the position of the object in space is easily calculated. In each of these cases, from the measurement of multiple points, a three dimensional image of the object and its position can be determined. The Laser Range Finder is based on an existing piece of commercial hardware and appears to be well suited for use in tanks. Data fusion capability is being developed to combine data from the Laser Range Finder and Structured Light systems to provide an integrated map, which will aid in path planning and collision avoidance. Each of these systems will be radiation-hardened to allow for longer in-tank time and therefore, higher resolution. Along with the hardware, software is being developed to aid in data acquisition and control.

Deployment Platform

These characterization and surveillance instruments will be deployed and positioned within the tanks using a remotely operated robotic arm. This Light Duty Utility Arm (LDUA) (Fig. 2) will operate as a versatile platform in a high-level waste environment, and therefore will be hardened against the hostile elements (radiation, caustics, water vapor, etc.) present in the tank surroundings. The LDUA will be an integrated robotic deployment system to perform inspection, surveillance, waste analysis, and small-scale retrieval tasks in underground storage tanks. The LDUA system will provide a mobile multiaxial positioning system that will access the tanks through existing 12 in. diameter riser penetrations located in the tank dome. Other ancillary system equipment will be deployed through existing 4 in. diameter riser penetrations. The system provides the capability to deploy remotely operated end effectors at multiple elevations and positions within the tank using a robotic manipulator arm mounted on a telescoping mast. The arm will provide seven degrees of freedom with a 13.5 foot reach. The LDUA has been designed and is currently being fabricated by SPAR Aerospace, the Toronto-based company known for constructing the Space Shuttle Arm. The support system design has been completed and a trailer is being outfitted for field use. Most of the control and data acquisition systems have also been procured. The trailer instrumentation will be installed and tested, and the first LDUA system completed by the fall of 1995. Currently, in addition to the LDUA demonstration at Hanford planned by the Tank Waste Focus Area, LDUA systems are being obtained for use at Idaho and Oak Ridge.

Fig. 2.

Leak Detection

An acoustic emission method is being developed to detect and locate leaks in underground pipeline systems. This technology will be employed also as a process monitoring tool during the tank waste retrieval campaign. Leaks in underground pipeline systems can be located using the characteristic soundwaves generated by the flow of fluids (either liquids or gases) through a hole. The difference in primary or secondary line pressure and the environment causes flow into the secondary line or the environment (depending upon the location of the leak). Sound-sensitive sensors placed along the entire length of the pipeline transform sound (mechanical)

energy from the leak to electrical energy. These electrical impulses are evaluated by a signal analyzer to determine the distance of the signal source from a known reference point. Various types of acoustic sensors intended for use in the proposed system have been tested to determine their anticipated life expectancy when subjected to radiation fields. This established a service life cycle for sensors attached to radioactive waste piping.

WASTE DISLODGING AND CONVEYANCE

Unlimited sluicing, adding large quantities of water to suspend solids, is the baseline method for sludge removal from DOE's tanks. However, since the consistency of the waste varies from low viscosity liquids to peanut butter-like sludges to hard, concrete-like solids, this process may not be capable of removing all of the material from the tanks. Also, the resulting dilution increases the volume of waste significantly. Most importantly, dilution may increase the risk due to the existing and potential leaks of hazardous and radioactive liquids from corroded and deteriorated tanks into nearby soils and groundwater.

The Waste Dislodging and Conveyance task of the Tank Waste Focus Area is developing tools for the removal of materials from these tanks. The working tools and removal devices being developed include rubblizing devices, cutting and extraction tools, water and air jets, grit blasting devices, waste conditioning devices, transport and conveyance devices, suction devices, monitoring devices, and various mechanical devices for the recovery or repair of waste dislodging and conveyance tools. For some retrieval operations it may be necessary to add small amounts of water to facilitate waste dislodging and removal. However, systems are being optimized to minimize the amount of water added to tanks, and to position the conveyance equipment deployed with the dislodging equipment to remove solid waste and free liquid as promptly as possible. Dislodging and conveyance tools will ultimately be deployed as end effectors on a remotely operated, articulated arm called the Long Reach Arm, being developed by the Office of Technology Development's Robotics Crosscutting Program.

The systems being developed use water jet technology to dislodge the waste while simultaneously conveying the waste and the used water to the surface by pneumatic or air conveyance systems. The deployment of high-energy and low flow-rate tools with concurrent suction will minimize secondary waste. Since no additional stress is placed on the structure of the tanks, these technologies are ideally suited for use in older tanks which may have weakened or damaged structures. Currently, two such tools are being demonstrated. The scarifier uses ultra-high-pressure water jets (40,000 to 50,000 psi) in a dislodging tool coupled with a pneumatic transport system. The miner uses medium pressure water jets (5,000 to 11,000 psi) coupled with jet pump transport. These waterjet technologies have been used to meet compliance milestones in the TPA at Hanford. The program has demonstrated a half-scale high-pressure scarifier at a waste removal rate of 2 ft3/min., half that of the target full-scale retrieval rate of 4 ft3/min.

The Tank Waste Focus Area, in cooperation with the Robotics Crosscutting Program, is developing an Integrated Hydraulics Testbed at Hanford. This system will characterize aspects of waste dislodging and conveyance processes, evaluate process equipment performance, and address integration issues associated with deployment of dislodging and conveyance systems by a manipulator arm. A fundamental understanding of the mechanics of dislodging and mobilizing waste will also be developed . The Testbed will enable longer duration, multiple-pass tests on a large area of simulated waste. The components can be deployed over all three dimensions to facilitate modeling of the behavior and response of various deployment control systems. A final test plan has been completed and the gantry robot for the testbed has been received by Hanford, and mechanics, hydraulics, and simulant testing will begin in 1995.

WASTE PRETREATMENT AND PROCESSING

The treatment and disposal of DOE's high level waste (HLW) is expected to be costly and presents many difficulties due to the high radioactivity, the variability in composition, and the number of different chemical components. A primary objective of DOE's technology development for tanks is to remove radionuclides and minimize the volume of the HLW final waste form (e.g., glass). The remaining LLW will then be eligible for near-surface disposal. Since the radionuclides constitute only a small portion of the waste, separations technologies offer promise to accomplish this goal. Performance of new and improved waste forms for the solidification and

disposal of LLW will also be evaluated.
Separations
In retrieving, processing and disposing of these wastes, separation technologies are the primary means to concentrate the radionuclides so that the volume of high level waste to be vitrified is minimized and the activity of the residual waste is reduced to make it eligible for near surface disposal. It is estimated that the volume of waste to be vitrified can be reduced by a factor of between 10 and 200 using advanced separation technologies. Removal of the radionuclides also reduces shielding requirements and makes further processing of the hazardous components of the waste safer, more efficient, and less expensive. The Tank Waste Focus Area is demonstrating separations systems and utilizes the technologies being developed in the Efficient Separations Crosscutting Program. Technologies will be demonstrated for the treatment of HLW supernate, sludges and acidic wastes.
The presence of radioactive cesium and strontium (e.g., 137Cs) is a major source of beta and gamma activities in high level wastes found in underground storage tanks. Ion exchange is one of the proposed technologies to remove cesium and strontium from these wastes. Due to the somewhat unique chemical characteristics of the tank wastes (i.e., extremes of pH and high ionic strength), most common ion exchangers are not suitable for this processing. The Savannah River Site (SRS) has developed a resorcinol-formaldehyde ion exchange resin for cesium removal. The resin has been demonstrated to have a high capacity for removing Cs and to be compatible with the extreme characteristics found in the Hanford and SRS tank wastes. In particular, the resorcinol-formaldehyde resin is found to have 10 times the capacity of the Duolite CS-100 phenol-formaldehyde resin. Currently, the resorcinol-formaldehyde resin is being demonstrated in column mode with a tank waste simulant. Engineering-scale tests will also be run on the Skid-mounted Ion exchange Demonstration (SKID) unit at the SRS. The resulting data will be critical in determining the optimum processing conditions (e.g., flow rate, elution volume and eluent). The data will also aid in the design of a modular, mobile ion exchange system for the extraction of cesium from alkaline tank waste, such as will be used for a Cesium Removal Demonstration. Recently concluded radiolysis studies on the resin suggest that the resin is stable up to a dosage of 5 x 108 rad, and a higher dosage causes some loss of extraction efficiency in a high pH environment, but not in water.
The Tank Waste Focus Area is planning a Cesium Removal Demonstration in 1996, using the high salt content supernate stored in the Melton Valley Storage Tanks (MVST) at Oak Ridge. The project will demonstrate the use of a modular, mobile, ion exchange system, using existing facilities for secondary containment, the offgas system, and support functions such as utilities. Decontamination for contact maintenance and the ability to transfer of the unit to other DOE facilities will also be demonstrated. The ion exchange material will be chosen based on its effectiveness in laboratory batch tests and small scale column tests using both simulated and actual waste supernates.
If the content of Transuranic elements (TRU) in TRU waste streams can be reduced to below 100 nCi/g, the waste can be classified as non-TRU. The TRUEX process is a solvent extraction process developed to separate the TRU components from aqueous solutions. Use of the TRUEX process may also reduce the volume of high-level waste, resulting in high cost savings during disposal. The Generic TRUEX Model (GTM) is a user-friendly computer software that models and predicts the TRUEX extraction behavior, calculates TRUEX flowsheets, and estimates space and cost requirements for installation. The GTM runs on a personal computer (IBM-compatible or Macintosh) using commercial spreadsheet software. The objective of the current task is to validate and refine the GTM's ability to design flowsheets for specific feeds and process goals and to predict extraction behavior of feed components and potential processing difficulties. This task will also discover and identify R&D needs for preparing TRUEX for broader implementation. The TRUEX processing of actual HLW and TRU wastes at Oak Ridge National Laboratory (ORNL) will validate the GTM.
The Tank Waste Focus Area has initiated a technical interchange between the US DOE and the Commissariat l'Energie Atomique (CEA) in France continues to examine the technical problems associated with the removal of actinides from acidic waste streams. In addition to the acidic wastes at INEL, large volumes of acidic wastes may be produced if Hanford, Savannah River, and Oak Ridge pursue acid dissolution of their sludges. This exchange is focusing on the French DIAMEX process which uses diamides as extractants to remove the actinides. This process will be evaluated

against comparable US extraction technologies, such as the TRUEX process.
Pretreatment
Fine particulates suspended in liquid wastes have a harmful effect on radioactive waste treatment processes such as ion exchange. These effects include ion exchange bed fouling, reduced removal efficiencies, higher shielding requirements, premature breakthrough of radioactive particulates, and interference of downstream waste disposal processes. Treatment processes for aqueous high level wastes are expected to require pretreatment to separate insoluble solids from the liquid. Crossflow filtration has been demonstrated as an efficient method to remove insoluble solids. The Tank Waste Focus Area is integrating laboratory scale filter testing with actual full scale operation to permit correlation between units and comparison between simulant and actual waste. The filtration needs of the proposed pretreatment processes for the underground storage tank wastes at Hanford and Oak Ridge will also be evaluated, to determine the best applicable filtration technologies.
The only tanks at Oak Ridge that comply with the Federal Facility Agreement between DOE, EPA and the Tennessee Department of Environment and Compliance are the double shell tanks. Evaporators offer the possibility of removing water from the waste in the double shell tanks, so waste from noncompliant tanks can be moved to compliant tanks. The alternative to evaporation would be to solidify the waste in noncompliant tanks, which produces a waste form for which there is currently no permitted storage. The Tank Waste Focus Area is developing an evaporator to remove excess water from a 4 to 5 molar sodium nitrate solution contaminated primarily with 137Cs and 90Sr at activity levels of about 8 mCi/l and 0.2 mCi/l, respectively. The project will evaluate the relationship between the performance of pilot scale evaporators processing surrogate solutions, and full scale units processing actual waste. It will also determine processing capabilities (such as decontamination factors), identify potential operating/maintenance problems for remotely operated evaporators, and evaluate the feasibility of decontaminating evaporator systems for hands-on maintenance and the possibility of moving the unit to another site.
An evaporator demonstration will be conducted at ORNL in late 1995 to process 25,000 gal of liquid LLW supernate from the Melton Valley Storage Tanks. The demonstration will return 19,000 gallons of concentrated waste to the tanks, and 6,000 gallons of condensate to the lab's process waste system. Technical specifications for the evaporator system were developed and the contract was awarded to Delta Thermal in October, 1994. Modifications to the System Safety Analysis for the facility that will house the evaporator have begun. The Oak Ridge Engineering Division has begun work on a feasibility study to define requirements for the installation of the evaporator system.
Solidification
The reference waste form for LLW at most sites has been a cement-based grout; however, this increases the volume requiring disposal, does not effectively retain certain contaminants, and has questionable long-term stability. Alternate waste forms must therefore be tested and evaluated. The Tank Waste Focus Area is testing ceramic waste forms for their ability to decrease waste volume, reduce processing and disposal costs, and enhance performance and long-term stability. Hanford has recently changed its reference LLW form from grout to glass.
The Nitrate To Ammonia and Ceramic (NAC) (Fig. 3) process involves chemical reduction of the nitrate contained in the mixed (hazardous and radioactive) tank waste and solidification of the waste. In this process, aluminum metal is used to reduce the nitrate, and silica is added to produce an alumina-silica ceramic. The nitrate-free ceramic product is then calcined, pressed, and sintered into a final waste form that is expected to pass leach tests. The NAC process can achieve concentrations of nitrate below drinking water standards. It is expected that radioactive species such as plutonium and strontium will enter the solid ceramic phase during the reduction of the nitrate anion. Sodium will be retained in a nepheline ceramic phase. In the process, radioactive contaminated scrap aluminum from various DOE sites could be shredded and used as feed to the NAC reactor. An additional advantage of the process is a reduction in the final waste volume of 55 to 75 vol% (as compared to a 30% to 40% volume increase by grouting). Laboratory experiments have demonstrated the feasibility of this process. The process was initially developed by the Oak Ridge National Laboratory, and Florida International University (FIU) is working with ORNL on further development.
Fig. 3.

CONCLUSION

The Tank Waste Focus Area, in collaboration with the National Laboratories and with university and industrial partnerships, is developing, demonstrating, and incorporating innovative technologies into DOE tank remediation baseline planning. Current demonstration activities are designed to meet regulatory milestones and ensure that proven technologies are available to meet the remediation requirements throughout the DOE complex. Where possible, demonstrations are progressing from non-radioactive testing using surrogate wastes to the use of lower radioactivity samples, e.g., at the Oak Ridge site, with eventual demonstration in the harsh, high radioactivity environment of the wastes at Hanford, INEL, and Savannah River. Besides solving DOE underground storage tank problems, the tools and technologies being developed and demonstrated are designed to facilitate their transfer to the private sector. Most of the technologies being evaluated have private sector partnerships and commercial applications of these technologies are being explored. At least four improved technologies/systems will be made available for transfer to industry during FY 1996. By bringing together the technology developers, customer organizations, and stakeholders, the Tank Waste Focus Area is establishing a streamlined development program that will provide cost effective solutions for use throughout DOE and at private facilities.

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59-1

SEGREGATION OF SOLID LOW LEVEL RADIOACTIVE WASTES AND UNCONDITIONAL RELEASE FROM CHALK RIVER LABORATORIES

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ABSTRACT

Currently in Canada, a number of programs have been designed to reduce the volume of Low Level Radioactive Wastes (LLRW) that must by stored in licensed radioactive waste management facilities. These wastes arise from operation of nuclear generating plants and research facilities, medical and industrial uses, and the decommissioning of nuclear facilities. The waste volume reduction programs are based on exempting certain LLRW from further licensing upon transferral for disposal, or on segregating wastes into either "releasable" or LLRW categories based on criteria that have obtained prior regulatory approval.

The design and construction of a pilot-scale study is now underway to demonstrate the cost-effectiveness and technical feasibility of segregation and unconditional release of solid wastes from AECL Research Chalk River Laboratories (CRL). Prior to program design at CRL, a study was conducted to examine three cases of de minimis/LLRW Segregation Programs that have been successfully implemented in Canada. For each case, the release criteria adopted, the procedures and instruments used in segregation or release, and the nature of the regulatory approval process were studied. Based on these cases, waste scanning procedures and a pilot plant layout were proposed for CRL.

REGULATORY POSITION

The nuclear regulatory body for Canada is called the Atomic Energy Control Board (AECB). The current position of the AECB on the exemption of radioactive materials from further licensing upon transferral for disposal is summarized in Regulatory Document R-85. R-85 states that when the circumstances of such disposal are considered to represent a negligible, or de minimis risk, expenditure of additional regulatory resources, or continued licensing of the material, is not justified. The AECB uses a de minimis dose of radiation to individuals of 0.05 millisievert in a year for deciding such exemptions on a case-by-case basis, provided that the radiological impact will be localized and the potential for exposure of large populations is small. (AECB, 1989)

The primary de minimis dose criterion of 0.05 millisievert per year follows from the acceptance of a corresponding de minimis health risk. The choice of a de minimis dose criterion of 0.05 millisievert to an individual per year represents an

extrapolation from a fatality risk from cancer of 10-6. The secondary requirement, that the potential for exposure of large populations be small, is intended to restrict undue reliance on dilution as a means of attaining compliance with the de minimis dose criterion (AECB, 1989).
R-85 is applicable principally to the disposal of solid low-level radioactive wastes. Exemption from further licensing of miscellaneous liquid wastes is determined on a case-by-case basis. R-85 does not apply to liquid or gaseous emissions currently designated as effluent discharges in the operating licences issued by the AECB for various nuclear facilities.

RESULTS OF CASE STUDIES

Case 1: Bruce Nuclear Power Development (BNPD)

In March of 1987, AECB approval was granted for the segregation of active and inactive solid wastes from Bruce Nuclear Generating Station (BNGS) "A" using a waste bag monitor. This approval was granted prior to the issue of R-85, but represents nonetheless a de minimis-type approach to waste disposal in that the application for exemption was based on an assessment of radiological dose calculated from pathways analysis.
The conditions of the approval were:
1. the value of 11 kBq gross gamma activity, as measured with a bag monitor, was to be used to segregate active and inactive waste
2. the inactive waste must remain within the property boundaries of BNPD
3. regular random selection of sample bags and survey of their contents in detail as a quality control check of the process.
Bruce NGS "A" conducted a test operation in 1985/86 to demonstrate the viability of waste categorization using a Waste Bag Monitor. The test operation used National Nuclear Corporation's WCM-11 waste bag monitor, which contains 4 large plastic scintillators. The alarm setpoint used was 11 kBq per waste bag. Over 48,000 waste bags from all station Zones were surveyed. When handfrisked on the exterior of the bag for contamination, none of the bags classified as inactive by the waste bag monitor contained detectable contamination. The volume of waste classified as radioactive was reduced by 27%, or 360 m3/a. A further check on the efficiency of the waste bag monitors was conducted by incinerating all wastes classified as inactive. Just prior to incineration, all wastes were again handfrisked on the exterior surface of the bag and no contamination was found. The incinerator ash was compared to ash from inactive incinerations prior to the startup of the program, and no differences in radionuclide levels were found (Lemkay, 1986).
A modified waste handling procedure was proposed. Waste bags were initially screened with a medium range gamma meter. Bags with dose rates detectable on that meter, or bags that had fields greater than 10 mSv/h, were not put in the waste bag monitor. Waste bags that contained less than 11 kBq/bag according to the waste bag monitor were classified as inactive. Bags of inactive waste were incinerated. Data bases were used to record waste bag data and the inactive incinerator ash data. The data were summarized monthly to ensure that the residual radioactivity of the inactive waste remained at or below a level of 0.3 Bq/g (Lemkay, 1986).
Other data submitted to the AECB prior to approval included a simple public exposure pathways analysis and a method to infer the concentrations of pure beta-emitting isotopes (primarily 3H, 14C, and 90Sr), which would not be detected by the standard screening procedures (Lemkay, 1986). The calculated maximum exposure to a member of the public was 20 mSv/a, which is below the regulatory criterion of 50 mSv/a.
This program for the segregation of active/inactive solid wastes at BNGS "A" was still in place as of December 1, 1994 (J. Hudson, personal communication).

Case 2: Darlington Likely Clean Waste Management Program

Prior to the fall of 1992, all wastes originating in Zone 3 of the Darlington Nuclear Generating Station (DNGS), which consists of areas that contain radioactive systems and material, were treated as contaminated (Ontario Hydro, 1990). This procedure resulted in large volumes of materials, which were in fact at least 75% inactive, being sent to the BNPD Radwaste Site for storage as low level radioactive waste. In the fall of 1992, the Likely Clean Waste Management Program obtained regulatory approval and was initiated to permit free release of Zone 3 Likely Clean Waste to offsite landfill or recycling if it displayed no activity above background.
An important distinction between the Likely Clean program and the AECB criteria for exemption from regulatory control is that AECB R-85 allows for a de minimis dose rate of 0.05 millisievert per year to an individual, while wastes designated as

clean by the Likely Clean program are uncontaminated to the limit of detection of the instruments and procedures in use (Hudson, 1993).
The first action in implementing the program was a source term analysis to establish the radionuclides potentially present in the waste streams. This allowed the selection of correct instrumentation for monitoring, and defined areas of the Station that were excluded from the collection of Likely Clean waste due to a higher risk of generating contaminated wastes. Periodic review of the source term characterization will be conducted.
Workplans, flowsheets, procedures and training materials were prepared for the Program, and staff dedicated to waste collection and processing/monitoring. A waste collection/segregation system for Likely Clean waste was set up in selected areas of Zone 3, and station staff were trained to include only items for which they knew the radiological history and which were expected to be free of contamination. At waste collection stations, waste bags are surveyed with a gamma meter, monitored for tritium and barcoded. A barcode reader/transaction manager is used to record each bag's barcode, including its point of origin, date and time of collection and name of the waste collector. Bags with gamma activity in excess of 10 mSv/h are rejected and enter into the active waste stream, and an investigation is conducted as to their point of origin and reasons for exceeding the criteria for Likely Clean.
A separate room was organized for the handling of Likely Clean waste, with a low background radiation level of approximately 50 nGy/h. All Likely Clean bags are deposited in the Clean Sorting room and stockpiled for 24 hours to allow any tritium in the wastes to reach an equilibrium air concentration inside the bag. The concentration of tritium inside the bag is then checked with a Triton tritium monitor, and any waste bags with tritium levels in excess of 0.5 Maximum Permissible Concentration in air , or MPC(a), are monitored in the bag monitor and then redirected to the active waste stream. Bags with <0.5 MPC(a) tritium are placed in the waste bag monitor to determine eligibility for hand sorting. The monitor reads the barcode, measures total and specific activity, and instructs the operator to redirect the bag to the active waste stream if the specific activity is greater than 74 kBq/kg. Each eligible waste bag is opened on a stainless steel sorting table that is equipped with a HEPA air filtration system. Wastes are handfrisked inside and out with an NE Technology CM7A contamination meter, which can simultaneously monitor alpha and beta/gamma contamination. The alarm setpoint on the CM7A is set at 29.6 Bq/100 cm2 for 137Cs. Any items with internal surfaces not directly measurable, materials displaying a radiation symbol, or items with activity above background levels are treated as active waste. Hand frisking is done at a maximum rate of 3 cm/s in order to ensure a detection capability of 55.6 Bq/100 cm2 at scanning speed. Handfrisking is limited to a maximum of two hours per person per day, and personnel must pass a written examination and practical training before they are qualified to work in the program. Items displaying no activity above background are sorted into recyclable or landfill categories. Landfill wastes are compacted and rebagged. A final quality assurance check is performed using a waste bag monitor with the setpoint at the intsrument's current Minimum Detectable Activity (MDA), which is usually about 240.5 Bq/bag for the 10 kg compacted bags. The landfill wastes are then released to offsite landfill. This program was still in operation as of December 1, 1994 (J. Hudson, Ontario Hydro, personal communication).
Case 3: Unrestricted Release Of Decommissioning Wastes, AECL CANDU Decommissioning Group (CDG)
AECB approval for unrestricted release for wastes arising from the decommissioning of Tunney's Pasture in Ottawa, Ontario, a former radioisotope production facility, was granted in March of 1992. Release criteria were based on unconditionally exempt levels for identified radionuclides extracted from the IAEA Working Document "Exemption from Regulatory Control Recommended Unconditional Exempt Levels for Solid Radioactive Materials"(IAEA, 1991). These levels are indicated in Table I.
TABLE I
The unconditionally exempt levels were established by the IAEA to apply to solid materials irrespective of the use to which they are put or to their destination after control has been relinquished. The levels are based on a review of the results of the exemption studies conducted on landfill disposal and incineration of wastes, and the recycle and reuse of ferrous and non-ferrous metals and concrete. These studies identified limiting radionuclides for detailed consideration in scenarios and exposure pathways. For the purpose of exemption, the level of individual dose

due to the sum of exposures from exempt practices of some tens of microsieverts per year was considered trivial. The dose to an individual from each exempt practice was to be in the order of 10 mSv/a. The general allowable surface contamination levels were 1.0 Bq/cm2 for beta/gamma emitters and 0.2 Bq/cm2 for alpha emitters. The averaging area was 1 m2, or the total surface area for smaller objects. A Ludlum scaler with an HP260 pancake detector was used for measuring small surface areas. A Berthold LB122 with a gas proportional detector was used for measuring large surface areas. As well, a Ludlum 2350 with a gas proportional detector was used. Loose contamination was detected from smears using a scaler with a shielded beta/gamma detector and another scaler with an alpha detector (J. Stapleton, AECL CANDU Decommissioning Group, personal communication).

In January of 1994, AECB approval was granted to the AECL CANDU Decommissioning Group for the segregation of radioactive wastes stored in a number of 28 m3 containers at Gentilly-1 Nuclear Generating Station in Quebec. These wastes were comprised mainly of non-compactible waste, primarily steel pipe, that were wrapped in plastic, and compactible wastes, primarily protective clothing and other consumable material. Wastes cleared for unconditional release according to the contamination limits agreed upon by the AECB underwent compliance inspection by an AECB inspector prior to release to offsite landfill. The compliance inspections involved one or more visits, sometimes unannounced, for verification of monitoring results, and were followed by approval to continue with offsite release without further inspection. However, metals cleared for unconditional release and destined for offsite recycling were to be stockpiled and verified by the AECB prior to all releases. For unidentified radionuclides, the release limits were <1.0 Bq/cm2 for beta-gamma emitters (137Cs equivalent), and <0.2 Bq/cm2 for alpha emitters (241Am equivalent). For identified radionuclides, the release limits were again based on values specified in the IAEA Working Document (IAEA 1991). Identified radionuclides known to be less radiotoxic, such as the pure beta emitters 14C, 63Ni, and 36Cl, were to be released to the approved levels stated in Table I (W. Gutzman, AECL CANDU Decommissioning Group, personal communication).

IMPLICATIONS FOR NEW PROGRAM DESIGN

The examination of these three cases has indicated key considerations for program design and for seeking regulatory approval. The major decision that has to be made prior to program design is whether the preferred strategy is to prove wastes are "clean" to the limit of detection of the instruments and procedures used, or to base waste segregation upon levels of radioactive contamination that meet de minimis risk and dose criteria or recommended unconditional exempt levels.

With the pending enactment of the Canadian Environmental Assessment Act, and a potentially larger role for public input and acceptance, a low-level radioactive waste program based on the "clean" principle may have greater chances of obtaining necessary approvals (Lipsett et. al., in prep). Another advantage of this approach is it's independence of release limits, the values of which may evolve through time and research. However, a disadvantage of this approach is that, as detection equipment and practices improve over time, distinguishing between measurable radiation levels due to radioactive contamination and those due to natural background levels (ie. due to naturally-occurring radionuclides, cosmogenic radionuclides, weapons testing fallout etc.) for some radionuclides in some waste forms may prove to be a major challenge (IAEA, 1993). As well, this approach does not explicitly define limits that may be used as worst-case source term values for performance assessment. One approach to modelling the impact of these practices may be to perform a dose assessment corresponding to the detection limit of the equipment and procedures used.

The advantages of establishing release levels based on the de minimis risk and dose criteria, or recommended unconditional exempt levels, are that the presence of variable background levels of radioactivity in wastes, particularly decommissioning wastes, will rarely pose a problem as background levels are usually far below unconditional exempt levels. The impact of unconditional release will be easier to model given that the levels will define worst-case source term values. Unconditional release of these materials means that they become eligible for offsite recycling/reuse or landfill, which eliminates incurring long-term liability for these wastes within a licensed site. The reduction in long-term liability results in a large cost saving, and resolution of the waste problem by the same generation that generated the waste. Some cost recovery for these programs may be possible through

the reuse/recycle route. However, consideration must be given to removing radiation/prior ownership symbols from reusable materials, and to defacing uniquely nuclear items for offsite landfilling. One disadvantage is that the release level approach may have a more difficult time gaining public acceptance.

Design and construction are currently underway at CRL for a pilot-scale program to segregate and unconditionally release wastes. This pilot program will be jointly undertaken by the AECL CANDU Decommissioning Group, AECL Waste Management Systems, and the CANDU Owner's Group. The goal of the pilot program is to prove that waste segregation is cost effective and technically feasible within the Active Area of the CRL site. The Chalk River site has been in use as a nuclear research and isotope production facility since the mid-1940's, so the possibility of the presence of a full suite of radionuclides, including alpha emitters and low-energy beta emitters, must be planned for. LLRW at Chalk River include active-area office wastes, research laboratory wastes, LLRW generated in reactor buildings, and decommissioning wastes. Careful consideration of the lessons learned in the three cases presented in this document has resulted in the decision to adopt a policy of unconditional release of materials based on criteria with prior regulatory approval. AECB approval has been sought to unconditionally release segregated wastes to Unconditional Clearance Levels consistent with the draft IAEA recommendations (K. Cotnam, AECL Research, personal communication). A program is currently underway to refurbish the CRL site, which will result in a large volume of building-emptying and demolition waste. These buildings have been exposed to up to 50 years of atmospheric fallout of radionuclides from cosmogenic sources and weapons testing, and therefore some of the wastes will have measurable radioactivity levels above the background one would expect from new construction materials. However, these radioactivity levels would be found on any buildings of similar age from this region of Canada because their source is atmospheric deposition. As a result, using the "non-detectable above background" approach would result in unacceptably high volumes of waste being handled as LLRW.

Waste handling procedures have been designed based on a combination of the approaches of the AECL CANDU Decommissioning Group and the Darlington Likely Clean Waste Management Program. A draft floor plan for the pilot can be seen in Fig. 1.

Fig. 1.

Personnel entering the pilot will first use gas-flow proportional hand and foot monitors to ensure that no contamination enters the facility through that route. Airflow into and out of the pilot will pass through HEPA filtration systems. After an education program provided by the pilot staff, waste generators will segregate the wastes at source into "Active" and "Likely Clean" bags. Wastes will be barcoded at their point of collection to allow tracking through the system. Likely clean wastes entering the facility will be stored for a period of time in a "Tritium Laydown Area" to allow any tritium in the wastes to reach an equilibrium concentration in the bag air. Each bag will be checked with a hand-held tritium monitor, and those bags that meet the pilot criterion for tritium, which is currently in the process of being finalized, will be eligible for further processing. The eligible bags will be placed in an initial Waste Curie Monitor with an alarm setpoint designed to ensure that active wastes are not inadvertently entering the facility, and to allow for barcodes to be entered into the pilot database. The wastes will then enter the segregating/monitoring room, which had been constructed with radiation shielding in the walls to ensure as low and as stable a background radiation level as possible. Waste bags will be opened onto an enamel-covered steel sorting table, where a gas-flow proportional hand scanner will be used to make near-contact measurements for alpha, beta and beta-gamma contamination. The alarm setpoints on the hand scanners will be determined to reflect the unconditional release criteria. Wastes will be sorted into two categories: wastes that meet the criteria for unconditional release, and wastes that will be handled as LLRW. Wastes suitable for unconditional release will be rebagged and barcoded, and a final check of radioactivity levels will be made in a second Waste Curie Monitor, which will also enter the new barcode into the pilot database to facilitate tracking and/or retrieval of the releasable wastes for compliance monitoring.

Pending AECB approval, the Waste Segregation Pilot will become operational in the first quarter of 1995.

CONCLUSIONS

A review of the Canadian case studies has indicated that LLRW waste segregation programs can result in significant reduction in the volumes of LLRW. The results of the pilot waste segregation program at Chalk River Laboratories will indicate whether a segregation program will be technically feasible and cost effective for the Active Area of the CRL site.

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AN ALTERNATIVE APPROACH TO THE RCRA "CONTAINED-IN" RULE FOR THE CHARACTERIZATION OF INVESTIGATION DERIVED WASTES

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ABSTRACT

Since 1990, the DOE/RFFO and EG&G have been discussing the management of IDW with EPA and CDPHE. These discussions have centered on the development of a comprehensive management program that would characterize IDW in accordance with regulatory guidelines and risk assessment methodologies, rather than using the RCRA "contained-in" rule.

The definition of RCRA hazardous waste is codified in Title 40 Code of Federal Regulations (CFR) 261. Under RCRA, the definition of hazardous waste focuses mainly on process wastes from a variety of chemical processes. However, in the strictest sense under RCRA, any material that is mixed with, derived from, or that contains any amount of RCRA listed contaminants also becomes a RCRA hazardous waste. Therefore, environmental media such as soil or groundwater that receives spilled, discharged or abandoned hazardous waste also becomes hazardous waste by definition, as long as the media contains hazardous constituents in detectable concentrations. The regulation of hazardous waste is primarily intended to prevent contamination of groundwater by regulating the storage, treatment and disposal of process waste and focusses on the prevention of using dilution as a form of treatment. Therefore, it is difficult to apply the definition of hazardous waste to environmental media that has received historical contamination, especially if the event occurred prior to the promulgation of RCRA regulations defining hazardous waste.

At RFETS, approval was obtained from the Region VIII EPA and CDPHE to characterize soil and sediment IDW using a risk-based determination, rather than using the Resource Conservation and Recovery Act (RCRA) "contained-in" rule, to determine if the IDW were hazardous waste. The impact of requiring IDW generated at RFETS to be classified as RCRA hazardous waste under the "contained-in" rule would result in the treatment, storage and disposal of a large volume of assumed hazardous waste. Specifically, over 4000 drums of IDW at RFETS had been generated from remedial investigations by the late spring of 1994, with additional investigations ongoing. The risk-based methodology developed for the characterization of IDW at RFETS is a

unique combinations of the traditional RCRA waste characterization process and an innovative risk assessment evaluation processes. This methodology allows IDW to be evaluated based on a review of associated chemical analytical data obtained during field investigations instead of assuming that all IDW contains process hazardous waste. Through the application of the risk-based methodology 79% of the drums evaluated were determined to be chemically non-hazardous (i.e, uncontaminated or containing very low concentrations of contaminants). Therefore, the material is exempt from RCRA waste management requirements and may be returned to the point of generation or disposed of as a non-hazardous waste.

INTRODUCTION

The Rocky Flats Environmental Technology Site is a DOE facility managed and operated by EG&G Rocky Flats, Inc. The primary mission of the Site, from its inception until the early 1990s, was to produce metal components for nuclear weapons. Both radioactive and non-radioactive wastes were generated during the research and production processes, and past waste handling procedures involved both on-site storage and disposal of wastes. In 1992, the primary mission of RFETS changed from weapons production to stabilization of the radioactive material remaining on site and environmental restoration.

In the late 1980's, the DOE/Rocky Flats Field Office (RFFO) began environmental investigations to determine the extent of site contamination. As a result of these investigations, by 1993 a backlog of over 4,000 drums of IDW had accumulated on site. Originally, the IDW generated during environmental investigations was to be stored of generation and disposed of in accordance with each Record of Decision (ROD) for the individual Operable Units (OUs). However, due to lengthy investigation period leading to each ROD, alternative means of handling the IDW were required. The alternative selected was to package the IDW and manage the drums according to appropriate waste handling requirements.

In March, 1994, DOE/RFFO received a warning letter from the CDPHE. The correspondence was a follow-up to an inspection that was performed of the soil and sediment IDW drums and subsequent review of the management protocol which had been established for the drums (CDPHE letter to DOE/RFFO, March 1994). Among the concerns expressed by CDPHE was that waste characterization had not been fully completed for IDW drums in 90 day storage areas. Many of the drums had also been stored in the 90 day areas more than the allowable time frame. The State specifically cited violations of 6 CCR 1007-3, Section 262, which requires a generator of a solid waste to determine if the material is a hazardous waste. The DOE/RFFO was advised by CDPHE to "take actions that will bring characterization, labeling, management, and storage of the soil and sediment IDW drums into compliance with Sections 262.11 and 262.34 and Part 100 of the Colorado Hazardous Waste Regulations." Inclusive to this effort was the preparation of "additional standard operating procedures that ensure completion of a properly conducted hazardous waste characterization for IDW containers that contain solid waste." To this end, the DOE/RFFO directed EG&G to initiate development of a procedure which could be used to properly characterize the soil and sediment IDW.

Purpose

This paper presents a procedure developed in 1994 by Woodward-Clyde Federal Services (WCFS) and EG&G to chemically characterize IDW using regulatory guidelines and risk assessment methodologies. The procedure is currently used to characterize chemical constituents of IDW only. A risk-based methodology for the characterization of radioactive constituents of IDW is under development.

Background

Since 1990, the DOE/RFFO and EG&G have been discussing the management of IDW with EPA and CDPHE. These discussions have centered on the development of a comprehensive management program that would characterize IDW in accordance with regulatory guidelines and risk assessment methodologies, rather than using the RCRA "contained-in" rule.

In December 1993, WCFS and EG&G began developing an operating procedure that would address the final regulatory disposition of IDW generated during environmental investigations at RFETS. EPA delegated their approval authority for this procedure to CDPHE. In April 1994, the operating procedure was approved by the State.

Investigation Derived Wastes

Environmental field investigations result in the generation of waste materials. These investigation derived wastes may include drilling muds, soil drill cuttings,

well development purge water, contaminated personal protective equipment (PPE), and solutions used to decontaminate non-disposable protective clothing and equipment. According to the EPA, "the management of investigation derived wastes must ensure the protection of human health and the environment and comply with certain regulatory requirements that are applicable or relevant and appropriate"(EPA 1991).

State and Federal Guidance Concerning IDW

The RCRA contamination assessment process requires that facilities with historical contamination perform investigations to thoroughly evaluate the nature and extent of the release of hazardous waste and constituents to the environment. These investigations, known as RCRA Facility Investigations (RFIs) include sampling of groundwater, soil, and any other potentially contaminated media. The RFI process generates drill cuttings and excess soil from sampling. Since the soil comes from a potentially contaminated area it must be contained and eventually disposed of.

The definition of RCRA hazardous waste is codified in Title 40 Code of Federal Regulations (CFR) 261 and 6 Colorado Code of Regulations (CCR) 1007-3. Under RCRA, the definition of hazardous waste focuses mainly on process wastes from a variety of chemical processes. However, in the strictest sense under RCRA, any material that is mixed with, derived from, or that contains any amount of RCRA listed contaminants also becomes a RCRA hazardous waste. Therefore, environmental media such as soil or groundwater that receives spilled, discharged or abandoned hazardous waste also becomes hazardous waste by definition, as long as the media contains hazardous constituents in detectable concentrations.

The regulation of hazardous waste is primarily intended to prevent contamination of groundwater by regulating the storage, treatment and disposal of process waste and focusses on the prevention of using dilution as a form of treatment. Therefore, it is difficult to apply the definition of hazardous waste to environmental media that has received historical contamination, especially if the event occurred prior to the promulgation of RCRA regulations defining hazardous waste.

During an environmental investigation, soil is displaced by sampling efforts, especially drilling, and the total volume may not be used for analytical samples. This soil, if removed from a potentially contaminated area, using a strict interpretation of the definition of hazardous waste, contains hazardous waste and cannot be returned to the location of origin and therefore must be contained. The impact of this interpretation of the regulations may result in a large volume of waste soil, known as investigation derived waste, that must be characterized and disposed. Waste characterization requires either using process knowledge or analytical results to determine the concentration of hazardous constituents. It may be difficult to use process knowledge to characterize potentially contaminated soil. It is also difficult to use process knowledge to confirm that soil is not contaminated. Regulating agencies will generally not accept process knowledge to characterize environmental media as uncontaminated. On the other hand, taking and analyzing samples from every container of soil can be extremely time consuming and costly. Therefore, alternatives to sampling every container and using only process knowledge needed to be explored and negotiated with regulating agencies.

State and Federal Guidance Concerning IDW

Guidance for the management of IDW issued by the EPA and the State of Colorado are based on different assumptions about the origin of the IDW. The EPA's guidance is based on IDW generated from a CERCLA field investigation. It states that "site managers should not assume that a waste considered to pose a potential risk at a CERCLA site is a listed or characteristic RCRA hazardous waste" unless there is positive evidence that the IDW contains a such waste (EPA 1991). However, the guidance also states that for proper protectiveness, or if RCRA requirements are considered relevant and appropriate to the site, IDW may best be managed as a hazardous waste. One of the management options for soil IDW allowed by the EPA is to return it to the boring location immediately upon generation. Other management options include consolidation within an area of concern, sending the IDW to an on-site or off-site treatment and disposal unit, or storage of the IDW for future treatment or disposal. Use of these management options are based on site-specific information and must be protective of human health and the environment.

Individual states are not given authority to implement CERCLA, but can be given authority for RCRA. A state's guidance for management of IDW may assume the material is the result of a RCRA corrective action or RFI and potentially contains RCRA hazardous waste. The State of Colorado's guidance states that environmental media is

not a solid waste. Therefore, by definition the media itself cannot be a hazardous waste. However, environmental media may contain hazardous waste and must be managed as such. To determine if the environmental media contains a hazardous waste, state guidance suggests that the media be analyzed for a suite of analytes based on available site knowledge. If no site knowledge is available, a complete Appendix VIII (40 CFR 261 and 6 CCR 1007-3, Section 261) analytical suite is required. Once the analytical results are validated, the results are compared to the following criteria:
1. non-detect for organic compounds except for those that are naturally occurring and/or
2. less than or equal to background level for inorganic and naturally occurring organic compounds.
If the IDW does not contain constituents above detection limits or above background and does not exhibit one of the hazardous waste characteristics (ignitability, corrosivity or reactivity) the State of Colorado no longer regulates the IDW and it may be returned to the location of origin or disposed of in the manner of choice. As an alternative to the State characterization scheme previously discussed, if the IDW contains constituents that are either detected organic compounds or inorganic compounds detected above background, then the concentrations of contaminants may by compared to site-specific risk based criteria.
If the IDW "passes" the risk assessment criteria, even though it failed the analytical criteria, and does not exhibit a hazardous waste characteristic, then the IDW is considered to not contain a hazardous waste. However, the IDW may still need to be managed in accordance with the state's Solid Waste Regulations because of the low levels of contaminants present.
If the IDW fails the risk-based criteria, it contains hazardous waste and must be managed as such.
The State's Interim Final Policy and Guidance for Corrective Action at RCRA Facilities (CDPHE 1993) provides guidance for performing risk assessment for IDW and will be discussed in more detail in the following paragraphs.
When a site is both a RCRA and CERCLA site, the more conservative guidance will generally prevail. State-specific RCRA guidance is usually more conservative then EPA's CERCLA guidance for the management of IDW.
Colorado's Risk Assessment Guidance
The State of Colorado allows the use of risk assessments to determine allowable levels of contaminants in IDW. This is an alternative to assuming that any detectable level of a contaminant makes the IDW hazardous. However,the risk methodology is based on extremely conservative assumptions. First, the level of acceptable risk for chemical carcinogens is any concentration below 1x10-6 risk level. This risk level is more conservative then the range allowed by the EPA of 1x10-4 to 1x10-6 for CERCLA investigations. Second, a long-term resident must be assumed to be the exposed individual, regardless of the planned or proposed future use of the site. Third, multiple exposure pathway must be evaluated including: soil ingestion, dermal contact with soil, inhalation of particulates, inhalation of VOCs, and ingestion of homegrown produce grown in contaminated soil. Fourth, there is no provision for source depletion or attenuation even though the source may only a be 55 gallon container of soil. Finally, although modelling is not required, the corrective action must take in to account migration and leaching potential of contaminates in soil.
Pros and Cons of IDW Management Options
Returning IDW to its location of origin is the most cost effective management option allowed by the EPA, but often is not considered protective of human health and the environment. Of the other waste management options, site-specific factors need to be taken into consideration to determine which management option is most cost effective. Assuming that IDW from potentially contaminated areas is hazardous without analysis may be cost-effective if the area is heavily contaminated with known contaminants. Characterization by process knowledge in this case will save analytical and data evaluation costs. However, if the IDW will ultimately be treated and disposed of off-site, the receiving Temporary Storage and Disposal (TSD) facility will likely require a thorough analysis of the IDW.
Most states will require analysis of the IDW. It may be possible to group drums by sampling location and composite the drum samples. At RFETS, the State allowed borehole samples to be used to characterize the drill cuttings in each drum. This

approach may be possible as long as there are records to correlate each container with the corresponding sampling location and is cost-effective since drum sampling may not be necessary in most cases.
Performing a site specific risk assessment may also be time-consuming and expensive, especially if toxicology information must be researched for a large number of chemicals but there is only a small volume of IDW to be characterized. The State of Colorado allows the establishment of alternative action levels below which a waste is no longer considered a threat to human health and the environment. These alternative action levels include Colorado Water Quality Standards, federal drinking water standards, and federal Water Quality Criteria. These standards are likely to be more conservative than risk-based levels for soil, but may be more cost effective than developing a site-specific risk assessment.

IDW CHARACTERIZATION AT RFETS

Data Availability

As mentioned previously, the State allowed the use of borehole data in lieu of drum samples. This was because detailed records of the sampling location, sample numbers and depth of samples were recorded for the drill cuttings in each container. However, in some cases, samples were not taken in the interval represented by the drum, certain types of analysis were not requested for that interval, or the sample results were never recorded for a variety of reasons. In those cases, other corresponding samples were used to characterize the drum. These samples were from

- An interval above the drummed interval in the same location,
- An interval at or above the drummed interval in a nearby location,
- An interval at or above the drummed interval in a location within the same individual hazardous substance site (IHSS, also known as a solid waste management unit (SWMU)), and/or
- An interval below the drummed interval in the same location

When more than one sample was used to characterize a drum, the results were averaged by analyte in accordance with the approved standard operating procedure.
If no data were available for the drum and no appropriate associations could be made, the drum was sampled.

Risk Assessment Methodology

The State of Colorado required RFETS to use the conservative risk assessment methodology described in its Policy and Guidance on Risk Assessment for Corrective Action at RCRA Facilities The guidance assumes a long-term residential multiple pathway exposure to a non-depleting source.
Most of the intake parameters used in the risk assessment methodology developed by RFETS for IDW are standard default values prescribed by the EPA for long-term residents. In addition, the state required that time-weighted averages be used for all pathways to account for the sensitive subpopulation of children under age 6. The intake parameters that apply to all pathways include:

- Exposure Frequency = 350 days or events per year
- Exposure Duration = 24 years for adults and 6 years for children
- Body Weight = 70 kg for adults and 15 kg for children
- Noncarcinogenic Averaging Time = 8760 days for adults and 2190 days for children
- Carcinogenic Averaging Time = 25550 days for adults and children

The pathway specific intake parameters are outlined below:

Soil Ingestion

- Ingestion Rate = 100 mg for adults and 200 mg for children

Dermal Contact with Soil

- Exposed surface area = 7100 cm2 for adults and 4600 cm2 for children
- Absorption Factor = 0.5 for adults and children
- Adherence Factor = 1 mg/cm2/event

Inhalation of Particulates (inorganics only)

- Inhalation Rate = 0.83 m3/hr for adults and 0.73 m3/hr for children
- Particulate Emission Factor = 4630 m3/mg

Ingestion of Homegrown Produce (metals only)

- Ingestion rate = 122,000 mg/day for adults and children
- Fraction from contaminated source = 0.36
- Soil-to-produce partitioning factor = chemical-specific

RFETS was not required to evaluate the exposure due to inhalation of Volatile Organic Compounds (VOCs) because models for calculating their concentrations in air had not been approved when the waste characterization was performed.

Performance of the Risk Evaluation for RFETS IDW
At the time the original risk evaluation was performed, over 4000 drums of IDW had accumulated at RFETS. To streamline the waste characterization process, some global assumptions were made, and part of waste characterization the process was automated by writing a computer program to perform the calculations and comparisons required by the procedure.
The basic steps of the waste characterization procedure performed by RFETS are as follows;
Step 1:
Determine which samples can be used to evaluate each drum.
Step 2:
Perform an evaluation of the analytical data including data useability, review of B-qualified data, and flagging or discarding any unusable or unnecessary data. A step was included to determine if any of the available flashpoint or pH data exceeded the regulatory standards for the hazardous waste characteristics of ignitability or corrosivity. If either of these standards is exceeded, the waste is a characteristic hazardous waste and must be managed accordingly. The risk assessment is still required to be performed for those drums
Step 3:
Determine background values to be used in the comparison of detected values to background. Data and statistical tests to determine background were dictated by the state. For RFETS, a chemical was determined to be above background if it exceeded the background mean plus two standard deviations value.
Steps 5 through 9 were performed by a computer program written specifically for RFETS data and the approved characterization methodology. The computer program processed the data on a drum-by-drum basis and provided as output a form for each drum. The form showed the drum and sample number, detected analytes and their average concentrations, risk and regulatory computations and comparisons, and the final disposition (i.e. RCRA hazardous, non-hazardous, etc).
Step 4:
Determine which chemicals were detected and calculate the average concentration of those chemicals in each drum.
Step 5:
Compare the average concentration of the inorganics to background values. If the chemical exceeds background, continue with the risk assessment. Inorganics not exceeding background values need not be evaluated further.
Step 6:
Compare the average concentrations of detected chemicals to corresponding RCRA Toxicity Characteristic (TC) values. The TC values in 40 CFR 261.24 are maximum allowed concentrations that leach from a solid medium. However, since the toxicity characteristic leaching procedure (TCLP) analysis was not performed on most of the samples and the values are expressed as liquid concentrations, the TC values were multiplied by an approximation value of 20 to be compared to total concentrations in soil. The TC list includes only 40 chemicals. Consequently, not all chemicals in the IDW had a corresponding TC value.
If any of the chemical concentrations exceed their corresponding TC value, the drum contains RCRA hazardous waste and must be managed accordingly. The remainder of the risk assessment must still be done to determine the appropriate waste codes for the IDW. If all of the chemical concentrations were below the modified the TC value, continue with the risk assessment.
Step 7:
Calculate risk-based concentrations (RBCs) for each chemical using given intake parameters and toxicity. Such information is available from the EPA on the Integrated Risk Information System (IRIS) and the Health Effects Assessment Summary Tables (HEAST). For the RFETS characterization the chemical list included all those that appeared in the environmental database. The list was then split into a RCRA and Inclusive. The State of Colorado required that the risk from all detected chemicals that appeared on the RCRA Appendix VIII list be summed to determine the risk due to RCRA constituents. If the drum passed the RCRA test all additional detected chemicals would be added to the RCRA risk to determine if the drum passed the Inclusive test. These tests will be described in more detail later.
Step 8:
Determine which chemicals are on the RCRA Appendix VIII list. For these chemicals

divide the average detected concentration by the corresponding RBC values (there may be a carcinogenic RBC and a noncarcinogenic RBC for each chemical). Sum the carcinogenic concentration to RBC ratios and noncarcinogenic concentration to RBC ratios separately. If either the carcinogenic or noncarcinogenic ratio sums exceed one, the drum contains RCRA hazardous waste and must be managed accordingly. At this point the risk assessment is completed. If neither of the ratio sums exceed one, continue with the risk assessment.

Step 9:

Calculate the carcinogenic and noncarcinogenic ratios for the remaining chemicals and add these ratios to the previously calculated ratio sums for each RCRA constituent. This step is known as the Inclusive risk analysis. If either the carcinogenic or noncarcinogenic ratio sum exceeds one, the waste must be managed until the proper disposition can be determined. At RFETS, the IDW that fails this Inclusive risk analysis test is being managed similarly to the RCRA waste until the CERCLA ROD, at which time, the treatment and/or disposal options will be determined. If neither the carcinogenic ratio sum nor the noncarcinogenic ratio sum exceed one, the IDW is deemed nonhazardous and disposal is not unrestricted at the site. At RFETS, this IDW will likely be used as clean fill on the plant site.

Since RFETS is a nuclear facility, a risk evaluation for radionuclides also needs to be performed before any of the IDW is truly considered non-hazardous. An extension to the above procedure is currently being developed that will characterize the radiological risk from each drum of IDW.

COST COMPARISON

In order to demonstrate the value of the chemical risk-assessment methodology, a cost comparison has been conducted. The comparison is limited to funding expenditures that were or may have been required for the chemical characterization of the IDW drums. Specifically, the costs of performing the risk-assessment methodology was compared to an estimate of the cost for obtaining RCRA characterization samples for each drum. The costs for the sampling project are best estimates of efforts required to obtain the samples and laboratory analytical results. The costs for the risk-assessment methodology are based upon actual costs incurred and are limited to the characterization of the IDW.

The cost estimate for the IDW drum characterization by direct sampling for contaminants of concern, is based upon the actual field sampling efforts. The scenario assumes a 14 week period in which to sample all 4200 IDW drums, using two field crews. Field crew members would be typical of those required to obtain samples, provide health and safety monitoring, package and ship all samples, and accurately document all activities. It is assumed that each field crew samples 30 drums per day, five days per week, for 14 weeks. Labor rates are $85.00 per hour for each field crew member. Laboratory analysis performed for each sample is 1) Contract Laboratory Procedures (CLP) for total metals, 2) Volatile Organic Analysis (VOAs), 2) Method 80/80 for drums suspect of pesticides or Polychlorinated Biphenyls (PCBs), 3) and radiological screens for 10% of the samples. The total estimated cost for the drum sampling and laboratory analysis would be $4.68 million dollars. The cost estimate is summarized in Table I, Cost Estimate for IDW Drum Characterization by Direct Sampling for Contaminants of Concern.

TABLE I

The cost incurred for the Risk-Assessment methodology for the characterization of the IDW drums was roughly $1.9 million dollars. This figure includes an accelerated sampling effort of 300 drums in a 72 hour period with rush charges an all samples sent to contractor labs. The remaining $600,000.00 was used for the procedure development, writing the computer code, and performing the risk-assessments of the 4200 IDW drums. Another benefit from the expenditure of the funds in the development of the program, is its continued usefulness. The same method is available for application to soil and sediment IDW which has been or will be generated from future field activities. Therefore, RI characterization samples may continued to be used for the dual purpose of site and waste characterization.

When these costs are compared, an estimated savings of $2,780,000.00 is indicated. This represents a substantial cost savings. Actual cost savings may be greater since costs of data management and final waste characterization were not included in the estimate. These additional costs would be expended for data management and validation, data review, and a final waste determination. Such activities require detailed review and can result in significant costs. Economic savings are not the

only benefit of the approved method. By not requiring RCRA sampling of the IDW drums, a substantial decrease in the potential exposure to personnel to chemical and/or radiological contaminants was achieved. This results in compliance to As Low As Reasonable Achievable (ALARA) requirements for occupational chemical and radiological exposures.

CONCLUSIONS

The Risk Assessment Methodology was applied to the characterization of 4200 soil and sediment IDW drums in the spring of 1994. The method complied with the EPA's and CDPHE's requirements to ensure protection of human health and the environment while complying with regulatory requirements that were applicable or relevant and appropriate. By carefully reviewing federal and state regulations that address IDW and risk assessment criteria, WCFS and EG&G were able to cultivate an innovative and technical process for characterization of the waste. By the application of the methodology, a reduction in waste and cost were realized. Based upon the result of the work, 79% of the drums were determined to be chemically nonhazardous and 4% of the drums were determined to contain CERCLA hazardous substances or other hazardous constituents in levels high enough to present a risk to the human health and the environment. Only 17% of the drums were determined to contain RCRA hazardous waste in levels high enough to present a risk to human health and the environment or in excess of RCRA regulatory levels.

A comparison of the waste characterization alone has indicated cost savings of at least $2.78 million dollars. This does not include cost savings based upon long term storage and or treatment of the soil and sediment IDW. Estimates show that an additional $4 to $8 million dollars were saved by performing the risk-based determination and not being required to respond to the strict requirements of the RCRA "contained in" rule. The cooperative efforts of WCFS and EG&G with CDPHE resulted in a characterization program for the IDW which resulted in cost saving and waste minimization. Most importantly, this was accomplished while maintaining the appropriate approved level of protection to human health and the environment.

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DWPF COLD CHEMICAL RUN SAFETY ENVELOPE ACCEPTANCE STRATEGY

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ABSTRACT

The startup process for the Defense Waste Processing Facility (DWPF) at the Savannah River Site (SRS) is a phased effort that includes three startup test phases prior to radioactive operations: Cold Chemical Runs (CCRs), Melter Heatup, and Waste Qualification Runs. While these three pre-radioactive test phases do not introduce radiological hazards to the facility, each does introduce chemicals in addition to the normal industrial hazards that pose potential risks to onsite personnel. The DWPF Final Safety Analysis Report (FSAR) scheduled to be completed in CY 1995, will define the safety basis for DWPF radioactive operations but will not address the specific hazards presented by the pre-radioactive startup tests. The FSAR will evaluate both radioactive and chemical hazards present during radioactive operations. It will not address the specific chemical hazards present during the test phases nor the configuration of the DWPF and operations during the test phases. These scope and schedule mismatches were recognized by the DWPF project team and addressed by the preparation of the CCR Safety Envelope Strategy. A review of the safety-related DOE Orders and Notices directed at facility operations found little guidance in the area of preparing an interim safety basis for facility pre-radioactive testing with significant chemical hazards. It became apparent that an interim safety basis was needed for the DWPF during the three test phases. A committee of DOE-HQ, DOE-SR and contractor representatives was formed, to draft a strategy for establishing an interim safety basis.

Development of a strategy was necessary to allow "buy-in" by the DOE line management

and interested parties, including the various oversight groups within the Department of Energy, since such an approach had not been used for other facility startups. Some of the more salient elements contained in the Strategy include:

Pre-radioactive Operation Authorization Basis Process - provides a description of the purpose, scope and roles of the CCR Safety Envelope (SE) Document, Safety Programs, and the Operational Readiness Reviews by the Contractor and DOE.

Safety Envelope Boundaries - defines the elements and boundaries (limits) of the SE to assure safety of onsite personnel and the public.

Operational Safety Requirements (OSRs) Significance Criteria - defines the thresholds for onsite and offsite chemical dose for determining the need for (Preliminary) Operational Safety Requirements.

Safety Envelope Acceptance Strategy - outlines the plan for preparation, review, approval and revision process for the CCR SE. It defines the roles for numerous DOE Oversight Groups.

Unreviewed Safety Question Process- requires the development and use of an Unreviewed Safety Question process to assure the limits of the CCR SE are not exceeded.

Graded Application of DOE Orders for CCR - defines the objectives of the Orders and Notices that are applicable during CCR.

The Strategy was approved by the SRS Manager, the Assistant Secretary for Environmental Restoration and Waste Management (EM-1) with concurrence by the Assistant for Environment, Safety and Health (EH-1) in December 1992. Since the Strategy's approval, it has been used to establish each test phase's safety basis within the DWPF SE and has become an essential element in gaining DOE's (both program and oversight organizations) authorization to begin each test phase. To date, two of the three test phases have been successfully completed at DWPF, and the last nonradioactive tests, Waste Qualification Runs, have started. The DWPF CCR Safety Envelope Strategy has been a significant contributor to their success. Acceptance of the Strategy by the DOE community as an approach to establishing the safety basis for pre-radioactive testing with chemical hazards as well as the Strategy's use and reliance on the project's Safety Programs (e.g., fire protection, industrial hygiene, emergency preparedness) are considered unique in the DOE.

Background

The Defense Waste Processing Facility (DWPF) is designed to accept the high level radioactive waste (insoluble solids and processed supernate (i.e., sludge & precipitate)) from the tank farms at the Savannah River Site, mix it with glass formers (frit), melt the slurry to form a borosilicate glass, and pour the molten glass into a canister which is subsequently decontaminated and seal welded. These canisters will be temporarily stored at DWPF and will be transferred to a permanent geologic repository. DWPF is being tested at various design parameters to assure product quality and safety. Simulated sludge and precipitate feeds are prepared and processed in lieu of radioactive waste for startup testing. A simplified process overview sketch is provided in Fig. 1, DWPF Process/Hazard Overview. This sketch also identifies the chemical hazards associated with the pre-radioactive simulant testing.

Fig. 1.

The startup process for the DWPF is a phased effort that includes three test phases prior to radioactive operations: Cold Chemical Runs (CCRs), Melter Heatup, and Waste Qualification Runs. While these three pre-radioactive test phases do not introduce radiological hazards to the facility, each does introduce chemicals in addition to the normal industrial hazards that pose potential risks to onsite personnel. The DWPF Final Safety Analysis Report (FSAR) scheduled to be completed in CY 1995, will define the safety basis for DWPF radioactive operations but will not address the specific hazards presented by the pre-radioactive tests. The FSAR will evaluate both radioactive and chemical hazards present during radioactive operations. It will not address the specific chemical hazards presented by the test phases nor the configuration of the DWPF and operations during the test phases. These scope and schedule mismatches were recognized by the DWPF project team and were addressed by the preparation of the CCR Safety Envelope Strategy (1). A review of safety-related DOE Orders and Notices directed at facility operations found little guidance in the area of preparing an interim safety basis for facility pre-radioactive testing with significant chemical hazards It became apparent that an interim safety basis was needed for the DWPF during the three test phases.

A committee of contractor, DOE-HQ and DOE-SR personnel developed a strategy paper which included contractor, DOE-SR and DOE-HQ activities. The committee prepared an action memo, approved by EM-1, which outlined the SE Strategy and sought DOE oversight concurrence from the Office of Environmental Safety & Health (EH). DOE-HQ also reviewed and accepted the SE through a contracted, independent Technical Review Group (TRG). The Strategy also required Operational Readiness Review (ORR) assessments by the contractor and DOE to assure effective implementation of the Safety Programs. Approval of the Strategy at the EM-1 level was appropriate to commit all affected groups to the program.

The Strategy was approved by the SRS Manager, the Assistant Secretary for Environmental Restoration and Waste Management (EM-1) with concurrence by the Assistant for Environment, Safety and Health (EH-1) in December 1992. Since the Strategy's approval, it has been used to establish each test phase's safety basis within the DWPF Safety Envelope and has become an essential element in gaining DOE'S (both program and oversight organizations) authorization to begin each test phase. To date, two of the three test phases have been successfully completed at DWPF and the last nonradioactive tests, Waste Qualification Runs, have started. The DWPF CCR Safety Envelope Strategy has been a significant contributor to their success. Acceptance of the Strategy by the DOE community as an approach to establishing the safety basis for pre-radioactive testing with chemical hazards as well as the Strategy's use and reliance on the project's Safety Programs (e.g., fire protection, industrial hygiene, emergency preparedness) are considered unique in the DOE.

Discussion

Some of the more salient elements of the Strategy follow:

The Pre-radioactive Operation Authorization Basis process establishes two distinct elements.

The first element is the preparation of a CCR Safety Envelope (CCR SE) (2) which is reviewed and accepted by the contractor, DOE-SR and the DOE-HQ. The CCR SE defines the safety basis and bounds the chemical hazards for DWPF nonradioactive testing.

The second element is the establishment of safety programs at DWPF. The safety programs include Fire Protection, Industrial Hygiene, OSHA, Operating Procedures, Training and Process Hazards Reviews, and Emergency Preparedness Programs. The safety programs were assessed by the contractor and DOE Operational Readiness Reviews prior to authorization to start testing and are not elements of the CCR SE.

Safety Envelope Boundaries - The two safety basis program elements define the program limits. The combined, but distinct elements assure effective safety controls to provide protection for the worker (i.e., Safety Programs) via Process Requirements (PR), and onsite personnel/public via the Safety Envelope which includes the Preliminary Operational Safety Requirements (POSRs). A schematic is provided in Fig. 2, DWPF CCR Authorization Basis Flowchart, to show the process and relationship between the two Authorization Basis elements.

Fig. 2.

CCR Safety Envelope - The Safety Envelope is established by assessing and using existing safety documents (e.g., Preliminary Safety Analysis Report, Hazards Analysis and Process Hazards Reviews). Where chemical consequences exceed the Operational Safety Requirements Significance Criteria (defined in the Strategy), the hazard is documented and controls are established as Preliminary Operational Safety Requirements (POSRs). These controls include Safety Limits, Limiting Conditions of Operation, and Administrative Controls The POSR term is established to distinguish it from radioactive OSR, or Technical Safety Requirements (TSRs) that will be required for radioactive operations.

POSRs are an integral part of the SE. Structures, Systems and Components (SSCs) used to prevent or mitigate the hazard are designated SSCs Important to Safety and listed in the SE accordingly.

Principal Design Criteria, Process Descriptions and Facility Design also form part of the SE. POSRs unique to the testing phases include locking out or limiting selected hazardous inventories that will not be required during radioactive operations and controlling simulant feed specifications to limit hazards.

The Safety Envelope includes the following sections:

a. Identification and description of the Chemical Hazards
b. The Safety Analysis associated with the Hazards
c. Administrative Controls appropriate to the Hazards (i.e. POSRs)
d. List of Structures, Systems and Components Important to Safety

e. Principal Design Criteria
f. Facility Design
g. Process Description
Operational Safety Requirements (OSRs) Significance Criteria - are provided in the following
TABLE I
These criteria were chosen based on Westinghouse Savannah River Company policy and good industrial safety practice. DOE-SR and DOE-HQ line management concurred with the criteria.
Unreviewed Safety Question (USQ) Process - The Strategy prescribes a USQ process, per DOE Order 5480.21, that includes a contractor review process to assure that design, operational or chemical process changes are reviewed against an approved Safety Basis (the SE) and that the Safety Basis limits are not exceeded without review by the DOE. The process assures that design or process changes do not increase accident frequencies or consequences that exceed the previously accepted SE or introduce new accidents which may exceed the OSR Significance Criteria. The USQ program is applicable to all plant modifications, temporary modifications, operating procedures and test procedures. From a contractor perspective, it is important to bound the accidents with conservative analysis and perform a careful review of the intended testing to assure the test activities are enveloped the SE. This review avoids review time of USQs for unanticipated events and the potential delay in aggressive testing schedules.
Graded Application of DOE Orders for CCR - The Strategy requires review of Orders and Notices to determine applicable requirements for the pre-radioactive testing program. The following orders and notices were recommended for consideration in this process. SEN-16B, Start or Restart of Nuclear Facilities, SEN-6E, Dept. Organization & Mgmt. Arrangements, Order 5481.1B, Safety Analysis & Review System, Order 5480.23, Safety Analysis Reports, Order 5480.22 including the implementation standard DOE STD. 3009, Technical Safety Requirements, Order 5480.21 Unreviewed Safety Questions. The DWPF Safety Envelope Acceptance Strategy and the Safety Envelope document have been invaluable tools during nonradioactive testing of DWPF. These documents defined the hazards unique to testing DWPF and allowed the operating contractor, DOE-SR, and DOE-HQ personnel to focus on the actions necessary to protect DWPF and Savannah River Site workers from those hazards. Employing a similar approach during the startup of new, large, and complex facilities like DWPF is highly recommended.
REFERENCES
1. K.G.Picha, Defense Waste Processing Facility Cold Chemical Run Acceptance Strategy, U.S. Dept. of Energy
2. D.F.Gehr, Defense Waste Processing Facility Cold Chemical Run Safety Envelope, WSRC-RP-92-975, Westinghouse Savannah River Company

59-4
REPOSITORY DISPOSAL CRITERIA FOR DOE SPENT FUEL
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ABSTRACT
A decision by the U.S. Department of Energy (DOE) in April, 1992, discontinued reprocessing and recovery of highly enriched U-235 (HEU) from irradiated fuels stored by the Idaho Chemical Processing Plant at the Idaho National Engineering Laboratory. The new mission of monitoring waste storage from previous reprocessing campaigns and of dealing with continued receipt and storage of irradiated fuels has prompted court actions by the State of Idaho requiring DOE to demonstrate efforts for preparing, packaging and shipping existing wastes out of the state to a national repository. These actions resulted in the formation of a multidisciplinary effort to evaluate options and approaches necessary for these wastes to qualify for geologic disposal. While the options being studied cover both high-level wastes (HLW) and spent nuclear fuels (SNF), dealing with a variety of HEU-type fuels presents special problems beyond those encountered with the commercial spent fuels now being studied for disposal in the proposed repository at Yucca Mountain, Nevada.

INTRODUCTION

The U. S. Department of Energy (DOE) has received irradiated nuclear fuel at the Idaho Chemical Processing Plant (ICPP) for interim storage since 1951, and for reprocessing to recover highly-enriched uranium-235 (HEU) since 1953. With the changes in world events, a decision by DOE in April, 1992 shutdown fuel reprocessing operations. At that time, efforts at ICPP were redirected toward evaluating options for safe interim storage and permanent disposal of both spent nuclear fuels (SNF) and the high-level waste (HLW) resulting from past reprocessing operations. With the characteristics and variety (over ninety types) of unprocessed irradiated DOE-owned fuels in storage at the ICPP and other DOE sites, development of treatment and packing criteria within the current regulatory guidelines is vital for defining interim storage and disposal options.

At Yucca Mountain (YMP), Nevada, the designated nuclear waste study site that may become the eventual geologic repository, the Office of Civilian Radioactive Waste Management (DOE-RW) has studied commercial fuels and some vitrified HLW generated by defense contractors. This paper reviews the current disposal criteria as it might affect disposal of DOE-owned SNF in a geologic repository.

HISTORY OF REPOSITORY DISPOSAL CRITERIA

Several regulatory standards identified disposal requirements for SNF and HLW. The primary regulations are 10 CFR 60 (NRC standard) and 40 CFR 191 (EPA standard). In addition, based on the Energy Policy Act of 1992, EPA contracted the National Academy of Sciences (NAS) to conduct a study to provide, no later than December 31, 1993, " . . . the (EPA) Administrator shall, based upon and consistent with the findings and recommendations of the National Academy of Sciences, promulgate, by rule, public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site." The EPA is to promulgate the standards not later than one year after the Administrator receives the findings and recommendations of the NAS. No later than one year after the EPA Administrator promulgates the standards, the NRC shall, by rule, modify its technical requirements and criteria, as necessary, to be consistent with the Administrator promulgated standards.

At the writing of this summary, the NAS had held seven meetings on various topics and had completed a draft of the findings and recommendations; the draft report is currently undergoing review by the National Research Council. The plan was to issue the final report by March, 1995; thus, the earliest EPA standard promulgation date would be March, 1996. So, the current technical requirements are used as the basis to determine the repository disposal criteria.

The development of the criteria for disposal of nuclear wastes in a geologic repository focuses on both the waste packaging (size, weight, package materials, etc.) as well as the individual waste form(s) themselves. Inclusion of DOE-owned spent fuels into a national repository engenders certain other considerations in developing specific disposal criteria that will be somewhat different from the commercial fuels under the auspices of DOE-RW. Certain DOE SNF (and HLW) materials may be classified as hazardous wastes under the Resource Conservation and Recovery Act (RCRA). Depending on the final determination as to the applicability of RCRA on these materials, 40 CFR 268 Land Disposal Restrictions (LDR) may have to be considered and implemented into the disposal criteria.

Unanswered questions arise with respect to criticality safety and fissile material accountability in a storage/disposal environment. Regulations, such as 10 CFR 60.131(b)7, are very specific governing conditions of criticality safety for operating facilities, including geologic repository operations. Additional work may well be needed to clarify implementation of criticality safety requirements in a post-closure repository environment. Similarly, 10 CFR 70.51 identifies certain requirements relative to accountability and safeguards associated with the handling, storage, and disposal of HEU and other weapons-grade, fissile materials.

Finally, DOE-RW has in the past two years focused on the development of a multi-purpose canister (MPC) for the storage, transportation, and disposal of commercial spent fuel. This standardized design should be used when ever possible for the disposal of DOE-owned SNF. Such an approach would provide significant savings through the use of common container design, handling, and transportation equipments. However, due to the variety of DOE-owned SNF, certain modifications of the MPC (design of internal features) may be required and must be resolved with the DOE-RW organization prior to defense waste acceptance into a national repository.

The following material references the Preliminary Waste Acceptance Criteria for the ICPP Spent Fuel and Waste Management Technology Development Program (WINCO-1157, September 1993) and its replacement, Waste Form Product Characteristics (draft-November 1994). These documents detail the basic waste criteria that, through computer simulations done in support of the performance assessment, identify waste forms and packaging needed to meet existing regulatory requirements for defense waste and DOE-owned SNF disposed of in a geologic repository.

ACCEPTABLE WASTE PACKAGING AND FORM CRITERIA

At the time much of the disposal criteria was being established for high-level wastes (HLW) generated as the result of fuel reprocessing activities, lesser efforts seem to be directed toward intact, spent fuel disposal. As a result, many of the attributes identified in federal regulations for HLW waste form/packaging and performance were not originally intended for direct application to spent fuels. A list of both waste package and waste form criteria as they are expected to apply to DOE-owned spent fuels is shown below.

Waste Package Criteria
Waste Package Materials Temperature Limits
Package Weight Allowable Void Space
Dimensions Package Labeling
Heat Generation Waste Package Handling Features

Waste Form Criteria
Solids Leach Rates
Liquids Corrosion
Explosiveness, Pyrophoricity, Combustibility Gas Generation
Chemically Reactive Mechanical Properties
Neutron Absorbers Radiolysis
Criticality Safety Surface Dose Limits
Safeguards and Material Accountability Radionuclide Inventory
Solubilities Organics
Inert Gases

As an example, liquid exclusion from waste packages is identified in 10 CFR 60.135(b)2: "The waste package shall not contain free liquids in an amount that could compromise the ability of the waste packages to achieve the performance objectives relating to containment of HLW . . . " However, given an argument that there is a potential for free liquids in some packages, can some free liquid be allowed if it can be demonstrated that it won't compromise the package? What amount of free liquid might compromise the package? On what basis might liquids be allowed without concern toward package damage? Should an absolute or quantitative value of allowable liquid be identified, without which there will always be a case for arguing against any package disposal?

Waste Package Materials

The selection of 304L stainless steel is suggested as a basic material for nuclear waste packages (1,2), although various combinations of both common and exotic materials may be used in conjunction with the overall containment. Series 300 (austenitic) stainless steels offer a good combination of corrosion resistance, fabrication, weldability (304L), availability, and favorable costs. Other barriers such as overpacks (Incoloy 825, carbon steel) and the geology/hydrology characteristics of the repository, should allow designed waste packages to meet the retrievability requirements - and allow the waste forms to meet the release limits - for the prescribed 10,000 year containment in the repository.

Package Weight

Selection of either the Multi-Purpose Canister (MPC) [125-ton] or Legal-Weight Truck (LWT) canister [25-ton] for DOE SNF will be dictated by the handling facilities of the producer responsible for fuel packaging. In most cases, the LWT would be used for shipment to and repackaging in an MPC prior to final disposal in the repository. It is conceivable that a small number of LWTs could end up as a final disposal package in a repository

Dimensions

Dimensional information for both MPCs and LWTs planned for storage/disposal containers of DOE-owned fuels is available in various documents (3,4). Overall

external configuration of the package will copy the design used for commercial fuel. There will necessarily be changes to the package internals to accommodate the variety of different fuels types, both in terms of packing density and the lengths of the fuel elements. External dimensional standards adopted for containers used in disposal of DOE-owned spent fuels will adhere to the criteria established for commercial fuels by the DOE Office of Civilian Radioactive Waste Management (OCRWM).

Heat Generation

Heat generation rates are currently quoted at ~800 watts for individual HLW canisters (1) (a Multi- Barrier Waste Package [MBWP] would contain four canisters). Current design for YMP (if a 'hot' repository concept is approved) will allow ~14kW/MPC with a commercial nuclear fuel load. The limitation on allowable heat generation for any MBWP or MPC in the repository will be a function of the ability of the geologic media to dissipate the heat, specifically, heat transfer away from any canister will be a function of the repository conditions (5).

MPC/MBWP configuration will affect heat retention within the canister, which in turn affects centerline or skin temperatures of HLW glass and spent fuels, respectively. In all cases, the controlling item will be the internal heat generation of either the fuel or glass waste form itself. Design will necessarily require an iterative approach to identify allowable heat generation based on known characteristics of the waste form, the package for the waste form, and the ambient conditions found in a specific repository.

Temperature Limits on Waste Forms

There are three basic concerns that potentially impact the temperature or thermal limits imposed on the waste packages destined for storage/disposal. They are: 1) fuel cladding temperature limitations, 2) phase changes in HLW borosilicate glass, and 3) collective temperature effects of the individual waste package on the repository media itself.

Originally, fuel cladding design was based on temperatures in operating, water-moderated reactors. The ability to quickly carry heat away from the fuels thereby allowed higher power levels in the controlled environment of the reactor without concern for overheating. Ambient conditions expected for fuels stored in dry conditions associated with MPC/geologic disposal will generally be more restrictive because of the thermal resistances encountered in the canister design and implied conservatism needed to establish or meet waste acceptance criteria. Of the typical fuel cladding materials encountered in DOE-owned fuels, zirconium and aluminum predominate. Burnup and cooling time (time-out-of-reactor) for each of the individual fuel types, along with criticality safety concerns, will determine the allowable fuel loading in each canister to avoid over-temperature conditions for the cladding or repository limits. Current limits for the aluminum clad fuels have been established at 150C, while zirconium and stainless steel clad fuels will allow for temperatures up to 340C (6).

Collectively, each waste package destined for the repository will have a calculated rate of thermal output. Any given waste package, when combined with the repository environment, will result in a steady state or equilibrium temperature both inside the package and within the geologic medium. Hence, the thermal output limits imposed on any package destined for the repository will be determined by the conditions in the repository itself, namely host rock temperature, predominant mechanism(s) of heat transport, 'thermal' package density within the repository, thermal output of each waste package, and individual waste package backfilling. Not until the actual, detailed characteristics of the repository are identified can the waste package loadings (both thermal and fissile) be finalized.

Allowable Void Space

The void space limitation was intended to encourage producers to maximize packing efficiency in canisters. However, this may be in direct conflict with the packaging required to assure criticality safety when multiple, HEU fuels are contained in a single package. Costs associated with 'excess quantities' of canisters (both production and storage/disposal) created by inefficient packing should ensure optimal packing. At least in the case of spent fuels to be disposed of in salt, void volumes should be filled with inert materials so that the canister emulates a solid body that might better withstand the crushing pressures experienced with salt creep over time. Other considerations favoring addition of other inert materials to a fuel cask with its significant void space might be: 1) enhanced heat transfer, 2) in situ stabilization of fuels, 3) added moderator displacement, and/or 4)

retardation of fuel 'reconfiguration' inside the MPC over time.

Package Labeling

Issues relative to waste package placement within the repository may be dependent on the content of the package (SNF or HLW) and/or the thermal considerations in terms of balancing or even concentrating heat loads for a 'hot' repository configuration. The ability to stage/position waste packages within the repository will be dependent on traceability of data back to the original packages and knowledge of their contents (source term inventories). Any subsequent need for retrieval within the 50-year time frame, should that prove necessary, would also rely heavily on traceability back to a data package and the linkage with any given waste package.

Waste Package Handling Features

Adoption of MPCs as a standard package will allow utilization of the developed technology and standardized equipment design without a large investment in design and qualification of a new canister.

Solids

Prohibition of particulate solids (7) is intended to minimize the spread of contamination, either through a handling accident (canister rupture), or subsequent leaching of materials due to the much greater surface area exposed to leachate upon waste package failure. The need to consolidate particulate material is also dictated by the adoption of best demonstrated available technology (BDAT) as a treatment path for nuclear wastes. Ultimately, the Department of Transportation (DOT) is expected to invoke standards governing the nature of the transported solids.

Consolidation of scrap materials such as TMI fuel debris, may not justify treatment beyond the canisterization already done. Similarly, fuels with known, damaged cladding (whether through destructive testing or corrosion failure induced by wet storage) will likely be 'canned' prior to packaging in an MPC.

Liquids

Free-standing liquids are prohibited from wastes packages destined for geological repositories for several reasons. The combination of internal corrosion (an overriding concern), gas generation, and leaching of the waste form(s) prior to canister breach can all contribute to an accelerated failure of the waste package. Current requirements are ill-defined and totally subjective at this time, with no quantitative technical basis identified (8). Assumptions made for HLW 'glass logs' may be totally inappropriate for spent fuels because of either the nature of packaging, processing, or preceding storage conditions. HLW canisters will experience temperatures in excess of 400C during production, and it is difficult to postulate under these conditions or events any accumulation of any liquids (free or otherwise) in an HLW package.

There is a greater likelihood of liquid availability in waste packages associated with spent fuels than with HLW waste forms. Fuels formerly stored wet and then dried while in interim 'dry' storage have a chance to dry out due to self- heating; canned fuels (formerly stored wet) may or may not dry out while in dry storage, depending on both the can and storage conditions. Fuels stored wet and loaded wet into an MPC offer the greatest chance of water introduction and retention in an MPC. Treatment under these loading conditions is expected to rely on both (self)heating and vacuum dryout after underwater SNF loading into the waste package.

One approach to establish the 'allowable liquid' limits may center on an allowable weight % based on the waste mass, and that in turn may be dependent on the type of waste under consideration. As an example, the Waste Isolation Pilot Plant WAC standards are based on certain assumptions associated with essentially undocumented packaging of materials in 55-gallon drums; these assumptions relative to typical residues that could remain in containers led to a 1% by volume limit. The method of developing a quantifiable acceptance criteria must be technically based, perhaps an allowable wt% (liquid) of waste loading that might result in: 1) canister overpressurization [steam generation?], 2) internal corrosion [allowable mils thickness degradation of the canister inner walls], 3) gas generation [H2 formation], or 4) some other definable value.

Fuels being loaded directly from 'wet storage' into an MPC may require a drying step that invokes some form of testing. Testing may consist of a combination of vacuum with water vapor sampling, dip-tube suctioning, etc. One approach (for purposes of calculation) may assume/allow the equivalent of a 100% saturated volume of any empty MPC in the form of water vapor. As condensed moisture (~385 gm), it would likely be visible were physical inspection possible. With this amount selected as an upper

limit, it would be possible to calculate: 1) maximum pressurization that could occur, given limiting conditions inside the waste package for temperature and pressure, 2) estimates of total metal mass reaction and gas production (M + H2O--->MO + H2), 3) estimates of uniform or perhaps localized corrosion amounts. Most of the conceivable events involving water in an enclosed MPC should be scalar over the time frame of interest.

Explosiveness, Pyrophoricity, Combustibility

Restriction on pyrophorics is to minimize potential for creating 'combustion', given an ignition source in a waste package. Studies at the Rocky Flats Plant (9) have shown that small quantities of pyrophoric plutonium can be accommodated in other nonpyrophoric materials without generating an unacceptable hazard. A 1% limit has been established in the criteria for the WIPP as an acceptable level of pyrophoric material in a transuranic waste package. The 1% is used instead of 3% (demonstrated acceptable at Rocky Flats) since TRU waste forms may not be as uniform or homogeneous as the materials in the Rocky Flats Plant study. Analysis as to the applicability of this limit to SNF will have to be further addressed. The presence of pyrophorics actually does not present a problem in most packages, where the amount of free oxygen would be limited. At best, exposed pyrophoric material at elevated temperatures might act as an oxygen scavenger in the repository environment, with perhaps a fast but controlled rate of oxidation and a limit on the extent of reaction.

Explosives, if contained in a waste package, present a potential hazard to operating personnel during shipment and handling and provide a source for failure of the waste container. Explosive materials would consist primarily of gaseous species (hydrogen) and/or nitrated organics. Through the exclusion of organics and chemically reactive species (nitric acid), and nitrated organics themselves, the issue of explosives should be adequately addressed. In the case of HLW treatment, much of the treatment occurs at temperatures which would destroy any explosive materials or components which might recombine over time. Credit should be allowed for any process that thermally treats material with explosive potential. Hydrogen generation, whether through corrosion reactions with metal in a package or by radiolysis, becomes the single, most significant source of a potentially explosive condition over extended time. However, this situation would also require an ignition source and a source of oxygen. Carbon in the graphite fuels is in itself combustible, but in a monolithic form, limited oxygen, and lack of a significant ignition source would qualify it as non-combustible.

Chemically Reactive Species

Intentionally excluding chemically reactive materials from any waste package is to: 1) prevent formation or development of new reactive materials in the package over time, and 2) avoid reactions that lead to accelerated degradation of the waste form or the containment package.

Reactive materials could generate new compounds or species within waste form or package that were not considered or analyzed in the performance assessment. These new species might alter the characteristics of the waste package in terms of solubilities or breakdown of the form itself, leading to accelerated leaching. Abnormal gas generation rates due to elemental sodium/water or uranium carbide/water reactions could accelerate transport away from the package. Packaging requirements might adopt a two phased approach. Oxygen displacement using an inert atmosphere within the waste package will initially promote decreased reactivity of untreated metallic fuels. Additionally, it may be possible to take credit for limited oxygen entry into a waste package such that 'controlled' oxidation occurs without concern for combustion, and where leachability of the metal oxide is less than that of the pure metal. For spent fuels with special concerns brought about by failed cladding or special, chemically reactive materials, such fuels may require treatment beyond mere canning.

Most wastes, to stand a chance of being acceptable in a geologic repository, will have a predominant amount of oxides or robust, non-reactive metals associated with them. In almost all conditions, the oxides themselves are stable; however, the leachability of technetium and neptunium are both highly dependent on their valence state in the final waste form. Although subject to some leaching, in general low solubilities of the oxide waste forms provide retardation against transport in a groundwater environment.

Neutron Absorbers

Use of neutron absorbers is applicable only to the tuff media. Earlier studies ('93 PA) addressed the use of either fissile mass limits (granite: 0.7 kg U-235/package) and spacing, or water exclusion (salt:10.0 kg U-235/package) and spacing as the two contingencies required for criticality safety in repositories containing fissile materials (10). In a tuff environment, neutron absorbers were considered as one of the contingencies for packages with high fissile loading where credit was not taken for water exclusion (4).
Reference to boron (in its natural isotopic occurrence) as a suitable neutron absorber is not intended to exclude other materials from consideration; it is merely illustrative. Allowance for other materials, such as cadmium, gadolinium, europium, would be based on an ability to have some assurance that the absorber would remain in proximity to the fissile material it is intended to 'poison'. Many factors need to be considered relative to neutron absorbers that might be used in a waste package containing SNF: differential separation of materials, method of fixation inside a waste package, cost, availability, environmental regulations. The same considerations apply whether they are associated with high-enriched uranium (HEU) or low-enriched uranium (LEU) packages.
Criticality Safety
In general, fissile material loadings in individual canisters will be dictated by the geological medium surrounding the canister at the time of placement in the repository. The canister contents need to remain subcritical under all credible conditions likely to be encountered at the Producer's site, including any interim storage array (11). The calculated effective neutron multiplication factor, keff, must be shown to be less than or equal to 0.95 (at initial package loading) after allowing for bias in the method of calculation and uncertainty in the experiments used to validate the method of calculation. The Producer needs to describe the method of compliance in the Waste Form Compliance Plan (WCP) and provide supporting documentation in a Waste Form Qualification Report (WQR). The WQR should also include sufficient information on nuclear characteristics, such as fissile material density and enrichment, of the canistered waste form to enable subcriticality to be confirmed under the repository storage and disposal conditions found in the repository. Where mixed fissile species are present, fissile gram equivalents (FGE) U-235 will be used to determine allowable fissile loadings in the canisters.
Application and enforcement of double contingency barriers to criticality safety beyond the post-closure phase of the repository is unclear. Under current regulations, providing continued assurance as to the presence and adequacy of these barriers in a post-closure environment will require both monitoring and/or periodic inspections to verify the continued adequacy of said contingencies, and remedial action if one or more of the barriers is compromised.
The issue of criticality safety in any repository is more likely a political issue than a technical one. In the PA for the salt and granite repositories, waste packages containing 0.7 to 10.0 kgs U-235 equivalent loadings, respectively, are intended to make a near-field (single package) criticality impossible. The adoption of this approach, while assuring against criticality, created an inordinate number of waste packages to deal with all the spent fuels throughout the DOE complex. Subsequent studies (12) associated with tuff media were to allow for a criticality at some small probability and evaluate the consequences in the repository if it did occur. At issue in this case study is the initial condition of the repository as a 'dry' location. Given water exclusion and fixed neutron absorbers as the two contingencies in place at the time of repository closure, double contingency protection would be in effect. However, without installed monitoring, there will be no convenient way to determine if one or more of the contingencies remained in place, nor would any remediation effort to correct failed barriers likely be possible.
Safeguards and Material Accountability
In accordance with International Atomic Energy Agency (IAEA) policies and treaties to which the United States government is a signatory, the United States has a compliance responsibility regarding materials contained in many of the DOE-owned SNF packages intended for repository disposal. Interpretation of current documents suggest there will be reporting requirements at least up to the time the material is disposed of in a geologic repository, if not beyond that point unless modifications or reinterpretation of controlling documents are made. Regulatory guidance (13) indicates that SNF materials with greater than 20% U-235 enrichment that are

packaged for disposal in a geologic repository must include documented accountability of fissile materials. Such controls establish the guidelines for material balance areas (MBA), security issues, and inventory and other record requirements. The impact of these requirements on geologic disposal of DOE spent fuels has yet to be evaluated in any detail.

Solubilities

With no more detail than found in the statement "the design shall include consideration of solubility", 10 CFR 60.135(a)2 prescribes actions that should be considered without identifying a quantified measure of compliance. Waste treatment issues relative to HLW should in most cases either fix materials of interest in an essentially non-leachable form, or the treatment process (conversion from a nitrate to an oxide or perhaps a change of valence) significantly reduces the solubility of the material. Selection and qualification of chemical species found within the HLW matrix is intended to minimize solubilities. Except in cases where a product form would be chemically more reactive, soluble, or corrosive, oxide forms of the radionuclides are the preferred species.

Spent fuels (intended for geologic disposal under the same regulatory constraints) need not typically be packaged or treated before insertion into the multi-purpose canister unless it is a designed portion of the packaging because of material concerns, such as failed cladding, sodium bonding, or reactive metals. Performance assessment analyses take into account the solubilities of the various constituents comprising the waste package. The materials selected for analysis in the PA are based on their predominance in the waste, their 'mobility' under normal conditions, and the expected effects on the biosphere should they transport past the repository boundary.

Leach Rates

Specific leach rates for radionuclides from SNF are based on the combined barriers provided by the fuel element matrix itself, cladding, and any canister(s) containing the material. Allowable leach rates for packaged SNF will have to be qualified under the regulatory release limits for the geologic repository(ies) as determined by performance assessment modeling efforts. In an analogous situation, an HLW borosilicate glass inside a 304L stainless steel canister would equate to the fissile material (UO2 , UAlx , etc.) inside cladding (Al, stainless steel, zirconium), both of which end up in an MBWP. A great deal of effort has been expended to develop a qualified HLW borosilicate glass form with high resistance to leaching. Emphasis for fuels has been directed toward SNF 'package' criteria, since the SNF materials will have to qualify to the same regulatory release limits prescribed for HLW waste forms without any treatment or conditioning of the fuels.

Corrosion

Corrosion performance of any waste package will be pure conjecture until the actual repository is identified, along with the characteristic makeup of the ground water in that area. Once ground water characteristics associated with a given repository are identified, then there will need to be a development program to predict material behavior in that ground water environment. Predicting corrosion behavior of materials after some 3-5 years of studies, and then extrapolating that information to performance for 50, 300, 1000 or 10,000 years into the future is pure speculation. The performance assessment models use extrapolation techniques and statistical sampling for corrosion rate ranges to estimate probable times to package failure, given current knowledge about failure mechanisms.

Gas Generation

Gas generated by any waste package located in a geologic repository may be of concern either from its behavior while in the repository or its effect on material transport through and beyond the repository boundary (14). Gas generation from external corrosion of the waste package will be a design condition associated with the metal barrier(s) of the MPC/LWT and any overpacks.

However, greater variability is expected within the SNF waste packages (both in type and amount) because of the wider variety of materials encountered with the mix of fuels of differing compositions. Initial loading of canistered waste forms (HLW or SNF) will likely be governed by the requirement for an inert cover gas in the waste package at the time it is seal-welded. Any delays in oxidation within the waste package should translate to increased longevity of metal components within that package, providing additional barriers to liquid transport once the outer package breaches.

Gas generation within any 'intact' canister is basically limited to: 1) the amount of hydrogen that might be formed due to the corrosion reaction caused by the residual water in the MPC/LWT, and 2) the buildup of helium from the alpha decay of any transuranics. In either case, a 'puff' off-gas release might be a concern at a canister breach. In case 1, hydrogen offers a potential flammable mixture, given the presence of both oxygen and an ignition source.

Mechanical Properties

Variability associated with the various components expected in the waste forms themselves suggests the requirements to consider mechanical properties (14) is focused on the packaging materials for the waste forms themselves. Mechanical properties of the waste package should address the issue of package deformation due to creep in the salt repository. For geologic disposal in a tuff repository, thermal effects may govern package design depending on the concept selected for allowable thermal heat loading.

Radiolysis

Consideration of radiolysis in a repository (14) was intended to focus on how water chemistries external to the waste package might be affected by the radiation fields created in a repository.

Surface Dose Limits

Stringent standards will be imposed on the shipment of any waste package containing SNF or HLW. The greatest concern will focus on the measured, external levels (< 200 mrem/hr) (15) that would impact workers involved in canister handling during all phases of transport between the producer site and the geologic repository. Lesser levels (<2 mrem/hr) will be permitted in the cab of transport vehicles. The shielding requirements for virtually any container with high level wastes suggest almost exclusive use of rail transport.

Radionuclide Inventory

Knowledge of the curie inventories of the radionuclides contained within the various waste packages serves several purposes. The average and maximum values for any package allow determination of heat loading, surface dose limits, and calculating materials available for release upon package breach (at any point in the future) in the post-closure repository environment. Additionally, curie inventories of transuranics (and especially fissile materials) are needed for the calculations necessary to assure criticality safety in the various SNF packages.

Reporting requirements are generally stated in a fractional percentage (>0.05%) of the total curie inventory in a package, and for those radionuclides with half-lives greater than 20 or (in some cases) 10 years (16,17). The regulatory guidance found in 40 CFR 191 gives reporting requirements, using a 20 year cutoff for any radionuclide without regard to a fractional percentage. Invoking a requirement for reporting radionuclides down to 10 year half-lives appears to have no basis other than to demonstrate the producer can 'do better' than what the current regulations stipulate. The self-imposed fractional percentage for reporting concentrations is totally arbitrary. As an example, 54 of the 63 radionuclides/transuranics listed with Shippingport (HEU) fuels fall below the >0.05% cutoff (18). Yet several of the radioisotopes, including Tc-99 and Np-237 which are considered very mobile once released into groundwater, fall below the threshold of the Producer imposed reporting requirements.

Organics

The presence of organics associated with SNF and HLW materials is much less likely based on the known properties of both types of waste. Much of the processing HLW experiences is at conditions that virtually precludes the presence of anything more than negligible concentrations of organics. The intention in writing the exclusion requirement was to prevent the addition of any organics in the treatment or preparation of the waste packages. The exclusion of organics simplifies package qualification from the standpoint of minimizing moderator in SNF packages, a potential gas generator in all packages, and elimination of reactive materials such as residual nitrates that might combine with the organics to form explosive materials within the waste packages.

Inert Gases

The allowable use of inert gases is to facilitate the welding procedure(s) that may be used to close the various waste canisters as they are produced. Of some concern may be the ambient temperature of the waste canister at the time of closure, and whether trapped gases may increase in pressure. While in a remote cell with a

controlled temperature environment, pressures inside the canister might well be less than in a repository environment where heat retention in the surrounding geological structure can increase the canister temperature above what was experienced in the remote cell at canister closure.

CONCLUSIONS

Although waste acceptance is the responsibility of the repository, disposal criteria based on existing regulations are examined here to provide an initial indication as to the technical, regulatory, and programmatic issues that remain to be resolved prior to acceptance of DOE-owned SNF in a repository. Furthermore, in the near term, these efforts will provide guidance in the development work and facility requirements needed to provide a waste package suitable for geologic disposal.

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DISPOSAL OF CLASSIFIED WASTES FROM NUCLEAR WEAPONS DISMANTLEMENT

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ABSTRACT

The DOE Nuclear Weapons Complex (NWC) has a policy for disposition of weapons components, but there is not currently any established policy or defined process for management of classified waste generated from weapons dismantlement. The Atomic Energy Act (AEA) governs classified material management, but does not contain specific requirements for the disposal of hazardous or mixed wastes. The Resource Conservation and Recovery Act (RCRA) governs disposal of hazardous and mixed waste but does not specifically address disposal of hazardous and mixed classified wastes. Further, DOE Orders do not currently provide guidance on disposal of classified wastes. Each NWC site presently has its own approach to management of classified wastes.

There are several significant issues associated with disposal of classified waste:
1. Long-term capacity restrictions and costs of classified waste disposal;
2. Obstacles to demilitarization and sanitization (processes which render the waste free from military value and unclassified);
3. Dual requirements between the AEA and RCRA;
4. Inconsistent terminology and procedures between sites; and
5. Stakeholder and security concerns.

Concern about these issues has led to informal discussions among several organizations in the Albuquerque Operations Office (AL) regarding management of classified waste. As a result of these discussions, DOE is evaluating current classified waste management practices at NWC sites. The primary goal is to examine the generation and ultimate disposal of classified wastes and to develop a DOE waste disposal policy or defined process for classified wastes generated from nuclear weapons dismantlement.

The AL group is encouraging the NWC to minimize classified waste generation and disposal and to develop formal practices and procedures to effectively manage the classified wastes that are generated.

Any policy or procedures implemented by DOE should:

Ensure that ultimate disposal of classified waste is protective of public health and the environment.

Ensure that hazardous, mixed, and/or radioactive classified wastes are disposed of in full compliance with all applicable regulatory requirements;

Define practices and procedures to effectively manage classified wastes destined for ultimate disposal, while ensuring site-specific flexibility;

Further mitigate the problem by generating classified waste only as a last resort; and

Be consistent with DOE's Disposition Policy.

INTRODUCTION

As part of its defense mission, the Nuclear Weapons Complex (NWC) within the U.S. Department of Energy (DOE), generates classified materials from dismantlement of nuclear weapons. These materials may be hazardous, radioactive, or non-hazardous and non-radioactive. Ideally, these materials should undergo demilitarization* and sanitization.* There are cases, however, when sanitization has not been a viable option and generation of classified waste results. This classified waste requires secure disposal.* Currently, DOE does not have a policy or formal process for management and ultimate disposal of classified waste. This has been complicated by the fact that classified waste is regulated by the Atomic Energy Act (AEA), which governs radioactive and classified materials, and, if hazardous, may be regulated by the Resource Conservation and Recovery Act (RCRA). The complexity of the management of classified waste and compliance with both the AEA and RCRA are of great importance to DOE, and as a result, the Albuquerque Operations Office has initiated action to address this issue.

The DOE Disposition Policy, issued in May 1993, provides guidance for disposition of classified materials prior to being considered waste. This policy lays the foundation for management of nuclear weapons, subassemblies, components, hardware, or ancillary equipment once they have no further use. This policy requires sites to work with approved practices and procedures for managing these components in accordance with the AEA and RCRA and other authorities as appropriate. Demilitarization and sanitization must be addressed per AEA requirements prior to declaring the components mentioned above as waste.

If the decision is made at the site not to sanitize, and instead to dispose of disassembled parts as classified waste, a policy or, at a minimum, defined processes consistent with the Disposition Policy must be developed to justify that the site has addressed the need for sanitization and determined that it is not feasible, and that disposal plans for the resulting waste do not violate any regulatory or DOE requirements. It is DOE's intent to integrate such a "classified waste policy" with the already existing DOE Disposition Policy.

For the purposes of this paper, the term classified waste is defined as classified material that will not be sanitized based on technical, Environmental Safety and Health (ES&H), or economic justifications. Classified waste is ultimately disposed of at DOE approved disposal sites which meet AEA and disposal site requirements to include RCRA and low-level waste requirements, as appropriate. The term ultimate disposal is defined as the fate of a classified material once all programmatic and

statutory concerns have been addressed and the dismantlement program is no longer concerned with it.
This paper does not attempt to establish a DOE policy or procedures regarding management of classified wastes from dismantlement activities. The contents of this paper reflect the opinions of the authors and do not necessarily reflect the position of DOE.
SIGNIFICANT ISSUES
The following sections discuss the significant issues associated with classified waste management and ultimate disposal resulting from dismantlement.
Long-Term Capacity and Costs
Classified waste must be disposed of in secure classified waste landfills. These landfills must be guarded in perpetuity, or until the waste is declassified* or unearthed and sanitized. Continued generation of classified waste will require DOE to evaluate long-term landfill capacity and security. As part of DOE waste minimization objectives and as a result of diminishing budgets, DOE should reevaluate disposal of classified waste as a preferred option. With capacities for all waste disposal within DOE slowly being absorbed, classified waste disposal should be reviewed as part of a site's waste minimization program, as well as part of a DOE-wide waste minimization effort.
The two primary DOE facilities to which most NWC sites send classified wastes for disposal are the Nevada Test Site (NTS) and Savannah River Site (SRS). However, several other sites have on-site classified disposal capabilities.
A potential problem arises if it becomes necessary to close and remediate a classified waste landfill. In that situation, all of the classified waste would have to be retrieved and sanitized or re-disposed of in another secure classified landfill. The potential costs of this process must be reviewed when analyzing trade-offs of sanitization versus disposal.
Although at an individual site, disposal of classified waste may appear to be more cost-effective than sanitization in the short-term, DOE should weigh those costs against the cost of infinite protection of the classification of the waste at the disposal site and any future disposal site to which the waste may be transferred. By including waste minimization into the budget equation, DOE will likely be able to effectively justify sanitization versus disposal in most cases. If classified waste generation is not minimized, DOE budgets must begin to project for outyear classified waste capacity requirements.
Demilitarization and Sanitization
Ideally, demilitarization and sanitization of nuclear weapons components should occur prior to ultimate disposal. Once a classified component has been demilitarized and sanitized, it is no longer classified and can be disposed of with other unclassified wastes, as appropriate. Currently, however, sanitization of classified components is often not determined to be the best option. At present, disposal of these materials as classified wastes is the option chosen at most NWC sites.
declassification is an administrative determination that a component is no longer classified.
Within the NWC, various obstacles exist associated with sanitization of classified materials. These obstacles include:
- Lack of on-site technical capability to perform sanitization;
- Lack of available financial or personnel resources to support sanitization activities;
- Lack of an appropriate facility in which to sanitize classified materials; and
- Difficulties and expenses associated with ensuring worker safety and protecting the environment during the sanitization process.

DOE sites must consider these potential obstacles when deciding whether or not to sanitize classified material prior to disposal. As noted above, the near-term versus long-term cost to dispose of waste generally has been the deciding factor to dispose of parts as classified waste instead of demilitarizing and sanitizing. Budget cuts are becoming commonplace throughout the NWC and it is often difficult to be visionary and consider the long-term DOE budget picture while attempting to manage the current fiscal year budget at one site. DOE must play a balancing act by strategically planning for the future while concurrently managing the complex on a day-to-day basis.
Statutory RequirementsManagement and control of nuclear weapons components, including dismantlement operations, fall under the authority of the AEA. The AEA

limits control of nuclear weapons, components, and materials; and the dissemination of "restricted data," and specifies that:

"It shall be unlawful, except as provided in Section 91, for any person to transfer or receive in interstate or foreign commerce, manufacture, produce, transfer, acquire, possess, import, or export any atomic weapon" (42 U.S.C 2122).

"It shall be the policy of the Commission to control the dissemination and declassification of Restricted Data in such a manner as to assure the common defense and security" (42 U.S.C 2161).

As a result of these AEA provisions, any classified material destined for ultimate disposal must be guarded in perpetuity or until such time that it is declassified or sanitized.

Disposal of solid and hazardous wastes are subject to the provisions of RCRA and respective state waste management regulations.

With regard to the AEA, RCRA provides that:

"Nothing in (RCRA) shall be construed to apply to (or to authorize any state, interstate, or local authority to regulate) any activity or substance which is subject to the Atomic Energy Act of 1954 except to the extent that such application (or regulation) is not inconsistent with the requirements of such (Act)" (42 U.S.C 6905(a)).

Interpretation of this clause of RCRA varies to the extent that many states that administer RCRA have determined that RCRA applies to any classified wastes that are hazardous or are considered mixed wastes (contaminated with both hazardous and radioactive constituents), while at least one state has determined that this provision of RCRA exempts classified waste from RCRA requirements. This inconsistency in interpretation will likely continue until such time that EPA or the courts issue an interpretation on RCRA applicability to classified waste. While the current interpretation may vary from site-to-site, the NWC general interpretation has been that RCRA does, in fact, apply to hazardous and mixed classified wastes and that an inconsistency need not necessarily exist.

Another area subject to interpretation is whether classified materials should be considered waste. RCRA contains statutory requirements that apply to solid waste and includes definitions of solid waste. It has become apparent that within DOE there are varied interpretations of terminology associated with classified material destined for ultimate disposal. For example, one site may not consider a classified hazardous material that is sent for burial in a classified landfill a waste because it still requires protection and cannot be considered discarded. Another site may consider the material to be waste because it has no future use and consider its burial in a landfill as discarding. DOE is working to develop definitions associated with the term "ultimate disposal" as it relates to classified material for which the NWC has no further use.

Stakeholder and Security Concerns

As DOE and states further cultivate their partnerships to evaluate and prioritize site issues, the states and other stakeholders have become increasingly involved in DOE decision-making processes. If classified waste is determined to be subject to RCRA by a state that has received authorization from EPA to implement RCRA, DOE is then required to comply with the law regardless of the fact that the waste is classified. It is DOE's responsibility to manage classified waste at each site in compliance with all applicable regulations, including AEA and RCRA. Although states may not have access to the classified waste, it is DOE's responsibility to assure the states that classified waste is managed safely and compliantly and not to withhold RCRA characterization or disposal information from the states due to waste classification.

Inconsistent Procedures

The decision process regarding disposal of classified wastes versus sanitization is different from site to site within the NWC. Factors such as funding, technical capability, worker safety and protection of the environment are all considered in disposition decisions. There is not, however, a consistent manner across NWC sites to assess whether the site's justification for ultimate disposal versus sanitization is the best option, or whether classified waste destined for ultimate disposal is managed effectively.

CURRENT PRACTICES AT NWC SITES

The NWC sites currently have a variety of approaches to management of their classified waste. The classified waste generated encompasses the following waste

types:
  hazardous;
  non-hazardous and non-radioactive; and
  low-level radioactive.
Current classified waste disposition activities at NWC sites include disposal of hazardous; radioactive; and non-hazardous and non-radioactive classified waste, including sanitary waste at on-site disposal facilities; as well as disposal of low-level radioactive classified waste off-site at approved DOE facilities.
General obstacles to sanitization were discussed previously in this paper. Specific obstacles being encountered at NWC sites at this time include:
  Lack of equipment to sanitize radioactive contaminated components.
  Lack of facilities in which to sanitize radioactive contaminated components.
  Costs of sanitization, in particular for certain small quantities of classified wastes.
  Lack of available personnel to perform sanitization activities.
  Potential worker hazards.
  Lack of awareness of sanitization options.

AVENUES FOR RESOLUTION

Efforts have been initiated within the NWC to develop a policy statement that will minimize the generation and disposal of classified waste while providing guidance for development of practices and procedures to effectively manage the classified wastes that are generated.
The Albuquerque Operations Office (AL), as the lead NWC office, has selected representatives with experience in weapons programs, environmental regulations, waste management, and the AEA to develop a draft policy statement, or at a minimum defined procedures, which will be forwarded to DOE/HQ for consideration. It is not the intent of this group to prohibit NWC sites from generating any classified waste, but rather to provide a comprehensive review of current classified waste management practices and how they reflect DOE's strategic goals of waste minimization and effective management of classified waste while meeting all regulatory requirements.
The AL group is working toward developing a policy or procedures that will both encourage the minimization of classified waste generated and disposed, and develop practices and procedures to effectively manage the classified wastes that are generated.
Any policy or procedures that are implemented by DOE should:
  Ensure that hazardous, mixed, and radioactive classified wastes are disposed of in full compliance with all applicable regulatory requirements;
  Define practices and procedures for NWC sites to effectively manage classified wastes destined for ultimate disposal, while ensuring site-specific flexibility;
  Further mitigate the problem by generating classified waste only as a last resort; and
  Be consistent with DOE's Disposition Policy.
A flow diagram illustrating a potential site decision-making process regarding sanitization versus generation of classified waste is shown in Fig. 1. This diagram emphasizes DOE involvement in site decisions to sanitize or dispose as classified waste. Criteria to review each site's classified waste management practices will not merely be based on economics, but will also include factors such as schedule, capability, health and safety concerns, regulatory requirements, and available disposal capacity.
Fig. 1.
The AL team approach has been beneficial in ensuring that both the environmental and weapons program concerns are objectively addressed to the benefit of DOE. It is envisioned that this team approach will overlap at the DOE/HQ and site level to effectively integrate all stakeholders.
In developing a classified waste management policy, DOE is faced with making potentially difficult decisions that affect long-term versus short-term goals by encouraging sanitization as the first option, and classified waste generation and disposal as a last resort.

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APPROACH FOR ADDRESSING ENVIRONMENTAL JUSTICE IN ENVIRONMENTAL STUDIES

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ABSTRACT

This paper discusses the inclusion of environmental justice in environmental documents that are prepared in compliance with the National Environmental Policy Act (NEPA). Environmental assessments (EAs) and environmental impact statements (EIS) contain sections which describe the affected environment and impacts on the affected environment. This paper discusses an approach that would be used to incorporate environmental justice in these standard sections of environmental documents. It also addresses basic concepts such as "disproportionately high and adverse," and the relationship to the traditional concept of significant. Some practical examples are given of the types of data and analyses which can be included in the description of the affected environment and in the evaluation of environmental impacts.

DISCLAIMER

Opinions and conclusions expressed in this paper are those of the authors only. Unless so stated in writing in other documents, these opinions and conclusions should not be construed to represent the positions or policies of any organization or government agency.

INTRODUCTION

Environmental justice has assumed an increasingly prominent role in the environmental movement over the past decade. In general, the term "environmental justice" refers to fair treatment of all races, cultures, and income levels with respect to environmental laws, policies, and government actions. In February 1994, Executive Order 12898 titled Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations, was released to Federal agencies. This order requires each Federal agency to incorporate environmental justice as part of its mission. Federal agencies are specifically ordered to identify and address disproportionately high and adverse environmental effects of its programs, policies, and activities on minority and low-income populations. In a related memorandum to heads of all Federal departments and agencies, released concurrently with Executive Order 12898, the President underscores provisions of existing laws that are intended to help ensure the environmental quality of communities throughout the nation. This memorandum further states that mitigation measures identified in environmental documentation should address significant and adverse environmental effects on minority communities and low-income communities.

In addition to describing environmental goals, Executive Order 12898 directs the Administrator of the Environmental Protection Agency to convene an Interagency Federal Working Group on Environmental Justice (referred to below as the Working Group). The Working Group provides guidance to Federal agencies for identifying disproportionately high and adverse human health or environmental effects on minority and low-income populations. The Working Group also coordinates with each Federal agency during development of an environmental justice strategy. Since definitions developed by the Working Group are not finalized, draft definitions are used in the analysis below. In coordination with the Working Group, the Department of Energy and other Federal agencies are developing guidance on implementation of the executive order.

CONCEPTS AND DEFINITIONS

This paper deals with the inclusion of environmental justice in EAs and EISs prepared in compliance with NEPA. The executive order which advocates this inclusion does not explicitly define key terminology. Rather, the development of definitions for terms such as "disproportionately high and adverse," "minority," and "low-income" are delegated to the Working Group. As of this date (January 1995), the development of definitions by the working group is not complete. In this paper we will present and discuss draft definitions, and discuss applications of these definitions to the preparation of environmental documentation.

The concept of "disproportionately high and adverse" effects is central to the implementation of the President's Executive Order. Attempts to define "disproportionately high and adverse" can be controversial because the terminology

is subject to individual perceptions of fairness, and it is difficult to define these terms in a meaningful and unambiguous manner. Analyses presented in this paper use the following definitions:

Disproportionately high and adverse human health effects - Human health effects, including cumulative or synergistic effects, on minority or low-income populations which exceed generally accepted levels of risk.

Disproportionately high and adverse environmental effects - Deleterious effects on biological and other resources within low-income or minority communities significantly exceeding those within the community at large.

Substantially affect human health - To impact human health such that there is a measurable incidence of any specific physical illness, disease, or disorder significantly higher than the national average.

Substantially affect the environment - To impair air quality, ground or surface water quality, animal or plant life, real property values, or other socioeconomic factors, such that there is a measurable impact significantly greater than that for the nation as a whole.

Regulations issued by the Council on Environmental Quality (CEQ) focus environmental documents on effects which are environmentally significant. These regulations specifically require that "Impacts shall be discussed in proportion to their significance. There shall be only brief discussion of other than significant issues." (40 CFR Section 1502.2b). The CEQ Regulations provide a rather detailed description of "significant" in terms of context and intensity (40 CFR Section 1508.27). However, the benefits or burdens of insignificant effects can be disproportionate. For example, if an action results in an insignificant risk of adverse health effects in an area with predominantly low-income households, the action may be perceived as inherently unjust, regardless of the magnitude of the risk. Both the letter and spirit of the CEQ Regulations discourage detailed discussion or mitigation of insignificant environmental effects. In such cases, treatments of environmental justice require a balance between reasonable allocation of resources devoted to effects which are not environmentally significant but which concern basic concepts of fairness. In this paper, it is assumed that in the preparation of NEPA documentation, an environmental effect must be found to be significant prior to any detailed evaluation of disproportionately high and adverse effects. A further discussion of "significant" verses "disproportionate" is given in Section 4 below.

Two other terms are of central importance because the subject executive order addresses environmental justice for two specific groups. One is "minority population" and the other is "low-income population." In this paper, we adopt the following definitions:

Minority - Individuals classified by the U.S. Bureau of the Census as Negro/Black/African American; Hispanic, Asian or Pacific Islander; American Indian, Eskimo, or Aleut; and "Other Races." The minority population in a potentially affected area is the number of individuals residing in the area who are members of a minority group.

Low-Income Household - A Household for which the median household income is 80 percent or below the median household income for the metropolitan statistical area (urban) or county (rural). The low-income population in a potentially affected area is the number of low-income households residing in the area.

We use the Office of Management and Budget Directive No. 15 as a basis for presentation of racial and ethnic data. This directive advocates the collection and reporting of the following minimum designations:

Race:	American Indian or Alaskan Native
	Asian or Pacific Islander
	Black
	White
Ethnicity:	Hispanic Origin
	Not of Hispanic Origin

Data of this format are available from the U.S. Bureau of the Census. In this paper, we specifically used Table P-12 of the Census Bureau's Standard Tape File 3A (STF3A) to describe the racial and ethnic composition of minorities (1).

Executive Order 12898 specifically identifies low-income populations as one of the

two populations designated for consideration of environmental justice. Using the above definition for low-income households, Table P-80 of STF3A can be used to identify low-income populations. One alternative is to use Table P-121 of STF3A which provides data for individuals who are below the poverty level. Both sets of data are indicators of low-income status. We use the definition above and Table P-80 of STF3A because the executive order specifically cites low-income rather than poverty status.

METHODOLOGY

Description of the Affected Environment

A description of the affected environment is standard for EISs and EAs (40 CFR Section 1502.15). As a consequence of Executive Order 12898, the affected environment sections of EAs and EISs now include a characterization of the potentially affected minority populations and low-income populations. For example, if the proposed action were to potentially affect urban seaports in the United States, the minority populations residing near the ports could be described as illustrated in Fig. 1. Estimates of the minority populations residing within 16 km of the port were obtained from a Geographical Information System (GIS) and 1990 census data extracted from STF3A. For comparison, Fig. 1 also includes a similar description for the U.S. population. In this figure, the ethnic designation "Hispanic" includes Hispanics of all races. Populations shown for American Indian, Eskimo, or Aleut; Asian or Pacific Islander; Black; and Other Races include only individuals who are not of Hispanic origin. For this particular example, the east and west coast ports differ noticeably in the racial and ethnic composition of minority populations residing near the ports. In terms of racial and ethnic percentages of the total population, both ports differ noticeably in composition from that for the total U.S. population.

Figure 2 shows a comparison of low-income households as a percentage of total households surrounding the ports and for the total U.S. This information was also extracted from 1990 census data contained on STF3A. These graphs show the percentage of low-income households residing within 16 km of the urban ports of Philadelphia and Long Beach, respectively. Data for surrounding counties refer to the percentages of low-income households residing in all counties which lie at least partially within a circle of 16 km radius centered at the port. The graphs also show percentages of low-income households for the states which contain the ports and for the U.S. For the port of Philadelphia, the percentage of low-income households residing within 16 km of the port exceeds those for surrounding counties, the state, and the nation. The opposite is true for the port of Long Beach. Data such as that shown in Fig. 2 provides a perspective for comparison of low-income households at the local, state, and national levels.

Fig. 1.

Fig. 2.

In summary, GIS software and U.S. Census Bureau data can provide a useful description of the minority populations and low-income populations residing in geographical areas within the U.S. and its territories. The racial/ethnic composition and income characteristics of the population can be described in some detail. This type of description is useful in the preparation of documents which comply with both NEPA and Executive Order 12898.

ENVIRONMENTAL IMPACTS

Descriptions of the environmental impacts of a government action are central to EAs and EISs (40 CFR Section 1502.16). This section discusses the inclusion of environmental justice in the impacts section for cases in which there are no significant impacts and for cases in which there are significant impacts.

We first consider the case in which there are no significant environmental impacts. As discussed in Section 2, CEQ regulations advocate detailed treatment only of those effects which are environmentally significant. Exhaustive treatment of effects which are not significant is specifically discouraged by the CEQ Regulations. When discussing negligible environmental effects, portions of the impacts section which deal with environmental justice should summarize the evidence that the effects are not significant and cite specific parts of the environmental document that support the findings.

If the government action results in significant environmental effects, then the executive order requires analyses to determine if the action results in disproportionately high and adverse effects on minority populations or low-income

populations. This evaluation in turn requires the identification of minority and low-income populations residing in the affected area. Use of census data combined with block group or census tract boundaries in the Census Bureau's TIGER LINE files can be an especially powerful tool in this evaluation (2).

The approach used in the evaluation depends on whether or not the environmental effects are quantified. In past studies of environmental equity in the siting of hazardous waste or other unwanted facilities, analyses often do not quantify adverse effects on the surrounding populace. Instead the burden of unwanted effects is assumed to fall equally on all residents in an affected area (3,4). Such studies have been concerned with disproportionate burdens rather than disproportionately high and adverse impacts. In order to determine whether or not an effect is disproportionate, it is necessary to answer the question: Disproportionate relative to what? A common approach is to define benefitted and burdened populations and compare statistics such as the percentages of minority residents or mean incomes between the populations (3,4). For example, in a study of sites for Waste-to-Energy Facilities, burdens of the facility were assumed to fall on all residents of the town in which the facility is located (3). The benefitted population was defined to be all persons residing within the service area of the facility - usually the entire county. If the adverse effects are not quantified, then the definition of benefitted and burdened population are imprecise, and presence alone in a given area is assumed to result in a burden. Such studies necessarily focus on disproportionate results, because the lack of quantification permits no comparison of minority and non-minority effects or low-income and non-low-income effects within the burdened population. The use of census data and GIS technology can still provide a useful tool for such studies in cases where the geographical distribution of unwanted facilities relative to low-income or minority populations is of central interest (2).

On the other hand, if the health, social, or other environmental effects are quantified, there is no need to define distinct benefitted and burdened populations in the evaluation of environmental justice. Rather, one can estimate the disproportionality between minority and non-minority or low-income and non-low-income populations within the burdened population. For example, radiological health effects are usually a central concern in the preparation of environmental documentation for actions involving radioactive materials. Models for health effects due to the transportation or storage of radioactive materials allow the quantification of impacts on the population. When combined with census data, these models allow a detailed evaluation of the impacts on minority and low-income populations. Figure 3 shows the low-income population residing within 16 km of the Military Ocean Terminal at Sunny Point, North Carolina. This port recently received urgent-relief shipments of spent nuclear fuel from foreign research reactors (5). Figure 3 was obtained from GIS software and U.S. Census Bureau data. The GIS provides a quantitative display of demographic data within a spatial context. Although estimates of radiological health effects were found to be negligible (5), this type of data could be used to estimate the health effects on populations residing near the port. As illustrated in Fig. 3, GIS processing of census data provides a technique for the spatial resolution of demographic data.

Fig. 3.

CONCLUSIONS

Disproportionately high and adverse effects on burdened populations are the central concern in evaluations of environmental justice. In NEPA documentation, significant environmental effects, if any exist, receive the primary consideration. This paper addresses the position that only those environmental effects that satisfy the requirements for significance given in the CEQ Regulations are appropriate candidates for detailed analysis under environmental justice concerns. Implementation of the analyses with GIS software and Census Bureau data provides a useful tool for characterization of minority and low-income populations, as well as quantification of environmental effects on minority and low-income populations.

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STREAMLINING THE NATIONAL ENVIRONMENTAL POLICY ACT COMPLIANCE PROCESS

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ABSTRACT

In June 1994 Secretary of Energy Hazel O'Leary issued the Secretarial Policy on the National Environmental Policy Act. In the introductory memorandum, she stated, "I am directing a number of actions to streamline the NEPA process, minimize the cost and time for document preparation and review, emphasize teamwork, and make the process more useful to decision makers and the public." This paper evaluates the streamlining strategies that several DOE sites are using to meet the challenges presented in the Secretary's policy. Emphasis is on time- and cost-saving measures, quality improvement, and stakeholder involvement. Several NEPA compliance personnel from various DOE sites were interviewed to gain a better understanding of present conditions and goals for the future.

INTRODUCTION

In June 1994, Secretary of Energy Hazel O'Leary issued the Secretarial Policy on the National Environmental Policy Act (NEPA). In the introductory memorandum, she stated, "I am directing a number of actions to streamline the NEPA process, minimize the cost and time for document preparation and review, emphasize teamwork, and make the process more useful to decision makers and the public." Her policy statement instructed the U.S. Department of Energy (DOE) to seek out and implement innovative strategies to streamline the NEPA compliance process and called for the completion of each DOE environmental impact statement (EIS) in just 15 months (1). This new time allotment is an ambitious goal. The average time needed for the preparation of a DOE environmental assessment, a much less detailed NEPA document than an EIS, has been 15 months (2).

In reviewing the DOE's EIS process, it is evident that improvement is needed. At the November 1992 meeting of the Weapons Production Complex Area Managers, plant managers, and design laboratory managers, the plant and laboratory managers identified the need to reduce the review and approval time for documents prepared under NEPA as one of their top five issues (2). In response to their concerns, a Total Quality Process Management Team was formed to address problems in the NEPA process. Their main focus was to improve the process for preparing environmental assessments; however, the recommendations contained in the Report of the Environmental Assessment Process Improvement Team also apply to the preparation of EISs.

In November and December of 1994, a survey was circulated among six DOE organizations that are responsible for NEPA document preparation. These organizations consisted of the Rocky Flats, Hanford, Oak Ridge, Savannah River, Waste Isolation Pilot Plant, and West Valley sites. Responses were received from five of the six organizations. In addition, some of the surveys were followed up by personal interviews for clarification.

Survey and interview results indicated that the DOE has welcomed the challenges proposed by the Secretary of Energy and the process improvement team to " ... make NEPA work better and cost less." DOE organizations are striving to implement and share creative strategies for streamlining the NEPA compliance process.

The survey participants are from organizations with varied missions, but they all have similar expectations from the NEPA documentation process--initiating projects on schedule; getting the maximum value added in return for time and money spent; making informed, environmentally sound decisions; and having stakeholder support for decisions made. EIS document managers are tired of unnecessary project delays, "midstream" change in scopes, waste of time and money expended, "truck-load" volume EISs, and dissatisfied stakeholders.

To date, studies have been done, groundwork established, and a new NEPA policy issued; however, system-wide changes are not easily achieved. The following discussion examines some of the methods being implemented by the DOE to make the transition from an era of slow-moving and costly environmental analyses to a better, faster, less expensive approach to the NEPA process. Changes being initiated include reducing the average length of EISs, increasing reliance on reference documents, getting the most out of EIS contracts, creating EIS management teams, and maximizing stakeholder involvement. Lessons learned at the various facilities are being shared so that the DOE may make fundamental, positive changes to the way its NEPA business is conducted.

EIS TIME AND COST REDUCTION

One of the ways the DOE intends to meet the 15-month EIS schedule and reduce document preparation costs is to reduce the volume of the documents. If documents are smaller, it logically follows that they will cost less and will take less time to produce. Brief EISs have always had the approval of the Council on Environmental Quality (CEQ). Their regulations state that "Environmental Impact Statements shall be analytical rather than encyclopedic." The CEQ also recommends that the text of final EISs " ... shall normally be less than 150 pages and for proposals of unusual scope or complexity shall normally be less than 300 pages." (3)

The DOE is getting back to basics, carefully prioritizing those issues that require analyses. Sometimes this involves a re-evaluation of traditional EIS formulas. The Savannah River
Site's F-Canyon Plutonium Solution Environmental Impact Statement provides only a limited discussion of the following:

Geologic resources	Noise impacts
Aesthetics/scenic resources	Socioeconomics
Ecological systems	Cultural resources

The Environment Impacts Section justifies the narrow discussion of these topics by indicating that minimal impacts would be expected to impact these resources. This section then focuses analyses on areas deemed worthy of closer attention such as health effects, air and water resources, utilities, waste management, land use, and transportation (4). By omitting sections of "boiler plate" information from the Environmental Impacts Section, the preparers have succeeded in reserving comprehensive analysis for legitimate issues (5). Readers wanting more information are directed to several other site environmental documents that are available in public reading rooms.

DOE is currently planning the preparation of their second supplemental EIS (SEIS-II) for the Waste Isolation Pilot Plant (WIPP). To reduce cost and document preparation time, the DOE plans to curtail the use of new and original environmental analyses that are not needed (6). This will be accomplished by incorporating, by reference, pertinent analyses contained in other WIPP compliance documents such as the WIPP No-Migration Variance Petition, the WIPP RCRA Part B Permit Application, and the WIPP Safety Analysis Report.

The DOE also hopes to reduce costs for the WIPP SEIS-II by providing the contractor with an annotated outline (7). Use of the annotated outline should reduce contract costs and ensure a better quality product by providing clear objectives to the contractor for each section that will be included in the document.

NEPA personnel at the Hanford Site state that they have reduced EIS preparation costs by dividing the project into several small pieces. For their Safe Interim Storage of Hanford Tank Wastes Environmental Impact Statement (8), NEPA contractors carefully evaluated staff assignments and utilized lower cost staff for less important aspects of the EIS. Furthermore, no staff members were assigned full-time to the EIS, so only productive hours were billed to the EIS budget (9). The DOE plans to take a similar approach for the WIPP SEIS-II, contracting out distinct analytical sections to the contractor so that only productive hours are billed (6).

Another strategy now being employed by the DOE to slash months from EIS schedules is the creation of project-specific management teams that are responsible, start to finish, for the collective preparation of EISs and for cooperative comment resolution. These management teams, with participants from all of the required organizational units, hold interactive internal review meetings, sometimes resolving comments for an entire EIS in just two weeks (5).

QUALITY IMPROVEMENT AND STAKEHOLDER PARTICIPATION

Quality improvement and stakeholder involvement go hand-in-hand for an EIS. DOE sites nationwide are making a greater effort to seek out stakeholder views prior to making EIS decisions. Increased up-front communication with the public results in EISs that display a more balanced range of alternatives. Another benefit is that the scope of the EISs is being agreed upon early in document development; fewer changes are occurring midstream in the process.

Savannah River Site NEPA personnel are finding that changing the format of their public meetings has made a significant improvement in the quality of the communication between DOE and the interested public. Savannah representatives stated that the public has been very receptive towards informal meetings. DOE representatives are leaving the platform and lectern behind and meeting face-to-face with the public for one-on-one question-and-answer sessions. For the F-Canyon Plutonium Solutions Environmental Impact Statement, even the draft hearings had a large informal component. For the first few hours of the meeting, stakeholders were asked if they had any questions or concerns that they would like to submit to the DOE representatives "off the record." At the conclusion of this informal comment period, the participants were given the opportunity to provide formal statements to a court recorder. Many of the attendees were so pleased with the informal discussion that they no longer felt their issues required formal response and resolution (5).

The WIPP has prepared a draft Stakeholder Involvement Plan which outlines the public outreach efforts that are planned for the SEIS-II. The SEIS-II preparation formula will include conducting intensive public participation meetings and will focus on individuals who have historically held interest in the WIPP project (10). By obtaining key stakeholder input at the beginning of the EIS process, the DOE expects to reduce later changes to the document's scope and to have greater public support for the alternatives that are chosen for evaluation. The DOE also plans to increase public accessibility to the SEIS-II development by using people instead of machines to answer the comment phone lines during business hours and by accepting comments via fax line and electronic bulletin board. These steps will reduce costs associated with the SEIS-II while enhancing opportunities for public involvement.

While the DOE plans to enhance opportunities for public involvement with the SEIS-II, it wants to ensure that time and money are expended only in response to genuine public interest. DOE representatives will hold fewer formal meetings at fewer locations for the SEIS-II. This approach responds to the lessons learned during the last SEIS preparation process. Attendance was low at some of the locations where previous hearings were held (DOE representatives outnumbered the attendees five to one), and few of those that did attend had a genuine interest in the WIPP project. Accordingly, the draft and final SEIS-II will also be provided to fewer public reading rooms than the previous SEIS.

CONCLUSION

The DOE has begun a journey that will lead to a streamlined NEPA process. However, no single strategy will result in the needed improvements, and the solutions now being attempted will not result in overnight success. Some of those interviewed felt that changes are not occurring quickly enough and that the DOE needs to make an even greater commitment to internal cooperation. The DOE should ensure that EIS management teams are composed of those with decision-making authority, that the EIS scope is agreed upon, and that contractors are provided with clear performance expectations.

The 15-month EIS envisioned by the Secretary is within the grasp of the DOE. The DOE is beginning to " ... make NEPA work better and cost less," finding success in reducing review cycle time, focussing environmental analysis on worthy issues, and strengthening communication with stakeholders. Improvements being made in one area of the NEPA process are triggering positive changes to other areas, and lessons learned with each EIS are being shared for the benefit of future preparers.

The DOE has taken the Secretary's challenge to heart, restructuring the way its NEPA business is conducted. Creativity, perseverance, and shared responsibility will ensure that the new approaches succeed.

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COMPUTER-AIDED WASTE MANAGEMENT STRATEGIC PLANNING AND ANALYSIS*

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ABSTRACT

A computational model called WASTE_MGMT has been developed to assist in the evaluation of alternative waste management approaches in a complex setting involving multiple sites, waste streams, and processing options. The model provides the quantities and characteristics of wastes processed at any facility or shipped between any two sites as well as environmental emissions at any facility within the waste management system. The model input is defined by three types of fundamental waste management data: 1) waste inventories and characteristics at the point of generation; 2) treatment, storage, and disposal facility characteristics; and 3) definitions of alternative management approaches. The model has been successfully used in the preparation of the U.S. Department of Energy (DOE) Environmental Management Programmatic Environmental Impact Statement (EM PEIS). Certain improvements are either being implemented or planned that would extend the usefulness and applicability of the WASTE_MGMT model beyond the EM PEIS and into the strategic planning for management of wastes under the responsibility of DOE or other agencies.

INTRODUCTION

The process of "waste management" can be as simple a task as packaging and properly disposing of a single waste stream at a single site, or it can be a very complex undertaking involving many sites with multiple waste streams subject to a variety of pretreatment, treatment, storage, packaging, transportation, and disposal options. Such complex waste management issues are being faced by the U.S. Department of Energy (DOE) and other federal agencies such as the U.S. Department of Defense. To provide an overall management strategy for all radioactive and hazardous waste generated or stored at its many installations around the country, the Office of Environmental Management (EM) of DOE is preparing a Programmatic Environmental Impact Statement (PEIS), often referred to as the EM PEIS. An internal review draft of the EM PEIS was issued in late November 1994. After being reviewed by the various DOE headquarters and field offices, the EM PEIS will be revised in response to comments and will be distributed for public and other agency (other than DOE) review. The current schedule calls for this distribution to take place in late March 1995.

The EM PEIS evaluates alternative management strategies for DOE wastes at approximately 50 sites and for five types of waste: high-level (HLW), low-level (LLW), transuranic (TRUW), low-level mixed (LLMW), and hazardous (HW). For each waste type, up to several thousand waste streams can exist at a given site. The number of alternative strategies evaluated varies by waste type from a few to more than 10. Wastes generated as part of on-going operations, wastes in storage from

past operations, and wastes that will be generated as part of environmental restoration activities are considered.

In such complex cases involving multiple sites, waste streams, and processing options, a computer model can be invaluable in analyzing the various waste management strategies and selecting the alternatives that meet certain predefined criteria. Argonne National Laboratory (ANL) scientists have developed a computational model called WASTE_MGMT for use in assessing the impacts associated with alternative strategies for management of the wide array of wastes being considered in the EM PEIS. The model has been used successfully for the EM PEIS and has the potential to be a valuable strategic planning tool for future management of DOE wastes. The model can also be adapted for similar use by other agencies.

This paper provides a brief description of the WASTE_MGMT computational model, explains how the model was used in the EM PEIS project, and discusses the ongoing and planned improvements to the model. The sources of data used as input to the model for the EM PEIS application are listed. Potential future uses of the model are discussed. More information about the WASTE_MGMT computational model can be found in Refs. 1 and 2.

DESCRIPTION OF WASTE_MGMT COMPUTATIONAL MODEL

In its current state of development, the WASTE_MGMT computational model calculates the following quantities for a given waste management strategy:

Annual quantities and characteristics of the wastes processed by each treatment, storage, and disposal (TSD) module (A module is a facility that houses one or more processes for TSD of a certain type of waste and can be characterized by material and/or energy flow into and out of it.);

Annual quantities and characteristics of the wastes shipped among sites;

Annual emissions of radionuclides and hazardous chemicals to air and water resulting from the operation of the TSD modules; and

Annual generation rates and characteristics of secondary waste streams resulting from the operation of the TSD modules.

To estimate the above, the WASTE_MGMT model requires three types of input data: 1) waste inventory and characteristics at each site; 2) characteristics of the technology modules used to sort, package, treat, store, and dispose of the waste; and 3) definitions of alternative management approaches. Waste inventory and characteristics data include the inventory in storage, estimates of future generation rates, radionuclide and hazardous chemical concentrations in the waste, and physical form of the waste. For the EM PEIS application, these data were collected for each waste type, site, and treatability group. Treatability groups were defined in such a way that they would encompass the large number of waste streams at a given site and so that the wastes within each group would be subject to the same kinds of TSD options. For example, 10 treatability groups were defined for LLW, 8 for TRUW, 23 for LLMW, and 9 for HW.

The TSD modules are characterized by parameters that relate the output quantities and resource requirements to unit quantity of waste processed by each module. Parameters are developed not only for the bulk waste quantities (such as volume and mass), but also for the contaminants (radionuclides and hazardous chemicals) in the waste.

Alternative waste management approaches are defined by specifying the processing paths followed by wastes in each treatability group from site of origin to disposal site, including the identity and location of TSD modules. The fate of the secondary waste streams generated during treatment operations is also defined.

The model is implemented in Microsoft FoxPro 2.5 for MS-DOS, extended version, and consists of approximately 4,200 lines of FoxPro code with comments. An additional 2,800 lines of code were written for preparation of the model's base tables. The processing time required to obtain results for a given strategy is strongly dependent on the complexity of the strategy and the processing speed of the computer used. Using a 90 Mhz INTEL Pentium processor, the computation times vary from less than a minute for a simple case up to 15 minutes for a very complex case.

APPLICATION OF WASTE_MGMT COMPUTATIONAL MODEL IN THE EM PEIS

The WASTE_MGMT model was used for evaluating alternative strategies for the management of three of the waste types considered in the EM PEIS: LLW, LLMW, and TRUW. Because of the unique features of HLW and HW, the alternatives for these two waste types were evaluated with approaches that were specifically designed for them.

The application of the WASTE_MGMT model for the evaluation of LLW, LLMW, and TRUW alternatives is discussed below.

Input Data and Sources

As discussed above, three types of input data were needed for the EM PEIS application. The nature and sources of these data items are discussed below.

Waste Inventory and Characterization: The waste inventory and characteristics data were collected by ANL from the best available sources within the DOE system. Data bases such as the Integrated Data Base (IDB) (3), the Waste Management Information System (WMIS) database (4), and Mixed Waste Inventory Reports (MWIRs) (5,6) were used. When the available data were not sufficient for the needs of the EM PEIS, the DOE sites were contacted to obtain additional data. In the event that the needed data could not be obtained, engineering judgment and approximations were used as necessary.

The waste inventory and characteristics data used for the analysis of LLW alternatives in the EM PEIS are described in Reference 7. The LLMW data were processed from the MWIR data bases with the help of Byron Palmer at Los Alamos National Laboratory (8) and Burdon Musgrave (9). The LLMW data used for the EM PEIS are provided in Refs. 8, 9, 10, 11, and 12. The TRUW data are available in Hong et al. (13). All these data will also be discussed in the EM PEIS.

TSD Module Characterization: The characterization parameters for the TSD modules considered in the EM PEIS were obtained primarily from the available literature. The material reviewed included documentation such as reference design reports, the Functional and Operational Requirements Report prepared by the Mixed Waste Treatment Project (14), trial burn test results from facilities permitted under the Resource Conservation and Recovery Act (RCRA), and various reports and publications in the open literature. The DOE or commercial sites operating the various facilities were contacted as needed to obtain additional information. Engineering judgment was used when the needed information was not available. Reference 7 provides the TSD characterization parameters used for LLW alternative evaluation in the EM PEIS. The parameters for LLMW and TRUW are provided in Refs. 8, 9, 10, 13, and 15.

The technologies employed for the TSD wastes in the EM PEIS are those that are either currently available or will soon be available. The characterizations of these technologies are conducted in a generic sense, and the same parameters are used for any site where the technology is located.

Definition of Alternative Strategies: The alternative strategies for managing each waste type were developed by META/Berger in consultation with DOE. The strategies were developed in terms of defining the treatment and disposal options and the names of the installations where the TSD facilities are located. The specific locations of the facilities at a given installation were not defined. The EM PEIS will provide descriptions of these alternative strategies.

Output Data and Uses

The output data currently calculated by the WASTE_MGMT computational model are listed above. For the EM PEIS, these results were subsequently used to calculate the human health risks due to facility operations (by Oak Ridge National Laboratory [ORNL]), human health risks due to transportation (by ANL), costs (by EG&G, Idaho and MK Corporation), ecological risks (by ORNL), facility accident source terms (by ANL), and air and water quality impacts (by META/Berger). These impacts were calculated outside of the WASTE_MGMT model by manual transmission of the output data from the WASTE_MGMT model to other models or spreadsheet applications. For example, the data obtained from WASTE_MGMT on air emissions of radionuclides were used to calculate the radiological risks to on-site workers and the off-site public by multiplying the released quantity of each radionuclide by an appropriate unit-risk factor for that radionuclide. The unit-risk factors, which were developed by ORNL, take into account site-specific conditions such as meteorology, population distribution, and site terrain. A separate factor was calculated for each radionuclide and receptor by assuming that a unit quantity of the radionuclide was released from a certain point at a site. Similarly, the transportation risks were calculated from 1) the WASTE_MGMT model output data on quantities and characteristics of the wastes shipped among sites and 2) transportation unit-risk factors (risk per unit distance traveled) developed by ANL.

FUTURE USE OF WASTE_MGMT COMPUTATIONAL MODEL

The fragmentation of analyses for the EM PEIS was necessitated by the division of labor among the organizations participating in preparation of the document and by

the tight schedules involved. Under different conditions, this fragmentation could be avoided, and WASTE_MGMT could be expanded into an integrated model that could calculate costs and environmental impacts associated with a given waste management strategy. For example, site-specific and radionuclide-specific unit-risk factors (risk per unit release) could be incorporated into the WASTE_MGMT model as a data file. The risks resulting from facility operations could then be calculated within the model by simple multiplication of the released quantities and the unit-risk factors.

Cost curves could be developed for the generic TSD facilities using the EM PEIS cost estimating methodology. These curves would define the costs of constructing and operating a facility as a function of the size of the facility (i.e., the quantity of waste processed at the facility). Such cost curves could be incorporated into the WASTE_MGMT computational model, and the model could then be used to estimate the facility costs associated with a given strategy. This process would be accomplished by interpolating on the cost curves for facility waste loads that correspond to the given strategy. Such curves could also be developed and used in the model for estimating facility resource requirements (e.g., staffing, energy, materials, and land).

In its fully developed form, the WASTE_MGMT model could be used to estimate the costs, risks, and resource requirements for a given waste management strategy at DOE installations. The model would be useful for DOE strategic planning efforts beyond the EM PEIS. In addition, the model was designed such that, given the waste inventory and characterizations data, it can also be used for installations other than those owned and operated by DOE.

Finally, the model could be equipped with a user-friendly graphical interface to facilitate its use by individuals who are not familiar with the design of the model. Alternatively, the results of complex cases from prior runs could be cataloged, and these could be used, through the user interface, to provide quick responses to questions frequently asked by the public or DOE strategic planners. Such improvements would greatly increase the usefulness of the model to future EM PEIS work, as well as to DOE waste management strategic planners. These improvements are either currently being worked on or will be initiated in the near future.

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ASSESSMENT OF TRANSPORTATION RISK FOR THE U.S. DEPARTMENT OF ENERGY ENVIRONMENTAL MANAGEMENT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT*

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ABSTRACT

In its Programmatic Environmental Impact Statement (PEIS), the Office of Environmental Management (EM) of the U.S. Department of Energy (DOE) is considering a broad range of alternatives for the future management of radioactive and hazardous waste at the facilities of the DOE complex. The alternatives involve facilities to be used for treatment, storage, and disposal of various wastes generated from DOE environmental restoration activities and waste management operations. The evaluation includes five types of waste (four types of radioactive waste plus hazardous waste), 49 sites, and numerous cases associated with each alternative for waste management. In general, the alternatives are evaluated independently for each type of waste and reflect decentralized, regionalized, and centralized approaches. Transportation of waste materials is an integral component of the EM PEIS alternatives for waste management. The estimated impact on human health that is associated with various waste transportation activities is an important component of a complete appraisal of the alternatives. The transportation risk assessment performed for the EM PEIS is designed to ensure through uniform and judicious selection of models, data, and assumptions that relative comparisons of risk among the various alternatives are meaningful and consistent. Among other tasks, Argonne National Laboratory is providing technical assistance to the EM PEIS on transportation risk assessment. The objective is to perform a human health risk assessment for each type of waste relative to the EM PEIS alternatives for waste management. The transportation risk assessed is part of the overall impacts being analyzed for the EM PEIS to determine the safest, most environmentally and economically sound manner in which to satisfy requirements for waste management in the coming decades.

INTRODUCTION

The U.S. Department of Energy (DOE) Office of Environmental Management (EM) is responsible for the treatment, storage, and disposal (TSD) of various wastes generated within the facilities of the DOE complex from environmental restoration activities and waste management operations. For the EM PEIS that is being prepared (1), alternatives are considered relative to the decentralized, regionalized, and centralized approaches. The types of radioactive waste being evaluated include high-level, transuranic, low-level, and low-level mixed. Hazardous waste is also being evaluated. The magnitude of the transportation activities varies with the alternative and ranges from minimal transportation for the decentralized alternative to significant transportation for the centralized alternative. Various aspects of the impact of transportation are radiological versus hazardous, rail versus truck, routine versus accident, worker versus public, population versus individual, and aggregate risk (i.e., probability times consequence) versus consequence. Because of

complex-wide considerations, large amounts of waste in some alternatives are expected to result in unprecedented large-scale transportation activities for DOE or other federal agencies; the potential cumulative impacts of these activities may warrant special attention.

The magnitude and complexity of the scope of the EM PEIS present the following unique features regarding transportation risk assessment: 1) large amounts of waste requiring shipment result from programmatic considerations that are unprecedented for DOE and other federal agencies, 2) consistency exists between the radiological and hazardous risks, and 3) the magnitude and complexity of the routes for shipment require an integrated and expedient computational approach. The EM PEIS method represents a comprehensive and integrated approach to assessing risk regarding the transportation of radioactive, hazardous, and mixed waste.

TECHNICAL APPROACH

The technical approach for conducting the transportation risk assessment was developed following a thorough and critical review of the literature and existing documentation prepared for major federal actions under the National Environmental Policy Act (NEPA). Consideration was also given to recent commitments arising from public awareness and litigation. The approach has also received DOE internal review and external review outside of DOE.

The transportation of radioactive and hazardous wastes involves risk to both crew members and members of the public. Part of the risk results from the nature of the transportation operation itself, which is independent of the waste contents or characteristics of the cargo; for instance, traffic accidents during transportation may cause direct injuries or fatalities. Similarly, pollution from vehicle exhaust emissions may also affect human health. These risks are characterized as "vehicle-related." On the other hand, the transportation of waste may pose an additional risk because of the characteristics and potential hazards of the cargo (i.e., waste) itself. These risks are therefore considered "cargo-related."

The technical approach is conceptualized in Fig. 1 for the EM PEIS transportation risk assessment. A more detailed description of the method has been provided by Monette et al. (2) for transportation of radioactive waste and by Hartmann et al. (3) for transportation of hazardous waste. Potential risks were estimated for the collective population, as well as for the maximally exposed individual, during routine transportation and accidents. For radioactive wastes, the cargo-related risks include exposure to external radiation from the waste package during routine operation and potential exposure to dispersed waste contents in accidents. The RADTRAN 4 (4) computer code was used to estimate the collective population risk. The approach of RADTRAN has been used extensively in previous NEPA assessments and has received wide acceptance. The collective population risk is a measure of the radiological risk posed to the society as a whole by the alternative being considered. As such, the collective population risk is a reasonable primary measure for comparing different alternatives. Supplemental analyses were provided by using the RISKIND (5) computer code to address areas of specific concern to individuals or population subgroups. The supplemental analyses are primarily meant to address the "what if" scenarios frequently raised in public comments. Examples of such scenarios are "What if an accident happens near my community?" or "What is the risk to me if I live adjacent to the access road?"

Fig. 1.

The general methodology for characterizing risk from exposure to hazardous chemicals has many parallels to the methodology for characterizing risk for radioactive waste transportation; however, no standard computer code exists in the chemical area that is a direct counterpart to RADTRAN. One of the major tasks of the EM PEIS transportation risk analysis is to develop the details and coding of the general chemical risk methodology that parallels the steps in the RADTRAN method. Included in such details are a realistic treatment of the chemical accident itself, as well as development of a variety of appropriate chemical risk end points. The treatment of fires (leading to toxic gas releases) and water immersion is also to be included in the implementation of the general chemical risk methodology. Unique to this approach is the assessment of the impacts of water immersion.

Another unique feature of chemical risk assessment is that routine operations of hazardous chemical transportation do not lead to cargo-related risks. Chemical risks occur only from accident conditions because small spills or seepages of hazardous or chemical waste during routine operations are kept to a minimum by existing

regulations and packaging of that waste. Consequence modeling of chemical spills during accidents is performed by using the Areal Locations of Hazardous Atmosphere (ALOHA) (6) air dispersion code in conjunction with several end points for human health effects (discussed in the following paragraphs). The ALOHA code has been used extensively by the U.S. Environmental Protection Agency (EPA) for assisting emergency field personnel in planning for accident release consequences and in implementing emergency response measures.

DEVELOPMENT OF END POINTS FOR HEALTH EFFECTS

The potential exposures from transportation of radioactive materials, either from routine operations or from postulated accidents, are usually at a low dose, such that the primary adverse effect is the induction of latent cancer. The correlation of the radiation dose and human health effects for low doses has been traditionally based on what is termed the "linear, no-threshold hypothesis," as described by the International Commission on Radiological Protection (7).

In addition to latent cancer (which is assumed to be a linear, no-threshold effect), exposure to toxic chemicals may also cause threshold, nonlinear effects. These effects are often of an immediate nature (i.e., acute). The severity of the immediate health effects therefore depends strongly on the toxicity and exposure concentration of the specific chemical(s) released and can range from slight irritation to potential fatality for the exposed individuals. Thus, for the EM PEIS human health risk assessment, three end points were assessed: 1) potential for life-threatening effects, 2) potential for other adverse effects, and 3) increased cancer risk. The first two end points are acute. The potential life-threatening effects are specific only to toxic chemicals identified as "poison inhalation hazards" by the U.S. Department of Transportation (DOT) (Title 49, Parts 173.115 and 173.132-133 of the Code of Federal Regulations [CFR]). Estimates of these effects are derived from the 50% lethal concentration (LC50) or other appropriate toxicity values. The toxicity values have been derived from the following sources: 1) the Registry of Toxic Effects of Chemical Substances (RTECS) database (8), and 2) Dangerous Properties of Industrial Materials (9). Estimates of other adverse effects are based on the inhalation reference dose values developed by the EPA, when available. Such data are derived from the EPA Integrated Risk Information System (IRIS) (10) database and the Health Effects Assessment Summary Tables (HEAST) (11). For the effects of latent cancer, assessment is performed for a carcinogenic risk of one in one million (10-6) or higher. The concentration values corresponding to this risk level have also been derived from the IRIS and HEAST databases. A description of the derivation of the concentration values for these end points for health effects is provided in Hartmann et al (3).

PARAMETERS AND ASSUMPTIONS

Major input parameters and assumptions used in the EM PEIS transportation risk assessment are discussed in the following paragraphs.

Waste Inventory and Characterization

A radioactive waste inventory computational model (WASTE_MGMT) (12,13) has been developed by Argonne National Laboratory to support the EM PEIS analysis of risk and cost. The model combines information on the waste inventory and characterization of waste across the DOE complex, on characterization of the TSD modules, and on definitions of the EM PEIS alternatives. Waste properties are provided in terms of isotope- or chemical-specific concentrations. Physical forms of waste are generally classified into a small number of categories such as vitrified waste, liquid waste, metal waste, and heterogeneous solid waste. Hazardous waste is inventoried and characterized with a separate database of shipping manifests collected from the entire DOE complex for the year 1992. From those manifests and associated data, physical and chemical characterizations were obtained, from which data on treatment and disposal could be obtained or estimated.

Packaging And Shipping Configurations

For the transportation of radioactive materials, the basic types of packaging required by the regulations are designated as Type A, Type B, or "strong and tight." The regulatory requirements are those specified by the DOT and the U.S. Nuclear Regulatory Commission (NRC) (49 CFR 173 and 10 CFR 71). For the EM PEIS, all transportation of radioactive waste has been assumed to take place in certified containers and exclusive-use vehicles. Low-level waste and low-level mixed waste are assumed to be transported in Type A packages. The high-level and transuranic wastes require the use of Type B packaging. The transuranic waste will essentially consist

of Type A waste containers within reusable, certified Type B shipping containers, such as TRUPACT-II for contact-handled waste and NuPac 72B for remote-handled waste. For hazardous waste, packaging is assumed to follow the DOT specifications for containers (49 CFR 173).

Shipping Routes

Representative shipping routes have been determined for all possible pairs of DOE sites that generate or store wastes. For each origin-destination pair, representative highway and rail routes were generated and analyzed by using the routing models HIGHWAY (14) and INTERLINE (15), respectively. The routing models are updated periodically to reflect current road and track conditions and have been benchmarked against the reported shipping distances and the observations of commercial truck and rail firms. The routes calculated conform to current routing practices and all applicable routing regulations and guidelines; however, the routes do not necessarily represent the actual routes that will be used to transport waste in the future. For risk analysis, the important routing characteristics include total shipping distance and the fractions of travel in rural, suburban, and urban zones of population density. The hazardous waste methodology uses only HIGHWAY because no such waste is currently or is planned to be sent by rail.

External Dose Rate

Because all shipments are assumed to take place on exclusive-use vehicles, the maximum external dose rate from a radioactive waste package is limited by the regulatory value of 10 mrem/h measured at 2 m from the lateral surfaces of the conveyance. Because of the complex nature of the EM PEIS alternatives and the different types of waste, a representative external dose rate appropriate for each type of waste was determined for the EM PEIS analysis. This dose rate was based on the average of the entire waste inventory appropriate for the EM PEIS analysis and, therefore, was not intended to represent site-specific values. External dose rate concepts are not applicable to hazardous waste.

Traffic Accident Rates

For calculation of the accident-related risk, accident rates for truck and rail transportation were taken from a compilation by Saricks and Kvitek (16). Data include the accident rate, injury rate, and fatality rate. State-specific accident rates for truck shipment were based on statistics compiled by the DOT Office of Motor Carriers for 1986-1988, specifically for heavy combination trucks involved in interstate commerce. These data were benchmarked against the existing nuclear commercial carriers. State-specific rates for rail accidents were based on statistics compiled by the Federal Railroad Administration for 1985-1988. Rail accident rates include both accidents on main lines and those occurring in rail yards. For hazardous materials, data on accidents were based on 1979-1983 California highway accident data for shipment of hazardous waste.

Accident Severity Categories

A range of potential severities for radioactive waste transportation-related accidents has been described by the NRC (17). The NRC scheme for classifying accidents categorizes accidents as a function of the magnitudes of the mechanical forces (impact) and thermal forces (fire) to which transported packages may be subjected. For each category of severity, a conditional probability is also assigned. The RADTRAN accident risk assessment considers the entire range of accident categories as defined by the NRC, including accidents with low probability but high consequences and those with high probability but low consequences. On the other hand, for consequences to maximally exposed individuals and to population subgroups, scenarios analyzed by the RISKIND code include those representing the accident scenarios with the highest release of waste. Hazardous waste accident scenarios were based on accident release data compiled from the DOT Hazardous Materials Incident Reporting System (HMIRS) (18) database. The contents of each DOE shipment were assumed to have a "breach fraction" (expected fraction of containers ruptured) and a "release fraction" (expected amount of each chemical released from those breached containers) during a transportation-related accident. The amounts of the chemicals released were used in the consequence modeling.

Accident Release Fractions

The release fractions relevant to the risk assessment include three major components: the fraction of the waste contents that could be released to the environment for a given severity of accident, the fraction of the released material that can be dispersed to the atmosphere, and the fraction of the dispersed portion

that is respirable. Most solid materials are difficult to release in particulate form and, therefore, are nondispersible even if they are released. Gaseous materials, on the other hand, are relatively easy to release when the container is breached. Release fractions under accident conditions have been estimated by the NRC (17) for Type A and Type B containers under various categories of accident severity. The dispersible fractions and respirable fractions have been suggested by RADTRAN (4) for various forms of materials. Breach and release fractions for hazardous waste are based on analysis of thousands of accidents in the HMIRS database (18).

Atmospheric Conditions

Because predicting the specific location of a transportation-related accident is impossible, generic weather conditions were used for transportation risk assessment. For accident risk analysis, neutral weather conditions (represented by Pasquill stability class D) were assumed. Because neutral meteorologic conditions constitute the most frequently occurring atmospheric stability conditions in the United States, these conditions are most likely to be present during an accident. For maximally exposed individuals, however, both neutral (represented by Pasquill stability class D) and stable (represented by class F) conditions were considered; the stable condition was used to represent a conservative condition intended to maximize the consequence.

Scenarios For Maximally Exposed Individuals

For routine transportation, risks to maximally exposed individuals were estimated for a number of hypothetical exposure scenarios. The receptors include transportation crew members, inspectors, and members of the public exposed during traffic delays, while working at a service station, or while living near a waste site. Parameters typical of such scenarios have been described in previous DOE documents (19,20). For accidents, the maximally exposed individuals are assumed to be downwind. The accident severity category with the highest release was used to represent "worst-case" scenarios for the accident analysis.

For hazardous waste, worst-case accidents were evaluated in a similar manner and were chosen on the basis of the shipments with the maximum quantity, toxicity, and volatility of the hazardous waste within those shipments.

DEVELOPMENT OF AUTOMATED COMPUTATIONAL SYSTEM

Because of the magnitude and complexity of the EM PEIS transportation risk analysis, an automated computational system was developed by Biwer et al. (21) to facilitate the computation of risk for radioactive waste. The system uses the unit-risk factor concept; that is, unit-risk factors derived from the RADTRAN 4 calculations form the basis of the system. Key parameters providing a foundation for the unit-risk factors include transport mode, external dose rates from waste packages, transport speeds, and radionuclide-specific information. Shipment risks are estimated by the combination of the unit-risk factors with state-specific population zone distances and accident rates, package release characteristics, and other relevant factors. For the purpose of the EM PEIS analysis for DOE facilities, the system also incorporated routing information derived from HIGHWAY (14) and INTERLINE (15). The system is designed to incorporate such routing information and unit-risk factors to produce risks for a particular alternative. The system is now equipped with a menu-driven feature for analyzing transportation-related risks for large and complex shipping campaigns required by the EM PEIS. For hazardous waste, a semiautomated system has been set up that links the hazardous waste database, the ALOHA model, and a spreadsheet program that computes risk.

ASSESSMENT OF ALTERNATIVES

On the basis of the technical approach and the assumptions discussed previously, the transportation risk assessment was performed for the EM PEIS. Potential transportation-related health risks were calculated for both all-truck transportation (radioactive and hazardous wastes) and all-rail transportation (radioactive wastes only). For shipments between each origin-destination pair, the collective risks to workers and the public for incident-free conditions and for accidents were assessed. For each alternative, the total collective risks represent the aggregate of risks from the set of origin-destination pairs. For comparison, the vehicle-related impacts (i.e., the total number of traffic accident fatalities) from collisions were also estimated and are presented along with the cargo-related (waste-related) impacts.

CONCLUSIONS

The transportation risk assessment conducted for the EM PEIS is a comprehensive and

integrated approach that can be applied to all types of radioactive waste, as well as to hazardous waste, generated within the DOE complex. By combining a traditional method (for radiological risk assessment) with some developmental components (for hazardous risk assessment), the magnitude and complexity of the analysis are addressed. The results of the assessment constitute an important component of the overall evaluation of waste management alternatives of the EM PEIS.

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RADIOLOGICAL ACCIDENTS POTENTIALLY IMPORTANT TO HUMAN HEALTH RISK IN THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAM*

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ABSTRACT

Human health risks as a consequence of potential radiological releases resulting from plausible accident scenarios constitute an important consideration in the U.S. Department of Energy (DOE) national program to manage the treatment, storage, and disposal of wastes. As part of this program, the Office of Environmental Management (EM) is currently preparing a Programmatic Environmental Impact Statement (PEIS) that evaluates the risks that could result from managing five different waste types. This paper 1) briefly reviews the overall approach used to assess process and facility accidents for the EM PEIS; 2) summarizes the key inventory, storage, and treatment characteristics of the various DOE waste types important to the selection of accidents; 3) discusses in detail the key assumptions in modeling risk-dominant accidents; and 4) relates comparative source term results and sensitivities.

OVERVIEW OF APPROACH

The objectives, scope, and various aspects of the approach to accident analysis in the U.S. Department of Energy (DOE) Environmental Management Programmatic Environmental Impact Statement (EM PEIS) have been reported earlier (1,2,3). As a consequence of evolving directions in the EM PEIS, these considerations have been somewhat refined and are briefly reviewed here. The EM PEIS currently calls for separate evaluations of the risks that could result from managing five different waste types: hazardous (HW), high-level (HLW), low-level mixed (LLMW), low-level (LLW), and transuranic (TRUW). The last four wastes present radiological risk and are addressed in the paper. Since the process details of final disposal alternatives are not being addressed in the EM PEIS, waste disposal or repository accidents are not addressed here.

The most recent guidance (4) from the Office of National Environmental Policy Act (NEPA) Oversight within DOE calls for consideration of the spectrum of accident scenarios that could occur in activities encompassed by the actions evaluated in the EM PEIS. This guidance also calls for a graded approach in which the risk-dominant scenarios are emphasized. Determination of risk dominance requires assessment of both the likelihood and the severity of plausible accident scenarios that could present a significant health hazard to either the workforce or the public. The spectrum of accident scenarios includes all accidents important to risk, from low frequency events with potentially high consequences (as typified by accident sequences associated with natural phenomena such as earthquakes) to relatively high-frequency events with very low consequences (as typified by routine industrial accidents).

To address the broad scope of the EM PEIS and to comply with the recent NEPA guidance, a phased approach was developed that includes the following interrelated elements: 1) selection of potentially risk-dominant storage and treatment operations and related facility configurations across the DOE complex; 2) selection, development, and probabilistic evaluation of a uniform set of risk-dominant sequences of accidents; and 3) determination of the evolution and final compositions of source terms predicted to be released from these sequences. This paper focuses on accidents important to risk, as determined by elements 1 and 2 above, and discusses the major source term modeling assumptions used in element 3. A personal-computer-based computational framework and database (5) have been developed to automate these elements and provide source terms. The source terms were subsequently used by Oak Ridge National Laboratory in the EM PEIS to assess the radiological health affects and risks to the general public and to the workforces.

RADIOACTIVE WASTE TYPES

The inventories and salient storage and treatment characteristics of the radioactive waste types considered in the EM PEIS are summarized below. The waste management alternatives discussed in the EM PEIS include the identification of siting options for storing and treating each waste type before disposal. Storage inventories and treatment throughout for each site affected by a given alternative are then defined by the current inventories, existing and projected waste generation rates, and then disposition of the waste. The volume and radio nuclide composition of each waste are tracked in a relational database (6) as the waste is processed to final disposal.

Low-level Waste

Several million cubic meters of LLW currently exist in the DOE complex. This waste ranges from low-activity waste that can be disposed of without treatment by engineered, shallow land disposal techniques to higher activity waste requiring the use of treatment and disposal techniques that provide greater confinement. LLW includes contaminated equipment and maintenance waste from operations; dry solids and solidified sludges from processing; and miscellaneous wastes, including neutron-activated reactor vessels and surface-contaminated concrete walls from decommissioning and decontamination (D&D). The waste is generally packaged in drums or containers and stored on outdoor concrete pads or in weather-protective sheds awaiting shallow land disposal or treatment. LLW is generated at more than 30 sites; 5 sites generate more than 80%: Hanford, Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL), and the Savannah River Site (SRS).

Two representative treatment philosophies are assumed in the EM PEIS: minimum treatment (stabilization of liquids and fines) and maximum volume reduction. The treatment technologies the depend on the physical characteristics of the waste and the final waste form as defined by the site-specific waste acceptance criteria (WAC) are 1) incineration, 2) solidification, 3) vitrification, 4) compaction and supercompaction, 5) size reduction (e.g., shredding, metal cutting, and shearing), 6) evaporation, 7) general aqueous treatment, and 8) various waste packaging techniques.

Low-level Mixed Waste

About 180,000 m3 of LLMW is currently stored at approximately 50 DOE sites. Generally, LLMW results from the same processes that generate LLW and includes aqueous liquids, organic liquids, sludge and particulates, soils, debris, special wastes, and inherently hazardous materials. LLMW is generally packaged in drums or containers and stored in Resource Conservation and Recovery Act (RCRA)-compliant weather-protective sheds before treatment. Another 250,000 m3 of LLMW is expected to be generated over the next 20 years, excluding that derived from environmental restoration activities. More than 99% of this waste has been or will be generated at 11 sites; sites generating the largest amounts are Hanford, ORNL, Rocky Flats Environmental Technology Site (RFETS), and SRS. The EM PEIS treatment technologies are 1) organic destruction, 2) aqueous liquids (wastewater) treatment, 3) metal (wastewater metal) removal, 4) neutralization (acid or base additions to neutralize waste streams), 5) stabilization, 6) metal recovery, 7) mercury recovery, 8) decontamination, and 9) deactivation of reactives.

Transuranic Waste

At the end of 1991, there was approximately 70,000 m3 of retrievably stored TRUW; about 96% of this waste is stored at Hanford, INEL, LANL, ORNL, and SRS. The TRUW

includes solid materials, such as contaminated clothing and equipment, and liquids and sludges resulting from chemical processing. A significant fraction contains hazardous components. TRUW is generally packaged in drums or containers and stored in concrete structures, in weather protective sheds, in earthen berms, or, in the case of remote-handled TRUW, in below-grade caissons. Most contact-handled TRUW, which dominates the total TRUW inventory, is stored in facilities with minimal containment, although DOE sites are increasingly moving toward qualified TRUW storage.

EM PEIS alternatives consider 1) minimal treatment (liquid absorption, compaction, immobilization, and repackaging) to meet the current Waste Isolation Pilot Plant (WIPP) WAC; 2) intermediate treatment (shredding, grouting, and changing containers) to reduce gas generation by waste in the repository; and 3) treatment to meet RCRA land disposal restrictions, which involves thermal destruction technologies similar to those for LLMW. For all alternatives, aqueous liquid is treated on-site before shipping. The treatment technologies are the same as those identified for LLMW.

High-Level Waste

HLW includes 1) liquid waste produced directly in reprocessing spent nuclear fuel and weapons production targets and any solid material derived from this waste that contains fission products in sufficient concentrations and 2) other material from the power reactor fuel cycle as determined by the U.S. Nuclear Regulatory Commission. HLW contains transuranic elements an fission products that are highly radioactive, heat generating, and long-lived. DOE HLW management follows six implementation phases: current storage in underground tanks, retrieval, pretreatment, treatment (generally high-temperature vitrification to produce glass logs to be sealed canisters), interim canister storage, and geologic repository disposal. However, the EM PEIS alternatives only address expanded interim canister storage at Hanford, SRS, and West Valley where the facilities will be sized to accommodate a production rate on the order of several hundred canisters per year.

SELECTION AND MODELING OF RISK-DOMINANT ACCIDENTS

Review of the operations and facilities discussed in the EM PEIS for the various waste types led to the establishment of three broad classes of accidents according to release characteristics and the facilities and populations affected. These classes include 1) general handling accidents involving a breach of waste packaging, 2) severe accidents at storage facilities, and 3) severe accidents involving treatment (or pretreatment) processes and facilities. Risk dominant operations, facility configurations, dominant accident sequences, and associated frequencies within each broad class of accidents are described further below.

Radiological source term = MAR x DF x RARF x LPF, Eq. (1)

where

MAR = the quantity of material at risk,

DF = The damage fraction or fraction of MAR exposed to accident stresses capable of rendering the MAR airborne,

RARF = the respirable airborne release fraction or fraction of material subjected to accident stresses actually rendered airborne and respirable, and

LPF = The leak path factor or fraction of the respirable airborne inventory that escapes any containment or confinement barriers to reach the ambient atmosphere.

The development of the MAR and DF parameters for the selected accidents is described below. The RARF is a function of the various accident stresses (pressurized release, fire, explosion, etc.) That affect the waste and the physical form of the MAR (liquid, sludge, solid, ash by-product, etc.), which varies by waste type, storage site, stage of treatment, and type of treatment technology. Values for the RARF were adapted from Mishima (7) to account for this functionality in the identified sequences. The LPF is a function of the response of the confinement was assumed to be breached and an LPF of 1.0 was assumed.

General Handling Accidents: LLW, LLMW, and TRUW

For EM PEIS alternatives addressing these waste types, general handling accidents involving waste package breach are expected to dominate the radiological risks to workers because of the relatively high frequency of such accidents and the proximity

of the workers to any release. Operations considered include handling in storage and staging areas, packaging and unpackaging, movement of waste within treatment facilities, and some treatment operations. Handling accidents include container breaches caused by package drops, by forklift or other vehicular impacts, by crane drops or crushing, and by overpressurization. The risk from exposure to radiation from operational incidents, such as puncture wounds during waste sorting, minor contamination from glove failures, and minor spreads of contamination from treatment equipment overpressurization, were judged to be enveloped by this class of accidents.

In the majority of handling accidents or hands-on pressure incidents, the MAR would be limited to a single package. Although more severe sequences involving an array of several containers are plausible, the releases of greatest overall risk to the workforce were judged to involve a worker in contact with or very near to a single breached package. Thus, a MAR of 55 gal (208 L) corresponding to the contents of a standard plastic-lined, carbon-steel drum was specified for all contact-handled waste types. The physical and chemical composition of the MAR was defined by weighting the relative treatability category inventories at each site. The DF would depend on the location of the breach, the physical form of the MAR, and the severity of the accident stress. Liquids and volatiles would be free to flow out of breached container, whereas most solid material would remain inside. Breached containers were assigned a DF of 0.25 for solid waste or a DF of 1.0 for liquid waste.

On the basis of a review of recent safety analysis reports, a probability of 1.0 x 10-4 per operation for package breaches, which is consistent with the aforementioned estimates of source term parameters, were assigned. Since most areas are simply staging areas for treatment or disposal operations, two handling operations were assumed, one for receiving and one for removal. Thus, the expected annual frequency (fmb) of a container breach for waste product x caused by a handling accident is as follows:

$$fmb = 0.0002 \times nx \,, \qquad \text{Eq. (2)}$$

where nx is the number of waste containers of waste product x received annually. To convert this value to a throughput number, it was conservatively assumed that the complete inventory turns over each year. Then the expected annual frequency of significant mechanical breaches would be given by the following equation:

$$fmb = 0.0002 \times N \,,$$

Eq. (3)

where N is the capacity of the facility in number of drums.

General Handling Accidents: HLW

Review of available safety documentation suggests that the risk-dominant accident during interim glass canister storage is the breaching of a canister during handling operations or transfer. It was conservatively assumed that the breach was so severe that the entire contents of a canister were dispersed into the surrounding vault (MAR = 1 canister, DF = 1.0). Partially degraded facility filtration was assumed (LPF = 0.001). The frequency for a canister drop with the above release was estimated to be 1.0 x 10-5/canister/year, with facility frequencies derived from the throughput.

Severe Storage Facility Accidents: LLW, LLMW, and TRUW

Radiation releases from severe accidents in a storage area are expected to dominate the risk of releases to on-site personnel and the general public for many DOE sites. Numerous storage facilities with large quantities of waste provide little or no formal containment or provide containment that would likely be breached in the event of severe thermal or structural challenges. This analysis focused on releases from severe operational accidents and external events involving fires in centralized storage pads and facilities judged vulnerable to large-scale releases. Other accidents were also considered but generally dismissed as being clearly enveloped in importance to risk by the fire sequences.

Accidents for storage facilities having solidified, vitrified, or otherwise highly stable wastes awaiting disposal were generally not analyzed. Landfills or other underground burial areas were also excluded from analysis. Finally, the volume of

solid wastes and the number of handling operations associated with drum, box, or crate storage exceed those associated with liquids stored in tanks for all waste types except HLW. Consequently, tank storage of liquids was judged to have a low relative risk and no related source term analyses were performed.

Fires can be categorized as either being local and involving limited inventories of wastes or, at the other end of the spectrum, as major facility fores induced by events that provide a source of fuel (such as heating gas, vehicle gasoline, or aviation fuel) and that also disable or overpower any available safeguards. Design and operational safeguards are in place to prevent fire propagation from a localized source, such as a single package or drum or a rubbish pile, to a much larger inventory. Packages for combustible materials include steel drums, fire-resistant boxes, and fire-protected shipping containers and are generally required by RCRA to be segregated. Finally, most centralized facilities have fire detection and suppression capabilities, including normal operator surveillance, automatic sprinkler systems, fire barriers, and on-site fire department response (or some combination of these types of protection).

The MAR in all storage fire scenarios depends on the storage configuration. The DF is a strong function of the packaging, the physical form (and combustibility) of the MAR, and the detection of and response to the fires. Two categories of fires were considered: waste-container fores and facility fires. Because of the relative infrequency of a single-container fire and the much greater consequences of fully developed facility fires, only the latter were analyzed. The representative fire was assumed to encompass the spectrum of undetected or unsuppressed fires, and the entire facility inventory of combustible waste was assumed to constitute the MAR. A DF of 0.1 was assumed as a generic value to account for the segregation and separation of waste packages in the facility and for the nature of the waste packaging as described previously. The estimated annual frequency for a fully developed facility fire with these source term parameters is 1.0 x 10-4/yr, the product of a generic facility fire frequency of 1.0 x 10-2/yr and a conditional fire suppression system failure probability of 1.0 x 10-2 . This value is consistent with existing documentation and is judged to be reasonable in light of the existing preventative and mitigative safeguards discussed previously. External event sequence frequencies depend on the location of the sites.

Although the inventories, physical forms, and radiological compositions of waste stored at each site are characterized in the EM PEIS and stored in the waste management database (6), compilation of analogous information for individual facilities on each site is beyond the scope of the EM PEIS. Accordingly, a unit inventory approach was used to develop radiological source terms with radiological and physical compositions derived by volume-weighing the inventories of the treatability categories within each waste type at each site. Because of the minimal containment properties of most facilities storing packaged LLW, LLMW, and contact-handled TRUW, a generic confinement configuration was assumed that did not consider containment or filtration (LPF = 1.0).

Severe Storage Facility Accidents: HLW

Because of the scope of the EM PEIS, ruptures, fires, and explosions of current storage tanks, which probably dominate the health risk of overall HLW management, were excluded. Radiological releases from severe fires, explosions, or natural phenomena-induced events were considered for interim storage facilities. However, the relevant safety reports for the various HLW interim storage facilities do not elevate the risk of fire, in part, because there is no significant accumulation of combustibles to support fire propagation. Given this, the low frequency of severe external events, and the lack of plausible airborne release mechanisms in light of the high integrity of the HLW canisters, severe interim HLW storage facility releases were judged unimportant to risk and not analyzed any further.

Severe Accidents Involving Treatment Processes and Facilities

The focus here is on possible fires or process explosions from operational or external causes. Many treatment operations were excluded from detailed investigation of large-scale releases on the basis of either the lack of a sufficiently radiologically concentrated MAR or the lack of an energy source capable of creating a risk-dominant airborne release. These operations include evaporative processes and solidification operations, such as grouting and cementation. In general, benign operations, such as packaging and nonthermal size-reduction activities, were excluded from consideration for large-scale accidents.

Technologies for mercury separation were excluded because of their relatively low-energy operating characteristics. Thermal desorption of residues, sludges, and resins, or of debris wastes, involves combustible material. This process was excluded, however, because (1) it operates at lower temperatures and pressures then incineration, a competing technology, and (2) the output product is much less dispersible then the ash from incineration.

Other processes involving high temperatures or pressures were more closely reviewed in light of the potential energy source for dispersing airborne radioactive or toxic material, as well as for challenging the facility's integrity and capability for filtration. Similarly, operations involving or being performed in the presence of combustible materials or involving feed lines of natural gas or fuel were reviewed in light of the potential for ignition and subsequent fire or explosions. On the basis of these considerations, as well as consideration of the alternative-dependent volume of waste to be treated by each process, incineration, wet-air oxidation, and vitrification were identified for their potential for major airborne release. A final comparative review of the characteristics of the identified treatment processes, augmented by scoping source term calculations, led to the selection of incineration as the technology most likely to dominate risk to the staff of the facility and the site, as well as to the surrounding general populations, for LLW, LLMW, and TRUW. (Risks of HLW treatment technologies are not considered in the EM PEIS).

Detailed modeling of facilities is beyond the scope of the EM PEIS. Accordingly, a treatment facility with generic confinement characteristics was used to assess accidents to envelop the releases from accidents in the treatment process. A DOE Hazard Category of 2 and the associated performance requirements for its systems were assumed. Double high-efficiency particulate air filtration was assumed. The MAR at the time of the accident was based on the facility throughput at each site, which varies by alternative. The DF was based on the location of the Mar, which varied by accident sequence. The dominant sequences were generally unconfined operational facility fires that were assigned an annual frequency of 1.0 x 10-3 and external event driven fires, the frequencies of which are site-dependent.

COMPARATIVE RESULTS AND SENSITIVITIES

Cross comparisons of airborne source term releases from similar postulated accidents for comparably sized inventories of LLW, LLMW, and TRUW at risk at various DOE sites were made. LLW accidents would generally result in the smallest releases of radioactivity (as measured in curies) with analogous LLMW accidents producing somewhat greater releases of radioactivity, and, because of the higher concentration of heavy metals, would be expected to result in considerably more significant exposures than LLW or LLMW.

Primarily because of their high atmospheric dispersion potential, fire-related accident sequences initiated by either internal or external events produced the most significant radiological source terms for both storage and treatment facilities. Comparisons of internally and externally driven fires reflected tradeoffs between the lower expected frequencies of external event sequences and their higher expected Dfs of material at risk. The relative importance of fire (as well as other types of source term dispersion mechanisms in accident sequences) is, of course, sensitive to the process, facility, and site characteristics identified above and to the affected waste type combustibility characteristics and initiating accident.

Because of the broad scope of the EM PEIS and the spectrum sites, waste types, and facilities considered, generic accidents with representative source term parameters were used for analysis. The results were sensitive to two categories of uncertainty: (1) that occurring from the application of generic models to a range of real facilities, waste materials, and processes with somewhat different characteristics and (2) that from the modeling per se. One of the biggest sources of uncertainty in the results follows from the uncertainties in the physical and chemical compositions of the inventories. The uncertainties not only affect the MAR factor in the source term equation, but propagate through all the source term factors. Resulting differences in source term evolution could be reflected in threshold failure effects on accident mitigation or containment systems. Consequently, significantly different atmospheric releases would be expected.

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DOE'S EVALUATION OF CANDIDATE U.S. PORTS FOR RECEIPT OF FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL

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ABSTRACT

The Department of Energy (DOE) is preparing a Draft Environmental Impact Statement (DEIS) to evaluate the impacts of a proposed policy for the management of foreign research reactor (FRR) spent nuclear fuel (SNF). Among the policy alternatives being evaluated is United States acceptance of FRR SNF for storage and/or processing. The DEIS will address only FRR SNF that contains uranium enriched in the United States This paper discusses the DOE methodology being developed to identify potential ports of entry in the United States that would satisfy National Environmental Policy Act (NEPA) requirements and other requirements resulting from District Court decisions, Congressional guidance, recommendations of a U.S. Merchant Marine Academy (USMMA) Workshop, and suggestions from public scoping meetings. From these various sources, DOE was able to develop a list of basic screening criteria which all acceptable ports must satisfy. These included: 1) appropriate experience handling containerized cargo, 2) favorable transits from the open ocean to the port facility, 3) appropriate facilities for receipt, handling and transhipment of FRR SNF, 4) ready access to truck and rail transport, and 5) lowest human population around the port facility. These criteria are being used in the DEIS to screen a large number of possible candidate ports to narrow the search to a reasonable range of potential ports on the Atlantic, Gulf, and Pacific Coasts. In addition to commercial ports,

several military ports are being considered from which a few will be identified in the DEIS for transporting FRR SNF to the United States using chartered vessels. After the screening criteria applied to each of port beginning with those initially selected for screening, DOE will consider the potential usefulness of each remaining port relative to other ports for receipt and shipment of the FRR SNF to any of the five potential DOE storage sites [i.e., the ease of shipping FRR SNF to each port from numerous foreign nations, and whether or not the low population criterion will permit shipment to storage facilities at the Savannah River Site (SRS), the Idaho National Engineering Laboratory, the Oak Ridge Reservation, the Hanford Site, and the Nevada Test Site]. Next, DOE will use a series of desirable attributes that individually are not adequate to reject an otherwise acceptable port, but collectively will permit DOE to select the most desirable group of low population ports for the proposed action. These port attributes include emergency response planning, secure short-term storage, good separation from high populations, berthing options that permit avoidance of conflicting activities, and ports that do not present severe environmental concerns (including the likelihood of severe natural phenomena). Use of these attributes will enable DOE to select the most acceptable commercial and military ports for acceptance of FRR SNF shipments to the United States if such a policy alternative is adopted.

INTRODUCTION

The United States Department of Energy (DOE) and Department of State are jointly proposing an action to adopt a new policy for management of FRR SNF. Only SNF containing uranium enriched in the United States would be covered by the proposed alternative. The purpose of the policy is to support United States nuclear weapons nonproliferation policy objectives, specifically by reducing the amount of highly-enriched uranium used in international commerce. Among the alternatives being analyzed are: 1) acceptance of the FRR SNF by DOE for storage and eventual disposition in the United States, 2) chemical separation of a portion of the SNF in existing DOE facilities in the United States, or through development of a new technology followed by storage and disposition of the resulting products and waste in the United States, and 3) reprocessing of the FRR SNF at overseas facilities. A no action alternative is also being analyzed.

DOE is preparing a DEIS to evaluate the impacts of the proposed policy and alternatives. As part of the implementation of one proposed policy alternative, DOE is developing criteria for evaluation of candidate United States ports against certain requirements. This paper discusses the methodology being developed for identification of potential United States ports that are well-qualified for receipt, handling and transhipment of FRR SNF to DOE storage facilities.

BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) has the authority under the Atomic Energy Act of 1954, as amended, to regulate certain aspects of SNF transportation within the United States Of the thousands of shipments completed over the last 30 years in the United States and abroad, none has resulted in an injury due to the radioactive nature of the cargo (1). For the same period, about 1,200 overland SNF shipments (924 domestic and 293 foreign) took place without any injury attributable to accidents or incident-free transport.

Since 1979, when the NRC first began approving shipments of SNF in the United States, there have been 317 SNF shipments in robust "Type B" shipping casks transported safely into and out of several United States ports of entry (under a previous policy). These ports included Newport News, Norfolk and Portsmouth, Virginia (Port of Hampton Roads); Savannah, Georgia; Portland, Oregon; and Oakland, California (1). However, prior to the fall of 1993, DOE did not have any generally applicable criteria for identifying desirable ports of entry for receipt of FRR SNF. As a result, DOE is developing appropriate criteria for identification of qualified ports of entry for the potential receipt of FRR SNF. The port selection process for the Draft EIS will be based on a set of criteria being developed by DOE to identify those ports which are most capable of providing for the safe receipt, handling and transhipment of FRR SNF during the period of the proposed action. The final methodology to be used in identifying capable ports will be based on criteria that has evolved from several events:

1. A 1991 Federal Circuit Court ruling on a draft environmental assessment which requires DOE to consider a reasonable range of alternative ports, including low population ports that are near SRS (2).

2. The 1993 Notice of Intent for the proposed action listed a series of preliminary criteria which might be applied to identify capable ports, including: a) "adequacy of harbor and dock characteristics to satisfy the cask carrying ship requirements; b) availability of safe and secure short-term storage; c) adequacy of overland transportation systems from the ports to the storage site(s); d) experience in safe and secure handling of hazardous cargo; e) emergency preparedness status at the port and nearby communities; and f) proximity to the proposed storage sites" (3).
3. A DOE-sponsored workshop on port selection criteria for SNF was held at the USMMA at Kings Point, New York, on November 15-16, 1993. Participants included independent experts from the maritime industry in the areas of marine transportation, intermodal systems, marine insurance, admiralty las, U.S. Coast Guard Operations, U.S. Navy Operations, Military Sealift Command Operations, and national cargo, pilotage, and ships operations (4). The workshop resulted in the development of specific recommended criteria.
4. On November 30, 1993, the National Defense Authorization Act (NDAA) of 1994 was signed into law, requiring the use of specific criteria to be used if "economically feasible," and "to the maximum extent practicable" in selecting ports for both emergency and non-emergency receipt of FRR SNF at SRS (5).
5. Nine public scoping meetings were held in November and December, 1993 to elicit public comments related to the proposed scope of the EIS. As a result, DOE received numerous comments related to the port selection process, including recommendations for avoiding use of high population ports and using low population ports, including military ports (6). Other comments focused on use of the safest and most experienced ports that are closest to storage sites.

PORT IDENTIFICATION PROCESS

It was the opinion of independent maritime experts that any modern port could safely receive, handle, and tranship FRR SNF to any of the potential storage sites (4). Nevertheless, DOE concluded that by careful selection, superior ports for FRR SNF receipt could be identified. The proposed candidate port identification process involves several steps that are related to the recommendations, requirements, and events discussed above. First, DOE has identified several required (i.e., mandatory) criteria which are being used to screen all potential ports in the contiguous states. In general, failure to satisfy any one of these criteria would result in disqualification. The proposed criteria are:

The ports must have appropriate (routine) experience handling containerized cargo, since all FRR SNF is expected to be shipped in this configuration

The ports must offer favorable transits from the open ocean to the selected terminal(s)

The ports must have appropriate facilities for safe receipt, handling and transhipment of FRR SNF

The ports must have ready access to intermodal transport (i.e., truck or rail facilities at or near the selected terminal)

The combined human population surrounding each port and the transportation routes to each potential storage site must be low (compared to other ports) to the extent economically feasible and the maximum extent practicable

These criteria are being applied to an initial list of a large number of potential commercial and military candidate ports in the contiguous United States which were identified from several commercial publications (7-12), and military references (13-16). Through the initial screening process, the number of capable ports can be narrowed down to a reasonable range of potential ports of entry. A more detailed discussion of the screening criteria is provided below.

Criterion 1: Appropriate Experience

Appropriate experience, a requirement of the NDAA, was defined as that which results from routine handling of containerized cargo equivalent to unloading or loading a small container vessel at every week or two [on the order of 20,000 twenty-foot-equivalent units (TEUs) per year; a TEU is an International Standards Organization container size of about 2.4 m x 2.4 m x 6.1 m (8 ft x 8 ft x 20 ft]. This experience could derive from any mix of breakbulk/container vessels or self-contained ships, and assure a core of experienced, trained stevedores who are readily available (i.e., avoids use of less experienced, part-time workers). It effectively discriminates against ports which are used for handling bulk cargoes (oil, grain, coal, etc) or cruise ships, since these operations do not involve routine handling of containerized cargoes (e.g., everything from TV sets and machine

parts to toxic materials, flammable, or explosive cargos). As noted earlier, containerized cargo experience is considered a major factor in experience because most or all FRR SNF is expected to be shipped in containers [specially designed to safely hold FRR SNF casks in the range of about 10 to 25 mt (17)], and container handling is the same regardless of the cargo inside the container.

Screening of commercial ports under this criterion will permit DOE to identify ports that are considered to be reasonably representative of United States ports which might be used for FRR SNF shipments. Many of these ports will not meet this criterion because they are associated with bulk cargo shipping (oil, chemicals, grain, ore, etc) that uses specialized cargo handling equipment (e.g., conveyer or pumping systems) totally unrelated to container handling equipment.

This criterion will also permit rejection of all naval bases and shipyards in the contiguous United States that do not regularly handle containerized cargoes (almost all breakbulk cargoes). In general, most military ports experience great variations in annual cargo throughput that reflect periodic changes in military missions. As long as military ports have periodic annual experience comparable to qualified commercial ports, they will be accepted under this criterion.

Criterion 2: Favorable Transit from Open Sea

This criterion is based on a recommendation of the USMMA workshop participants who found that a short transit from the open ocean to port was necessary to maximize the safety of SNF shipments. However, it was clear that since this criterion focuses on ship safety, it is essentially synonymous with the requirement for a favorable ship transit. Thus, while a port might be within a few miles of the open sea, if the channel exposed ships to numerous hard shoals, ship wrecks, or reefs, transits might be more risky than longer but less difficult transits. Thus, DOE determined that ports meeting the intent of the Workshop recommendation would have relatively short trips from large, deep bodies of water that were oceans, seas, or extensions thereof, such as bays or sounds (e.g., the Chesapeake Bay, San Francisco Bay, or the Puget Sound), and present no special navigational hazards to ships. Long transits along narrow, winding, or crowded ship channels (e.g, the St. Lawrence seaway to a Great Lakes port, or far up the Mississippi River) were not considered to be consistent with the USMMA recommendation. In addition, it was concluded that a minimum channel depth of 7.6 m (25 ft) would be required for use by at least small to intermediate size vessels (7-12). Information on channel depth and navigational problems were identified in the U.S. Coast Pilot series (18-23).

Criterion 3: Appropriate Port Facilities

The NDAA of 1994 requires the use of ports with "appropriate port facilities" that allow safe handling of FRR SNF (5). The USMMA Workshop recommended as "necessary for safe shipment" that an acceptable port have "adequate port cargo facilities," including berthing options (i.e., to avoid conflicting activities at an adjacent berth or onshore if necessary) and onsite cranes with trained operators (4). To meet these requirements, DOE has determined that the ports should have the following minimum physical attributes: 1) adequate water depth alongside pier (at least 7.6 m [25 ft] for small to intermediate size vessels (7-12); 2) adequate piers with berthing options (4); and 3) at least one adequate crane for offloading containerized casks of FRR SNF on to ground transport (at least 30 mt (33.1 ton) capacity for the largest FRR SNF casks which weigh on the order of 10 to 25 mt). While the USMMA Workshop preferred purpose-built container cranes for handling containerized FRR SNF in ports, it did not conclude that such cranes were necessary to satisfy the criterion. Thus, acceptable alternatives include use of other types of shore cranes with adequate lifting capacity, or the use of shipboard cranes.

The DOE also examined the possibility that ports currently found unqualified might meet all requirements during the period of the proposed action. In general, many important port improvements were identified during the next five or ten years. However, in most cases, the improvements will occur in ports already found acceptable, or in ports which were rejected for other reasons. Therefore, future port improvements are not expected to significantly change the list of ports which currently are found acceptable under all the DOE port identification criteria.

Criterion 4: Ready Access to Intermodal Transportation

The USMMA Workshop found that it was necessary for SNF safety that a port have ready access to intermodal transportation (i.e., easy access to truck or rail transport). It is common practice for some ports to move containers from shipside to a nearby marshalling yard for loading onto trucks or railcars. Such moves are increasingly

done quickly using purpose-built container handling equipment which minimizes congestion in the offloading area and avoids additional occupational exposures (since the special systems in use do not require additional handling by other personnel). Therefore, DOE has concluded that lack of an intermodal rail facility immediately at the port is not sufficient basis for rejecting an otherwise acceptable port (all ports have truck access) as long as the rail facility is within a few miles of the port. Such additional transport risk is small compared to that which would result from normal overland transport of FRR SNF for hundreds or thousands of miles to a storage facility. No additional ports were screened out from application of this criterion.

Criterion 5: Low Human Populations

This criterion reflects a number of factors which were identified by DOE for consideration. First, a 1991 District Court decision on return of Taiwanese SNF indicates that the courts consider port populations to be an important part of the NEPA process for assessing the range of reasonable port alternatives (2). Second, the NDAA of 1994 requires that, if economically feasible and to the maximum extent practicable, ports of entry for FRR SNF bound for SRS have the lowest human populations in the area surrounding the port (5). While the NDAA of 1994 was written specifically for FRR SNF shipments to SRS, DOE elected to apply this criterion to identifying ports of entry for all five potential DOE storage sites, to the maximum extent practicable.

However, DOE found in its analyses of total impacts associated with receipt, handling, and transhipment of FRR SNF that public risk is driven not only by port populations, but by the populations within the immediate proximity of overland truck and rail shipments from the port of entry to each of the five potential storage sites (24). Since DOE considers the affected populations along the overland transportation route as important for protection of the public health and safety as those in the immediate vicinity of each port, the total unique population associated with the receipt of FRR SNF at a specific port and overland transport to a specific storage site is being treated as the affected population for each port. Since the NDAA did not define what was meant by populations surrounding ports, DOE examined three sets of populations circumscribed by the 1.6 km (1 mi) radius, the 8 km (5 mi) radius, and the 16 km (10 mi) radius. The 1.6 km radius would include members of the public in the immediate vicinity of the selected port terminal, including those members of the public outside the port boundary who would be expected to be most impacted by most accidents and incident-free transport from the port to the nearest interstate highway. However, for the most severe accidents (involving engulfment of SNF casks in fires beyond the design basis for Type B casks), plume rise would loft released activity high into the air, and peak ground concentrations would not occur until well outside the 1.6 km radius. In addition, the census figures do not include workers within the port boundaries whose numbers may be larger than the 1.6 km population outside the boundary for remotely located terminals. Therefore, the 1.6 km population was not considered adequate to reflect the intent of the criterion. The 8 km radius population was also found not to reflect the majority of the potential public impacts from severe accidents, and was determined to be inadequate. The potential impacts of severe accidents on the 16 km radius population, however, was found to include most of the total radiological impacts on the public for typical (mean) meteorology, and typical dry deposition and fallout patterns.

In addition, analyses of the potential impacts of severe accidents on a range of port populations indicated that the average per capita dose within 16 km, while small, was higher than any of the averages for more distant populations. As a result, DOE selected the 16 km radius population to represent the port populations most likely to be impacted by the entire range of potential port accidents (including severe accidents) and incident-free transport from the port to major interstates or rail systems.

The distributions for the combined populations of candidate ports and populations within 0.8 km of the transportation routes for each of the five potential storage sites for both truck and rail overland transport are somewhat skewed due to a few very high population ports. However, given the large uncertainty and variances for the small sample sizes for port/site combined populations, DOE assumes that the populations are normally distributed.

Very early in the EIS project, DOE realized that any proposed methodology imposed in the NEPA process would be suspect unless the methodology could be shown to be fair

and impartial. Therefore, DOE elected to apply a standard statistical model to these populations to assist in the selection of the lowest population ports practicable, so as to preclude any way to manipulate the data to reach a predetermined outcome. For normally distributed populations half of the combined port/site populations will be above the average, and half will be below. For purposes of selecting ports whose combined port/site populations (truck and rail) are as low as practicable, DOE will select all those populations that are below the average of the combined populations for further analysis.

USE OF DESIRABLE PORT ATTRIBUTES IN FINAL PORT SELECTIONS

For the ports remaining after the above screening criteria were applied, a number of desirable attributes were included in the port identification process which were not important enough individually to reject an otherwise acceptable port from further consideration. However, collectively, these attributes were useful in identifying a final group of ports which will be included in the DEIS. These desirable attributes included the following:

Relative usefulness by each storage site based on using low port/site populations,
Relative usefulness by FRR SNF owners based on available shipping lines serving each United States port,
Lack of significant conflicting uses at port terminal(s),
Adequate emergency preparedness capability,
Secure short-term storage,
Well-separated from high density populations,
No environmentally sensitive areas in ports,
FRR SNF handling experience, and
No severe natural phenomena of concern.

Another desirable port selection factor recommended by the USMMA Workshop (4) is the absence of local regulatory restrictions on receipt and handling of SNF. DOE elected not to use this factor for port selection, but will provide that information in the EIS for consideration by decision-makers. There were two reasons for doing so. First, it is well established that local restrictions on international or interstate commerce are void under the U.S. Constitution, and challenges have been rejected by the Federal courts. Second, if DOE were to avoid selection of ports with local restrictions, every port wishing to close its doors to receipt of SNF (or any other type of cargo) would simply promulgate an ordinance, defeating the port selection process.

"Usefulness," as used here, is a term in which the relative numbers of scheduled shipping lines and types of vessels that use a port, and the countries or regions that are served by those lines are compared for two or more otherwise acceptable ports for purposes of selecting the best of that group of ports. Also, using only those combined port/site populations which meet the low population criterion means that some ports will not be able to receive and transhipSNF either by truck or rail to one or more of the sites. This will also limit the usefulness of some ports.

The status of the low population ports for the remaining desirable attributes cannot be addressed at this time. However, some general observations are possible. Some conflicting uses at an otherwise acceptable port, such as loading or unloading explosive cargoes, can be avoided by judicious planning so that they do not occur when FRR SNF would be present. Other conflicting activities can also be avoided by judicious selection of berths which are well-separated from port areas where potentially dangerous and conflicting activities are underway (e.g., liquified natural gas or petroleum transfers, and loading or unloading of explosives or military munitions). Some severe natural phenomena, such as hurricanes, can also be avoided by careful planning. The Draft EIS will provide a final accounting of these attributes and, along with potential usefulness, will provide the basis for the selection of a final group of ports.

CONCLUSION

DOE has developed an impartial methodology for objectively screening and evaluating potential ports of entry for potential shipments of FRR SNF into the contiguous United States These criteria will be included in a DEIS to evaluate a policy for managing FRR SNF. Application of these criteria will assure that well-qualified ports of entry will be used if a policy is adopted for the United States to accept FRR SNF.

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60-7
ENVIRONMENTAL ASSESSMENT OF THE SINKING OF THE NUCLEAR-POWERED SUBMARINES THRESHER AND SCORPION
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ABSTRACT
Comprehensive deep ocean radiological environmental monitoring operations were conducted in 1983 and 1986 at the site of the sunken nuclear-powered submarine THRESHER and in 1986 at the site of the sunken nuclear-powered submarine SCORPION. Numerous sediment, water, and marine life samples were obtained from the sites. In addition, many in situ gamma radiation measurements were collected using sensitive gamma spectrometry techniques. None of the samples collected from the sites showed any evidence of release of radioactivity from the reactor fuel elements. Very low concentrations of radioactivity in the form of activated corrosion products (cobalt-60 and nickel 63) from the primary piping systems were detected in the sediments at both sites. Fission products were not detected above concentrations of world-wide fallout levels in sediment, water, or marine life samples. No cobalt-60 or nickel-63 was detected in the large number of fish and other marine life specimens or in undisturbed water samples at the sites. Samples of sediment, water, and marine life from the SCORPION site were analyzed for plutonium isotopes using very sensitive mass spectrometry techniques. Total plutonium radioactivity concentrations and the relative concentration of plutonium isotopes were typical of background concentrations due to fallout from nuclear weapons testing. Thus there is no evidence of leakage of plutonium from nuclear weapons that were on the SCORPION when it sank. The 1983 and 1986 survey results confirm that the THRESHER and the SCORPION have not had a significant effect on the radioactivity in the environment.
INTRODUCTION
Comprehensive deep ocean radiological environmental monitoring operations were conducted in 1983 and 1986 at the site where the U. S. Navy nuclear-powered submarine THRESHER sank in 1963 and in 1986 at the site where the U. S. Navy nuclear-powered submarine SCORPION sank in 1968. Both surface-ship-deployed and submersible-operated monitoring equipment were utilized during the course of these expeditions to obtain measurements and samples from the vicinity of the THRESHER and SCORPION sites. The objectives of these operations were:

To provide valuable follow-up radiological environmental monitoring data to that obtained from previous missions to the THRESHER and SCORPION sites.

To test and evaluate improved technology for conducting oceanographic radiological monitoring in the deep ocean.

The results obtained from the 1977 expedition to the THRESHER site and the 1979 expedition to the SCORPION site both utilizing the U. S. Navy's deep submersible TRIESTE, and the results obtained from the 1983 expedition to the THRESHER site utilizing DSV ALVIN, it's former support ship the R/V LULU, and the ORV CAPE FLORIDA, were useful in evaluating the potential sea disposal of obsolete submarines as described in the Final Environmental Impact Statement on the Disposal of Decommissioned, Defueled Naval Submarine Reactor Plants (U.S. Navy 1984). The 1986 expedition to both the THRESHER and SCORPION sites utilized DSV ALVIN with its support ship R/V ATLANTIS II. The enhanced capabilities of R/V ATLANTIS II compared with R/V LULU for conducting deep sea oceanographic operations precluded the need for an additional oceanographic research vessel during the 1986 expedition.

Background

The U. S. Navy nuclear-powered submarine THRESHER sank on April 10, 1963, 100 miles from land in water 8500 ft deep off the coast of Nova Scotia. Radiation measurements, water samples, bottom sediment samples, and debris collected from the area where THRESHER sank were analyzed for radioactivity with highly sensitive equipment shortly after the sinking and again in 1965 by various laboratories. None of these samples showed radioactivity above naturally occurring background levels or evidence of radioactivity released from the THRESHER. In 1977, follow-up samples of water, sediment, marine life, and debris were collected from the immediate THRESHER area by the submersible TRIESTE. None of these samples showed any evidence of release of radioactivity from the THRESHER'S reactor fuel elements. However, cobalt-60 released from the coolant system was detectable at low levels in sediment samples from localized areas that were not sampled during the original surveys. The cobalt-60 radioactivity in these sediment samples was small compared with naturally occurring radioactivity. Cobalt-60 was not detectable in the samples of water,

marine life, or debris. Nickel-63 was not detectable in any of the samples analyzed except by magnetically concentrating the radioactive corrosion products from a composite sample of sediments containing the highest concentrations of cobalt-60.
The U. S. Navy nuclear-powered submarine SCORPION sank on May 22, 1968 several hundred miles southwest of the Azores in more than 10,000 feet of water. The SCORPION carried two MARK 45 ASTOR torpedoes with nuclear warheads when it sank. The warheads were low-yield tactical nuclear weapons containing plutonium and highly-enriched uranium. Seawater and bottom sediment samples taken near SCORPION'S hull were analyzed for radioactivity with highly sensitive equipment shortly after the sinking. None of these samples showed radioactivity above naturally occurring background levels or evidence of radioactivity released from the SCORPION. In 1979, follow-up samples of water, sediment, marine life, and debris were collected from the immediate SCORPION area by the submersible TRIESTE. None of these samples showed any evidence of release of radioactivity from the SCORPION'S reactor fuel elements. However, cobalt-60 released from the coolant system was detectable at low levels in sediment samples from localized areas that were not sampled during the original surveys. The cobalt-60 radioactivity in these sediment samples was small compared with naturally occurring radioactivity. Cobalt-60 was not detectable in the samples of water, marine life, or debris. Nickel-63 was not detectable in any of the samples analyzed except by magnetically concentrating the radioactive corrosion products from samples of sediments containing the highest concentrations of cobalt-60.
It was the low concentrations of radioactivity present in the sediment near the THRESHER and SCORPION that provided a means to assist in the evaluation of improved deep ocean monitoring devices. The monitoring program conducted at both sites included the following elements:

In situ gamma spectrometry measurements of sediment and submarine debris.

Measurement of currents and temperature in the bottom water adjacent to the submarine debris.

Collection of representative bottom water and sediment samples for radionuclide analysis.

Collection of disturbed sediment for chemical and radionuclide analysis.

Collection of bottom-dwelling fish and marine life for radionuclide analysis.

Operation of an acoustic transponder navigation network to ensure proper placement of oceanographic sampling and monitoring devices.

MONITORING EQUIPMENT

A number of sampling devices were utilized to obtain samples of seawater, sediment, and marine life, and to make in situ gamma spectrometry measurements. This included instrumentation specifically developed for this purpose and modifications to existing oceanographic sampling equipment. A detailed description of this equipment is contained in Table I and is described in detail in Sheldon and Michne 1993a and b.
Equipment operated from DSV ALVIN included sediment core tubes and box corers used to obtain samples of sediments and interfacial seawater; Niskin bottles, pumping systems, and passive chemical monitors used to obtain large samples of seawater for subsequent analysis; and several types of in situ gamma spectrometers to make direct measurements of gamma-emitting radionuclides.
Surface-deployed equipment operated from R/V ATLANTIS II during the 1986 expeditions to both the SCORPION and THRESHER sites and from ORV CAPE FLORIDA during the 1983 expedition to the THRESHER site included various biological samplers including tetrahedral marine life traps and a two-chambered fish trap; water samplers including Niskin bottles, GO-FLO samplers, and a pumping system (Winget et. al. 1982); sediment and interfacial water samplers including box corers, a tripod sphincter corer (Burke 1968), and a quadropod sampler; two types of wire-deployed or free vehicle in situ gamma spectrometers; current meters; and acoustic transponder navigation equipment to precisely identify the locations of samples obtained from the sites.

METHODS

As indicated previously, the expedition to the THRESHER site in 1983 utilized the DSV ALVIN with it's support ship R/V LULU, both operated by the Woods Hole Oceanographic Institution, and the oceanographic research vessel ORV CAPE FLORIDA, operated at the time by the University of Miami. The 1986 expedition to both the THRESHER and SCORPION sites utilized the Research Vessel (R/V) ATLANTIS II, with ALVIN, both operated by Woods Hole Oceanographic Institution. ATLANTIS II is 210

feet long, displaces 2300 long tons, and is driven by two 1100 horsepower diesel engines with a twin screw configuration. She is outfitted with a stern mounted A frame utilized for the efficient launch and recovery of ALVIN. Scientific facilities aboard the ship included a deep submersible control and monitoring station equipped with a UQC underwater telephone and electronic equipment. This station includes computers to permit tracking of ALVIN and other equipment outfitted with acoustic transponders on the ocean floor.

The basic navigational network consisted of setting three transponders in a triangular configuration with the main target near the center of the triangle. The transponders were anchored approximately 100 meters above the ocean bottom. One acoustic signal was periodically transmitted from the ship to the three transponders, which replied at different frequencies. The time for this round trip signal was measured at the ship using an acoustic processor controlled by an IBM computer. After all transponder anchors had settled to the ocean bottom, a survey was conducted using Loran C navigation. The ship was driven around the network transmitting to and receiving from the transponders. These signals were then processed in the computer to give XYZ coordinates of each transponder and the exact distance between them. These numbers were then put into the computer's acoustic navigation program.

The navigation program was then used to control the ship's position relative to the transponder network in XY coordinates. The navigation program could locate other packages within the network besides the ship. This was the basic method used to deploy monitoring equipment on the bottom at predetermined targets from the surface ship. The ship was maneuvered as the monitoring equipment was tracked to hit the bottom at the specified target. The equipment was released at the desired location from a depth of approximately 50 to 100 meters off the bottom and the exact position of the instrument in relation to the target was determined. A description of this underwater acoustic navigation system is contained in Loud and Scheer 1984.

Six ALVIN dives were conducted at the THRESHER site in 1983, in which various monitoring devices were utilized. Thirteen deployments of various oceanographic monitoring devices were conducted from CAPE FLORIDA. These included the current meter, fish traps, box corer, sphincter corer, quadropod sampler, gamma spectrometer, and the water sampling pump described previously.

Two ALVIN dives were conducted at the THRESHER site in 1986 utilizing the submersible-operated and deployed instrumentation discussed previously. Monitoring efforts were concentrated near those areas of the THRESHER that were not heavily sampled in 1983. Two deployments of oceanographic monitoring devices were conducted from R/V ATLANTIS II, including the wire-deployed gamma spectrometer and the water sampling pump. Numerous core tubes and in situ gamma spectrometry measurements were collected by ALVIN.

Ten ALVIN dives were conducted at the SCORPION site in 1986, in which various monitoring devices were utilized. These devices included the gamma spectrometer, water pump sampler, sediment core tubes, and GO-FLO water samplers described previously. ALVIN operations were conducted during the daytime with precision navigation by the computer system aboard ATLANTIS II. Nighttime operations aboard ATLANTIS II included the surface-deployment of monitoring devices, the recharging of ALVIN's batteries, sample processing, and dive preparations for the next day. ALVIN's principle objectives were to survey and sample the SCORPION area, locate and provide coordinates for the target to be used by ATLANTIS II for nighttime deployment of monitoring devices, and to verify the positions of monitoring devices the day after they were deployed. Twenty seven deployments of various monitoring devices were conducted from the surface. These included the current meter, fish traps, box corer, gamma spectrometer, and the water sampling pump described previously. Numerous samples were collected and measurements obtained.

At each site, a current meter was deployed as a free vehicle during the first full day on station and was moored approximately 2 m off the bottom in the vicinity of the submarine. Current direction, velocity, and temperature were recorded continuously. The meter was successfully recovered after it was released from the bottom by an explosive type timer.

Shipboard Gamma Spectrometry

Selected samples of sediment, seawater and marine life were analyzed with shipboard intrinsic germanium detectors each coupled to a Canberra Series 10 multichannel pulse height analyzer. The detector systems were shielded inside lead forts to

reduce background radiation levels and to improve the sensitivity of detection. A sufficient supply of liquid nitrogen was stored aboard ship to keep the detectors operating around-the-clock during the course of the expedition. Results from these analyses were used to make preliminary assessments and to determine the locations for future samples. The same samples were later analyzed using more sophisticated laboratory analytical equipment and the results reported in Sheldon and Michne 1993a and b.

Radiological Precautions

Various radiological control procedures were implemented to minimize personnel radiation exposure and the potential for radioactive contamination in the event that excessive radiation or contamination was encountered. Previous radiological monitoring data were reviewed. Based on this review, the major longlived radionuclides of concern during submergence and surface recovery operations were identified as the activation product cobalt-60, the fission product cesium-137, uranium-235, and the transuranic plutonium-239. The detection of either cobalt-60 or cesium-137 would also imply the presence of other long-lived activation and fission products or core materials such as nickel-63, strontium-90, uranium-235 and plutonium-239.

Radiological control procedures were in compliance with the Woods Hole Oceanographic Institution (WHOI) Isotope User's Manual, which is based on Nuclear Regulatory Commission (NRC) regulations imposed by WHOI's NRC license. The control of personnel exposure during diving operations consisted of using personnel monitoring devices including thermoluminescent dosimeters (TLD's) and pocket dosimeters along with radiation survey instrumentation aboard ALVIN. Each dive crew carried two portable gamma radiation survey instruments for use in high and low radiation fields. On each dive ALVIN also carried an externally mounted radiation detector connected to a multichannel analyzer inside ALVIN. At least one person on each dive was instructed in the use of the radiation survey meters. All diving personnel were assigned and instructed in the use of a 0-200 MR self-reading pocket dosimeter for monitoring real time exposure, and an accompanying TLD to provide a permanent record of external exposure. It was also recognized that based on previous deep submergence operations involving radioactive sources that considerable shielding would be afforded by benthic sediment, seawater, and the 4.5 cm thick titanium sphere of ALVIN. No measurable radiation exposure above natural background was measured by any of the instrumentation or TLDs.

The potential spread of radioactive contamination was controlled by strict adherence to radiological survey and contamination control procedures for all equipment, including ALVIN, which returned to the ship after coming in contact with the THRESHER or SCORPION debris or surrounding areas. Access to all equipment returning to support ships was restricted until radiologically trained personnel had completed beta-gamma surface contamination surveys, radiation surveys, and swipe surveys which were immediately dried and checked for alpha as well as beta-gamma surface contamination. Oceanographic monitoring devices, ALVIN, and the swimmers used to aid in ALVIN recovery operations were all surveyed prior to being released to unrestricted areas. All survey instruments were calibrated by GE-KAPL personnel prior to departure. Each instrument was also source checked daily during shipboard use. None of the personnel monitoring devices or radiological survey instruments measured any detectable radiation or radioactive contamination in or on ALVIN, support vessels or any of the recovered oceanographic equipment. Only very low levels of radioactivity as measured by the in situ gamma spectrometers or sensitive laboratory instrumentation were detectable in any samples obtained from the THRESHER or SCORPION sites.

Analytical Techniques

Samples of sediment, unfiltered seawater and marine life were placed in 500 cc Marinelli beakers or 250 ml plastic containers, as appropriate, for counting using intrinsic germanium detectors coupled to multichannel analyzers. Mass spectrometry analysis was conducted on samples found by gamma spectroscopy to contain detectable levels of uranium-235, and on selected samples obtained from the SCORPION area. Samples of seawater, sediment, and marine life tissues underwent chemical separation procedures for subsequent analysis for uranium and plutonium by isotope dilution mass spectrometry.

Prior to radiochemical analysis of seawater samples, salinity measurements were made on the discrete water samples to verify that they were truly bottom samples and had

not been compromised in any way by malfunction or handling. The samples were then analyzed radiochemically for strontium-90, cesium-137, and plutonium-239 and 240 by the standard procedure contained in Wong et. al. 1970 and Suprenant et. al. 1983. Nickel-63 is the predominant beta-emitter present in the corrosion products inside reactor coolant systems and contained in the activated metal structures inside the reactor vessel. Nickel-63 analyses were conducted on those sediment samples that contained detectable concentrations of cobalt-60, and on seawater, selected marine life samples, and corrosion products magnetically removed and concentrated from a sediment slurry. Corrosion products were magnetically separated from those sediment samples having detectable concentrations of cobalt-60.

Biological samples of fish and other marine life were preserved for subsequent enumeration and identification to the lowest taxonomic levels possible. All of the fish were initially frozen. Crustaceans and other organisms were fixed in 8% formalin and later transferred to 70% ethanol.

Thirty-three of the rattails (Coryphaenoides armatus), three codlings (Antimora rostrada), and the eel pout (Zoarcidae sp.) from the THRESHER site and the twenty rattails (Coryphaenoides armatus), five Ophidiids (Barathrites iris), and the two eels (Histiobranchus bathybius) from the SCORPION site were thawed and dissected on a clean bench for subsequent trace metal analysis. Triplicate samples of various tissues including muscle, skin, kidney, bone, gonads and liver from each fish were analyzed.

Samples of stomachs, intestines, and anal sacs of the twelve fish from the THRESHER site and the twenty-seven fish from the SCORPION site were selected for gut content analysis to determine food web interactions. The primary advantage of the immunological analysis method is its ability to qualitatively characterize the soluble proteins present in an otherwise unidentifiable mass of gut contents or fluids.

RESULTS

Radiological and Environmental Evaluation

The radioactivity concentrations measured in the sediment samples agree with expected results when compared with decay-corrected values for previous samples, including spacial variations in radioactivity levels. When compared with 1977 and 1979 results (U.S. Navy 1984), a smaller percentage of the core samples exhibited detectable concentrations of cobalt-60 in the surface (0 to 5 cm) sediment segment. Similarly, a larger percentage of the cores exhibited detectable concentrations of cobalt-60 in the deeper segment. This indicates a dilution of radioactivity in the surface sediments that cannot be accounted for by radioactive decay alone. This dilution may be due to mixing of the sediments caused by benthic storms, sedimentation, and bioturbation resulting in the movement of radioactivity to greater depths. The main source of the radioactivity was determined to be the corrosion products released from the reactor coolant at the time of the accidents. Subsequent corrosion of activated structural material has not resulted in a significant increase in the radioactivity present in the sediment. The cobalt-60 concentrations measured in the sediment are small compared with the potassium-40 and other naturally occurring radionuclides present. Cobalt-60 was not detectable in any of the seawater samples or marine life including the 63 fish. By gamma counting samples for a nominal 1000 min, it was possible to achieve typical detection sensitivities of 0.01 pCi/gm for sediment and marine life and 0.005 pCi/liter for seawater samples using solid-state intrinsic germanium and Ge(Li) detectors coupled to multichannel analyzers. Levels as low as 0.001 pCi/liter were obtained for seawater samples using radiochemical extraction. Cesium-137 concentrations in excess of fallout levels were not measured in any of the seawater, sediment, or marine life samples analyzed. This indicates that the integrity of the reactor fuel inside the core has been maintained.

The highest concentrations of cobalt-60 in the sediment (< 0.3 pCi/gm at the THRESHER site and < 0.5 pCi/gm at the SCORPION site) were found in the upper 5 cm. Cobalt-60 was detectable in sixteen of the thirty core tubes collected by DSV ALVIN at the THRESHER site and in seventeen of the fifty-three core tubes collected at the SCORPION site. At both sites, the absence of cesium-137 concentrations in excess of those attributable to fallout indicates that the integrity of the reactor fuel inside the cores has been maintained and confirms the results from previous samples obtained by KAPL.

Results from the analysis of marine life samples indicated that the species

composition and abundances were within the range of variation expected for the depth and location of the sampling area. It is evident from these results that there are no significant ecological changes attributed to the sunken submarines and that there has been no significant "reef" effect although some colonization of the debris was observed. The immunological assay of the fish stomach contents indicates the apparent predation on deep sea invertebrates, including benthic gastropods, sea anemones, holothurians, and tanaids. Although little is known about the feeding habits and range of deep sea fish, the fact that none of the fish contained any radioactivity attributable to the THRESHER or the SCORPION and that the organisms found in the fish guts were predominantly benthic indicates that there has been no significant bioaccumulation or concentration of the radioactivity released at the time of the submarine sinkings. Marine animals could theoretically accumulate radioactivity by ingestion of contaminated sediments or prey organisms, by absorption from the seawater, or by the adsorption onto external tissues. Any bioaccumulation of the refractory radioactive corrosion products found in the sediment at the THRESHER or SCORPION sites was not observed. If one considers the nonradioactive cobalt and nickel present in the seawater and sediments combined with the inventory of nonradioactive cobalt and nickel in the fish, there exists a great dilution potential of stable elements to reduce the likelihood of any potential bioaccumulation of radioactive isotopes of these elements.

This monitoring program provided the necessary samples and information to evaluate the environmental impact of the loss of the THRESHER and SCORPION. The collection of seawater, sediment, and marine life samples and their subsequent analysis for radioactive corrosion products and fission products provided a check on the various environmental exposure pathways. Since most of the corrosion products, including those formed during previous reactor operations, released from corrosion resistant material are insoluble (Schmidt 1982), they are likely to have deposited on the sediment when they were released from the submarine debris. Some of these corrosion products may have been adsorbed into sediment particles or complexed by organic materials. The soluble corrosion products which were shown (Schmidt 1982) to represent only a small fraction of the chemical forms in deep ocean sediments, could have been transported through pore water to different locations within the sediment. By taking core samples, both sediment particles and pore water were available for quantitative radiochemical analysis. In situ gamma radiation measurements were useful to obtain immediate indications of any abnormal radioactivity concentrations but were unable to find any areas with contamination levels in excess of 3.6 pCi/g at the THRESHER site or 2.08 pCi/gm at the SCORPION site. If localized areas of higher contamination were found, the information would have been useful to direct subsequent core sampling locations. Because of the small quantities of radioactivity released to the environment, and the great amount of available seawater dilution, it was necessary to use the in situ pumping systems to concentrate and remove the radioactivity from the seawater in the immediate vicinity of the THRESHER and SCORPION. In spite of the highly sensitive detection capability achieved, no radioactivity attributable to either submarine was found in any of the seawater samples analyzed except for small quantities of particulate cobalt-60 resuspended from the sediment by ALVIN operations and two soluble phase cobalt-60 signals near the detection limit attributed to "bleed through" of particles from the prefilter. Elevated concentrations of plutonium were not detected and confirmed in sediment, water, and fish samples. Los Alamos National Laboratory reports that the plutonium in the nuclear weapons at the SCORPION site would have been exposed to seawater either immediately upon sinking or shortly afterward. Plutonium metal corrodes quickly in seawater to form a heavy insoluble material. Since any chemical or physical changes of the plutonium occurred shortly after the sinking and no elevated plutonium was found in the environment during this survey 18 years after the sinking, there is no reason to believe that the plutonium will be subject to accelerated release or dispersion in the foreseeable future.

Evaluation of Monitoring Devices

The radiological environmental monitoring conducted from the ATLANTIS II, ALVIN, LULU and CAPE FLORIDA yielded very useful information on the suitability of various types of equipment for this type of operation. In general, surface-deployed devices such as the box corer, which are heavier and more streamlined, are easier to deploy at a given target than those devices such as the fish trap that are more apt to drift on their way to the bottom. With careful maneuvering of the surface ship, it

was possible to place monitoring devices close to the target area.
These demonstrations of surface-ship and submersible-conducted radiological environmental monitoring involved two semi-independent means of accomplishing the required objectives. The selection of either method for use is dependent upon the specific requirements for the given site. For example, use of the submersible ALVIN at depths in excess of its maximum operating depth of 4500 m is not permitted. The surface-ship-deployed monitoring devices have depth capabilities in excess of 6700 m. Submersible-conducted operations are more appropriate where "close in" sampling from precise locations are required. Surface ships are more appropriate for general area monitoring.
Future deep-ocean radiological environmental monitoring expeditions should concentrate on utilizing those devices that proved to be the most productive during these expeditions. This would include the surface-deployed box corer, water pump, fish traps, current meter, and the submersible-operated sediment core tubes, box corers, water pump, water bottles, marine life samplers, and gamma spectrometer. If sediment samples with undisturbed surfaces are desired, then samplers such as the box corers, or submersible-operated core tubes should be utilized. The other types of sediment samplers either penetrate the bottom too rapidly or otherwise disturb the surface sediment. If high sensitivity is required for analysis of the radioactivity content of seawater, the water sampling pump can provide detection limits better than 100 times greater than that obtainable from collection and analysis of water samples by traditional means. The specific equipment used should be tailored to the objectives of the monitoring expedition. For surface operations, the oceanographic vessel should be equipped with a bow thruster to assist in the proper positioning of the ship and equipment within the acoustic transponder navigation network.
DISCUSSION
The radiological environmental monitoring conducted with the ATLANTIS II, CAPE FLORIDA, LULU and ALVIN demonstrated the capability to conduct deep-ocean monitoring with the equipment discussed above. ATLANTIS II and CAPE FLORIDA were able to place surface-deployed samplers and monitoring devices in close proximity to major pieces of the THRESHER and SCORPION by the use of an acoustic transponder navigation network. The relative positions of these instruments were determined by computer and confirmed by ALVIN to be satisfactory. The exceptional maneuverability of ALVIN, supplemented by the use of the remotely operated vehicle JASON, Jr. provided the most complete environmental monitoring of the THRESHER and SCORPION yet conducted. ALVIN was able to make in situ gamma measurements and obtain sediment, seawater and marine life samples. These expeditions demonstrated that comprehensive radiological environmental monitoring of localized deep ocean areas could be achieved at reasonable cost using existing oceanographic technology.
The samples of water, sediment, and marine life obtained from the THRESHER and SCORPION sites were analyzed for radioactivity. None of these samples showed any evidence of release of radioactivity from the THRESHER or SCORPION reactor fuel elements. Cobalt-60 in magnetic high temperature corrosion products released from the THRESHER and SCORPION coolant systems or internal surfaces of piping or components during the sinkings was detectable at concentrations lower than those measured in 1977 and 1979. Nickel-63 was detectable at low concentrations in corrosion products magnetically separated from sediment samples. The cobalt-60 and nickel-63 radioactivity present in the sediments is small compared with naturally occurring radioactivity. Cobalt-60 and nickel-63 were not detected at significant concentrations in samples of seawater or marine life. Cesium-137, strontium-90, plutonium-239, 240 and americium-241 concentrations in seawater were consistent with deep water concentrations expected at this location and depth from nuclear weapons testing sources rather than THRESHER or SCORPION debris. Thus, the loss of the THRESHER and SCORPION has not had any significant environmental impact.
The reactors and materials used in all U.S. Naval submarines and surface ships are designed to minimize potential hazards to the environment even under the most severe conditions such as the accidental sinking of a ship. The reactor fuel elements are made of materials that are extremely corrosion resistant, even in sea water. Based on deep sea corrosion data from submarine structural materials reported in Barth and Sheldon (1989), the reactor core could remain submerged in seawater for centuries without releases of fission products, while the radioactivity decays. Since the protective cladding on the fuel elements corrodes only a few millionths of an inch

per year, most of the fission product radioactivity (primarily cesium-137 and strontium-90, with nominal half-lives of 30 years) will be gone by the time the cladding is penetrated by corrosion. Since the corrosion rates of these materials at the bottom of the sea are so low, the maximum rate of release and dispersal of the radioactivity would be so low as to be non-detectable and would have no future adverse impact on the deep-sea environment.

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Session 61 -- Pollution Prevention/Waste Minimization
Co-chairs: David A. Zigelman, WSRC;
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61-1
OPPORTUNITY ASSESSMENTS: A TOOL FOR WASTE MINIMIZATION AND POLLUTION PREVENTION

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ABSTRACT
Waste costs, liabilities, and regulations have been and are today growing concerns for government and industry. Pollution prevention opportunity assessments (PPOAs), formerly known in the Department of Energy as process waste assessments (PWAs), are tools which assist a site in achieving pollution prevention benefits. A PPOA is an important tool in a site's Waste Minimization / Pollution Prevention (WMin/P2) Program which can lead to the identification of pollution prevention opportunities, prioritization of pollution prevention activities, implementation of cost-effective pollution prevention practices and technologies, and creation of a baseline from which to set waste reduction goals and measure progress. PPOAs are applicable to any process or planned activity which has the possibility of generating waste -- this includes activities or projects that handle and manage waste after it has already been generated.
This presentation will provide an overview of PPOAs and their role in WMin/P2. It will describe the importance of bringing WMin/P2 into the planning stages of any

process or project and how PPOAs can be used in Decontamination and Decommissioning and Environmental Restoration activities.

INTRODUCTION

Waste costs, liabilities, and regulations continue to be growing concerns for government and industry. Pollution prevention opportunity assessments (PPOAs) are a tool which proactively manages these concerns. The Department of Energy (DOE) has recognized the benefits of PPOAs in the DOE Waste Minimization Pollution Prevention Crosscut Plan (1) which states that the completion of opportunity assessments is of critical importance to DOE because they are an essential management decision-making tool that tell DOE:

1. how much waste and environmental releases can be avoided,
2. through what activity/process changes waste reduction can be achieved,
3. what it will cost to implement a pollution prevention opportunity, and
4. what will be the long-term savings in avoided waste management costs.

Before exploring what a PPOA is, a brief discussion of pollution prevention is necessary. Pollution prevention (P2) may also be referred to as waste minimization, source reduction, and/or recycling. The definition used in the DOE-sponsored PPOA training class is: "Pollution prevention reduces or eliminates material releases to air, water, and/or land." The benefits of pollution prevention include a proactive approach to waste management, compliance issues, and liability concerns; economic incentives; improved employee and public health; and environmental stewardship. Basically it all boils down to the phrase, "It's the right thing to do."

What is a Pollution Prevention Opportunity Assessment?

A PPOA is a tool with the objective of identifying opportunities and methods to reduce or eliminate all wastes (2). A material balance is performed around a specific process, project, or activity to qualify and quantify the materials entering and exiting. The exiting materials are separated into product, by-product, recycled, waste management (liquid and/or solid), and air release streams. Next, opportunities are identified and evaluated for their ability to eliminate or minimize the non-value-added output streams exiting the process. Therefore, the PPOA provides the basic tool for the identification of opportunities to eliminate/minimize the release of hazardous, nonhazardous, and radioactive wastes. This also provides a comprehensive baseline from which to measure P2 progress. The assessments also identify those processes, activities, and procedures that need to be improved or replaced to promote pollution prevention. Simply stated, PPOAs can be the foundation upon which comprehensive and effective pollution prevention rests.

Application of PPOAs to Nonroutine Activities

Opportunity assessments have commonly been completed on routine processes -- those which occur in a manufacturing plant. However, they can also be useful on nonroutine waste streams and at nonmanufacturing sites such as laboratories. By changing one's point of view, PPOAs can also be conducted on non-routine activities.

Some facilities or groups conduct activities on a project-oriented basis (3). This means that only one waste-generating activity is done once. Two examples are an engineering laboratory that builds prototypes and a research laboratory conducting experiments. In each case the projects have unique chemical inputs, a defined length of time, and very specific activities.

Project-oriented activities can be assessed in two ways. First, a macro-level view, looking for consistent, general activities that occur from project to project or experiment to experiment, can be taken. Such activities can include material procurement, cleaning techniques, or personal protective clothing use. Common waste streams can be reviewed using a standard PPOA, resulting in facility-wide waste reduction ideas, such as implementation of procurement controls or procedure changes.

The second way to approach pollution prevention is to include it in the planning phase of the project, experiment, or activity. The project scope and plan would be reviewed in the same way an on-going activity would be reviewed, except on paper in PPOA form. The steps of the project can be mapped, inputs and outputs identified, and brainstorming done to identify options with the potential for waste minimization. Applicable, cost-effective methods to reduce waste can then be written into the project plan. In addition, it is helpful to make sure that at least one member of the planning team is an advocate of pollution prevention.

The waste streams from Decontamination and Decommissioning (D&D) and Environmental Restoration (ER) are also often considered "nonroutine" and project oriented. Even

when these processes are performed on waste, they generate additional, so-called secondary, waste and there are opportunities to minimize this waste. Decontamination and Decommissioning operations usually involve removal of buildings and equipment, while ER usually includes treating or removing soil and returning the land to its natural state. Therefore, these activities can be addressed on a macro level or in the planning phases. The common waste streams such as personal protective equipment, deactivation, decontamination debris from surface cleaning, and material inputs can be assessed or PPOAs can be completed during the planning phase.
An alternate way to consider D&D and ER waste streams is to view them as part of a long series of routine processes.(See Table I)
TABLE I
An example is the D&D of buildings. If there are many buildings at a site selected for demolition, then the entire schedule can be viewed as a regular activity to be conducted multiple times. This perspective permits the consideration of equipment and initiatives which may only be cost effective when used on multiple projects, as well as develop lessons learned from one to another. Therefore, PPOAs should be applied to the entire site restoration program to determine common pollution prevention opportunities, since some opportunities may be economically feasible only when multiple projects are considered. By performing PPOAs on the routine sub-tasks within a D&D or ER project and revising these assessments as knowledge is gained, it is possible to continuously improve waste minimization activities.
Decontamination and decommissioning and ER projects are unique in that a large portion of the waste, commonly referred to as legacy waste, already exists. Since these activities involve the tearing down of structures or cleaning up of existing waste or structures, some people claim that no pollution prevention can be accomplished. However, extensive recycling opportunities with concrete and steel and other building materials are present. The key is to prevent as much secondary waste as possible (source reduction) and embrace recycling as an excellent and viable method for successful P2.
PPOA Methodology
The major steps involved in a PPOA are:
- Planning and Organization,
- Project / Activity Assessment,
- Project Description,
- Flow Diagram,
- Input/Output Summary,
- Option Generation,
- Option Evaluation, and
- Final Report.

In order to perform a successful PPOA, all employees involved in the project must participate, most importantly the operators and workers with the operation and/or program experience. Employee activities can significantly affect the amount of waste generated, and therefore, they can contribute greatly to the assessment. Successful pollution prevention requires actions and ideas from everyone.
After organization, the project needs to be assessed or evaluated. The end result is baseline knowledge of the activity and both qualitative and quantitative measures of the inputs and the outputs associated with it. In order to develop this knowledge, a detailed understanding of the wastes generated and operations performed is required. The assessment should begin by examining information about the process, operations, and waste management practices. Tools such as flow diagrams and material balances are used to achieve this baseline knowledge.
Once the project steps and sources of waste generation are understood, the PPOA enters the creative phase. Following the collection of data and site inspections, the members of the team will have begun to identify possible ways to minimize waste or prevent pollution. Identifying potential options relies both on the expertise and creativity of the team members. Much of the requisite knowledge may come from their education and on-the-job experience; however, the use of technical literature, contacts, and other sources may also be pursued.
Many pollution prevention options will be identified in a successful assessment. At this point, it is necessary to identify those options that offer real potential to minimize waste and reduce costs. Since detailed evaluation of technical and economic feasibility is usually costly, the proposed options should be screened to identify those that deserve further evaluation. Screening followed by detailed evaluation

serves to eliminate suggested options that appear marginal, impractical, non-cost effective, or inferior.
After the option evaluation is complete, a final report is required to document the team's suggested implementation plan. The final report for a PPOA is a compilation of essential facts characterizing the project or activity, identifying pollution prevention options, analyzing option feasibility, and projecting option implementation costs. The purpose of this report is to document the project baseline and pollution prevention options and to provide information to management on the funding requirements necessary to implement the feasible pollution prevention options.
Pollution prevention opportunity assessments have proven to be useful tools to evaluate input materials, identify sources of waste, and generate pollution prevention options for D&D and ER processes. Two techniques have been successfully used to break these processes down into management chunks. Assessments can be performed on common tasks such as equipment decontamination or sampling. Alternatively, assessments can be performed on subelements such as site characterization or screening of alternatives. Regardless of the method chosen, performing a PPOA results in a fuller understanding of the sources of waste and identification of options for pollution prevention.
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61-2
IMPLEMENTING WASTE MINIMIZATION
USING ROI PROJECTS
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ABSTRACT
The implementation of process changes for source reduction and recycling opportunities typically falls to a lower priority when compared to activities for environmental risk management, regulatory compliance, and Department of Energy (DOE) Order compliance. However, source reduction is still the best practice. How can waste minimization (WMin) facilitators convince waste generators to implement WMin projects? DOE has discovered that identifying high return-on-investment (ROI) projects provides an incentive for waste generators to implement WMin because the savings realized can be used to pay for compliance requirements. Initial projects implemented at several DOE sites have proved that WMin projects can achieve the cost savings expected. These cost savings result from avoided waste disposal costs, reduced waste handling costs, and reduced material and energy costs. There is currently an excess of commercial hazardous waste treatment capacity because WMin has been so successful in the private sector. Intangible savings may also be realized from improved safety and work environments, delay or avoidance of capital expenditure for storage and disposal facilities, and avoidance of waste storage and disposal liability.
WHY USE ROI PROJECTS FOR WASTE MINIMIZATION?
The Department of Energy (DOE) is becoming more aggressive in its implementation of waste minimization/pollution prevention (WMin/P2). While DOE sites have operating funds for training employees and tracking and reporting waste, few sites have the capital funds necessary to implement WMin/P2 projects, even for changes as simple as installing a bypass wastewater disposal system. Although some procedural and administrative improvements can be implemented using operating funds, it is usually small capital projects that have the biggest impact on waste reduction. Waste minimization is recognized as the best waste management program for the future due to increasing costs and potential liability, but most waste management programs at DOE sites must focus limited resources on regulatory compliance and risk avoidance.

Return-on-investment (ROI) projects that implement waste minimization can compete for waste management dollars if they offer an ROI that provides a high return on those waste management dollars in a short period of time. This facilitates making capital improvements that increase process efficiency, or recycling and reusing materials, consequently reducing waste. The waste management funds invested in WMin/P2 ROI projects generates savings within a year or two; those savings can be used for compliance activities.

Cost-effective WMin/P2 projects are needed so WMin programs can compete with other regulatory compliance activities. All large quantity waste generators are required by the Resource Conservation and Recovery Act (RCRA) to certify that they have a waste minimization program on each hazardous waste manifest (40 CFR 262, Appendix). The United States Environmental Protection Agency (EPA) issued interim final guidance on the elements of a waste minimization program in 1993 (EPA, 1993), but this guidance is extremely flexible compared to the more explicit rules governing waste management. As a result, waste minimization programs are the first to lose funding during periods of diminishing resources. The EPA is becoming stricter about auditing WMin programs for their effectiveness, but this program will probably not impact waste generating sites for some time.

WHAT ARE ROI PROJECTS?

ROI is a comparison measure used by private industry to evaluate whether to implement capital projects or invest profits in financial opportunities. The return rate is calculated as a percentage of profit earned from the implemented project divided by the capital invested. This return rate corresponds to an interest rate that would be earned on financial investments. If a company can earn a 10% return on a bond or other financial tool, why should it invest in a capital project that only returns 6%? ROI measurements ensure that a company earns the greatest possible profit on funds invested. It also can be used to prioritize projects. In a non-profit government environment, savings realized in a few years from investments creates a sense of urgency for making improvements.

Capital ROI projects involve purchase and installation of equipment to improve a process and reduce waste. An example is installing a pH meter on wastewater outlet piping and a bypass valve that routes the wastewater directly to an effluent point if the pH is within permitted range and requires no treatment (Fig. 1 illustrates existing and proposed effluent treatment systems). Some capital projects may completely replace a process with new technology. Many sites are purchasing digital printers to replace hazardous photochemical processes, so no chemical waste is generated. Other WMin/P2 projects include installing a distillation column to recover acid and recycle it back into a process, installing bulk oil storage tanks so individual oil containers are not needed, and recovering lead from soil for recycling.

Fig. 1.

ROI evaluation for capital projects measures the expected improvement in a process after the project is implemented. The profit earned is determined by deducting the expected operating costs and equipment amortization costs from the expected savings. This profit term is divided by the cost of implementing the project. This ROI calculation takes the following form:

Eq.

where

S - annual operating cost savings
A - annual operating costs
C - capital equipment costs
I - labor and material costs
L - useful equipment life

Annual operating cost savings (S) can be measured in materials and energy saved, waste service costs avoided, and waste handling labor eliminated. Expected annual operating costs (A) include additional labor, increased energy, new materials, and new waste disposal services required to operate the new process. The equipment amortization term (C+I/L) divides the project installation costs (C+I) by the useful life of the equipment. The project installation costs include capital, labor, and material costs. Capital equipment costs (C) include the funds needed to purchase pumps, valves, piping, or other items needed to complete the process change. Labor and material costs (I) are required for the initial installation of the capital equipment. These savings and cost terms are used to measure the performance of the

implemented projects.
ROI PROGRAM AT DOE/AL
DOE/AL based its WMin/P2 ROI program on a successful energy and waste reduction program at Dow Chemical (Nelson, 1993). Dow held a contest to solicit energy and waste reduction projects, implemented those with ROI values greater than 100%, and achieved audited savings of over $110 million per year for 575 projects. The Deputy Secretary of Energy was so impressed by these results that he decided to adopt the program. The DOE Pollution Prevention Executive Board identified high ROI WMin/P2 projects at the DOE sites. Twenty ROI projects have been funded by headquarters and will be audited in FY95 to determine if projected waste reductions and cost savings were achieved.
The Albuquerque Operations Office (DOE/AL) decided to accelerate the ROI program by using waste management funds that had been held back for productivity improvements to implement eight ROI projects during late 1994. DOE/AL developed an ROI program guide to assist sites in identifying projects, calculating the savings and costs associated with implementing the projects, preparing a project proposal package, and locating funding sources. The WMin/P2 projects submitted competed very successfully against other productivity projects. Three projects are now complete and operational, and five other projects are being implemented. Table I summarizes the ROI and the waste reduction expected from each project, estimated using the ROI calculation presented above.
TABLE I
INITIAL PROJECTS
These initial projects are not large or glamorous investments, but the annual savings is significant. Simple process changes can greatly impact the amount of waste generated. DOE/AL has found that the following elements lead to a competitive ROI project:
  use of existing and proposed process flow diagrams;
  preparation of background calculations identifying the savings and costs used in the ROI calculation;
  a thorough understanding of the waste generating process; and
  identification of potential funding sources.
A process flow diagram clearly defines the existing conditions by showing inputs, outputs, waste streams, labor and energy required, and process flow rates. Preparing the proposed flow diagram forces the project champion to consider any additional costs associated with the proposed process -- increased maintenance of equipment, increased energy use, new materials and waste disposal services required. Careful evaluation of a process flow diagram eliminates projects that convert one form of waste into another, which is not waste minimization. The background calculations provide a reality check on the basis for project costs and savings that are easily inflated by eager project champions. Although the WMin/P2 program is currently providing funding for these ROI projects, sites are encouraged to identify other funding sources for ongoing projects.
The following project descriptions give a brief synopsis of the simple and effective process changes implemented or proposed, as well as the basis for the annual savings.
Empty Paint Cans Press
An oil filter press was installed in a paint shop to crush empty and partially empty four-liter and twenty-liter paint cans. The press crushes the cans to one-eighth of their original size and the paint is collected separately. The volume of empty containers requiring disposal is decreased by 11 m3 per year, so the savings result from waste disposal costs avoided.
Portable Oil Analyzer
A portable oil analyzer was purchased for use on all vehicles and installed equipment to determine if an oil change is required. Previously, oil samples were sent off-site for analysis or oil was changed according to a conservative maintenance schedule. Use of the new analyzer saves oil because it is changed less frequently. Elimination of oil incineration costs and off-site laboratory costs are the basis of the savings. This new change-only-when-needed procedure is estimated to extend the life of all lubricants about 400%.
Lead Recovery
Lead-contaminated soil was formerly excavated, put in drums, and shipped as hazardous waste. A soil sifter was purchased to remove the lead, which can be

recycled, and the soil can be returned to its original location. The savings result from waste disposal costs avoided and recycled material sold.

Booster Water Heater

Currently, disposable styrofoam dishes and plastic utensils are used in cafeterias and create large volumes of waste annually. A booster water heater will be purchased at one site so the dishwasher can be used on reusable dishes and silverware. The savings are expected to come from eliminating the need to purchase plastic products and avoided waste disposal costs.

Bulk Oil Storage Tanks

The vehicle maintenance facility currently purchases motor oil in plastic liter and four-liter containers. Bulk oil storage tanks will be installed so oil can be purchased in bulk at a lower unit cost. It also eliminates the containers and saves their disposal costs.

Stainless Steel Rack

A stainless steel vacuum gas rack will be installed to replace the mercury glass rack in a laboratory. This eliminates mercury-contaminated oil and solid waste, so disposal costs are avoided.

Reject Water Bypass

This project will install conductivity and pH meters in an effluent pipeline in order to determine if the wastewater meets city discharge limits (see Fig. 1). If so, wastewater will be diverted through a new pipeline that bypasses the industrial wastewater pretreatment facility so it can flow directly to the discharge point. This saves money on the chemicals and energy needed to run a pretreatment facility, and disposal costs for the sludge generated by the facility on about 15% (80,000 m3) of the process outflow.

SUMMARY

These projects all achieve most of their savings through avoided waste disposal costs. This proves one of the basic tenets of waste minimization -- avoiding waste generation saves both present and future costs. The metrics to measure project performance are built into the ROI formula -- waste volume reduced, actual costs incurred, and material and disposal costs saved. The intangible savings associated with WMin/P2 projects are more difficult to quantify, but include reduced worker exposure due to improved safety and work environments, delay or avoidance of capital expenditure for storage and disposal facilities, and avoidance of waste storage and disposal liability. As process managers learn how WMin/P2 can improve their operations, they are more receptive to other WMin/P2 initiatives. DOE will continue to identify and develop ROI projects to implement WMin/P2 at DOE sites. Many off-the-shelf, short duration, quick turnaround projects are already being implemented. DOE is using other ways to promote projects that require technology development, longer installation periods, and larger capital investments.

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61-3

MIXED WASTE MINIMIZATION IN THE BIOMEDICAL COMMUNITY

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ABSTRACT

Research-related biomedical mixed waste is one of the major sources of commercially generated mixed waste. Data from the National Institutes of Health (NIH) were evaluated to characterize typical research-related mixed wastes and to determine which of these mixed wastes are difficult to manage. Two such waste streams include

fixes and wastes from electrophoresis gels and wastes from protein blotting procedures. Interviews with individual researchers were conducted to determine reasons for generating mixed waste and methods of minimizing or eliminating such generation. Treatment alternatives were also identified to improve management of biomedical mixed waste. Possible alternatives to generation of mixed waste include automated techniques using chemiluminescence. It is apparent, however, that mixed waste generation from biomedical research can be reduced but not entirely eliminated. The more difficult-to-manage mixed wastes tend to be those having radionuclides with relatively long half-lives and relatively low heat content. One treatment option for facilities having large volumes of such liquid mixed waste is treatment by phase separation, pH adjustment, and chemical oxidation, with subsequent release to a sanitary sewerage system regulated under the Clean Water Act.

BACKGROUND

According to recent surveys conducted by the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission, there is a large number of biomedical research facilities that generate low-level mixed waste (1). Much of these wastes, which are not liquid scintillation counting fluids, are difficult to dispose of. As a result, these wastes are being accumulated on site at research institutions until disposal options become available. Because of the difficulty that research institutions are having in finding adequate storage capacity and disposing of these wastes, they have been the target of increasing scrutiny. Described below are the results of a waste minimization program directed at the low-level mixed wastes (LLMW) produced at one of the largest biomedical research facilities in the United States, the main campus of the National Institutes of Health (NIH).

The NIH main campus is a sprawling site located in Bethesda, Maryland. At this site, basic and applied biomedical research is carried out by over 20,000 researchers using most of the methods, techniques, and procedures that are typical for work in this area. In recent years, these activities have produced a significant volume of LLMW which is in long-term storage on site. It is this volume of LLMW which has served as the impetus for the ongoing mixed waste minimization program.

Low-level and mixed waste is managed at the NIH by a centralized waste management group which is responsible for collecting, characterizing, storing, treating, tracking, and disposing of all LLMW produced at the NIH. This group maintains a sophisticated computer database tracking system which is used to manage information about the LLMW from the time of generation until it is treated, disposed of, and/or placed in long-term storage.

The Waste Minimization Program that Foster Wheeler Environmental Corporation identified for NIH was based on EPA's Waste Minimization Opportunity Assessment Manual (2) which defines waste minimization as "source reduction to reduce or eliminate waste generation, followed by recycling or reuse where wastes are unavoidable, and finally, minimization of the volume, toxicity, and mobility of waste generated through treatment and stabilization." The first step in a program such as this is to obtain management commitment to waste minimization, develop written program objectives, allocate resources, and establish goals. The second step is to assess the available data on the sources and types of waste being generated and identify options for waste minimization. The third step is to develop and evaluate the feasibility of alternatives for accomplishing waste minimization, as defined above. The final step is to implement alternatives that achieve the waste minimization goals and to document and communicate the results.

Foster Wheeler Environmental Corporation conducted the second and third steps of this program for the NIH. Thus, we assessed the quantity and types of LLMW that were being generated and the waste-generating processes. The NIH waste tracking database was used to develop profiles of the typical LLMW streams, their waste designations, and the treatment and disposal options that were being used for the waste. This database contained approximately 3,800 records in a dBASE II format describing LLMW generated between 1989 and June 1993. Each record contained 187 fields that were used to indicate such parameters as the type of waste-generating process, the radionuclides present, the chemical constituents present, the waste volume, the waste designation, and the treatment and disposal options which were used for managing the waste. This information was used to sort the waste streams into groups according to the waste-generating process, to characterize the volume and composition of the wastes produced by each of the process groups, and to determine

which process groups were producing the largest amounts of LLMW which had no disposal options. The Department of Energy's National Low-Level Radioactive Waste Management Program recently published the results of this data investigation in a report (3).
The significant findings from the initial database analysis were 1) relatively few waste-generating processes produced the large majority of LLMW; 2) wastes produced by these processes have similar properties and compositions; and 3) the large majority of the difficult-to-manage LLMW is produced by even fewer processes. Difficult-to-manage LLMW was assumed to be LLMW in storage for more than one year. These results suggested that a successful waste minimization approach might be to investigate in detail the processes that were generating most of the difficult-to-manage LLMW and look for options in these processes which would meet waste minimization objectives.

APPROACH

The approach for investigating the waste-generating processes at NIH was a two-part process. The first step was to use the database information to identify laboratory facilities and researchers that were generating the difficult-to-manage LLMW. The second step was to interview each of these generators regarding the biomedical research procedures they were using and the LLMW that they were generating as a result. During the interviews, each generator was asked to identify alternatives which would achieve LLMW minimization objectives. When the interviews were completed, comparisons were made between researchers using the same procedures to determine if the LLMW generation was comparable. The information obtained from the generators showed that substantial LLMW minimization was achievable for several procedures and that many researchers were already using techniques and procedures that reduced LLMW generation.

RESULTS

Of the difficult-to-manage waste streams identified, two waste streams represent most of the waste in storage for more than one year. These waste streams include fixes and wastes from electrophoresis gels and wastes from protein blotting procedures.

Electrophoresis Gel Fixing/Washing

The biomedical research process which generates the largest volume of LLMW in storage for more than one year is the fixing and washing of electrophoresis gels. This process is a valuable biomedical research tool for making separations of proteins or deoxyribonucleic acid/ribonucleic acid (DNA/RNA) fragments. The electrophoresis gel technique, used by researchers to separate, identify, and purify fragments, is often used to determine the sequence of bases in nucleic acids. A mixed waste is generated with this process when the DNA, RNA, or protein molecule is labeled with a radioisotope such as 32P or 35S for autoradiography or subsequent processing.
The gels used for electrophoresis are typically constructed of polyacrylamide or agarose. These chemicals act as porous media that behave like sieves by retarding or obstructing the movement of the protein or DNA/RNA molecules while allowing smaller molecules to migrate freely. The size of the gel pore determines the extent of the molecular sieving. The mobility of molecules through gels is sensitive to pH which, therefore, must be controlled. To control pH, the molecules being evaluated are dissolved in an aqueous buffer system that maintains the pH required for the separation. The molecules migrate through the gel media when subjected to an electric field. The mobility of the molecules through the electrophoresis gel is determined by the strength of the electrical field applied to the gel and the charge, size, and shape of the molecule.
After the molecule separation step, the gels are typically fixed and washed by soaking them in a bath of 10% methanol and 10% acetic acid in water. Another bath solution which is frequently used is 7% trichloracetic acid. These bath solutions become mixed wastes after their contact with the gels containing the radioactively labeled DNA, RNA, or proteins. Fixing and washing the gels is considered necessary to prevent tearing the gel and to remove contaminants in the gels such as urea that might otherwise prevent the gels from drying or cause them to stick to the photographic film. The fixed and washed gel is then placed in contact with photographic film and radiation emitted from the labeled DNA, RNA, or protein fragments exposes a characteristic pattern on the film. This technique is called autoradiography.

Several alternatives were identified for achieving waste minimization for the gel fixing/washing process. All of these are currently being used, to some extent, and all have advantages and disadvantages as compared with the standard approach described above. An obvious approach is to eliminate the use of radioisotopes for identifying the DNA and protein molecules being separated with gels. Other options include using fluorescent dyes such as ethidium bromide or visible dyes such as Coomassie Blue or silver-based staining. Several manufacturers supply kits and procedures using materials such as these as substitutes. While the use of fluorescent dyes such as ethidium bromide may avoid generation of a mixed waste regulated under the Resource Conservation and Recovery Act, ethidium bromide is considered to be a toxic chemical. Thus, overall risk minimization should also be considered in evaluating substitute processes.

A variation on these materials is the use of automated DNA sequencing equipment in which fluorescent dyes are laser activated, scanned, and analyzed by computers in real-time as the molecules migrate through the gels. This type of equipment is expensive (>$100,000) and is more efficient in high volume, standardized applications that can tolerate the required time needed for the initial setup, calibration, and standardization.

If autoradiography cannot be eliminated, the volume of gel fixing and washing bath solutions (mixed waste) may be eliminated by using gels whose manufacturers' claim do not require fixing. However, if the research question cannot be answered using these substitute gels, mixed waste can be minimized by using smaller volumes of solutions for the fixing step, or by recycling the bath solutions several times before discarding them. Volume reduction of as much as 90% has been achieved by some researchers. In other situations, the fixing volume can be reduced by obtaining more effective gel dryers which can quickly dry gels even though they have not been extensively washed prior to drying and would otherwise require lengthy drying times.

The generation of a LLMW during gel fixing/washing can also be prevented in some cases if ethanol is substituted for methanol. However, the effectiveness of such a substitution is case-specific and not always feasible.

Blotting Techniques

Another significant biomedical research LLMW generating process is called "blotting." In this process, radioactively labeled protein residues, which have been previously separated using gel electrophoresis, are transferred, using an applied electric field, from the gel to a membrane such as a glass fiber filter coated with a thin layer of noncovalently adsorbed polybase or other media such as sheets of polyvinylidene difluoride (PVDF). This is done because, typically, after electrophoresis, the protein molecule is trapped within the gel matrix and is relatively inaccessible. However, if the protein molecule is transferred to a membrane surface, it becomes easily accessible to many detection and analysis techniques. After the protein is transferred to the filter paper, it is washed using reagents such as methanol or ethanol to improve retention on the membrane and to remove impurities prior to further analysis. These washings can produce a LLMW if the proteins are labeled with radioisotopes and if reagents such as trichloroacetic acid or methanol are used.

The use of radioisotopes for protein identification and the generation of a LLMW may be avoided if chemiluminescent reagents are used. Most of these use the horseradish peroxidase (HRP)-mediated luminol oxidation reaction in which the oxidation of luminol is catalyzed by peroxidase to produce light that can be photographically recorded. Vendors claim that chemiluminescence offers advantages over radioactive methods due to its speed, sensitivity, and convenience. Another possibility for waste minimization is the substitution of ethanol for methanol for final membrane washing. This technique can avoid the generation of a LLMW provided that the ethanol concentration is sufficiently low that an ignitability-characteristic waste is not generated.

Another waste minimization possibility is to use a transfer procedure with an electrophoresis buffer consisting of tris-glycine buffer at a pH of approximately 8.3. While this procedure requires approximately 12 hours to complete, it does not generate a LLMW. A more recent procedure using a concentration of 20% methanol in the transfer buffer results in the generation of a LLMW. The advantage of this procedure, however, is that it is much more rapid and uses a smaller amount of electric current.

Equipment substitutions can also reduce the LLMW volume generated from blotting. One vendor supplies a multi-purpose fluorescence scanner for gels and membranes that can be used as a substitute for radiolabeling techniques. The vendor claims that the system can be used to detect and quantify a variety of complex protein separations by using fluorescent dye kits specific to the test being performed. The system uses an argon scanning laser with a 50-micron scanning width that is combined with a photomultiplier tube and computer analysis system. The time required for analysis is reduced from a total time of 72 hours for film autoradiography to as little as 2 hours with fluorescence. A drawback to this system is that the cost is approximately $57,000.

Other Mixed Wastes

A number of other LLMW-generating processes used in biomedical research were evaluated for waste minimization options. Other processes included high-pressure liquid chromatography (HPLC) scintillation counting wastes and cell culture/harvesting wastes. HPLC waste is generated when high-pressure liquid chromatography is coupled with liquid scintillation counting with output from both analyses assessed by computer. HPLC systems can identify and quantify specific molecular species with high speed and accuracy. HPLC liquid scintillation waste generation rates can be as high as 10 milliliters per minute, 24 hours per day. Another mixed waste that appeared to be difficult to manage was generated from cell culture/harvesting. LLMW is generated when radiolabeled cell cultures are lysed with trichloroacetic acid, followed by cell fragment washing using methanol. Mixed waste is generated from the lysing solutions as well as the washes and liquid scintillation counting.

Effective options were found to be available for most of these processes. Similar to the options described above, the alternatives usually required that generators make some compromises with respect to sensitivity, procedural time, equipment costs, and/or broadness of application.

LLMW Waste Treatment Options

Much of the LLMW produced by the procedures described above does not currently have a disposal option, although some options may become available in the future. These wastes typically contain methanol, acetic acid, or trichloroacetic acid at a concentration of approximately 10%. Other compounds that could be present at lower concentrations include phosphoric acid, chloroform, and ethanol. The pH of the wastes is typically in the range of 1-4 and the radioisotopes typically present are 35S, 32P, 3H, and 14C in relatively low concentrations. Disposal of such a waste stream to the sanitary sewer system would be possible in some areas if it is pretreated. The pretreatment objectives would be to remove compounds from the waste stream that would prevent it from being discharged, increase the pH to acceptable discharge limits, and create a smaller secondary waste stream that could be transferred to an incineration facility or other disposal outlet.

With the exception of chloroform, the reagents present in the LLMW are miscible in water. With the exception of phosphoric acid, they are also organic compounds that are potentially subject to further oxidation. The 32P and 35S radionuclides have relatively short physical half-lives (14.3 days and 87.9 days, respectively) and can be impounded for decay prior to disposal. Treatment technologies for this waste stream are, therefore, focused on removal or destruction of the organics and adjustment of the pH to acceptable discharge levels. Several technologies are available to accomplish these objectives including phase separation, chemical oxidation, biological degradation, carbon adsorption, solvent extraction, and pH adjustment.

One possible treatment system was evaluated by combining phase separation, pH adjustment, and chemical oxidation together in a sequential treatment train. The treatment process would begin with phase separation of immiscible organics, followed by pH adjustment using sodium hydroxide. Then the waste stream would pass through a chemical oxidation reactor where the organic compounds would be oxidized to carbon dioxide. This process would be followed by another pH adjustment and discharge to the sanitary sewer system. If significant quantities of 14C, or other radionuclides, are present, it is likely that an emissions control system would be required for carbon dioxide gas released from the chemical oxidation reactor.

One of the most commonly used chemical oxidation processes for wastewater uses ultraviolet (UV) light and hydrogen peroxide to oxidize organic compounds to carbon dioxide. The UV light activates organic molecules making them more amenable to

oxidation and assists in the creation of hydroxyl radicals from the hydrogen peroxide. The hydroxyl radicals are powerful oxidants that then act on the organic molecules present to produce carbon dioxide and water. Chlorinated compounds will contribute some acidity to the water that would require adjustment of the pH following the oxidation step. The hydrogen peroxide is added to the influent stream and the mixture is passed through a closed vessel equipped with powerful lamps that generate light energy in the UV spectrum. Typically, versions of this technology that use high-intensity lamps operate continuously at ambient temperature and require residence times of 1 to 5 minutes to effect near-complete oxidation of organic compounds.

An advantage of chemical oxidation is that no secondary waste is produced because organics are completely oxidized. The high concentration of the organics (100,000 ppm) in the anticipated waste stream would require a recirculating water stream of treated water to which the concentrated waste stream would be slowly added to maintain an influent organics concentration at a desired concentration of about 1,000 ppm. A 30-kW system is estimated to have the capacity to process 30,000 liters of concentrated LLMW per year at an initial capital cost of approximately $100,000. The annual operating costs for such a system are estimated to be approximately $35,000 for electrical power, chemicals, and UV bulbs. Operating labor would be an additional expense.

CONCLUSIONS

The results from this study indicate that waste minimization options are available for the typical procedures used in biomedical research. These management options have the potential, however, to require more time to perform each analysis, to increase capital expenditures for equipment, to change the quality of the final results compared to a standard published protocol, or to be useful in a more narrow application than traditional procedures. Because of these potential limitations, the individual preferences of researchers need to be both respected and challenged. Institutional conditions such as the availability of funds for new equipment, the costs for disposal services, and institutional tolerance toward the generation of LLMW can affect the LLMW generation. Vendors and equipment manufacturers have the resources to invest in the development of improved protocols and procedures for biomedical research. The development of these alternatives is currently a very active area.

Onsite waste treatment for some types of LLMW is an option for some facilities. This treatment is site-specific due to widely varying permit conditions, permit status, and the ability of an institution to support the operation and maintenance of waste treatment equipment. Providing aqueous mixed waste treatment services to several small generators may be feasible, however, for waste brokers.

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61-4

WASTE MINIMIZATION DURING TRANSITION AT THE SAVANNAH RIVER SITE REACTOR FACILITIES

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ABSTRACT

The Reactor Division at the Savannah River Site is actively pursuing deactivation from an operating mode to a cold shutdown condition with minimum surveillance and maintenance. In addition, the Reactor Division is conducting several transition activities to prepare its five separate reactor facilities for their eventual decontamination and decommissioning. During the planning of this transition, increases in waste generation, both in the types and quantities, was forecasted. The

Reactor Division, therefore, initiated several activities to minimize the waste generated during the transition. This paper addresses the many activities pursued and their effectiveness.

INTRODUCTION

The Savannah River Site (SRS) is a 815.8 square kilometer (315 square miles) Department of Energy (DOE) production facility located in western South Carolina. This facility has multiple operational areas which generate a variety of waste materials. Located on the site are five reactor facilities (K, L, P, R, C) in various stages of deactivation and transition.

In March 1993, the Reactor Division (RD) at STS received guidance from the DOE to halt operation of the K-Reactor and deactivate all reactor facilities to a cold shutdown condition. During FY-94, the RD aggressively pursued this objective by performing the transition activities listed below. Each activity presented a unique waste management challenge, in particular in the area of waste minimization.

- Discharge reactor tank components into the spent fuel pools, including cadmium control rods (mixed waste) from four of the five reactor tanks
- Drain all operational reactor systems, consolidate and clean-up heavy water moderator, and vacuum dry all reactor heat exchangers
- Remove all excess equipment and materials (clean and contaminated) from all five reactor facilities
- Consolidate nearly 40 years of chemical inventory in all five reactor facilities and their disposition
- Clean-up spent fuel pools and consolidate domestic and foreign fuel
- Demolish and remove reactor facility support buildings ranging from clean administrative buildings to a contaminated hot shop
- Remove underground fuel oil storage tanks
- Remove installed lead shielding (clean and contaminated) in four of the five reactor facilities

The activities described below address the minimization of waste generated from the above items. The waste minimization activities are divided into four categories (source reduction, recycling, treatment, disposal) corresponding to the four aspects of the Department of Energy pollution prevention strategy.

SOURCE REDUCTION

Source reduction activities are those which will prevent the generation of waste altogether, and therefore is the most important aspect of any waste minimization effort. Three separate activities established by the Reactor Division will be discussed. They are the formation of the Reactor Waste Management group, the review of work packages using a comprehensive checklist, and the inclusion of waste minimization into ALARA reviews.

Reactor Waste Management

The Reactor Waste Management (RWM) group was formed in June 1991 to respond to the growing difficulties of properly handling waste in K-Reactor (operating) and L and P-Reactors (standby). The group originally consisted of one manager, three supervisors and about seven non-exempt operators. Their charter was to coordinate the packaging and shipment of the vast quantity of operational waste being generated in the K, L, and P-Reactors. At the start of the deactivation/transition, the group had grown to one manager, 10 professionals/supervisors, and 24 non-exempt operators. Their revised charter included coordinating the package and shipment of all reactor facility transition waste in accordance with new and more stringent disposal facility requirements, all environmental field operations (sampling, closure of USTs, etc.), the design and operation of a state-of-art decontamination facility, and completion of transitional and decontamination projects.

Because of the dedicated waste handling group within the Reactor Division, waste minimization has not only enjoyed increased success, but the average education level of the Division regarding waste minimization has greatly increased. The latter effect was accomplished by formal classroom training for all members of the Division, but probably more importantly through the availability of expert help from the RWM group.

Work Package Review

Each work package developed in the Reactor Division is reviewed specifically for waste minimization using a checklist. The checklist asks the following questions (abridged) concerning the generation of waste.

- Will the activity use a hazardous chemical?

Will the activity involve the disposal of waste that is constructed of or comes in contact with hazardous constituents?

Will the activity result in generating greater than 1 B-25 (90 ft3) of radioactive waste?

Will the activity involve a new waste stream or the modification of an existing waste stream?

If any of the above predetermined amounts of waste generation are triggered, the work package must be reviewed by the RWM group. During the RWM review, a specialist works with the actual workgroup on how to effectively minimize the waste being generated. In addition, the RWM group is alerted to any generation of hazardous/mixed waste so that substitutes can be identified or if generation is unavoidable, the proper collection areas can be set up. It is important to note that the work group, not just the RWM group, is actively participating in waste minimization planning.

ALARA Reviews

Many jobs conducted during the transition have required formal As Low As Reasonably Achievable (ALARA) reviews because of significant exposure or contamination levels. As a part of these reviews, waste minimization is formally discussed and planned. Experience shows that jobs with these higher exposure and contamination levels generate more waste than jobs with lesser exposure and contamination. Once again the work group is actively participating in waste minimization planning.

RECYCLING

Recycling is the pollution prevention aspect where most waste minimization activity has taken place in the RD. Within the reactor facilities there are large quantities of recyclable metals and reusable equipment and materials. The recyclable metals include stainless steel from process piping and spent fuel pool components, carbon steel equipment and piping, and copper cable and piping. The stainless steel represents the most cost effective candidate for immediate recycling, therefore, was pursued as a part of the SRS's beneficial reuse program.

Much of the equipment and material left in the radiological areas of the five reactor facilities have the potential for reuse or to become clean salvage if it were decontaminated. Therefore, the RWM group is pursuing the design and start-up of a Central Decontamination Facility not only for excess RD equipment and materials, but also for the site's excess equipment and materials. In addition to the Central Decontamination Facility, the RWM group met the challenge of consolidating the excess chemicals of all five reactor facilities and centralizing them into a single computerized storage facility. The chemicals are then advertised for reuse throughout SRS.

Each one of the above recycling activities (stainless steel, Central Decon Facility, excess chemicals) are discussed in more detail below.

Stainless Steel Recycling

Within the reactor facilities, the process piping is constructed of 304 stainless steel. Also, several smaller items such as slug buckets within the spent fuel pools, filter frames within building filter compartments are also made of 304 stainless steel. SRS is pursuing the cost effective recycling of these stainless steel materials. The general idea is to deliver contaminated stainless steel to subcontractor's melter for recycling into stainless steel burial boxes and drums. As of today, three subcontractor's melter have been awarded a contract to demonstrate their capabilities on approximately 60 tons of contaminated stainless steel. Stainless steel has been collected in L-Reactor (process piping), and in R-Reactor (slug buckets). The L-Reactor piping was cut up into pieces suitable for feeding into the contractor's melter using plasma-arc torches. This process was costly as well as time consuming and served as additional emphasis to build the Central Decontamination Facility which will include a containment for the purpose of size reducing material. The Central Decontamination Facility is discussed in the next section of the paper. Each piece of process piping was bagged/tagged and loaded into B-25 containers controlled by the RWM Waste Inventory and Management Program (WIMP). The WIMP is a computerized program utilizing barcode technology to positively track the contents and disposition of every container of waste/material in the RD. The WIMP is discussed in detail in the "Disposal" section of this paper. The full B-25s were sealed and placed into "sea-train" type containers specifically procured to meet transportation requirements to the subcontractor's melter. The R-Reactor slug buckets turned out to be a perfect candidate for stainless steel recycling. They

required no size reduction and the buckets selected for initial recycling were contaminated to the point that decontamination for clean recycling was questionable. The slug buckets were directly loaded into the "sea-train" containers. At the writing of this paper several loads of stainless steel were being readied for shipment to the subcontractor's melter.

Central Decontamination Facility

The objective of the Central Decon Facility (CDF) is to operate a state-of-the-art decontamination facility using portable equipment in a building modified to handle a variety of changing decontamination equipment and techniques. The facility will offer a central location to stage salvageable contaminated materials, perform decon, and release them for uncontrolled use. Additionally, by utilizing portable to semi-portable equipment, the separate units can be moved to support local large scale decon work such as that resulting from the decontamination and decommissioning of a SRS facility.

The CDF is located in C-Reactor (see Fig. 1) and the Central Shops area of SRS. Within the C-Reactor building, there is a staging and sorting area, a large decon/size reduction hut and the decon area itself.

Fig. 1.

The staging and sorting area is located in the assembly area of the C-Reactor. It offers a large square footage area to stage contaminated materials for decon and also for deconned/clean materials awaiting salvage or reuse. A large "permacon" hut will be erected in the stack area for large item decon or size reduction/packaging of items deemed not cost effective to decon. The actual decon booths/units are located in the drum storage area and crane maintenance area of the C-Reactor building. Within this area a service ring header will be installed consisting of a ventilation exhaust trunk, 480V/120V power, service and breathing air, and radioactive liquid drain lines. The ring header will service portable decontamination booths/systems located around the service ring header. The concept of the service ring header allows for quick addition of new and effective decontamination equipment and techniques.

Also located within the C-Reactor complex is the CO2 blasting enclosure. CO2 blasting equipment is housed in a portable enclosure specifically designed to effectively contain all contaminates and particulate emissions in a fast moving HEPA exhaust stream. The CO2 pellets sublime into gas and therefore do not contribute to any secondary waste streams. The CO2 blast enclosure and equipment consists of the following items:

1. Two "sea-train" type containers modified to house a stainless steel lined blasting booth, a glovebox room, a counting/change room, a blast equipment room (CO2 pelletizer and air dryer), and a ventilation room. (See Fig. 2).
2. A portable (on wheels) CO2 tanker with refrigeration unit.
3. A portable (on wheels) air compressor.

Fig. 2.

The pelletizer, within the blast equipment room, produces solid carbon dioxide pellets from liquid CO2 from the portable tanker. These pellets are propelled using dry compressed air from the portable air compressor/air dryer and delivered to the blast booth of the enclosure. There they are trained onto the surface to be decontaminated. The pellets sublime causing a large local pressure change on the surface which removes contaminates, paint, and any other surface particulates. The contaminates are immediately entrained into a fast moving HEPA exhaust stream. All the entrained contaminates are then deposited onto a HEPA pre-filter or the HEPA filter itself. The CO2 enclosure is currently being procured with expected delivery in June 1995. Feed material for the CO2 enclosure include: 170 tons of contaminated lead collected from the removal of installed shielding in the reactor facilities, stainless steel slug buckets, carbon steel shielding, etc.

Located in Central Shops is the 728-N Vacuum Blasting Facility. This is a HEPA exhausted building used to perform decon via the LTC Vacuum Blaster. The LTC Vacuum Blaster is an all pneumatic operated grit blaster with a grit recycling capability to significantly reduce the amount of secondary waste generated. The LTC Vacuum Blaster is supported by a 750 cfm air compressor and a combination air cooler/air dryer. The vacuum blaster is a very aggressive technique suitable for work on "hard to decon" fixed contamination. The facility recently was put into operation. Feed material will mostly consist of contaminated tools and contaminated stainless steel requiring an aggressive means of decontamination. At the writing of this paper, the

Vacuum Blasting Facility has proven highly successful in the decontamination of tools and stainless steel slug buckets. Approximately 11.5 tons of stainless steel has been cost effectively released as clean salvageable 304 stainless steel.
To support decision making at the Central Decontamination Facility a computer database is being developed which will take information concerning a candidate item to be decontaminated, and using history developed through operation, make a decision if the candidate can be cost effectively decontaminated.
For each decontamination candidate the data base must either contain or calculate the following information.
Unique identification number (barcode)
Determine salvage metal price based on the material the item is made of and its weight
Determine the disposal cost avoided if the item is not handled as radioactive waste, but reused or recycled
Determine the cost to decontaminate the item
Determine if it is cost feasible to perform the decontamination
The data flow diagram (Fig. 3) describes the flow and general connection of data for the Decon vs. Disposal database. The flow of data starts with a candidate and the capture of pertinent information concerning the candidate. The following table describes the information needed on the candidate.
Fig. 3.
TABLE I
If the candidate is cost feasible to decontaminate, other information must be captured, including a storage location, cost data, and decontamination history. If the candidate is not cost feasible to decontaminate, then it is disposed of.
Several reports are necessary to be generated from the Decon vs. Disposal database. First, there is the Decon item file. This is a history report on each candidate that also serves as an input form. It is a convenient form to accumulate costs and other information while the item is being decontaminated.
In addition several other reports such as Decon Method Cost Reports, Time to Decon Reports, and Cost Benefit Reports are required on an as needed basis.
Excess Chemical Reuse
The large quantities of surplus chemicals collected over 40 years of operation in the five separate reactor facilities had to be properly handled. Many of these chemicals were hazardous and would have to be disposed of as hazardous or mixed waste which has high storage and disposal costs. The RWM group implemented a chemical excess program which has significantly reduced not only the hazardous and mixed waste generated, but also has saved other SRS Divisions new chemical purchase costs.
More than $200,000 in hard dollar waste disposal costs and new chemical purchase costs have been avoided since the start-up of the Reactor Division Excess Chemical Program. The program, which includes a building converted for storage of excess chemicals, barcoding computer database equipment for management, and use of the site all-in-one network as advertisement, was started-up for less than $1,000 by utilizing surplus equipment and buildings. Excess chemicals were collected in each reactor facility and centrally stored. Each chemical was barcoded and entered into a data base file which is periodically downloaded to the site-wide all-in-one network for reuse advertisement. Customers utilizing RD excess chemicals include the Department of Energy (DOE), all other SRS Divisions, and the SRS Environmental Protection Department. SRS is currently reviewing the program as a model for a new initiative to handle excess chemicals on a sitewide basis. In this way the lessons we have learned can benefit the site program.
TREATMENT
The RD is a generator of waste at SRS and does not necessarily treat waste. However, in one case, it has proved beneficial for the RD to take on the waste treatment role. Currently there is only one centralized waste compactor (253-H compactor) used for treatment of compactable waste at SRS. The 253-H compactor is not capable of receiving tritiated waste, which is generated in all five reactor facilities as well as at the heavy water processing facility. The tritiated compactable waste historically was disposed of in B-25s which were directly buried in the disposal facility without treatment in a compactor. Because of the draining of the process systems in the five reactor facilities, a large amount of tritiated compactable waste was generated. The RWM group responded to this by installing and starting-up a

small compactor in the L-Reactor for the express purpose of compacting this tritiated compactable waste stream. Installation into a reactor facility, which is permitted to release tritium, simplified the start-up requirements. Substantial waste minimization has been realized; however, the actual cost savings are minimal because of the health protection requirements to operate the compactor. In order to compact tritiated waste, operators must wear plastic suits with supplied air which drives costs up. Although the operation of the compactor has proved successful and cost effective, it failed to produce the promise of substantial cost savings.
A small compactor located in close proximity to a compactable waste stream can prove very cost effective if costs to operate are minimal (i.e., minimal HP requirements) and no other large treatment facility is available (economies of scale).

DISPOSAL

Two activities are worthy of mention in the area of disposal. Although disposal activities are not directly associated with waste minimization, the following activities prove that when faced with the final option, disposal, there are still ways to minimize the waste actually shipped for disposal. The two activities described below are the Waste Inventory and Management Program and the RWM Self Monitoring Program.

WASTE INVENTORY AND MANAGEMENT PROGRAM (WIMP)

The WIMP system is a relational database with integrated barcode technology to track and inventory radioactive, hazardous, and clean waste. WIMP is currently configured to track and inventory radioactive waste with future plans to track and inventory hazardous and clean waste. WIMP is made up of three major functions with several minor support functions. The three major functions are: Issue Container, Waste Tag Entry, and Return Container. Each function is explained below.

Issue container

The issue function is used to issue a B-25, B-12, 55 gal. drum, etc. to a responsible individual within a work group. The individual must have completed and passed a four hour training session with exam in order to take custody of a container. WIMP is loaded with the individuals allowed to take custody of containers and will not allow a container to be issued to non-authorized personnel. WIMP will generate an issue form which captures all the pertinent information about the individual and the job being worked. The container is issued with a unique barcoded ID number. If greater than 3 B-25s are issued to one individual or to one particular work package, WIMP will automatically request a waste minimization surveillance be conducted. The issue form is generated with the rules associated with use of a B-25 and requires the signature of the individual accepting custody. One of the rules is that only non-compactable waste can be placed in a B-25. Since the institution of the WIMP system, compactable waste generation has doubled while non-compactable waste has reduced leading to a significant minimization of waste.

Waste Tag Entry

As waste is generated and bagged, the individual sealing the bag fills out and attaches a Radioactive Waste Tag on the bag. The waste tag captures all the pertinent information concerning the bagged waste including: type of waste (non-compactable, compactable), contents, location generate, packager, waste stream number, ID number (barcode) of container waste is placed in, etc. The waste tag is signed by the packager ensuring no hazardous or other prohibited items are present in the waste. The bottom of the waste tag is separated and collected to be entered into the WIMP system. Each waste tag is entered into WIMP to form the contents, "waste cuts," of each B-25 container.

Return Container

After the B-25 container is full, custody is returned to the RWM group for storage and shipment. The individual returning the container must review a return container form, which lists the complete contents of the container, and signs that no hazardous or other prohibited items were placed in the container. Operators inspect the container for fullness and then seal and store the container. Periodic inventory is conducted using a portable barcode reader as well as setting up the shipment of containers.

Other minor functions include manifesting, sorting capability, report generations, and querying.

RWM Self Monitoring Program

The RWM Self Monitoring program allows trained RWM Operators to perform Health Protection (HP) coverage of most RWM jobs including monitoring for other personnel

and writing up official HP surveys. The program has been more successful than originally anticipated and is credited with not only increasing the productivity of the RWM group, but also alleviating some of the work load constantly facing HP. Waste minimization has also been increased by combining the waste minimization knowledge of the RWM Operator with the freedom gained from having HP knowledge and radiological control at your fingertips.
Currently our memorandum of agreement with HP concerning the RWM Self Monitoring program allows monitoring up to the following limits.

Transferrable Surface Contamination: 100,000dpm/100cm2 beta-gamma
or
1 x 106 dpm/0.1m2 beta-gamma
2,000 dpm/100cm2 alpha
Radiation: 5 mrem/hr
Airborne Tritium: 25 microcuries/hr (confirmed activity)

The memorandum of agreement lists the following low risk general activities where a HP trained RWM Self Monitor can perform the surveys necessary to conduct the activity.
- In-process Decontamination
- Waste Removal From the Contamination Areas
- Low Level Waste Trailer (LLWT) Activities (liquid waste removal)
- Sampling Known Drums, Tanks, etc.
- L-Area Compactor Operations

CONCLUSIONS
The effective management of radioactive waste, and especially the minimization of that waste, is largely contingent on the dedication of each worker within the organization. This idea holds true in many situations and is not particularly eye opening, but the way the RD has accomplished this dedication is unique and proved effective in the span of just one year.
The formation of the RWM group not only allowed waste issue to be centrally controlled, but provided every worker with the RD ready access to correct information concerning waste. No longer are workgroups "hiding" hard to handle materials or even disposing of the material because it is easier, or they lacked information on how to properly handle the material. The programs established by the RWM group (Excess Chemical Reuse, Central Decontamination Facility, and WIMP) provides an easy mechanism for the handling of excess chemicals, equipment, scrap material, and waste.
A new mentality has developed. The average worker actively looks for opportunities to reduce waste at its source, recycle metals, or minimize waste by segregation. Workers now plan for waste minimization during the creation of work packages and ALARA reviews. They understand what can and cannot go into waste containers and maximize their waste going to the compactors. Through the institution of the WIMP, ownership of waste containers is transferred to individuals who now take pride in effectively controlling the contents of their containers.
The formation of the RWM group has resulted in turning a whole division of personnel into active champions of waste minimization.

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POLLUTION PREVENTION AND WASTE MINIMIZATION IN ENVIRONMENTAL RESTORATION
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ABSTRACT
In recent years, increasing attention has been given to pollution prevention and waste minimization (PP/WMin) within the Federal government as a result of the Pollution Prevention Act of 1990 and Executive Order 12856, Federal Compliance With Right-to-Know Laws and Pollution Prevention Requirements. During the same time, one of the Department of Energy's (DOE) main missions has changed from weapons production to environmental restoration which includes decommissioning of facilities. Restoration activities, by their nature, will result in the removal of a large quantity of waste requiring treatment and disposal. It is anticipated that

waste resulting from restoration activities will become DOE's largest waste source. No Federal or State guidance exists to assist environmental restoration program and project managers in preventing or reducing waste resulting from environmental restoration activities. Therefore, DOE Office of Environmental Restoration has developed program guidance to incorporate PP/WMin principles into the restoration process. This paper discusses the program guidance for environmental restoration.

INTRODUCTION

DOE's primary mission has changed from weapons production activities to environmental restoration. Environmental restoration includes remediation and decommissioning (formerly referred to as decontamination and decommissioning). As a result of environmental restoration activities, DOE is producing a greater quantity of material requiring treatment and disposal. However, treatment capacity is not always available, and disposal is becoming more difficult and costly as a result of increasing regulatory requirements and limited disposal capacity. Therefore, there is a greater need to prevent or minimize waste generation. Pollution prevention and waste minimization (PP/WMin) activities reduce the demand for treatment and disposal capacity resulting in less regulatory involvement and reduced costs. It is important for PP/WMin principles to be incorporated in environmental restoration activities to ensure the greatest environmental and financial benefits.

Much of the waste to be processed by environmental restoration activities (e.g., contaminated soil, water, building materials) was generated as a result of past production activities. This waste is referred to as primary waste. In the course of performing restoration activities, new waste (e.g., drilling cuttings, solvents used in cleaning equipment or treating waste) is generated. This waste is referred to as secondary waste. PP/WMin can be applied to both types of waste. For primary waste, recycling and reuse are the most significant PP/WMin opportunities (particularly for decommissioning activities). For secondary waste, source reduction results in the greatest benefits. For both types of waste, segregation of hazardous from nonhazardous or radioactive from nonradioactive can result in substantial benefits.

The 1994 Waste Minimization/Pollution Prevention Crosscut Plan (1), DOE's principal PP/WMin program planning document, requires each site to develop and maintain site-wide and generator-specific PP/WMin programs. In addition, the Crosscut Plan identifies the key elements which sites must include in their PP/WMin programs. The purpose of this environmental restoration guidance document is to provide supplementary information for sites to utilize when developing PP/WMin programs for environmental restoration activities. This document provides:

- a summary of regulatory requirements for applying PP/WMin principles to waste generating activities;
- supplementary guidance to the Crosscut Plan addressing several key elements of an environmental restoration PP/WMin program;
- a summary of various phases of environmental restoration work and target PP/WMin opportunities for each activity; and
- a brief description of several PP/WMin tools which can be applied generically throughout the restoration process.

Many of the PP/WMin principles discussed in this guidance are engineering and administrative practices commonly applicable to any type of work. This guidance is intended to briefly describe these practices and provide examples of the application of these practices as they relate to environmental restoration activities. The document is not intended to provide detailed guidance for conducting PP/WMin activities. For broader, general guidance on site pollution prevention programs, sites should refer to the 1994 Waste Minimization/Pollution Prevention Crosscut Plan, or guidance prepared by the Environmental Protection Agency (EPA) or State regulators.

REQUIREMENTS

A variety of requirements exist for establishing and implementing a PP/WMin program. The most extensive requirements exist in the Pollution Prevention Act of 1990. These requirements, as well as other requirements applicable to DOE sites, are discussed below. It should be noted that no requirement excludes restoration activities from the application of PP/WMin.

Federal and State Laws and Regulations

All significant environmental laws and regulations have requirements for pollution prevention and/or waste minimization to be incorporated into compliance activities. The following is a summary of some of the more significant requirements:

Pollution Prevention Act (PPA) of 1990 - established a national policy that waste generators implement a program which emphasizes, in decreasing order of preference: a) source reduction, b) recycling, c) treatment, and d) disposal. The PPA also requires facilities which must report toxic releases under Section 313 of the Emergency Planning and Community Right-to-Know Act (EPCRA) of 1986 to identify and implement source reduction and recycling activities for those toxic chemicals reported. Pollution prevention is defined in the PPA as "source reduction." Source reduction includes: a) reducing the amount of hazardous substances, pollutants, or contaminants entering a waste stream or otherwise released into the environment; and b) reducing hazards to public health and the environment from the release of such substances, pollutants, or contaminants. Pollution prevention does not include most recycling or waste volume reduction (e.g., evaporation or other waste concentrating activities which do not reduce the quantity of hazardous substances, pollutants, or contaminants in a waste). These activities are typically considered waste minimization by EPA.

Resource Conservation and Recovery Act (RCRA) of 1976 - requires hazardous waste generators to have a program in place to reduce the volume and toxicity of hazardous waste and report PP/WMin efforts undertaken to EPA annually. RCRA amended the Solid Waste Disposal Act of 1965.

Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) of 1980 -includes as one of the nine criteria for evaluating the acceptability of an action the reduction in the toxicity, mobility, or volume of a waste through treatment. CERCLA was substantially amended and expanded by the Superfund Amendment and Reauthorization Act of 1986.

Requirements also exist in the Clean Water Act of 1972 and the Clean Air Act of 1963 (most recently amended in 1990), and state laws and regulations. In addition, states and EPA incorporate PP/WMin on a site-by-site basis into permits and compliance agreements.

Executive Order

While a variety of Executive Orders require pollution prevention, the most significant to restoration activities is Executive Order 12856, Federal Compliance With Right-to-Know Laws and Pollution Prevention Requirements (2). This Executive Order requires Federal agencies to comply with the reporting requirements of EPCRA and PPA. The Executive Order also requires Federal agencies to establish a goal of reducing the release or off-site transfer of toxic chemicals by 50% by the end of 1999 and requires sites to develop a pollution prevention plan.

DOE Orders

DOE has several Orders requiring PP/WMin to be incorporated into all Departmental activities. DOE Order 5400.1, General Environmental Protection, requires sites to establish programs for the minimization of all types of waste, including hazardous and mixed waste and to develop pollution prevention plans. DOE Order 5820.2A, Radioactive Waste Management, requires sites to develop waste reduction programs for radioactive and mixed wastes.

To assist sites in meeting these requirements, the 1994 Waste Minimization/Pollution Prevention Crosscut Plan, issued by Secretary O'Leary on February 25, 1994, provides guidance on the development of site-wide and generator-specific PP/WMin programs. In addition, a variety of other Federal and State guidance is available (e.g., EPA's Facility Pollution Prevention Guide (EPA/600/R-92/088) (3), Texas Natural Resources and Conservation Commission's Pollution Prevention Assessment Manual for Texas Businesses (4)).

PP/WMIN PROGRAM FOR ENVIRONMENTAL RESTORATION

As previously noted, each site should have site-wide and generator-specific PP/WMin programs as outlined in the 1994 Crosscut Plan. Seven program components, which are key to a successful environmental restoration PP/WMin program, are discussed below. Although the discussions are not prescriptive in nature they are useful to environmental restoration managers as a guide to developing a PP/WMin program for environmental restoration work.

Goals/Objectives

Establishing goals/objectives is key to ensuring that a PP/WMin program is successful, and where the goals are quantifiable, success is measurable. However, as noted in the 1994 Waste Minimization/Pollution Prevention Crosscut Plan, developing quantitative PP/WMin goals for environmental restoration activities is difficult because the nature of environmental restoration work, in most cases, is to remediate

as much contaminated media as necessary to protect human health and the environment. Therefore, unless specific activities lend themselves to quantitative goal setting, sites should, as a minimum, establish qualitative goals for eliminating or reducing waste during environmental restoration activities. A goal may be a simple statement of the site's intent to identify and implement activities which eliminate or reduce the generation of waste in all phases of environmental restoration work.

In addition to setting goals, sites should develop PP/WMin objectives. Objectives are specific actions which assist project managers and/or workers in understanding and meeting PP/WMin goals. For example, an objective for pollution prevention would be to require environmental restoration project managers to seek alternative non-hazardous chemicals as substitutes for chemicals which result in the generation of hazardous waste.

Resources/Budgets

To meet goals and objectives, adequate resources and funding must be available. The 1994 Waste Minimization/Pollution Prevention Crosscut Plan discusses the PP/WMin activities for which budgets should be established. The EM Budget Formulation and Activity Data Sheet (ADS) Development Field Guidance for the Fiscal Year 1997 Planning and Budget Cycle (5) describes how the funding should be identified in ADSs. Beginning with the FY 1997 budget cycle, EM-40 will meet the WMin/PP reporting requirements of Executive Order 12856 and Office of Management and Budget Circular A-106 (Federal Agency Pollution Abatement and Prevention Planning) by incorporating PP/WMin budget information in the section of the ADS designed for pollution abatement or prevention projects. These projects include all PP/WMin activities with the exception of pollution prevention opportunity assessments (see section below), or research, development, and demonstration projects.

PP/WMin Expertise/Training

Each DOE Operations Office has a WMin Coordinator. The Coordinator is responsible for developing and implementing the site PP/WMin program and providing expertise to all site organizations to develop generator-specific PP/WMin programs. Environmental restoration managers should seek assistance from the WMin Coordinator in establishing a restoration PP/WMin program and identifying opportunities to eliminate or reduce waste generation.

Each site should have a PP/WMin training program available to all site personnel. At most sites, this training will be geared to production processes and not restoration activities. The environmental restoration organization should work with the site WMin Coordinator to develop training specific to restoration activities. Training should target environmental restoration project managers, design engineers, and field workers.

Additional general training is available through the Kansas City Plant (KCP). KCP conducts training in Pollution Prevention Opportunity Assessments (see next section). KCP is developing a PPOA process for restoration activities (expected to be complete in 1995). Training is conducted periodically at KCP or arrangements can be made for training at another location.

Pollution Prevention Opportunity Assessments (PPOA)

A Pollution Prevention Opportunity Assessment (previously referred to as Process Waste Assessment) is an analysis of a process or activity to identify opportunities to eliminate or reduce the generation of waste (including air and water emissions) or consumption of raw materials, water, or energy. Once identified, the opportunities are evaluated and compared to determine the most efficient and cost effective option to implement.

A variety of PPOA processes exist. The Environmental Protection Agency has outlined a PPOA process in the Facility Pollution Prevention Guide (EPA/600/R-92/088) (3). Kansas City Plant has developed a separate process sponsored by DOE's Waste Minimization Division (EM-334). In addition, individual state regulatory agencies may provide guidance on PPOA processes. These methods emphasize source reduction as the primary activity and recycling/reuse as secondary activities. While the general concepts of these processes can be applied to restoration activities, it should be noted that the greatest PP/WMin benefits in environmental restoration activities will result from emphasizing recycling/reuse for primary waste and source reduction for secondary waste. Sites should choose a particular PPOA process and apply it consistently to all activities on site. This will result in consistent site-wide data on waste prevented and cost savings.

PPOAs should be included as a routine aspect of restoration activities, as

appropriate. The level of effort associated with a PPOA will be dependent on the environmental restoration project/activity to be implemented. For example, a PPOA for sampling operations will require a less detailed analysis than a PPOA for the implementation of a remedial alternative. Project managers should determine the depth of assessment appropriate for each activity. In addition, all PPOAs should be well documented (documentation is discussed later in this paper). In general, the following points should be considered relative to applying PPOAs to environmental restoration activities:

A PPOA should be applied to the entire site restoration program to determine common pollution prevention opportunities.

PPOAs should be applied as early in the planning process as possible and throughout each step of an individual restoration project including execution. For example, in the CERCLA remedial action process, PPOAs should be applied during the Preliminary Assessment/Site Inspection (PA/SI) when gathering historical data, before the Remedial Investigation when characterizing the site, before the Feasibility Study for evaluating alternatives, and before the Remedial Design and Remedial Action for designing and implementing the preferred remedial alternative. Likewise, PPOA's should be applied through all stages of a CERCLA removal.

Teams should be established for conducting PPOA's. The teams should include personnel with expertise in conducting PPOA's (e.g., site WMin Coordinator) and personnel with direct responsibility for, and knowledge of, the activity (e.g., design engineer, construction coordinator).

All media (e.g., air, water, soil) potentially impacted by the activity should be included in the PPOA.

An economic analysis of alternatives should be conducted. The analysis is essential to comparing the alternatives and justifying actions taken.

As a result of conducting a PPOA, it is possible that a process which produces a large quantity of waste, or a waste that is difficult to treat, is identified as a pollution prevention opportunity, although no substitute process is readily available. In these instances, information on the process in need of technology change should be identified to EM's Office of Technology Development (EM-50). EM-50 collects and prioritizes the pollution prevention technology development needs identified throughout the EM organization and targets its pollution prevention technology development efforts in the areas of identified needs.

Prioritization of Opportunities

Since funding and resources are limited, sites must develop a method to prioritize PP/WMin activities. A variety of systems exist and are usually included as part of a PPOA process. However, it is essential that a site choose a method and consistently apply that method to all activities. Factors usually considered in prioritizing activities are listed below. All factors are tied to waste reduction or prevention.

Regulatory drivers: Regulatory drivers such as Land Disposal Restrictions (LDR), Emergency Planning and Community Right-to-Know Act (EPCRA) Section 313(c) toxic chemical reduction requirements, or compliance agreement obligations are all relevant to restoration activities.

Financial considerations: Sites should analyze the lifecycle costs associated with PP/WMin projects and utilize the analysis to compare different projects for prioritization.

Risk: Utilizing such factors as permissible exposure limit or threshold limit value as a ranking of risk posed to worker or public safety and health could be considered for prioritization.

Documentation/Information Exchange

It is important that all PP/WMin activities (including PPOAs) be documented and the information made available to other DOE sites. Documentation should include information such as a brief description of the PP/WMin activity and its applicability to environmental restoration, approximate amount of pollution avoided or the amount of waste minimized, any obstacles overcome to implement the activity, cost information, if available, and a site contact. Documentation should be consistent across all projects on a site. The information for PPOAs should be recorded in appropriate documents (e.g., Remedial Investigation/Feasibility Study (RI/FS) work plan, Record of Decision (ROD), etc.) and should be combined for a site-wide report of all PP/WMin activities performed during a year.

To ensure that information is available throughout the DOE complex, documentation should be provided to DOE's Pollution Prevention Information Clearinghouse (EPIC).

The system provides a single source of DOE PP/WMin activities and provides access to EPA's Pollution Prevention Information Exchange System (PIES). EPIC and PIES provide technical information on PP/WMin and information on PP/WMin conferences, workshops, etc.

Identification of PP/WMin Issues

In the process of conducting environmental restoration activities, sites will identify issues which prevent PP/WMin. This may include DOE policies which inhibit activities that may result in PP/WMin or situations where adequate technologies have not been developed to treat a restoration activity waste. These issues, upon being identified, should be discussed with the site WMin Coordinator. For DOE-wide policy issues, the site PP/WMin coordinator will interact with the Waste Minimization Division (EM-334). EM-334 will address this issue through appropriate channels (e.g., the Waste Reduction Steering Committee or the Pollution Prevention Executive Board). For issues which are limited to EM-40 activities only, sites may wish to discuss the issue with the appropriate EM-40 program manager. The program manager will interact with the EM-40 Pollution Prevention Core Group for resolution. For technology issues, the site WMin Coordinator will interact with the Site Technology Coordination Group (STCG). It is the function of the STCG to identify programmatic needs to the Office of Technology Development (EM-50).

ENVIRONMENTAL RESTORATION ACTIVITIES

The basic steps in environmental restoration are similar whether the activity is remedial or decommissioning. The following is a brief description of the stages of environmental restoration, and some PP/WMin opportunities associated with each stage.

Negotiations

In most cases, sites will meet with regulators to discuss the requirements for conducting restoration activities prior to beginning the activity and periodically during the activity. EPA has established a policy to include PP/WMin in all Agency activities including restoration. Therefore, site representatives should take advantage of negotiations to incorporate PP/WMin in initiatives related to restoration activities. Example PP/WMin opportunities include obtaining agreement that historical data is acceptable to limit sampling, thereby limiting sampling waste; establishing future land use as other than the most conservative assumption of residential; testing innovative technologies which will result in less waste; and limiting monitoring requirements after closure. Though these are difficult measures on which to obtain regulator agreement, several successes have been realized within the DOE complex.

Interim Action

These are actions taken to ensure that there is no imminent threat of hazards to the environment, public, or workers. Interim actions include many or all of the following stages (i.e., preliminary assessment, characterization, evaluation of cleanup alternatives, etc). However, in an interim action, the time and effort spent on each individual stage is less than for a full remediation or decommissioning.

Preliminary Assessment and Inspection

This is the process of beginning to identify potential areas of concern by reviewing historical information and inspecting the site. Sites should collect as much historical information as possible to preclude extensive sampling in later stages and should use this pre-planning stage to begin identification of PP/WMin opportunities.

Characterization

Based on information obtained in the preliminary assessment and inspection, sampling is conducted to determine the extent of hazardous or radioactive contamination. PP/WMin opportunities should be identified and incorporated in the workplan and any appendices. For example, sites should target the large waste producing activities associated with characterization such as borehole drilling and sampling.

Under CERCLA, the process of adequately determining the type and extent of contamination at a site and identifying and evaluating cleanup alternatives is an iterative process. To make this process more efficient, and thereby reduce the cost and potential for waste generation, DOE and EPA have developed more efficient approaches to the RI/FS process. DOE has developed the Streamlined Approach for Environmental Restoration (SAFER), and EPA has developed the Superfund Accelerated Cleanup Model (SACM). Sites should consider applying a streamlining approach to the RI/FS process to facilitate PP/WMin.

Evaluation of Cleanup Alternatives
Alternative cleanup options are identified and screened. This process requires PP/WMin to be incorporated in the selection of an alternative since the quantity of waste generated is a factor considered in the evaluation of alternatives. In addition, project managers should ensure that PP/WMin principles are incorporated in treatability studies.
Documentation of Preferred Alternative
For each restoration process, a document is produced which formally discusses the actions to be taken to complete the remedial process or decommissioning. While no PP/WMin actions are taken during this stage of the restoration process, the documentation of the preferred alternative should discuss PP/WMin actions taken and considerations made in this area during the process of selecting an alternative.
Design of Cleanup Action
Engineering and planning are conducted to implement the approved alternative. A PPOA of all activities in the cleanup process should be conducted to identify all PP/WMin opportunities. This is an important stage to identify all opportunities, since PP/WMin activities can be included in workplans and contracts. PP/WMin concepts should be applied to the design of structures or equipment to be used in the restoration activity to reduce or eliminate waste generation when the structures or equipment are decommissioned at the end of the restoration activity.
Implementation of Cleanup Action
The approved alternative is executed. PP/WMin opportunities identified in the design stage should be implemented. In addition, cleanup activities should be reviewed periodically to identify additional PP/WMin opportunities not identified in the design stage or opportunities which may be appropriate as a result of changes which may occur in the implementation of the cleanup action. Once identified, these new PP/WMin activities should be incorporated into the cleanup activity. Sites should ensure that contractor/subcontractor personnel have been trained properly in PP/WMin principles.
Closure
Upon completion of the cleanup action, final administrative steps are taken to ensure that the action has been completed successfully and documented fully. Depending upon the action taken, some ongoing activities may be necessary (e.g., continued sampling or maintenance). PP/WMin opportunities identified for sampling or maintenance during the characterization stage can be implemented during the closure stage.
POLLUTION PREVENTION/WASTE MINIMIZATION TOOLS
There are a variety of standard PP/WMin activities which can be applied throughout the environmental restoration process. Several of these activities are discussed below. It should be noted that, in some cases, there is no difference between applying these concepts to production processes or restoration activities (e.g., the operation of a treatment technology).
Operating Practices
Changes to operating practices may yield a significant reduction or elimination of waste. In addition, these changes are usually very easy and inexpensive to implement.
Housekeeping: Work areas should be kept clean and equipment properly maintained to reduce the chance of breakage or leaking. Spill response plans should consider cleanup methods which reduce the generation of spill cleanup waste. Equipment should receive regular preventive maintenance to ensure efficient operation.
Material segregation: All materials should be handled or stored to prevent commingling. Commingling of materials can result in larger quantities of waste being produced. In addition, recycling may be reduced or precluded if contaminants are not segregated from recyclable materials, since recycling value decreases with decreasing purity. Materials which should be segregated include: hazardous and nonhazardous wastes, radioactive and nonradioactive wastes, liquids and solids, chemically incompatible substances, and recyclable materials and nonrecyclable materials.
Administrative changes: In some cases, administrative criteria result in the increase in waste generation. For example, the location of boundary lines established for radiologically controlled areas may result in unnecessary classification of materials as radiologically controlled waste.
Process Changes

Process changes can include changes to equipment or changes to materials used in the process. More efficient equipment may result in less use of resources or less waste generation. Substituting non-hazardous materials for hazardous inputs (e.g., solvent substitution) will result in the reduction or elimination of hazardous waste. Utilizing nonhazardous or less hazardous materials also results in reducing the potential for worker exposure.

Recovery/Recycling/Reuse

Although these activities are not considered pollution prevention, they will result in minimizing waste. This activity will probably result in the greatest benefits for decommissioning activities. Recycling building materials such as scrap metal, timber, and concrete can result in significant waste reduction and cost savings. Recycling is easier if the materials are not radiologically contaminated. However, if there is surface contamination, sites should use the release criteria provided in DOE Order 5400.5, Radiation Protection of the Public and the Environment, and Nuclear Regulatory Commission Regulatory Guide 1.86 (6) to facilitate recycling. Volumetric release criteria are not available currently. In addition to building materials from decommissioning, consideration should be given to recycling, reusing, or reclaiming unspent cleaning solvents or lubricants used in all activities.

CONCLUSIONS

By incorporating the concepts of pollution prevention and waste minimization into environmental restoration activities, DOE can significantly reduce the amount of waste which must be treated and disposed. This will reduce risks to the environment, workers, and the public; lower waste management costs; and assist in maintaining compliance. Sites should develop environmental restoration PP/WMin programs as defined in the 1994 Waste Minimization/Pollution Prevention Crosscut Plan. The programs should include qualitative and quantitative (where possible) reduction goals and ensure that adequate resources are available to meet these goals. PPOAs should be applied in all stages of environmental restoration activities. Budget information for environmental restoration PP/WMin activities should be trackable and retrievable for inclusion in the A-106 reporting process. Program offices requiring assistance in establishing an environmental restoration PP/WMin program should contact the site WMin Coordinator or the program manager at DOE Headquarters.

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IMPLEMENTATION OF WASTE MINIMIZATION AT A COMPLEX R&D SITE

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ABSTRACT

Under the 1994 Waste Minimization/Pollution Prevention Crosscut Plan, the Department of Energy (DOE) has set a goal of 50% reduction in waste at its facilities by the

end of 1999. Each DOE site is required to set site-specific goals to reduce generation of all types of waste including hazardous, radioactive, and mixed. To meet these goals, Argonne National Laboratory (ANL), Argonne, IL, has developed and implemented a comprehensive Pollution Prevention/Waste Minimization (PP/WMin) Program.

The facilities and activities at the site vary from research into basic sciences and research into nuclear fuel cycle to high energy physics and decontamination and decommissioning projects. As a multidisciplinary R&D facility and a multiactivity site, ANL generates waste streams that are varied, in physical form as well as in chemical constituents. This in turn presents a significant challenge to put a cohesive site-wide PP/WMin Program into action.

In this paper, we will describe ANL's key activities and waste streams, the regulatory drivers for waste minimization, and the DOE goals in this area, and we will discuss ANL's strategy for waste minimization and it's implementation across the site.

INTRODUCTION

Pollution Prevention and Waste Minimization implementation at a large research and development facility can be a difficult process. The key steps to implementation are pollution prevention awareness and training, knowledge of waste streams and practices, and a judicious approach to committing resources. Cultural change and buy-in on the part of the user community are critical steps in program implementation. Argonne has developed an approach that has begun with employee awareness and training and source reduction and recycling. A Waste Minimization Advisory Committee selected from a cross section of internal stakeholders has also been formed. The next step in the process is the development of interlinks between ANL's data management systems to identify areas of concern. All of these steps have demonstrated success in their initial phase.

ARGONNE R&D AND WASTE GENERATION

Argonne National Laboratory (ANL) is a multiprogram national laboratory with sites near Chicago, Illinois, (ANL-East) and Idaho Falls, Idaho, (ANL-West). ANL was established by the Atomic Energy Act of 1946 as the successor to the Metallurgical Laboratory of the Manhattan Project. Its initial mission was the development of fission reactor technology for power generation and the supporting science and technology; Argonne conceived most of the reactor systems in use today. The Laboratory's focus expanded to include high-energy physics research in the mid-1960s and environmental R&D in the 1970s.

ANL's major national research facilities include: Advanced Photon Source, Structural Biology Center, Intense Pulsed Neutron Source, Argonne Tandem Beam Linear Accelerator System (ATLAS), and High-Voltage Microscope/Tandem Accelerator. In addition, there are a host of other facilities such as the Fuel Cycle Facility, Hot Fuel Examination Facility, Transient Reactor Test Facility, and the National Battery Test Laboratory. The Integral Fast Reactor (IFR) Program, including the Experimental Breeder Reactor-II, are based at Argonne West (ANL-W) near Idaho Falls.

ANL's areas of supporting and basic research include: materials science, chemical sciences, computing and information sciences, mechanistic biology, environmental research, nuclear physics, and high energy physics. Technology development activities are focused in the areas of energy and industrial technologies and advanced reactors. Even though in January 1994 the DOE announced the termination of the IFR program, the DOE is proposing to redirect Argonne's advanced technology capabilities to the areas of nonproliferation, spent nuclear fuel and waste treatment, reactor and fuel cycle safety, and decontamination and decommissioning programs. The Laboratory's Institutional Plan (FY1995-FY2000) recognizes the major mission areas to be energy and environmental technologies, national research facilities, basic research, industrial and manufacturing technologies, technical evaluation, and education. ANL's core competencies include fission and other advanced energy systems; accelerator-based research, facilities, and technologies; industrial and transportation technologies; modeling, simulation, and advanced computing, environmental R&D, partnership R&D with industries and universities; education and training. ANL has a total staff of 5,210 over 1,100 of which represent the scientific and engineering staff. The Laboratory budget for FY94 was in excess of 450 million dollars.

The diversity of mission activities, the facilities, the R&D programs, and D&D projects generates a wide variety of radioactive and hazardous waste streams. Table

I summarizes the recent ANL waste generation data (in this paper, we focus on ANL-E, referred to as ANL, unless otherwise noted, and the data presented are for ANL-E, even though a similar waste minimization program is in place at ANL-W). The R&D waste generation accounts for approximately 40-60% of the low level, hazardous, and special waste streams. This waste is generated by many different projects and the quantities may vary from negligible to liters or drums per year. Approximately 90% of the radioactive waste generated at ANL
can be classified as low level. Hazardous wastes represent the largest segment of the waste inventory. In Table I, the increases from 1992 to 1993 reflect increased project activities particularly in the D&D area and an increased knowledge of the waste in the legacy waste area. The 1994 data are currently being compiled and are likely to reflect mild increases in the two areas mentioned above.

REGULATORY DRIVERS AND DOE DIRECTIVES

Waste Minimization and Pollution Prevention are driven by a complex set of regulatory drivers, in ANL's case, administered by the EPA, state laws, disposal site criteria, Presidential Executive Orders, and the DOE Orders. The list of regulatory drivers is summarized in Table II.

By far, the PPA and the FFCA had the most impact on DOE facilities including ANL. The FFCA waived the sovereign immunity for federal facilities, and it also amends the RCRA by requiring DOE to prepare facility plans that provide for the development of treatment capacities for mixed wastes. The PPA establishes a hierarchy of pollution prevention activities that must be followed.

The Department of Energy (DOE) is committed to waste minimization and pollution prevention at its facilities. The first Waste Minimization/Pollution Prevention Crosscut Plan, published in 1992, introduced waste minimization strategies and key objectives. Under its 1994 Waste Minimization/Pollution Prevention Crosscut Plan, the DOE has set a goal of 50% reduction in waste by the end of 1999, and each DOE site is required to set site-specific goals to reduce generation of all types of waste including hazardous, radioactive, and mixed. To meet these goals, ANL has developed a comprehensive PP/WMin Program Plan that is applicable to all activities and all waste generators at the site.

WASTE MINIMIZATION PLANNING AND IMPLEMENTATION

In compliance with the Pollution Prevention Act hierarchy, waste source reduction is a main element of waste minimization strategy at Argonne. The pollution prevention hierarchy can be briefly summarized as the following order: source reduction, recycling, waste treatment, and disposal.

A PP/WMin Program Plan and a PP/WMin Awareness Plan have been prepared. The purpose of the two ANL plans is to attain for this site the goals of the DOE's 1994 Crosscut Plan and to ensure compliance with the federal and DOE requirements discussed earlier. The strategies outlined in these plans are now being implemented.

To meet the challenge of PP/WMin at Argonne, several major initiatives have been taken:

1. Implementation of source reduction and recycling practices.
2. Employee awareness and training programs.
3. Tracking of waste through Waste Management Data System (WMDS)
4. Implementation of a Chemical Management Systems (CMS).

The program is being expanded to include: 1) linking the CMS with the WMDS to provide information and reporting capability, 2) development of training videos, and 3) implementation of site-wide waste reduction goals.

ANL has also formed a Waste Minimization Advisory Committee that includes representatives of the research and development areas, operations, and compliance. It also includes representatives of DOE's Argonne Area Office as well as the DOE Chicago Operations Office. A representative of the New Brunswick Laboratory, which is also located at the ANL site, is also on the committee. The purpose of the committee is to review pollution prevention program implementation, waste generation practices, and provide direction and integration of the program into the laboratory culture.

Waste source reduction and recycling have become the principal elements of waste minimization strategy at ANL. The Laboratory has committed to specific waste reduction goals of 50 % for Toxic Release Inventory (TRI) wastes and 25% for other waste streams over a five-year period. The baseline for these reductions is the 1993. Concerted efforts are being made for source reduction through materials exchange and recycle of excess materials such as metals. Initiatives in this area

include the use of nonhazardous scintillation fluors, alternative solvents, and chemicals. A Chemical Exchange Program is also in effect at ANL to exchange chemicals not being used by generators. Technical expertise is provided to waste generators and several technologies are being evaluated to further reduce the volumes of waste generated.

Separation technologies developed by ANL researchers have allowed the extraction of radionuclides such as Pu, Am, Np, and U out of certain waste solutions. For stored and operational liquid waste from site-wide activities, another ANL-developed ion-exchange process can allow the separation of actinides.

ANL's waste minimization strategy features awareness and training of the employees as a first step. The waste generators are also being trained on proper techniques for readying waste for pick up by waste handling specialists. A bar code computer system allows waste management personnel to retrieve information about individual waste containers from a central data base with a hand-held, battery-operated bar code reader. This information determines the waste handling requirements. After employee awareness and training, the next important step is the general waste stream analysis and review and identification of priority areas. The final step is to work directly with D&D and environmental restoration projects to identify opportunities at the outset. The D&D and restoration projects are the single largest generator of hazardous and radioactive waste.

The ANL Awareness Program began in 1993 with a poster program similar to the safety posters presented in many locations. The program started with developing a character that people could identify with the recycling program, individuals could suggest a name for the character, and receive a recycling container for their work place. They voluntarily emptied this into common collection bins. As an immediate result of this program, white paper recycling increased by 62% (over 1993), which represents approximately 40% of the available paper on site. In 1994, ANL recycled 190 tons of white paper, which as standard sheets laid end to end could stretch 6,500 miles. This program achieved a plateau of recycling that was limited by the number of people willing to bring their paper to a central receptacle. This hurdle is being overcome by instigating a custodial collection once per week in lieu of garbage collection. This is being tried in several buildings to identify problems before going laboratory wide. The principal problems identified so far include inadequate collection equipment and the need for a container that can be transported up and down stairways. One building generates so much general trash that a trash pick-up cannot be avoided. The problem is being addressed by conducting a second sort of facility waste to identify particular problems. The awareness program is being expanded to feature metal recycling and sharing of materials in the near future.

Training efforts are starting with a simple video to be used in all employee training sessions. The theme will be to identify the problem and focus on simple solutions that individuals can use every day to impact pollution prevention. Topics will feature housekeeping, documentation, control of quantities, and sharing of materials. Specific training of waste generators is also planned.

Awareness and training will only take the program to a certain stage. The next major hurdle is to identify specific waste streams for review and action. The collection of data is the first major step. With over 1100 technical staff generating very small to large quantities of waste, the problem is in identifying priority targets. For instance, methylene chloride may be a very high priority pollutant, but if it is generated in very small quantities, scarce review resources may be dedicated elsewhere.

ANL currently has three data bases that are relevant to the pollution prevention effort. The procurement system processes purchase requests and is currently based on the military code system. It is relatively easy to review purchase orders for cost or quantity but not for specific material or user. Fortunately, this system is currently being phased out. The new system is ORACLE-based and will have capabilities more friendly to the pollution prevention effort. The second data base is the Chemical Management System (CMS). This system was created in 1992 to serve as a repository for material safety data sheets (MSDSs)and inventory control. It is ORACLE-based and also contains all of the regulatory tables plus a capability to identify surplus material for exchange. The final data base is the Waste Management Data System (WMDS). This system is currently CLIPPER-Based and manipulates all of the waste data from the time of documentation. The present data system interlinks are shown in Fig. 1.

The initial steps in the data management process occurred in FY93 when data were developed for the DOE Annual Waste Reduction report and the (TRI) Report. This process was only partially effective because the first data only reflected waste as shipped. A second review of data was necessary to identify specific sources and quantities of TRI chemicals. This effort brought several problems to the forefront: 1) the data did not necessarily reflect material entering the system; 2) data quality was unreliable; for instance some data did not reflect concentrations but only the absolute container size; 3) access to the data was extremely limited, and 4) the waste management data base did not directly access either the regulatory lists or the MSDSs contained in the CMS data base; and 5) units of measure were not always consistent.

The need to link pollution prevention to WMDS was immediately apparent. The limited data available from the 1993 TRI report indicated that the largest H_2SO_4 generator contributed 1000 L of waste. A review with the generator indicated that half of this was dilute rinse water that could be neutralized.

Fig. 1.

The planned data linkages will provide for real-time data availability to the pollution prevention staff. Because waste data can provide a powerful driver for managers to incorporate pollution prevention in their programs, the second step in the data management process will be to generate two reports for laboratory managers to use. The first will be a simple report summarizing individual division waste generation. This report will be provided to division management. The second will be a detailed report of specific waste generation practices and will be available electronically to the researchers. Linking of the data bases is currently underway. The user community, requests chemicals through the procurement chain. The original systems had the user independently ordering materials. With the establishment of the CMS, the first step was for an individual to prepare a purchase requisition and a data sheet for chemical management and inventory control. Procurement is developing new software for its system and as a part of that effort, the link between the user, purchaser and the inventory control by chemical management will be electronic and automatic. The next step in this process will be to prompt a user to determine if they can use materials from inventory and provide them with the link to inventory. A user that cannot use inventory will not be required to do so. Finally, quantity approval limits will be applied to limit excess purchase.

The final links will be between the waste management system, chemical management system, and the pollution prevention group. The final data management configuration will resemble Fig. 2.

Fig. 2.

The advantages offered by the linked system are many:

Users will have access to CMS including MSDS and regulatory information at the outset of a project or experiment.

Procurement and inventory information will be fed to CMS and subsequently to pollution prevention and waste management in a real time frame resulting in knowledge of future activities.

Users will be able to generate waste requisitions electronically and transmit the information to waste management resulting in a more detailed knowledge of waste and its timely disposal.

This planned linkage will provide use of regulatory tables, inventory trends, procurement and waste documentation. This information will be used by pollution prevention group to determine target waste generators, waste generation trends, and documentation of successful source reduction efforts. The current schedule calls for the initial links between waste management and pollution prevention to be completed the spring of 1995. The final links will, however, be determined by the budget availability
for these activities.

CONCLUSION

Argonne has implemented a comprehensive waste minimization program with significant initial success. We have found that institutional buy-in at a complex R&D site, such as Argonne, is essential to implementation of a consistent waste minimization effort. Awareness and training are also essential components and are in fact responsible for source reduction in major part but they have limitations that are impacted by buy-in and commitment on the part of the user community as well as, both the physical and financial commitment. Awareness and training have been effective at

identifying and capturing the "low hanging fruit" but have limitations at prioritizing waste streams. A detailed knowledge of waste generating practices is essential at a large research facility because of the varied nature of the materials. Overall, it is anticipated that Argonne will be able to meet its waste minimization goals through a multi-faceted but consistent approach applied to all facilities and activities at the site.

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WASTE MINIMIZATION APPLICATIONS AT A REMEDIATION SITE

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ABSTRACT

Applying waste minimization principles at a remediation site is a challenge for many companies. We are required to practice waste minimization and pollution prevention (WM/PP), but how is it achieved at a site where waste generation is thought to be a key product?

The real challenge is to shift the waste minimization paradigm from focusing efforts on source reduction to focusing on recycle/reuse by inverting the EPA waste management hierarchy. A fundamental difference at remediation sites is that source reduction has limited applicability to legacy wastes but can be applied successfully on secondary waste generation. The bulk of measurable waste reduction will be achieved by the recycle/reuse of primary wastes and by segregation and decontamination of other wastestreams. Each effort must be carefully measured in terms of being economically and ecologically beneficial.

Structuring a successful, dynamic WM/PP program produces a mechanism to set the paradigm shift in motion. The program must be comprehensive, integrated, and proactive in order to capture opportunities early in the CERCLA/RCRA process.

The structure of a successful WM/PP Program will be illustrated by a case study of a Decontamination and Decommissioning (D&D) project (Plant 7) which was completed at the Fernald Environmental Management Project during the summer of 1994.

Key elements of the program were successfully integrated into the remediation project by including waste minimization personnel in all phases of the project to coordinate, promote, identify, plan, and implement in the field all waste minimization opportunities. Design, Engineer, Construction (DEC) Teams offered invaluable access to all aspects of the remediation project.

Source reduction opportunities were identified for secondary wastes generated from decontamination activities, radiological area access control, waste packaging activities, and procurement of materials/chemicals. Implementation of these actions dramatically reduced the amount of wastes generated during the D & D activities.

Recycle/reuse opportunities were identified for the majority of primary wastes including structural steel, lead flashing, transite, and concrete. The decision to recycle and based on the life-cycle cost of the waste including disposal costs and on the decision to be responsible, ecologically, for waste management.

Segregation techniques were essential for dispositioning wastes as sanitary, low-level, mixed, or recyclable and greatly reduced costs for waste disposal. Segregation also allowed for the responsible management of wastes. Volume reduction in conjunction with segregation techniques accomplished a further reduction in disposal costs.

Management personnel of the Plant 7 D & D project effectively utilized the shifting focus from source reduction to recycle/reuse to accomplish a measurable reduction in primary wastes shipped for disposal and to reduce the amount of secondary wastes which were generated during the project.

The success of this benchmark effort communicates to other facilities gearing-up for remediation that waste minimization can be practiced at a remediation site and can be accomplished in an ecological and economical fashion with stakeholder acceptance.

INTRODUCTION

The Fernald Environmental Management Project (FEMP) is owned by the Department of Energy and was used for the processing of uranium. The facility is a 1,050 acre site in southwestern Ohio. In 1989 Fernald suspended production of uranium metals and was

placed on the National Priorities List (NPL). Since production at Fernald formally ceased in 1991, the site's mission has changed from one of production to environmental restoration. A Remedial Investigation and Feasibility Study (RI/FS) is being conducted, along with other response actions under the 1990 CERCLA Consent Agreement with the EPA. This change in the mission led to many changes in Management's operating philosophy. Many groups which were necessary for producing a product were deemed irrelevant for remediation work, including Waste Minimization. At first sight, Waste Minimization does not readily appear to be applicable to remediation work. Environmental remediation is designed to correct adverse impacts to the environment from past operations and generates significant amounts of waste requiring management. The premise of pollution prevention is to avoid waste generation, thus remediation is in direct conflict with this premise. Although greater amounts of waste will be generated during environmental remediation, treatment capacities are not always available and disposal is becoming more difficult and costly. This creates the need for pollution prevention and waste minimization.

Applying waste minimization principles at a remediation site is an enormous challenge. If the remediation site is also radiologically contaminated it is even a bigger challenge. Innovative techniques and ideas must be utilized to achieve reductions in the amount of waste that must be managed or dispositioned.

One concept utilized at Fernald was to shift the waste minimization paradigm from focusing efforts on source reduction to focusing efforts on recycle/ reuse by inverting the EPA waste management hierarchy. A fundamental difference at DOE remediation sites is that source reduction has limited applicability to legacy wastes but can be applied successfully on secondary waste generation. The bulk of measurable waste reduction will be achieved by the recycle/reuse of primary wastes and by segregation and decontamination of secondary wastestreams. Each effort must be measured in terms of being economically and ecologically beneficial.

In Operable Unit 3, which consists of all the facilities and structures on the site, nearly 87% of the material to be dispositioned can be recycled. This includes heavy and light gauge metal such as equipment, lead flashing, ductwork, steel structures etc. The remaining waste will be disposed at either a designated DOE disposal facility, commercial disposal facility or in an on-site disposal cell. The exact amount to be disposed is driven by effective project management.

During the decontamination and decommissioning (D&D) of these facilities, secondary wastes will be generated. These secondary wastes include such things as anti-contamination clothing, wastewater, compactible trash, packaging materials, used equipment, tools, used chemicals, and a variety of other materials. The amount of secondary waste generated is dependent on how efficiently the project is planned and managed. Segregation techniques must be employed by project personnel in order to take materials and equipment into radiologically controlled areas and survey the materials back out as radiologically clean. By utilizing these techniques, materials are driven to the least-cost disposition, such as sanitary waste. Practicing segregation and practical decontamination will greatly reduce the amount of waste to be disposed, realizing a substantial cost savings.

Another effective concept employed at Fernald is the structuring of a dynamic Waste Minimization Program in order to set the paradigm shift in motion. The program must be comprehensive, integrated, and proactive in order to capture opportunities early in the CERCLA process. Waste Minimization opportunities must be identified during each phase of the CERCLA process by integrating waste minimization/waste management personnel into the project teams. Project and design engineers must also be trained in the application of waste minimization techniques.

Starting at the Remedial Investigation, waste minimization opportunities exist during sampling activities, lab analysis, and treatability studies. Waste characteristics, waste quantities and potential dispositioning options are identified during this phase. Waste minimization plays a key role in defining the number and types of options available for conducting these activities.

During the Feasibility Study phase, additional analysis are performed to identify waste treatment and disposition options. Secondary waste generation becomes important during this phase and must be properly quantified.

The more tangible opportunities for waste minimization occur during the Remedial Design/Remedial Action phase. Design/Engineer/Construction (DEC) teams should be formed which include waste minimization personnel. This allows involvement up front

in the design phase in order to coordinate, promote, identify, plan and implement all waste minimization opportunities. DEC teams offer invaluable access to all aspects of the remediation project.
The Waste Dispositioning Plan provided to the D&D contractor is another area where waste minimization opportunities can be identified. Segregation, decontamination, volume reduction and efficient packaging should all be discussed in the plan.
Finally, actual field presence of waste minimization/waste management personnel during project implementation is essential in order to assure all possible waste reduction techniques are utilized. This includes monitoring waste minimization boundaries, packaging activities, chemical usage, and segregation activities.
Effective project management has proven to be the most essential element in assuring waste minimization is applied to all remediation projects. Fernald has also determined that quantifying the actual amounts of wastes avoided or recycled/reused per project enables the effectiveness of the waste minimization program to be assessed. Quantifying these amounts was accomplished by developing specific performance measures for remediation activities. Typical waste minimization performance measures are based upon reductions in the amount of waste generated at a facility. At a remediation site this measure of effectiveness does not ensure proper accounting for reductions in secondary waste generation or recycle/reuse initiatives.
As an alternative to waste generation data, Fernald is tracking the initial (in situ) quantities of materials requiring management to the final quantities disposed, recycled, or reused from each remediation project. These three indices quantify the success of waste minimization applications and provide a benchmark for waste minimization in remediation activities. Remediation activities are tracked as nonroutine waste generating processes instead of as routine, on-going processes. Since each remediation activity is projectized, specific waste information is available. Activities are initiated with the approximate known quantities and types of waste in situ. This known quantity is used to calculate the following indices:
1. Recycle Index - This is calculated by taking the amount (volume or density) of the waste recycled over the amount of waste in situ.
2. Reuse Index - This is calculated by taking the amount (volume or density) of materials which are reused over the amount of waste in situ. This includes the reuse of equipment, supplies, and materials which can be quantified.
3. Disposal Index - This is calculated by taking the amount (volume or density) of waste that is disposed over the amount of waste in situ. This is tracked per waste type, such as demolition debris, metal, transite, and soil. This index enables the tracking of secondary waste generation, as well as packaging efficiencies and bulking factors.
Understanding what causes an increase or decrease in the amount of waste disposed leads to better project management. Tracking these three indices over time gives an accurate, graphical display for management to use in evaluating the waste management program. These indices should show a trend of more efficient management and cost savings as each project is completed and the lessons learned are applied to the next activity.
In conclusion Fernald has realized the importance of applying waste minimization principles to remediation activities. Shifting the focus from source reduction to recycle/reuse of primary wastes and minimizing secondary waste generation has accomplished measurable reductions in wastes shipped for disposal. The success of this effort communicates to other facilities gearing-up for remediation and D&D activities that waste minimization is relevant during remediation and can be applied through effective project management.

Session 62 -- Utility Experience in LLW Management
Co-chairs: C. Clint Miller, PG&E;
Mark Vandale, Yankee Atomic Electric
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NORTHEAST UTILITIES, MILLSTONE STATION EXPERIENCE WITH EICHROM INDUSTRIES' DIPHONIX SELECTIVE ION EXCHANGE RESIN IN LIQUID RADWASTE PROCESSING
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ABSTRACT
Radwaste system optimization efforts are intensifying as nuclear utilities are barred from low level waste (LLW) disposal sites and environmental and ALARA programs encourage a reduction in the curies discharged to the environment. The three nuclear units at Millstone station, GE BWR (unit 1), C-E PWR (unit 2) and Westinghouse PWR (unit 3), have completed a series of bench top and side stream pilot scale tests of Eichrom's Diphonix resin, a novel gel type ion exchange resin. This testing was part of an overall optimization of their radwaste systems including use of coagulants and cesium specific zeolite materials. By installing Diphonix resin, Millstone is addressing the goals of minimizing the spent resin generated from their liquid radwaste systems and minimizing the activity discharged into the environment from these systems while continuing to use existing capital equipment. Millstone Station's effluent summaries for 1993 show that greater than 50% of the activity discharged from each unit was from cobalt isotopes. Discharge of Zn-65 from unit 1 was the site's highest contributor to whole body dose. Each liquid radwaste system at Millstone station consists of a particulate removing filtration component and an ion removing ion exchange component. The ion removing component relies primarily on standard mixed hydrogen form cation and hydroxide form anion exchange resin beds and, in some cases, a synthetic cesium specific zeolite. The liquid radwaste processed by these systems contains sodium, calcium, chloride and sulfate ions at concentrations approximately one billion times those of the radioactive components in these systems*(1). The standard mixed bed resins are exhausted by exchanging the sodium, calcium, chloride and sulfate ions. The resin exchange capacity is overwhelmed by the common ions causing the radioactive components to pass through. The result is the need to replace resin beds at a much higher frequency than desirable.
Diphonix resin differs from typical cation exchange resin through its unique combination of diphosphonic acid and sulfonic acid functional groups which exhibit selectivity for Co, Zn and other transition metals over sodium and calcium*(2). Samples of radwaste liquids from each Millstone unit were passed through laboratory scale columns of Diphonix resin. These tests demonstrated that Diphonix resin was capable of removing all detectable cationic Co-58, Co-60 and Zn-65. In another test, a side stream was taken from the discharge of unit 1's radwaste system carbon bed effluent and passed through a column of Diphonix resin over a period of weeks. This test
is designed to measure throughput to cobalt or zinc activity breakthrough. Although the test is continuing, breakthrough has not been reached after passing in excess of 45,000 gallons per cubic foot of Diphonix resin. It is typical for standard cation resin to have cobalt or zinc activity breakthrough after passing 5000-10000 gallons per cubic foot. This significant increase in throughput for Diphonix resin over typical cation resin throughputs makes the use of Diphonix resin a preferred technology. Other key reasons Millstone has installed Diphonix resin are the reduction in curies discharged to the environment and the lower personnel exposure resulting from less frequent resin replacements and handling.
RADWASTE OVERVIEW
An increasingly important cost affecting domestic nuclear power plants is the handling and storage/disposal of low level radioactive waste, LLW. These increasing costs are triggering a conversion to products capable of processing greater radwaste liquid volumes per unit spent waste generated. Any change in processing technology should also reduce the curies discharged to the environment and the resulting personal exposure dose. Eichrom's Diphonix resin has these capabilities. Northeast Utilities Millstone Station has evaluated this resin for its radwaste systems using bench top and side stream pilot scale tests. The success of these tests led to the installation of Diphonix resin to process liquid radwaste at Millstone. Details of Millstone's evaluation are reported here.
LLW Disposal
As of June 30, 1994 thirty one states lost their access to the Barnwell repository for their LLW disposal. Barnwell is still open to Southeast Compact generators, but under South Carolina law the LLW facility is slated to close forever on December 31,

1995. Nuclear stations which do not have access to a disposal site are forced to store waste on site. Industry experts caution that this interim on-site storage is not long-term storage(3). The result of this situation is that each plant storing LLW on-site will likely have to handle the waste twice adding to the incentive to minimize the volume each plant produces. Current estimates for the cost of disposing spent ion exchange resin are $750 per cubic foot at Millstone Station. Other nuclear station's circumstances can result in a higher or lower cost ascribed to spent resin waste.

Radwaste Discharge Limits

Nuclear plants are well below the legal limits for activity discharged as specified by two parts of the Code of Federal Regulations (CFR). The radioactive materials release limits of Title 10 of the CFR, Part 20 (10CFR20) governs the concentrations which may be released, on an instantaneous basis, from the plant release paths at the site boundary. Appendix I of 10CFR50 contains additional release restrictions which govern cumulative releases throughout the year. This regulation requires the calculation of real exposure pathways to the general public outside the restricted area, based on actual radioactive releases to the environment. This regulation is more stringent than 10CFR20, limiting exposure to no more than 5 mrem a year, much lower than from background sources. Most nuclear plants desire to continue to reduce their radioactive discharges by employing new technology. An approach based on replacing existing resins with more selective resins could address the goals of reducing the quantity of activity discharged and reducing the amount of spent resin generated to do this.

Examples of metal isotopes in PWR radwaste are Co-58 and 60, Cs-137 and 134, Cr-51 and Mn-54. BWRs that are injecting zinc note that Zn-65 is a major component in their radwaste along with cobalt and cesium. Zn-65 is important because of its high relative contribution to whole body dose.

The Liquid Radwaste Process

During the operation of a nuclear power plant, various radioactive materials and chemical wastes are formed. These materials will be in the form of gases, liquids, or solids. These materials must be safely collected, processed and disposed of to minimize radiation exposure to plant personnel and to the general public. The liquid radwaste process essentially consists of collecting and treating the liquid from various operations including floor and equipment drains. The collection tanks feed treatment systems that typically include components of filtration and ion exchange. The ion exchange process is usually performed by a mixture of sulfonic acid functionalized polystyrene/DVB strong acid cation exchange resin in the hydrogen form and quaternary amine functionalized polystyrene/DVB strong base anion resin in the hydroxide form, hereafter referred to as the 'standard mixed bed'.

The selection of resins for radwaste processing can be improved in two key areas. First, the standard mixed bed is removing many ions that are not required to be removed such as sodium and calcium. The removal of these non regulated ions consumes the ion exchange resin's capacity. By comparison, a resin that is selective for the activated metal ions, present in much lower concentrations, would result in greater throughputs and thus generate less spent resin waste. Second, the sulfonic acid functional groups on the cation exchange resin are not always strong enough to remove the activated ions to the desired concentrations. Decontamination factors for activated cobalt range from 1 to 50 at Millstone Station using standard mixed bed resin. Millstone desires to improve on this level of decontamination by eliminating activated metals in their discharges to less than detectable levels whenever possible.

OBJECTIVES

The challenges faced by nuclear power station radwaste managers are clear:

1. Lower the radioactivity content of the plant's waste
2. Minimize the creation of solid waste requiring disposal, and
3. Accomplish both of the above without requiring expensive new capital equipment.

DIPHONIX RESIN

Diphonix resin represents a new technology specifically developed to accomplish these objectives. R&D Magazine recently recognized Diphonix resin as one of the 100 most technologically advanced products of 1994. The following highlight the chemical and performance attributes of Diphonix resin. (4,5)

Highly Effective Radionuclide Uptake

The primary exchange group in Diphonix resin is diphosphonic acid, known to form

very stable complexes with metal ions including zinc and cobalt. Table I shows that this functionality drives the concentration of metallic species to the lowest achievable concentration, on average 100 times lower than conventional cation exchange resins. This testing was performed with the resins treated with calcium to simulate resin that had been in use for some time. Distribution values for cobalt and zinc on Diphonix resin are 20,000 and 100,000 respectively. The distribution values can be considered to give an approximate idea of the selectivity of Diphonix resin and cation resin for the different ions relative to calcium from neutral solutions. The distribution value is defined in equation 1. Ao and Af are the metal concentration before and after contact with the resins. W is the weight (g) of the resin and V is the volume (mL) of the aqueous phase.

Eq. (1)

TABLE I

Selective Metal Capacity

The diphosphonic acid groups react selectively with metallic ions, and not with common, non-regulated species such as calcium, magnesium and sodium. Because radionuclides are present at extremely low concentrations, compared with typical sizable concentrations of common species, the ability of Diphonix resin to use capacity selectively makes a major impact on economics. Results of Millstone's side stream testing of Diphonix resin on unit 1's radwaste show significantly greater throughput per unit of resin than with conventional strong acid cation exchange resins. Details of this testing are provided latter in this paper.

Quick Reaction Kinetics

A second functional group, sulfonic acid, is also incorporated into Diphonix resin. This functional group provides for quick reaction kinetics, making the diphosphonic acid group readily available to target species. Excellent kinetics allows Diphonix resin to operate at high flow rates. This makes possible the use of existing plant resin vessels operating at current flow rates.

Gel Type Polystyrene/Divinylbenzene Polymer Matrix

This is the same base polymer combination currently used in most operating radwaste systems. Diphonix resin is compatible with existing resin vessels and underdrain systems. It is supplied as 20 to 50 mesh beads
for radwaste applications.

MILLSTONE RADWASTE CHEMISTRY

A review of the chemistry of the radwaste liquids to be treated is the appropriate first step in any radwaste system upgrade program. Figure 1 shows the total effluent activity in curries for each Millstone unit for 1993. The Fig. also shows the percentage that each isotope contributed to this total. The sum of Co-58 and Co-60 contributed greater than 50% of the effluent curies for each unit. Unit 1 is a 660 MWe GE BWR which went commercial in 1971. Its total 1993 liquid effluent activity was the lowest of the three Millstone units at 0.128 curies. Zn-65 was 23% of unit 1's activity and 1% of the site's activity but contributed 75% to the site's whole body dose of 0.1456 mRem. Due to environmental accumulation factors, Zn-65 contributes a proportionally larger dose than either cobalt or cesium nuclides. Unit 2 is a 875 MWe Combustion Engineering PWR which went commercial in 1975. Its total 1993 liquid effluent activity was 0.952 curies. Unit 3 is a 1149 MWe Westinghouse PWR which went commercial in 1986. Its total 1993 liquid effluent activity was 2.24 curies.

Fig. 1.

The activity data helps identify which metals to target for removal but does not establish what competing ions might be present and their relative concentrations. Table II provides data on the variation of ionic and radioactive constituents in Millstone unit 2 radwaste liquid. Variations in the content of liquid radwaste is caused by the wide variety of systems that flow into the radwaste system and the various operations that the unit goes through during the fuel cycle. The ratio of common ionic constituents to radioactive constituents is approximately one billion to one. It is this infinitesimal fraction of the total ionic content that requires removal prior to discharge.

TABLE II

The Multiple Challenges Of Radwaste Chemistry

Removal of all activated corrosion and fission products typically requires the availability of both an ion removing component and a particulate removing component. The focus of this paper is on the removal of ionic (dissolved) activity. It is

appropriate, however, to mention that Millstone has studied the use of coagulants as a means of removing colloidal constituents through a combination of cartridge filtration and organic carbon bed filtration. Unit 1 currently uses coagulant injection and carbon bed filtration as a means of enhancing activity removal. The ratio of activity present as a filterable solid to ionic species is variable. Current plant activities such as shut down and start up chemistry changes will likely impact this ratio. The key is to have a system in place that can handle either form.

The detailed listing of the constituents in liquid radwaste in Fig. 1 and Table II define the challenge of removing the activity. Whether an element is activated has no impact on the mechanism required to remove it. Diphonix resin should be a key component of any radwaste system desiring to remove ionic activity consisting of divalent, trivalent, tetravalent and hexavalent metals. The powerful complexing strength of Diphonix resin may also help remove complexed metals which might otherwise be unreactive with the standard mixed bed. An example of this is antimony. Some of the activity listed in Fig. 1, however, exists as anionic complexes which are not reactive with Diphonix resin. These anionic complexes require that an anion exchange resin continue to be used in the radwaste process. Millstone has also studied the use of cesium specific synthetic zeolite materials. Diphonix resin and standard cation resin both remove cesium. Millstone has concluded, however, based on laboratory and field testing, that the synthetic zeolite material results in greater throughput per unit material than either Diphonix resin or standard cation resin. The synthetic zeolite material has now become part of the radwaste systems in units 1 and 2 for cesium removal.

DIPHONIX RESIN TESTING

The data in Table I confirms that Diphonix resin possesses the necessary attributes to merit further study as a replacement of Millstone's cation exchange resin. The selectivity and reactivity of Diphonix resin for cobalt and zinc and other metals over sodium and calcium is superior to cation exchange resin. A test program was established by the Millstone chemistry department to evaluate the performance of Diphonix resin on radwaste liquid samples in laboratory scale columns operating at the same flux rate as the existing standard mixed bed system. These tests will focus on Diphonix resin's ability to completely remove activated metals and to process greater volumes of radwaste than conventional cation resins.

Test Column Scale Up

The initial work was performed using unit 2's radwaste. The Diphonix resin used was 20-50 mesh bead size. Bench scale parameters are as follows:

Radwaste Demineralizer Vessel: 42 in (ID) X 63 in height
Cross sectional area: 9.62 ft2
Influent Flux at 35 gpm: 35 gpm/9.62 ft2 = 3.64 gpm/ ft2 = 14.83 (mL/min)/cm2
Bench Test Vessel: 2.5 cm (ID) X 30 cm bed height
Bench Vessel Cross Sectional Area 4.91 cm2
Bench Vessel Flow Rate (14.83 mL/min/cm2)(4.91 cm2) = 72.8 mL/min

The test column bed height of 30 cm will likely result in faster breakthrough than would occur in the operating vessels. Bed depths of 90 cm are typical in operating vessels. The advantages of shorter test columns are less time and labor to achieve complete breakthrough curves.

Eichrom Cobalt Characterization

Initial tests of unit 2's radwaste showed that decontamination factors 10 to 20 for cobalt were less than expected based on earlier batch uptake work shown in Table I. A hypothesis was formed that a small non-cationic portion of the cobalt activity was unreactive with the powerful diphosphonic acid functional groups on Diphonix resin. Samples of this radwaste were sent to Eichrom for further study. Similar decontamination factors were achieved for the cobalt during the initial testing at Eichrom. Total decontamination of Mn-54 and Cs-137 through the Diphonix resin was achieved during the Millstone and Eichrom work. The Eichrom work (6) next characterizeed the species not complexed by the Diphonix resin and devised a procedure for its removal. This work concluded that all detectable cobalt could be removed by passing the radwaste through Diphonix resin followed by a strong base anion exchange resin in the hydroxide form. The

decontamination factor for this treatment was greater than 1000 based on the detection limits of the intrinsic germanium counter used. Other treatments using submicron filtration, macroporous resin, precipitation and digestion were incapable of improving the removal of this 5-7% fraction of cobalt activity that passed the Diphonix resin. The conclusion is that this radwaste sample contained both cationic and anionic forms of cobalt. To remove both cobalt species, Diphonix resin can be used as a replacement of the cation resin component of a standard mixed bed or, where multiple vessels are available, Diphonix resin can be used in series with anion exchange resin. Additional details on the use of Diphonix resin is found latter in this paper under the headings, 'System Design Upgrade' and 'Operational Action Plan'.

Unit 3 Test

Millstone also conducted laboratory column testing on a small volume of radioactive waste water from unit 3. Initial treatment was performed by coarse filtration and carbon. The water passed through the Diphonix resin contained activity from Mn-54, Co-58, Co-60, Cs-134, Cs-137 and Sb-125. In each case but antimony, Diphonix resin was able to remove all detectable activity. The sample from unit 3 apparently did not have any anionic cobalt present. A small fraction of the antimony was removed from the waste with the Diphonix resin.

Unit 1 Test

Laboratory testing of waste from unit 1 indicated that Diphonix resin was capable of removing Zn-65 from the waste stream. Because Zn-65 is the major contributor to off site dose, this Diphonix resin capability is a significant benefit. Approximately 20-25% of the zinc activity and 5-10% of the cobalt activity was removable on anion exchange resin. Together with Diphonix resin, total decontamination of the waste was accomplished.

A slip stream from the unit 1 radioactive waste system taken prior to the system's ion exchange vessels, was connected to a Diphonix resin column. This test was set up to quantify throughput to transition metal activity breakthrough. The same size test column was used as described earlier in the test column scale up section of this paper. The flux rate was 100 mL/min (equal to 5 gpm/ft2), slightly higher than with the laboratory test runs. Just as with the operating radwaste system, flow to the test column is intermittent based on processing batches of radwaste liquid. The test started on December 17, 1994. At the end of January, the Diphonix resin column had processed 45,000 gallons per cubic foot of resin without transition metal activity breakthrough. The test is continuing to be run. For comparison, standard cation resin typically processes between 5,000 and 10,000 gallons of unit 1 waste prior to transition metal activity breakthrough. The influent water conductivity during the test ranged from 140 to 1780 S. Figure 2 provides a graphical representation of the data collected from this test so far. Conductivity is reported as microsiemens 'S' and decontamination factors are listed as 'DF'. High conductivity water causes the effluent activity from the operating standard mixed beds to increase. This increase is due to influent activity not being held on the resin and the release of collected activity being bumped from the resin's exchange sites. Although the conductivity range indicates that other ions were competing for exchange sites, Diphonix resin removed 100% of cationic transition metals from the slip stream. This was verified by placing an anion resin column down stream of the Diphonix resin. No transition metal activity was measured.

Fig. 2.

SYSTEM DESIGN UPGRADE

Northeast Utilities is in the process of switching from two 40 cubic feet standard mixed beds to a system consisting of three 15 cubic feet volume, 24 inch diameter stainless steel vessels. Liquid radwaste is collected from various floor and equipment drains and fed through a spiral-wound cartridge filter prior to entering the series of three vessels. The primary vessel contains a cesium specific zeolite identified as D-230 by its supplier. The secondary vessel contains 10 cubic feet of Diphonix resin sandwiched between 2 cubic feet of 20 to 50 mesh granular activated carbon on the top and bottom. The final vessel in the series contains 10 cubic feet of Dowex SBR-C hydroxide form, type I, strong base anion exchange resin. Previously, when effluent concentrations of nuclides exceeded acceptable limits, the entire 40 cubic feet of mixed resin bed required disposal. The fact that the high effluent nuclide concentration was caused by exhaustion of either the cation exchange resin component or the anion exchange resin component was not consequential as it was

impractical to isolate the exhausted component of the mixed resin bed. The new system is more flexible and will generate less LLW.

OPERATIONAL ACTION PLAN

The new system provides Millstone with an improved ability to control radwaste costs and reduce curies discharged and personal dose. Routine laboratory analyses consist of a gamma isotopic analysis and conductivity measurements of the liquid prior to discharge. Table III outlines a series of additional tests which provide direction on which component of the system should be replaced once routine analysis shows isotopic concentrations exceeding desirable levels. These tests were developed based on the potential presence of a different species of activated elements. Gamma counting data does not provide information on the species of a particular isotope. The additional tests prevent the premature disposal of components of the radwaste system prior to the end of their useful life. The testing is performed on aliquots from the radwaste system discharge.

TABLE III

The samples from each test listed in Table III should be gamma counted. Decontamination factors (DFs) can then be calculated for each set of media tested. For instance, if the decontamination factor is maximized by test 5, it is likely that the anion exchange resin should be replaced. Chromium and iodine are usually present as anions. Their presence in the radwaste discharge indicate breakthrough from the anion exchange resin in the system. Cobalt can be present as an anion and may be the cause of reduced system DFs despite the fact that the Diphonix resin is removing all soluble cationic cobalt. If the DF in test 5 is higher than test 4 it is likely that the cobalt in the radwaste effluent is anionic.This result would signal the replacement of the anion resin if effluent activity justified this action.

SUMMARY

The Millstone testing of Diphonix resin for the treatment of liquid radioactive waste revealed several positive results:

The polystyrene-divinylbenzene base material and mesh size are the same as commonly used cation exchange resin, which is compatible with existing plant systems.

The effective capacity of Diphonix resin was much greater than standard cation resin because of demonstrated selectivity thus reducing the LLW generated.

Diphonix resin is capable of holding transition metals on the resin exchange sites during transients of high conductivity water.

Millstone focused their efforts to improve their radwaste systems by first identifying the major contributors to effluent activity and personal exposure dose. Non-activated elements present in the radwaste liquid were also identified. The data targeted cobalt and zinc activity for removal from waters containing high relative concentrations of common elements including sodium and calcium. These conditions favor the use of Diphonix resin for improving effluent water quality (lower activity) and reducing resin waste by treating more waste per cubic foot of resin then with standard cation resins. The testing led to the installation of Diphonix resin as part of a three bed radwaste system. This provides Millstone with a flexible design capable of handling a variety of radwaste streams. The authors conclude that a similar approach would be effective at other nuclear plants.

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LIQUID WASTE VOLUME REDUCTION WITH BITUMEN SOLIDIFICATION AT THE SURRY RADWASTE FACILITY

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ABSTRACT

In 1987, the JGC Corporation was contracted to engineer, design, construct, and operate an independent, dedicated radwaste treatment and decon facility at the Surry Power Station. Since beginning operations in 1991, more than 67.5 million liters (18 million gallons) of radioactive liquids have been processed with no detectable activity, excluding tritium, in liquid effluent. The key element in achieving this performance is the Bitumen Solidification System.

The Surry Radwaste Facility Bitumen Solidification System is capable of processing many types of radioactive liquids, sludge and resin slurries. The NRC approved Topical Report, "High Strength Asphalt Solidification Process for Low Level Radioactive Waste", and the Vendor Process Control Program (PCP) allows stabilization of Class B or C boric acid waste within a bitumen matrix. Because boric acid from the primary coolant let-down is not recycled at Surry, the Bitumen System processed approximately one thousand 200 liter drums of boric acid waste stream solids as Class-A waste during the first two and one-half years of operation.

To reduce the volume of waste generated, Virginia Power had a new PCP developed to process the waste as Class A. The Class A PCP has resulted in significant savings and benefits to Virginia Power.

INTRODUCTION

Virginia Power's Surry Power Station is a two unit Westinghouse PWR. Both units began commercial operations in the early 1970s. Within three years, the installed Liquid Waste Evaporator had failed and portable vendor supplied ion exchangers were placed in service to process liquid waste. These systems processed liquid waste in compliance with regulatory requirements, but resulted in high annual releases of radioactive materials to the environment. In 1985, Virginia Power implemented aggressive feasibility studies for the improvement of radwaste treatment processes. Visits and technical discussions were conducted with radwaste facility operators from the US, Europe and Japan. In these studies, Virginia Power focused on:

- Providing greater on-site storage capabilities;
- Reducing the volume of radwaste generated for disposal;
- Reducing releases of radioactivity and pollutants to the environment;
- Reducing radiation dose to operating and maintenance personnel;
- Providing reliable operation with proven state-of-the-art technologies; and
- Incorporating lessons learned from Virginia Power and other processors.

As a result of these studies and evaluations, Virginia Power completed a conceptual design for an integrated Radwaste Processing and Storage Facility. In 1987, the JGC Corporation was contracted to engineer, design, construct and operate the radwaste

facility. The facility systems were selected based on technical and economic evaluations of competing alternative technologies. The stand-alone, dedicated Surry Radwaste Facility (SRF) began operations in 1991. Since beginning operations, approximately 67.5 million liters (18 million gallons) of radioactive liquids have been processed and released with no detectable activity, excluding tritium. A key element to achieving this performance is the Bitumen Solidification System.

Why Bitumen Solidification?

Virginia Power selected the Bitumen Solidification System to process waste because:

The technology and operational experience proved very versatile and effective overseas and in the United States for processing boric acid, sodium borate, sodium sulfate, resins and various sludge,

The technology provided good volume reduction ratios and used inexpensive non-specification packaging materials,

The binder, High Strength Asphalt, is relatively inexpensive, leach resistant and has a high compressive strength. It is also readily adhesive, highly waterproof and durable,

The system design provided proven safety benefits for personnel and the environment by using moderate temperatures (under 230C) to produce a homogeneously mixed monolithic form and incorporating remote technology to support ALARA goals.

THE BITUMEN SOLIDIFICATION SYSTEM

The Bitumen Solidification System is composed of several subsystems, each performing a specific purpose to support processing the waste into a solidified product in accordance with Licensing Requirements for Land Disposal of Radioactive Waste, Title 10 of the Code of Federal Regulations, Part 61.

The subsystems include: (See Fig. 1)

Waste Feed Subsystem
Waste Pretreatment Subsystem
Bitumen Storage and Feed
Heating Fluid Subsystems
Thin Film Evaporator Subsystem
Distillate and Ventilation
Drum Handling Subsystem
Solvent Recovery Subsystem

Basic Principle

The System uses a chemical and mechanical process to uniformly disperse and solidify waste solids into a bitumen matrix. The Thin Film Evaporator (TFE) is the heart of the system. The TFE uniformly mixes waste and bitumen in the upper processing area at a predetermined ratio. Four mixing blades in the TFE rotate and cause the bitumen/waste mixture to be spread over the internal heated surface. The mixture is heated to 230C along its entire length with a synthetic heating fluid. The high temperatures evaporate free water from the liquid waste and bound water from the sodium borate compound. Heating also reduces the bitumen viscosity to ensure better mixing with the waste solids. Bow waves created by the rotating blades cause turbulent flow conditions which further improves evaporation (See Fig. 2). The mixture flows downward through the TFE by gravity to a bottom cone and discharge area. The cone provides a reservoir to hold the bitumen mixture while switching drums. The final product temperatures average 175C as it exits the TFE into a drum on a remotely operated conveyor. Solidification of the end product occurs upon natural cooling of the bitumen binder. The evaporated water flows upward through the blades to a moisture separator which strips liquid droplets from the vapor, returning them to the processing area. The vapor flows to a condenser.

WASTE PROCESSING EXPERIENCE

Virginia Power received NRC interim approval to solidify boric acid waste to a stable waste form in 1991. The final NRC approval followed in January 1993. Because boric acid from the primary let-down is not recycled at Surry, approximately one thousand 200 liter drums of Class A waste were generated to Class B standards during the first two and one half years of operation. While the bitumen system worked extremely well, it was believed that the process could be improved to reduce the volume of waste generated.

Virginia Power contracted the ADTECHS Corporation, owner of the approved topical report and Process Control Program (PCP), to develop a PCP to process the Class A waste to Class A standards and improve volume reduction. ADTECHS approached the NRC with the Class A PCP. The NRC's position was that the January 1991 "Technical

Position on Waste Form" Revision 1, provided adequate guidance on the characteristics expected of a Class A solidified waste form. The Class A PCP and waste processing requirements were reviewed by the State of South Carolina Department of Health and Environmental Control and Chem Nuclear and found to be in compliance with the Barnwell License. The Class A PCP was finalized in January 1994.

Class B PCP Experience

The Class B PCP:

Allowed a maximum solid-to-bitumen ratio of 38:62. The system operated at maximum target feed rates with the solids-to-bitumen ratio driving the system efficiency.

Processed an average of 946 liters of concentrated waste liquids at 10.4 percent solids to generate one bitumen drum. This resulted in an average volume reduction of 4.7.

Required the concentrated waste pH be maintained in the narrow range between 9.0 and 9.5.

Required 10 to 14 hours of cooling/hardening time for each bitumen drum. Resulted in the average contact radiation levels of the drums between 125-150 mrem/hr.

Class A PCP Experience

The Class A PCP:

Increased the maximum waste solid-to-bitumen ratio to 60:40 allowing the mechanical limitations of the equipment to drive the system efficiency.

Increased the average concentrated waste volume processed to generate one bitumen drum to 1325 liters at 11.1 percent solids. This resulted in a volume reduction of 6.6.

Expanded the concentrated waste pH range between 7.0 and 10.0.

Reduced the average cooling/hardening time to between 6 and 8 hours because of the increased solids loading. This resulted in less contraction of the final product.

Increased the average contact radiation level of drums to between 350 and 400 mrem/hr. Because processing evolutions are performed remotely, the higher dose rates did not appreciably increase dose to personnel.

Reduced the volume of waste generated by 25 to 30 percent, thus saving Virginia Power approximately 25 to 30 percent in shipping and burial costs. The waste reduction amounted to a savings of approximately 120 to 130 drums per year.

Effectively increased the drum storage area capability from 1.5 years (470 drums/year) to 2 years. The drum storage area can store up to 693 drums.

To ensure the waste solidified using the Class A PCP is Class A and not Class B, a comparison is made of the liquid waste batch isotopic analysis report to a list containing the highest isotopic activity of previous batches.

A simple calculation using these comparisons can determine potential Class B waste. The waste feed rate of the potential Class B waste can be reduced to ensure the solidified drum is Class A.

Other Experiences with Bitumen Solidification

Strong acids, such as sulfuric acid, tend to hold heat longer thus increasing the bitumen cooling/hardening rate.

A higher waste solids-to-bitumen ratio increases compressive strength and produces less contraction in the final product.

Solvent cleaning and hot water soaking is required every 80 to 100 drums. Extending the cleaning time may result in a higher product temperature, excessive TFE vibrations, high TFE Drive Motor amperage and high upper processing area dose rates.

Solvent cleaning reduces bitumen buildup in the distillate and ventilation pathways to improve evaporation. The solvent is non-hazardous. Each cleaning uses about 40 to 55 gallons of solvent. This solvent is reused several times and can be disposed of like oil.

Hot water soaking is performed after each solvent cleaning to dissolve boric acid and sodium borate crystals which plate out in the processing area. This reduces dose rates and also improves vapor flow out of the TFE.

SUMMARY

A Bitumen System is designed for personnel and environmental safety. It operates at moderate temperatures and incorporates remote technology to support ALARA goals. The Bitumen Solidification Thin Film Evaporator System is an excellent choice for processing concentrated liquid waste because of the economical packaging and binding materials used, the versatility of the types of waste that can be processed and the

competitive volume reductions achieved with both the Class A and Class B PCPs. Also, the processed drums can be easily stored and easily shipped. As operational experience continues and is further refined, additional economic advantages are sure to follow.

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NUCLEAR ELECTRIC'S APPROACH TO THE MANAGEMENT OF INTERMEDIATE LEVEL RADIOACTIVE WASTES

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ABSTRACT

Nuclear Electric (NE) is the major nuclear power plant operator in the United Kingdom (UK), and operates six Magnox and 5 Advanced Gas Cooled Reactor (AGR) stations all with twin reactors, a PWR, and has two Magnox stations in the early stages of decommissioning. Quite a range of waste types are produced, from lightly contaminated items suitable for shallow land burial to more active materials such as sludges and resins. Since 1959 Low Level Waste (LLW)* has been routinely disposed of to a shallow land burial site at Drigg in Cumbria, operated by British Nuclear Fuels plc, and this is the basis of the NE disposal strategy for this category of wastes. No disposal route is available at present for Intermediate Level Waste (ILW) and such wastes are currently accumulated in purpose built facilities. Eventually these wastes will be disposed of to deep land burial.

With the exception of some ILW which has already been treated and Magnox fuel element debris, the general approach for ILW is encapsulation in a cementitious matrix, either intimately mixed in the case of resins and sludges or grouted in the case of discrete solids. Wet slurry wastes are organic/inorganic ion exchange materials from pond water treatment, filter pre-coat material, sludges arising from fuel clad corrosion in storage ponds and from drains and laundry wastes. They are stored under water in special tanks, will be retrieved hydraulically and passed through a de-watering plant to ensure that the feed to the encapsulation plant has acceptable properties.

Magnox fuel assembly components are items which are removed at the station prior to transport of the fuel off-site for reprocessing, and are mainly magnesium alloys. As such they are amenable to a dissolution process from which the effluent can be discharged to sea following removal of insoluble material containing virtually all the activity. One plant is operating with this process.

NE is currently investigating a process in which ion exchange resins are destroyed by chemical means, resulting in a reduction in the predicted organic inventory of the repository and a significant reduction in the waste volume. The activity remains with the small amount of secondary waste residue which is encapsulated in cement as ILW.

NE is involved in the search for novel ion exchange materials which either show a marked increase in their adsorption capacity for particular chemical species and can therefore yield decreased arisings of ion exchange resin or show a high degree of specificity for a particular species. There is also the strategic element in guarding against the non-availability of currently used materials. Electrochemical

ion exchange is another area of interest in which the normal process is enhanced by the application of a potential. The process appears to be very versatile, however direct experience so far is limited. Ultrafiltration is being investigated as a means of removing colloidal or soluble radioactive species. These are either rendered insoluble by chemical addition or absorbed onto added seed material and removed by cross-flow filtration.

BACKGROUND

When the UK Government privatized the Electricity Supply Industry in 1990, the nuclear component of the Central Electricity Generating Board (CEGB) was retained in the public sector under the name Nuclear Electric (NE), with responsibility for operating Magnox and Advanced Gas Cooled Reactor (AGR) stations and for completion of the Sizewell B Pressurized Water Reactor (PWR). At its formation NE inherited from the CEGB a well developed strategy for the management of those radioactive wastes arising during station operation.

Subsequently the strategy was updated to take into account the new Company's objectives, and is kept under review to ensure that NE is able to take advantage of developments in waste management techniques and to respond to changes in legislation, etc. At all times NE has endeavored to produce a strategy which not only meets its own stringent requirements but which is consistent with the UK Government national policy. Reference (1) gives a recent statement on NE's radioactive waste and decommissioning strategy.

Previous documents relating to the NE waste management strategy deal primarily with waste management issues on a Company-wide or generic basis, though site specific aspects are mentioned as necessary, for instance in discussing potential shortfalls in accumulation capacity. A further and more detailed appraisal is currently under way and this will result in each power station having its own specific documentation detailing the waste management strategy for that site. This is being carried out partly at the request of the UK nuclear industry regulators (HM Nuclear Installations Inspectorate HMNII, Ministry of Agriculture, Fisheries and Food MAFF and Her Majesties Inspectorate of Pollution HMIP), but is timely for other reasons. Firstly great progress has been made in recent years in the development of technologies for the retrieval, processing and packaging of ILW and it is appropriate to consider the ramifications of this for individual sites. Secondly plans for the construction of a UK deep repository for the disposal of wastes are becoming more clear and it is useful to consider any effects this may have on current and future waste management proposals. Thirdly as the Magnox stations approach the end of their useful lives it is appropriate for NE to begin to plan for the retrieval and packaging of the radioactive ILW at its sites, converting strategies for the management of ILW into action plans.

INTERMEDIATE LEVEL RADIOACTIVE WASTE ARISINGS AT NE POWER STATIONS

Both wet and dry operational ILW arises as a result of spent fuel handling, plant maintenance and the treatment of gaseous and liquid effluent. To date these wastes have been accumulated in purpose built facilities, approved by the UK Regulator and in most cases sized to accommodate the predicted station lifetime arisings. The type of accumulation facility depends upon the waste type, with wet slurry waste being kept under water in tanks and other solid waste being stored dry. The initial characterization of ILW is into the following types:

- Wet slurry wastes
- Fuel assembly components
- Miscellaneous contaminated items (MCI)
- Miscellaneous activated components (MAC)
- Desiccant/catalyst/charcoal

Wet slurry wastes comprise organic and inorganic ion exchange materials, filter pre-coat materials often mixed with sand, sludge or ion exchange resins, sand and sludge materials which can contain detergents and other chemical agents from the laundry, wash areas, etc. and oily sludge. In the case of the Sizewell B PWR this category includes the primary and secondary ion exchange resins and boric acid concentrate from the evaporator. These wastes derive from the management of liquid effluent, which involves collection, monitoring, treatment for the removal of activity as appropriate, further monitoring and discharge against approved limits via the station cooling water outfall.

Fuel assembly components are items which are removed from the fuel, and for Magnox reactors this includes gas flow diverters, fuel spacing components and a variety of

end fittings. Most of these fuel components consist of magnesium with small alloying additions such as aluminum and zirconium, although other alloys and graphite are found at some stations. They are removed at the station prior to transport of the fuel for reprocessing. AGR fuel element debris consists of various steel and graphite components, many of which can be reduced in size for more efficient storage. Miscellaneous contaminated items (MCI) comprise a range of combustible and non-combustible materials including items of plant arising from maintenance and consumable items such as cartridge filters. Miscellaneous activated components (MAC) comprise irradiated reactor components including shield plugs, stand pipe assemblies, flux measurement probes, control rods and fuelling machine grabs. Desiccant is used to dry the reactor coolant gas on all AGR and some Magnox reactors, and catalyst only arises on AGRs. Alumina, silica gel and molecular sieve desiccants have been used and in the case of some AGRs desiccant containing platinum/alumina catalyst can arise as a mixture in the accumulation vault. The activity in the desiccant is dominated by tritium.

INTERMEDIATE LEVEL RADIOACTIVE WASTE MANAGEMENT AT NE POWER STATIONS

Active Effluent Treatment Plant (AETP)

All stations have plant items such as filters and ion exchange units which, together with the accumulation facilities and the discharge system, comprise the AETP. On some stations the AETP is located in a separate building connected to the reactor building by a pipe bridge or tunnel. Where appropriate collection of the primary wastes is with segregation on the basis of activity level and/or chemical composition, as this may dictate the treatment technique. The waste treatment produces secondary wastes.

In addition to the main purpose of filtering and monitoring all liquid effluent prior to discharge, the AETP will also provide a route for collecting backwashed sand or filter pre-coat material, used ion exchange resins from effluent treatment and pond water clean-up, sludges from tanks and similar materials, and directing them to storage tanks. When exhausted pond water ion exchange resin or sludge is sent to storage, the AETP would process any supernatant liquid prior to discharge. Active drain liquids are collected from areas such as the reactor cooling ponds and floor and sump drains from the AETP. These liquids would be treated to remove oil using a separator tank or coalescing plate separator, and particulate material removed in a sand pressure filter. The latter utilizes a bi-modal particle size distribution to give good filtration efficiency and long service.

Figure 1 shows a flow diagram for the liquid effluent treatment plant at the Heysham 2 AGR, which also has a number of waste accumulation facilities and a high degree of waste stream segregation.

Fig. 1.

Pond Water Treatment Plant (PWTP) Effluent

All Magnox stations have a pond water filtration loop to remove particulate material, and a bleed from this loop forms a second loop for chemical control. In the latter water is passed through cation and anion ion exchange units, which are preceded on some stations by an additional guard bed to remove caesium using a non-regenerable ion exchange material. AGRs have a similar pond water treatment loop with mixed bed ion exchange resins. These are not regenerated but replaced when exhausted and there is not in general a requirement for dedicated caesium removal plant due to the low incidence of AGR fuel clad failure.

Active Laundry Effluent

This is a high volume but low specific activity waste and treatment is usually filtration in deep bed filters containing anthracite, sometimes coupled with cartridge filters prior to monitoring and discharge. Ion exchange treatment is usually unnecessary for this effluent.

Tritiated Water

This arises from regeneration of the silica gel desiccant in the reactor coolant gas dryers on AGR stations. It is transferred to tanks or drums prior to sampling, monitoring and discharge against approved limits via the station cooling water system. No specific treatment is carried out.

Secondary Wastes

Wastes arise as a result of encapsulation plant operation, and consist of active drains, ion exchange and filter materials from the treatment of effluent before discharge, filters from the filtration of gaseous discharges and possibly residues of encapsulant. Also there will be sludge and residues from the treatment of Magnox

debris by the carbon dioxide dissolution process. These secondary wastes will generally be accommodated in the existing waste accumulation facilities, noting that some of these may be extended during station life.

Accumulation Facilities

Early accumulation facilities were simple shielded mild steel or concrete tanks, which often hold several related waste streams. More recent designs incorporate waste stream segregation, double containment and improved accessibility. Many of the accumulated sludges and resins on Magnox stations are notionally classified as ILW but it is likely that prior to retrieval they will be shown to be LLW and therefore disposable to the UK shallow waste disposal site after solidification. Since about 1990, a significant amount of these low level wastes have been successfully retrieved and immobilized in cement in a mobile plant, and sent for shallow disposal.

Historically the Magnox power stations have used a range of materials for effluent and pond water treatment, and these have not generally been segregated, but stored under water in shielded tanks. The AETP sludges, although nominally ILW may be LLW, and the radionuclides will be a mixture of activation and fission products with some actinides. The fission products, especially caesium 137, will dominate the activity in the resins.

Encapsulation Plant

NE policy is to use either fixed or mobile encapsulation plants depending on the particular circumstances at individual sites. Trawsfynydd Magnox power station, which, due to restrictions on discharges into an inland lake has always generated more active waste arisings during operation than coastal stations, has a fixed encapsulation plant (2) which operates with a polymer encapsulant. The other NE fixed plant is located at the Sizewell B PWR and is intended to encapsulate ILW arisings in cement on a routine basis for interim storage prior to disposal in the proposed deep waste repository. All other Magnox and all AGR stations are located on coastal or estuarine sites and therefore have lower secondary waste arisings due to the higher discharge authorizations. NE's strategy is that these stations will be served by mobile encapsulation plants, one for ILW and a smaller, simpler one for LLW.

It is anticipated that the mobile ILW encapsulation plant will be straightforward in operation, with emphasis on the quality of the product and the Quality Assurance (QA) record for each package produced. A cylindrical `liner', nominally of 2.6 m3 capacity, complete with a mixing paddle and lid will be removed from the store and moved into the active plant area through appropriate barriers. The waste is monitored into the drum and de-watered as necessary. With the mixing paddle rotating at the correct speed the ingredients of Ordinary Portland Cement (OPC), Blast Furnace Slag (BFS) and other additives will be metered into the drum and the mixing continued for the required time. When the mixing is stopped and the motor drive withdrawn, the drum is moved to another location within the active conditioning area for curing or setting. The drum contents are tested for set by the use of a weighted point and any surface (or bleed) water may be removed. The drum then passes to another location where an inactive grout is added to form a cap, and when this is set the lid is put in place and bolted or welded on. The surface of the drum may then be decontaminated if needed, the QA record checked and the drum removed to the on-site storage facility. The drum is identified by means of a laser etched alpha/numeric hexadecimal number which allows a high degree of internal cross checking to cover the possibility of operator error in entering the number into the computer.

Figure 2 shows the mobile ILW encapsulation plant with the transportable unit, liner handling equipment, the processing area and the processed waste storage facility. Figure 3 is a diagram of the NE large liner for the encapsulation of sludges and resins in the mobile plant. It is manufactured from thin stainless steel, the components being formed by rolling, spinning and pressing and welded together by full penetration welds. The materials, welding procedures, construction methods and inspection standards all meet the requirements for packaging radioactive wastes.

Operation of the mobile LLW encapsulation plant is similar in principle, but using 200 liter drums or large liners, and will not have to meet the stringent QA requirements of the ILW product. Experience has shown that the plant is reliable, easily decontaminated for transport, the product drums have very low external radiation levels and the dose to the operators is well within acceptable limits.

Fig. 2.
Fig. 3.
Inventory of Radioactive Wastes
Since the mid-1980s a national inventory of radioactive wastes has been prepared which is funded jointly by UK Nirex Limited and the Department of the Environment. The last completed inventory was in 1991 (3,4 and 5) and the data collection phase has been completed for the current inventory. For each identified waste stream, information is collected on existing and future stocks based upon current and estimated future arisings, physical and chemical properties, radiochemical content and conventional toxicity such as heavy metals and asbestos. All waste producers input to the inventory and the information is checked and validated as far as possible in order to form the basis of the quality assurance system under which wastes will be packaged for disposal. NE is responsible for nearly 600 separate waste streams, including future PWR sources and those which will arise during decommissioning.
NUCLEAR ELECTRIC EXPERIENCE IN MANAGING ARISINGS
Encapsulation in Cement
In common with many other waste producers the NE approach is to immobilize wastes in cement, typically a mixture of Ordinary Portland Cement (OPC) and Blast Furnace Slag (BFS). For slurry and wet wastes this is achieved either by metering the waste, in a pumpable form such as a slurry, into a waste container containing premixed BFS/OPC or alternatively adding the BFS/OPC to the waste, depending on the flow characteristics and the density of the waste material. In either case the contents of the waste container are continuously mixed by a rotating paddle which is left in the container and becomes part of the wasteform. For solid dry wastes, these are either grouted individually in a container or for material such as Magnox debris it can be compacted into a single billet and grouted. Grouting solid wastes sometimes requires a containment cage or restraint to stop items floating in the cement.
The main area of development work has been with wet slurry wastes, including ion exchange resins, sludges, slurries and filter pre-coat material. These wastes have a fairly wide range of chemical and physical properties, and have been addressed as individual wastes at present based on knowledge of their origin and operational practice on stations. However, as some of the wastes will arise as mixtures a methodology is being developed that will permit formulations to be specified for mixed wastes based on those for the individual wastes.
It is usually inappropriate to work at the development stage with actual radioactive wastes, and considerable effort is devoted to identifying suitable simulants. In some cases the wastes can be successfully simulated by non-radioactive unused material, however, this can represent a problem if the materials are no longer available. This is the case with some of the ion exchange materials used on Magnox stations. In the case of sludges from the AETP the composition is complex and a range of materials are assembled to simulate the waste successfully. However the materials may also suffer some form of chemical and/or physical degradation in use and this also has to be taken into consideration.
A formal procedure has been drawn up for formulation development work involving small and large scale testing of the formulation and evaluation of the product properties relevant for storage, transport and disposal. During the course of the development program a number of potential difficulties have been identified with certain wastes, such as the presence of boron which can delay the setting of cement and the presence of certain resins and filter aid materials which have a propensity to swell and cause the cement wasteform to break apart. All development work has followed a similar line, starting with small scale scoping trials to determine the overall performance of selected BFS/OPC mixtures. This may involve monitoring the time for a sample to set against pre-determined criteria of hardness, general appearance, the presence of free water on the surface (bleed water), and how long any water takes to be re-absorbed. The behavior of samples after prolonged immersion in water was also examined and samples prepared for long term monitoring. At this stage some optimization of the waste loading was carried out and any particular difficulties identified. This has on occasion led to the consideration of alternative formulations using specialized cements, materials other than BFS and additives to counter particular physical or chemical difficulties.
The next stage in the development work was larger scale production of samples for product evaluation testing, in which samples were subjected to a wide range of

standardized tests which have been developed for cement formulation development work. For example, the mixing behavior can be characterized by properties such as the torque required by the mixing motor and the viscosity of the mix. Samples were monitored for mechanical properties including compressive strength as measured by the transmission of ultrasound, elastic modulus, dimensional stability, and resistance to fragmentation under water immersion testing. Porosimetry has been carried out after typically 7 days and 90 days augmented with pore water analysis after 28 days and 90 days. Other samples were subjected to gamma-irradiation after 120 days curing, at a rate of about 2 x 103 Gy h-1 and to a total dose of 9 MGy. This represents an accelerated irradiation test in terms of dose rate but is to the estimated whole life dose to be received by the wasteform in the repository. All materials used in the development work are closely prescribed, and the radioactive wastes are represented by simulants which have been agreed to have the necessary physical and chemical similarities to the actual waste.

The underlying requirement in all the cement formulation development work has been to produce a formulation which is robust and can successfully encapsulate all the envisaged extreme compositions of a particular waste. The formulation should also provide the encapsulation plant with a satisfactory operational window within which the components for the encapsulation process can be accurately monitored over the required range. The achievement of these objectives will result in all the wastes being immobilized to form a strong monolith within the container, which is resistant to degrading effects of the environment and will therefore retain its integrity for the required period. This includes interim storage at the station, followed by transport to and possibly further storage at the repository, emplacement in the repository and backfilling. The integrity of the wasteform is also supported by the corrosion resistance of the container, and the overall Quality Assurance record of the package will demonstrate that it meets all regulatory requirements for transport to the repository.

Encapsulation in polymer

Polymer offers particular advantages including higher waste loading, increased processing rate and greater tolerance to variations in waste material and composition. Its major disadvantages are that it is much more costly than cement, and would increase the organic material content of the repository. For these reasons its use for wastes at sites other than Trawsfynydd is being kept under review.

Magnox Dissolution

There are significant amounts of Magnox fuel debris in accumulation facilities at Magnox stations, and some is in a partially corroded condition which could make retrieval more difficult. NE has developed a dissolution process in which the Magnox alloy is dissolved in carbonic acid, which is then filtered, checked for activity, treated by ion exchange where necessary and discharged to sea. The insoluble secondary waste, which represents several per cent of the original mass, contains most of the activity and is accumulated pending encapsulation in cement. A plant using this technology has been in operation for some time at Dungeness B Magnox power station, and Fig. 4 shows a cutaway view identifying the main plant areas. Reference (6) compares the dissolution process with encapsulation in cement and other options and (7) describes the process and equipment for the Dungeness B plant.

Fig. 4.

Safety Case for Continued Accumulation

As a condition of granting a site license the UK Regulators now require that all NE stations have a formal safety case and comprehensive safety case documentation for all safety related activities undertaken on site. This includes the radioactive waste processing and accumulation facilities at each station and covers both current operational safety and long term issues. For all facilities this requires the development of safety arguments for continued waste accumulation at power stations based upon Hazard and Operability (HAZOP) studies. In the case of Magnox stations the safety cases must be available on the Long Term Safety Review (LTSR) timescales laid down by the UK Secretary of State for Energy. In order to meet these timescales NE has developed a generic approach augmented by site specific documentation.

The CEGB strategy was based upon an assumed 30 year lifetime for Magnox plant and 25 years for AGR plant. NE aims to extend the life of these stations, possibly up to 40 years. A direct result of this extended lifetime is that for some ILW waste streams, where accumulation capacity would have been adequate, there is now the possibility

that action will need to be taken to provide more capacity or to undertake retrieval and processing.

Waste Sampling and Characterization

An essential input to the radioactive waste encapsulation development program and all aspects of waste management is the acquisition of sampling and analysis data to provide information on the physical, chemical and radiological characteristics of the waste. This will enable appropriate simulants to be developed for inactive test work, in addition to providing the database for the development of a methodology for the encapsulation of mixtures of wastes. An extensive program of sampling and analysis is both time consuming and expensive. Where several wastes have been accumulated in the same tank or vault over a period of time without mixing, there is also the question of correct interpretation of the data with respect to location within the tank particularly the depth. NE is therefore looking at operational records to identify any information which can be used to augment sampling and analysis data.

AGEING OF WASTES

The NE strategy for waste management takes account of ageing of both accumulation facilities and of the wastes themselves, noting that some wastes could age in a manner and to an extent that there may be difficulties to be overcome in the identification of a simulant or in the encapsulation formulation. Six main mechanisms are identified by which wastes may change their physical and chemical properties with time. Thermal degradation due to heating or thermal cycling of a waste can produce both physical and chemical changes. However, waste accumulation facilities remain below 50C and no evidence of alteration in waste phases has ever been noted by NE. Chemical degradation and corrosion can result from interactions between the waste components and degradation products from adventitious or naturally occurring chemicals such as water and oxygen. Metals will degrade by oxidative corrosion reactions whereas non-metallic inorganic compounds such as clays could degrade over a long period to form gels due to alkaline hydrolysis. Organic materials can degrade by a number of mechanisms. Radiolytic degradation of wastes such as inorganic ion exchange materials, desiccants and clays will produce minor structural changes and perhaps a small amount of oxygen. Organic ion exchange resins will show some loss of functionality and an increase in the level of cross linking, neither of which represents a particular problem. Common to both situations will be the radiolysis of water which will initially produce H2 and O2 as stable products although most of the latter may be scavenged by impurities. Microbiological degradation is not in general supported by inorganic materials and the enzymes that effect the degradation are more effective with simple organic molecules or those with a regular structure rather than those with irregular non-repeating structures. Physical mechanisms which result in damage to waste materials (which in this context constitutes ageing) would be known from plant operational records, and usually fall into the category of mechanical handling and compression due to the weight of accumulated waste in a facility. Both tend to cause the production of smaller particles. Physico-chemical mechanisms are processes such as Ostwald ripening which is mainly a characteristic of colloidal materials, coagulation and flocculation, crystallization, change in electrokinetic properties and loss of volatile species.

DEVELOPMENT AREAS

Desiccant Washing

The water content of AGR coolant gas is controlled using a silica gel desiccant with hot gas regeneration. Desiccant capacity falls during service and the drier beds are periodically replaced, giving rise to silica gel that may contain sufficient tritium to require classification as ILW. A process has been developed whereby most of the tritium is removed by water washing, allowing the desiccant to be classified and disposed of as LLW. The tritium is discharged through normal station discharge routes within authorization.

Ultrafiltration (UF)

Many radionuclides exist in aqueous streams in an insoluble form and under such circumstances UF can potentially play a significant part in their treatment. Unfortunately many of the species to be removed from NE's wastes are soluble or in microcolloidal form under normal operating conditions, and one possible solution is to alter the pH, for at higher pH's many metal ions are easily precipitated as their hydroxides. However there are disadvantages to this if the pH has to be raised significantly, both on the grounds of cost and ease of operation, and important

isotopes such as Cs137 would still not be removed. An alternative approach is to add small quantities of an additive or "seed" material onto which the radioactive species are absorbed, but which is itself insoluble.

A number of seed materials have been identified and are mostly inorganic substances with ion exchange properties typically used in a finely divided form at concentrations in the range 10-100 ppm. No single seed has been identified which can provide a satisfactory performance over a wide range of contaminants and the main thrust of the work has been to develop cocktails of seeds for this purpose. Laboratory work (8) has identified suitable chemical conditions in terms of seed mixture and composition, pH and the number of processing steps, by which PWR effluent might be treated in a UF plant. The results of a demonstration at pilot plant scale using selected seeding material has also been reported (9). The results have been substantiated using real effluent from the Doel PWR in Belgium and it is probable that the same methodology would yield satisfactory materials for Magnox effluent.

Wet Oxidation (WETOX) of Ion Exchange Resins

A number of methods have been suggested for reducing the organic content and volume of organic ion exchange resins, of which chemical oxidation is generally the most preferred because it minimizes secondary waste arisings. Development of a wet process is being pursued in the UK and the complete process involves a radioactive waste concentration step by evaporation and solidification of the residue. The process can therefore reduce waste volumes for disposal, improve encapsulated product quality and enhance the long term safety of geologic disposal by reducing the organic inventory of the repository.

NE is a partner in a process development program which includes small scale experiments with actual radioactive wastes (ILWs) as well as economic and safety assessments of the plant. However the main feature is the construction and operation of a mobile pilot plant that is undertaking radioactive demonstration runs in order to allow confident extrapolation of the process and costs to a full commercial scale. The program specifically addresses NE ILW ion exchange materials.

The adopted process employs an aqueous hydrogen peroxide feed to resins suspended in water at 100C and at atmospheric pressure. A dissolved metal ion catalyst, e.g. Fe2+ and/or Cu2+ is used. The main reaction products are CO2, H2O and mineral salts e.g. sulphate. The hydrogen peroxide process readily destroys about 95% of organic material in a batch process. The remainder can be destroyed but with increasing difficulty and expense. This is expected to be alleviated in the mobile plant by use of a semi-continuous approach. Other chemical, e.g. silver II oxidation, and physical, e.g. UV irradiation, processes may be applicable but tend to be expensive and/or at an early stage of development. On economic grounds a single step oxidation is preferred.

The active mobile pilot plant has been constructed within a single ISO container suitable for normal road transportation. The main plant items are the reaction vessel, batching vessel, condenser, condensate tank, scrubber, HEPA filter, pumps, agitator, air cooler and gas analyzers. Feeds of resin and peroxide will be semi-continuous but evaporative concentration of residue will be performed on a batch basis within the reactor vessel. The design throughput for resin is 0.2 m3 day-1 with an availability expected around 150 days year-1. Reference (10) describes the small scale and pilot scale work together with the conclusions reached for each main type of resin.

Electrochemical Ion Exchange (EIX)

This is an ion exchange process in which the movement of ions between solution and ion exchange material is controlled electrochemically, and is being developed for application to liquid waste streams arising at Magnox, AGR and PWR plants. The ion exchange material is regenerated simply by reversing the polarity of the electrodes and the volume of regenerant liquor is typically 1-2 bed volumes, suitable for direct encapsulation in cement. This contrasts with the behavior of conventional ion exchange materials which are often not regenerable (and hence give rise to appreciable volumes of spent waste which have to be disposed of) or require chemicals such as acid or alkali for regeneration. Each regeneration cycle produces at least a few tens of bed volumes of secondary waste and this requires further treatment before it can be encapsulated for disposal.

Novel Absorbers

Ion exchange materials currently used are quite adequate, however, there is a need

to monitor developments and identify alternative materials which can be specified should those presently employed cease to be manufactured, and to identify new materials with superior properties to those currently used. The important factors are an improved capacity such that rates of arising can be reduced, and improved removal of key radionuclides such as Co60 or Cs137 with ion specific absorbers to produce a stream with lower activity levels.
NE monitors these developments in order to be able to respond should any significant progress be made in an area of relevance to the Company, and also participates in a UK nuclear industry group which undertakes tests of new absorber or ion exchange materials. Such tests involve contacting the materials with several simulated radioactive wastes and measuring how much radioactivity remains in the solution at various times up to 24 hours. In the past five years a significant number of materials have been investigated, including a number of surface modified inorganic materials, complexing ion exchange resins and more general absorbers.
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WORKSHOP SUMMARY
Workshop Co-Chairs: J. S. Devgun, ANL;
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"HOW CLEAN IS CLEAN?"
CLEANUP CRITERIA AND PRACTICES
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This workshop was the third in the "How Clean is Clean?" series. Last year's workshop focused on cleanup standards, guidelines, and pathways analysis models, and the year before, on the statistical approaches to final release of cleaned-up sites. The number of sites requiring cleanup in the federal and private sectors runs into thousands, and the cleanup costs are projected to be in hundreds of billions of dollars. With cleanup actions at a number of the sites earnestly in progress, the objective this year was to focus discussion not only on the criteria but also on the practices, i.e., the application of the criteria in a cleanup project.
A panel of experts was convened to discuss and debate the issues related to the theme of the workshop. The panel included representatives from the U.S. Department

of Energy (DOE), U.S. Environmental Protection Agency (EPA), U.S. Nuclear Regulatory Commission (NRC), New York State's Department of Environmental Conservation (NYDEC), Canada's Atomic Energy Control Board (AECB), the International Atomic Energy Agency (IAEA), and the two workshop Co-Chairs.

This year a true workshop-like setting was also provided breaking out into three focus groups and then reconvening for a general discussion. Each focus group was led by one or two discussion leaders. The topics for group discussion were:

1) Risk-Based Criteria vs. Generic Concentration Limits
(Subtopics: Application in the Field - Pros and Cons, Site-Specific Criteria, Pathways Modeling, ALARA),
2) Criteria vs. Cost
(Subtopic: Impact on Project Cost of Cleanup Criteria Being Based on Risk, Technology, Background Level, Detection Limits, Regulations),
3) Who Should Decide?
(Subtopics: Federal Regulators, States, Local Communities, Technical Basis vs. Public Acceptance).

This new format provided an opportunity for the participants to fully contribute their own knowledge, experiences, and ideas, in addition to those of the panelists, discussion leaders, and workshop co-chairs. The primary purpose of this workshop was to provide a forum for discussion and debate to the approximately 70 people who registered for the workshop. A package of papers, notes, or viewgraphs was provided to each participant in advance to stimulate discussion.

In the panel discussions, Anthony Kluk of DOE discussed the DOE Order 5400.5 which lists the release guidelines used by DOE for decontamination of equipment, structures, and sites. Implementation of these guidelines is presented in DOE/CH 8901, "A Manual for Implementing Residual Radioactive Material Guidelines. A Supplement to the U.S. Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and SFMP Sites," June 1989. The release guidelines for surface contamination listed in DOE 5400.5 are generally consistent with standards established by the NRC and equivalent to Section 4 of Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors." The DOE applies these limits to non-reactor facilities as well as reactor facilities. These limits apply to interior equipment and building components that are potentially recoverable or reusable.

The guidelines for residual radionuclides in soil are divided into two groups. Generic guidelines exist for thorium and radium. These generic guidelines for Ra-226, Ra-228, Th-230, and Th-232 are: 5 pCi/g, averaged over the first 15 cm of soil below the surface; and 15 pCi/g, averaged over 15 cm-thick layers of soil more than 15 cm below the surface. (This guideline assumes secular equilibrium. If secular equilibrium does not exist, the limit is applied to the isotope with the higher concentration.)

Derived guidelines are used for other radionuclides for residual soil contamination. The basic dose limit, 100 mrem/yr, and the ALARA principle are the basis for these derivations and DOE/CH 8901 specifies the procedures to be used for this determination. External gamma radiation is limited to 20 R/h above background for an average level inside a building or habitable structure.

Practical implementation of the guidelines was discussed at two sites. One of these is Santa Susana Field Laboratory located in eastern Ventura County California. The other is the Laboratory for Energy-Related Health Research, which consists of several DOE-owned buildings located on a fifteen-acre site leased from the University of California, Davis.

The EPA representative discussed Risk Assessment Guidance for Superfund as described in report EPA/540/R-92/003. A demonstration calculation (by Steve Dean) with the risk assessment code RISKCALC (version 3.0) was presented for a hypothetical site with U-238 contamination.

Charles Willis of the NRC discussed the quarter-century-old guidance of Regulatory Guide 1.86 which continues to be used as new criteria are being developed. Despite its age, problems continue to be encountered with the interpretation and scope of this guidance.

Specifically, for Shoreham and for Fort Saint Vrain, the NRC staff concluded that, in accordance with the ALARA principle and for consistency with the criteria for other nuclides, the surface contamination criterion for tritium and for iron-55 should be increased and for Shoreham the maximum value was 200,000 dpm/100 square

cm. At Shoreham, the exposure rate above background from accessible surfaces was limited to 5 R/hr and bulk contamination was not allowed to exceed 8 pCi/g. For Shoreham, the DECON option was selected. The termination survey was conducted in accordance with the June 1992 draft of NUREG/CR-5849 and it entailed over 230,000 measurements.
Paul Merges (NYDEC) discussed cleanup guidelines [Technical and Administrative Guidance Memorandum (TAGM) 4003 Cleanup Guidelines for Soils Contaminated with Radioactive Waste] adopted by New York State in September 1993 for radioactively contaminated sites. The policy in the TAGM states that the total effective dose equivalent to the maximally exposed individual of the general public, from radioactive material remaining at a site after cleanup, shall be as low as reasonably achievable and less than 10 mrem above that received from background levels of radiation in any one year.
Four sites were reviewed: 1) EAD in Tonowanda, New York; 2) Off Site Properties in West Valley, New York; 3) Cintichem Site in Tuxedo Park, New York; and 4) Radium Chemical Site in Queens, New York.
Ivan Vovk of IAEA emphasized the need for internationally agreed criteria. Recently the IAEA has begun preparation of a report which intends to document the latest international approach to the development of cleanup criteria. The approach stems from ICRP publication 60 and the distinctions made between "practices" and "interventions." The objective is to develop a coherent framework through which a logical approach to decision making for all situations can be developed. An IAEA Working Group has been established and discussions are going on for developing criteria for "practice" and "intervention" situations.
Jas Devgun explained the relevant key recommendations of ICRP 60 and the distinctions between "practice" and "intervention." A paper on the topic is available in the Proceedings of Waste Management '92, Vol. 1, p. 499-503.
Bernie Zgola of AECB presented the Canadian approach to the material and site release based on case-specific dose optimization. Decommissioning case-studies discussed included Atomic Energy of Canada Limited (AECL) facilities at Tunney's Pasture, WR-1 Reactor, and the Elliot Lake decommissioning of mines and mills.
The discussion group format on the three topics provided an open forum for groups of participants to discuss and debate these topics. Donald Wood (Westinghouse Hanford Co.), Catherine Sigmon (ORNL), John Corley (Consultant), Adam Levin (TLG), and Lars Soholt (LANL) served as discussion leaders. Each group presented their key points when all participants reconvened for a general discussion. Risk-based cleanup criteria were generally favored by the participants rather than the generic limits. The Canadian approach of case-by-case was generally praised as a sensible approach. It was agreed that the public needs to be better informed of the cost to society of cleaning up sites to background levels for little or no additional benefits in terms of reduced risk. It was also highlighted that the local public at the site should be included in the decision making process from the very beginning of a cleanup project.
In the evaluations, a majority of the participants rated the workshop in the "very good" to "excellent" category.
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